UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C 20555-0001

May 13, 1997

MEMORANDUM TO: Stuart Richards, Chief
Operator Licensing Branch
Division of Reactor Controls
and Human Factors, NRR

FROM: George M. Usova
Operator Licensing Branch
Division of Reactor Controls
and Human Factors, NRR

SUBJECT: GFE RESULTS: APRIL 1997

The April 9, 1997 Generic Fundamentals Examination (GFE) was administered to 155 candidates at 25 facilities. The examination operated smoothly and without incident.

The summary statistical results follow:

	PWR	BWR
No. of examinees	84	71
Mean score	91.07%	88.73%
Median score	91%	90%
High score	98%	98%
Low score	82%	71%
Number/% of failures	0/00%	4/5.6%
Failures by facility:		Hope Creek - 1 (77%) Peach Bottom - 1 (71%) Clinton - 2 (77%, 76%)

The statistical results of this exam, e.g., mean scores and range are generally in line and stable with past exam performance. Overall exam difficulty level (i.e., mean score) is targeted at 87.00 and actual exam difficulty levels of 91 and 89 are sufficiently close to targeted goals and consistent with past GFE performance.

Some slight rise in mean scores should reasonably be expected given facility "last day" withdrawals. I note that in this examination all facilities combined withdrew, during the final week preceding the exam, a total of 22 candidates from their initial count of intended test takers. The initial count of intended test takers was 177 candidates; however, 155 candidates took the examination. The most common reason given for last minute withdrawals was that the candidates withdrawn were not prepared for the NRC/GFE given their performance on the respective facility audit examinations.

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ORB1

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Withdrawing a substantially large number of potentially poor performers from the GFE exam naturally excludes those potentially weak performers from affecting the GFE mean score since their inclusion in the exam would tend to have a depressing effect on the mean score. Conversely, the remaining examinees, who partook in the examination, likely represent a better prepared and more knowledgeable group of examinees. Hence, this more selective group of examinees will have the tendency to skew the mean score upward.

Facility Comments:

In summary, three BWR facilities made seven comments on five questions. Similarly, five PWR facilities made 18 comments on nine questions of which Beaver Valley, alone, accounted for 8 of the 18 comments.

Regarding the BWR facility comments, the contractor reviewed and researched each of the comments affecting five items. Based upon its own analysis (see Attachment 1 for a complete discussion of comments), the contractor recommended, for HOLB review and approval, that the grading of two items (60 and 86) be adjusted to accept two correct answers as follows:

	Question #	Change
BWR	60/88	Accept two answers, B or D
BWR	86/14	Accept two answers, A or B

The contractor further recommended that no answer key changes be made to the remaining three BWR items.

Regarding PWR items, the contractor again reviewed the comments affecting those nine items. Based upon its analysis (see Attachment 1), the contractor recommended, for HOLB review and approval, that the grading of four items (22, 37, 44, and 64) be adjusted to accept two correct answers and recommended that no answer key changes be made to the remaining five PWR items.

PWR	22/50	Accept two answers, B or D
PWR	37 (form B)	Accept two answers, A or D (form B only)
PWR	44/72	Change answer to B (for Beaver Valley and Vogtle only)
PWR	64/92	Accept two answers, A or C (for Ginna only)

In particular, items 44 and 64 above, were facility-specific exceptions that warranted an additionally correct answer for Beaver Valley, Vogtle, and Ginna only. On the hand, item 37 of Form B contained a mislabelled distractor option sequence where the letter "D" was in the "A" position. This unfortunate occurrence inadvertently caused only one candidate at Beaver Valley to transcribe "D" on to the answer sheet rather than his intended "A". In fairness, I agreed that the "D" response should be granted credit. As noted, however, this error only affected one candidate nationwide, Beaver Valley, and had no pass/fail consequence. See Attachment 2 for the

contractor's explanation of this occurrence. The contractor has committed to stronger quality control measures to ensure that errors of this type do not occur again.

NRC Resolution of Comments:

Headquarter's staff reviewed the contractor's BWR and PWR recommendations and concurred with the recommendations. The final answer keys were revised accordingly to accept answer key changes as recommended. See Attachment 3.

In keeping with our recent HOLB policy to provide comment resolution feedback to those facilities making comments, I directed the contractor to include, within the facility grade report, a copy of the specific NRC resolution(s) to comments made by that individual facility.

In summary, this examination administration was a successful one.

Attachments: 1. BWR/PWR Facility Comments/Resolution
2. Contractor explanation of mislabeled options
3. Final Answer Key

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ATTACHMENT 1

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—HOPE CREEK
EXAM—BWR FORM A/B
QUESTION: 60/88

A reactor has been operating at 100% power for eight weeks when a reactor scram occurs. The reactor is critical 6 hours later and power is increased to 100% over the next 6 hours.

How is core Xe-135 concentration behaving when power reaches 100%?

A. Xe-135 is building in toward equilibrium.

B. Xe-135 is burning out toward equilibrium.

C. Xe-135 is building in toward a peak value.

D. Xe-135 is burning out toward a minimum value.

ANSWER: B.

COMMENT:

Xenon concentration twelve hours after a scram with a subsequent startup and power escalation beginning six hours after the scram is difficult to predict. Xenon concentration for such a scenario would actually be determined by computer program. Hope Creek has limited experience with startups occurring within twenty hours of a full power scram. The student text (page attached) supports Selection "D" as the most correct answer. It also supports Selection "A" during a startup which quickly establishes a power level of greater than 10%. Depending on the exact circumstances, we believe "A", "B", or "D" could be considered correct. Given the stated scenario, we would not expect an operator to differentiate between selections at the recall level of knowledge.

RESPONSE:

Partially concur. Compared to the normal core xenon-135 behavior following a reactor scram from long term 100% power, this scenario would result in the following important deviations:

- Xenon would not reach its normal peak value following a scram
- Xenon would peak and turn sooner following the scram

Because of the ramp time associated with the power increase, the initial rate of xenon burnout would be less than the burnout rate if reactor power had been quickly increased to 100%. As a result, xenon may or may not decrease below the 100% equilibrium level before finally stabilizing.

In addition to the above, xenon would decrease for 4 to 6 hours after reactor power reaches 100% due to the differences in half-lives between I-135 and Xe-135. This means that xenon certainly will be burning out when power reaches 100%.

The knowledgeable examinee may justifiably conclude that, when power reaches 100%, xenon is either burning out toward a minimum value (if currently less than equilibrium) or burning out toward equilibrium (if currently greater than equilibrium). In either case, however, xenon will be burning out when power reaches 100%. Therefore, options B and D are both possible. On the other hand, option A is not possible because it states that xenon will be building in..

Based on the interim answer key, this question was answered correctly by 45/71 examinees and yielded a moderate positive discrimination index of +0.23. The answer key has been changed to accept both options B and D as correct answers.

FACILITY—HOPE CREEK
EXAM—BWR FORM A/B
QUESTION: 66/94

A reactor is operating at 80% power near the end of a fuel cycle. Which one of the following lists the typical method(s) used to add positive reactivity during a normal power increase to 100%?

A. Withdrawal of deep control rods and increasing recirculation flow rate

B. Withdrawal of deep control rods only

C. Withdrawal of shallow control rods and increasing recirculation flow rate

D. Withdrawal of shallow control rods only

ANSWER: A

COMMENT:

Near the end of fuel cycle, there may be no deep rods remaining in the core. This occurs just prior to coast down. Depending on how close the plant is to the end of cycle, shallow rods and recirculation flow rate may be used to add positive reactivity. Selection "C" could be considered as an alternate correct answer.

RESPONSE:

Do not concur. End of fuel cycle power coast down is an atypical operation that would have been identified in the premise of the question. Making the assumption that the plant was operating in coast down would be unwarranted and incorrect. Secondly, the premise of the question states that a normal power increase will occur. This is contrary to the assumption that coast down is occurring.

Based on the interim answer key, this question was answered correctly by 31/71 examinees and yielded a moderate positive discrimination index of +0.21. No answer key change is required.

FACILITY—HOPE CREEK
EXAM—BWR FORM A/B
QUESTION: 84/12

Which one of the following pairs of fluids undergoing heat transfer in identical heat exchangers will yield the greatest heat exchanger overall heat transfer coefficient?

A. Oil to water

B. Air to water

C. Steam to water

D. Water to water

ANSWER: C.

COMMENT:

When steam condenses to water the convection heat transfer coefficient is quite large. When no phase change occurs the convection heat transfer coefficient is very low. The answer on the examination key is "C" - steam to water. If the steam is saturated and rejects some of its latent heat of vaporization, "C" is correct. If the steam is superheated, which it must be if it is to reject energy and remain as steam, the correct answer is "D" - water to water. Selection "D" could be considered as an alternate correct answer.

RESPONSE:

Do not concur. The facility comment failed to provide an example of an application of a superheated steam-to-water heat exchanger. The typical BWR has several applications that use steam-to-water heat exchangers, e.g. feedwater heaters and the main condenser. In these applications, steam condensation causes a large overall heat exchanger heat transfer coefficient, thereby supporting option C as the correct answer. It would be unwarranted and incorrect for an examinee to assume a superheated steam heat exchanger application that is foreign to a BWR plant when several relevant applications exist.

Based on the interim answer key, this question was answered correctly by 29/71 examinees and yielded a nearly zero discrimination index of -0.02. No answer key change is required.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—LIMERICK
EXAM—BWR FORM A/B
QUESTION: 84/12

Which one of the following pairs of fluids undergoing heat transfer in identical heat exchangers will yield the greatest heat exchanger overall heat transfer coefficient?

A. Oil to water

B. Air to water

C. Steam to water

D. Water to water

ANSWER: C.

COMMENT:

The question should be deleted for three reasons:

1. There is insufficient information given. To make the calculation to determine respective overall HT coefficients (U), the candidate would need the film thickness and conductance for each of the conditions given. That data was omitted, forcing the candidate to make an estimate.

2. The answer "c" is not correct for all conditions. According to Granet (1980, pp 537) the estimated U for water to water can exceed the U for steam to water in the case of a tank heater. Since the type of heat exchanger was not specified, the candidate may make a reasonable assumption (See attached copy of the U estimation table).

3. The question is beyond the scope of required knowledge. Even given the tools and the data to make the determination, the K/As in Section 293007, HEAT TRANSFER and HEAT EXCHANGES do not support the question. The most closely related item, K1.06, requires the operator to "discuss the factors which affect heat transfer in a heat exchanger."

Given the need for supporting data, the lack of a clear tie to a K/A, and the reference information to the contrary of the answer, the question should be deleted.

Reference: Granet, I. (1980), Thermodynamics and heat power, Reston, VA: Prentice-Hall

RESPONSE:

Do not concur for the following reasons:

In response to facility comment 1, requiring examinees to calculate exact values of heat transfer coefficients is beyond the scope of the GFE. A general understanding of the heat transfer characteristics for the listed fluids is sufficient to determine the correct answer.

In response to facility comment 2, the question stated that ***identical*** heat exchangers were to be considered. The reference provided by the facility does not show comparison values for identical heat exchangers. Rather, the reference gives coefficient values for different applications. Consequently, the coefficients resulting from these applications cannot be directly compared. Instead, one can observe from the reference that steam-to-water heat exchangers can result in maximum coefficient values of 600 Btu/hr-ft^2-°F, whereas water-to-water heat exchangers can result in maximum coefficient values of only 275 Btu/hr-ft^2-°F. This is consistent with the text, Heat Transfer and Fluid Flow, General Electric, February 1985, page 7-27, which shows a typical overall heat transfer coefficient of 600 Btu/hr-ft^2-°F for BWR condensers and feedwater heaters, both of which are steam-to-water heat exchangers. Therefore, based on the relevant information listed in the reference the answer key (option C) is supported.

In response to facility comment 3, the referenced K/A does support the knowledge being tested by this question. This question is also supported by:

- 291006K1.03, *"Principle of operation of condensers"*
- 291006K1.10, *"Basic heat transfer in a heat exchanger"*
- 293007K1.07, *"Describe how the presence of gases or steam can affect heat transfer and fluid flow in heat exchangers."*

Based on the interim answer key, this question was answered correctly by 29/71 examinees and yielded a nearly zero discrimination index of -0.02. No answer key change is required.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—LIMERICK
EXAM—BWR FORM A/B
QUESTION: 92/20

Which one of the following describes the fuel-to-coolant thermal conductivity for a fuel assembly at the beginning of core life (BOL) as compared to the end of core life (EOL)?

A. Larger at BOL due to a higher fuel pellet density

B. Larger at BOL due to lower contamination of fuel rod fill gas with fission product gases

C. Smaller at BOL due to a larger gap between the fuel pellets and clad

D. Smaller at BOL due to a smaller corrosion film on the surface of the fuel rods

ANSWER: C

COMMENT:

The correct answer is "B" not "C." This question is essentially identical to an existing INPO exam bank question (Thermal Limits #238) which was also incorrect until December, 1996; the date of the issuance of the GF Bank update (See attached copy).

The question was corrected in response to feedback (See attached copy of the feedback form dated 3/31/95). The technical justification for modifying the answer for both the exam bank question and the GFE question is identical; the former having already been corrected.

RESPONSE:

Do not concur. The facility-provided reference material does not directly address how BOL fuel-to-coolant thermal conductivity compares to EOL thermal conductivity. It mentions factors that affect the MAPLHGR limit curve. The reference points out that thermal conductivity is a factor that influences the shape of the MAPLHGR limit curve. However, the curve is influenced by many factors, not just the change in thermal conductivity. Therefore, it cannot be presumed that a change in thermal conductivity alone is responsible for the observed change in the MAPLHGR limit curve. Likewise, it cannot be concluded that a decrease in the heat transfer coefficient of the fuel pin gases causes a reduction in the overall fuel-to-coolant thermal conductivity. Hence, this argument does not justify accepting a second correct answer.

The text, Heat Transfer and Fluid Flow, General Electric, February 1985, page 9-45 states "A short time after reactor startup the [new] fuel cracks radially and redistributes out to the cladding." When the fuel pellets come into contact with the cladding, thermal conductivity increases dramatically.

The above text, page 9-22, also states "The overall heat transfer coefficient *for regions the pellet is not in contact with the clad* is given by its conductance through the gases." This excerpt comes from a

section that is discussing the change in the MAPLHGR limit curve from MOL to EOL. This quote indicates that the fuel pellet maintains contact with the clad over core life. Because there is essentially no direct contact between fuel pellets and clad at the BOL, the direct contact that exists at the EOL causes overall fuel-to-coolant thermal conductivity to be higher.

Based on the interim answer key, this question was answered correctly by 46/71 examinees and yielded a small positive discrimination index of +0.08. No answer key change is required.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—QUAD CITIES
EXAM—BWR FORM A/B
QUESTION: 86/14

During full power operation, critical heat flux is most likely to occur in a

A. center fuel bundle with flow restrictions.

B. center fuel bundle without flow restrictions.

C. outer fuel bundle with flow restrictions.

D. outer fuel bundle without flow restrictions.

ANSWER: A.

COMMENT:

The provided answer key shows "A" as the correct answer. The facility requests that "B" also be considered as a correct answer for the following reasons. The question references the likelihood of critical heat flux occurring. The term "flow restrictions" is confusing and can be interpreted two ways, first as core orificing and second as some other form of flow blockage, such as debris. If "flow restriction" refers to additional flow blockage to an already orificed core then "A" is correct. If "flow restriction" refers to core orificing, then "B" is correct.

Reference(s): General Physics Corporation
BWR Generic Fundamentals
Thermodynamics
Chapter 8, Thermal Hydraulics, Rev. 1
Page 22

RESPONSE:

Concur. The text, Heat Transfer and Fluid Flow, General Electric, February 1985, page 8-40 refers to core orifices as "Local restrictions such as the inlet orifice...". Therefore, it is conceivable that an examinee might interpret the term "flow restrictions" as core orifices. If so, option B would be the correct answer.

Based on the interim answer key, this question was answered correctly by 47/71 examinees and yielded a small positive discrimination index of +0.17. All of the remaining 24 examinees selected option B. The answer key has been changed to accept both options A and B as correct answers.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—QUAD CITIES
EXAM—BWR FORM A/B
QUESTION: 92/20

Which one of the following describes the fuel-to-coolant thermal conductivity for a fuel assembly at the beginning of core life (BOL) as compared to the end of core life (EOL)?

A. Larger at BOL due to a higher fuel pellet density

B. Larger at BOL due to lower contamination of fuel rod fill gas with fission product gases

C. Smaller at BOL due to a larger gap between the fuel pellets and clad

D. Smaller at BOL due to a smaller corrosion film on the surface of the fuel rods

ANSWER: C

COMMENT:

The provided answer key shows "C" as the correct answer. The facility requests that "B" also be considered as a correct answer for the following reasons. The question references fuel to coolant thermal conductivity at BOL compared to EOL. The question refers to core life and not bundle life. Since a fuel bundle can remain in the core for several operating cycles, this can be confusing, causing answer to vary. If the time span referred to is bundle life, then "B" is correct since fission product gas contamination affects the fuel thermal limits at the end of bundle life. For a new fuel bundle "C" is correct. As the fuel ratchets and expands from neutron exposure it may come into contact with the clad. This would cause thermal conductivity to increase, but this occurs only early in bundle life.

Reference(s): General Physics Corporation
BWR Generic Fundamentals
Thermodynamics
Chapter 9, Core Thermal Limits, Rev. 1
Page 41

RESPONSE:

Do not concur. Regardless of which phrase is used (core life or bundle life), the intent of the question premise remains the same; to compare the fuel-to-coolant thermal conductivity of a fuel bundle before and after long term power production. Secondly, the comparison is made between only two specific reference points, BOL and EOL. The parameters that affect thermal conductivity ***during core burnup*** (e.g., contamination of fuel rod gases) are not relevant unless they support the final conclusion, that thermal conductivity is smaller at BOL. Option B states that fuel thermal conductivity is larger at BOL and , therefore, is incorrect.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

The references provided by the facility point out that thermal conductivity is a factor that influences the shape of the MAPLHGR limit curve. However, the curve is influenced by many factors, not just the change in thermal conductivity. Therefore, it cannot be presumed that a change in thermal conductivity alone is responsible for the observed change in the MAPLHGR limit curve. Likewise, it cannot be concluded that a decrease in the heat transfer coefficient of the fuel pin gases causes a reduction in the overall fuel-to-coolant thermal conductivity. Hence, this argument does not justify accepting a second correct answer.

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Based on the interim answer key, this question was answered correctly by 46/71 examinees and yielded a small positive discrimination index of +0.08. No answer key change is required.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—BEAVER VALLEY
EXAM—PWR FORM A/B
QUESTION: 14/42

Which one of the following will cause an upscale failure of a boron-trifluoride (BF_3) failed fuel detector operating in the proportional region?

A. The detector electrode high voltage power supply output has decreased 5% due to setpoint drift.

B. The detector chamber has become flooded with water due to leakage around the electrodes.

C. A power supply fuse in the amplifier circuit for the neutron monitoring instrument has opened.

D. A temperature rise has caused the gas pressure inside the detector to increase to within 5 psi of design pressure.

ANSWER: B.

COMMENT:

Most candidates had to answer this question by eliminating the distracters. No reference could be located that supports the key choice of B.

Most of the staff concur that a BF_3 proportional counter will fail if flooded with water but most would expect the instrument power fuses to open, causing a downscale meter response.

Recommendation: Delete from this exam. Modify for future use. Change choice B to read "The discriminator voltage level has decreased to 25% of its required level." OR "the detector develops a short between the electrodes.

Reference(s): Westinghouse, Radiation, Chemistry and Corrosion Considerations for Nuclear Power Plant Application, Page 5-22.

RESPONSE:

Do not concur. The proposed replacement options would be potentially correct answers for this question. However, each replacement option also has the potential of opening the instrument power fuses.

Second, this question should not be assumed to be faulted simply because the process of elimination was used by a number of examinees to identify the correct answer. Eliminating obviously wrong options can be an effective method for problem solving, especially for this question if the examinee

lacked the knowledge that borated water has higher conductivity than the inert gas used in the BF_3 detector.

Based on the interim answer key, this question was answered correctly by 63/84 examinees and yielded a moderate positive discrimination index of +0.25. No answer key change is required.

FACILITY—BEAVER VALLEY
EXAM—PWR FORM A/B
QUESTION: 22/50

A centrifugal pump is operating at rated conditions in an open system with all valves fully open. If the pump suction valve is throttled to 50% closed, pump discharge pressure will _____ and pump differential pressure will _____.

A. remain the same; remain the same

B. decrease; remain the same

C. remain the same; decrease

D. decrease; decrease

ANSWER: D.

COMMENT:

Too many assumptions are required to determine the correct answer. Specifying an open system without any system description presents choice "B" as possibly correct.

By assuming that the system determines pump discharge pressure, as it would discharging into the bottom of a large tank choice B would be correct. To determine that the pump differential decreases the candidate must assume suction pressure drops below the minimum NPSH.

Recommendation: Accept B and D for this exam. Avoid asking candidate to consider actions which are counter to "good operating practices" or include in the question" ... erroneously throttled...". Accompany the question with a diagram that better defines the pump/system arrangement. Possibly add initial suction pressures and temperatures that show a reasonable close proximity to saturation.

Reference(s): BV, Nuclear Operator Training, Thermodynamics, Chapter 4, System, Pumps and Valves.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

RESPONSE:

Concur. Depending on the assumption by an examinee both B and D could be correct. If the examinee assumed that partially closing the suction valve maintained pump net positive suction head (NPSH), then B is correct. If the examinee assumed that partially closing the suction valve reduced pump NPSH below required NPSH, then cavitation would occur and the answer would be D. Therefore, both B and D could be considered as correct answers.

Facility recommendations for question improvement are all valid.

Based on the interim answer key, this question was answered correctly by 29/84 examinees and yielded a near zero discrimination index of +0.01. The answer key has been changed to accept both B and D as correct answers.

FACILITY—BEAVER VALLEY
EXAM—PWR FORM B ONLY
QUESTION: 37

A centrifugal pump is taking suction from the bottom of a vented cylindrical storage tank that contains 100,000 gallons of water at 60 °F. A pressure gauge at the inlet to the pump indicates 40 psig. Over the next several days storage tank temperature increases to 90 °F with no change in tank water level and no change in head loss in the pump suction line.

Which one of the following is the current approximate pressure at the inlet to the pump?

D. 39.8 psig

B. 37.4 psig

C. 34.6 psig

D. 31.2 psig

ANSWER: A.

COMMENT:

Obvious Typographical Error

Choices lettered: D
B
C
D

Recommendation: Change first choice to "A"

RESPONSE:

Concur. This typographical error occurred only on the form B exam. It had the potential of causing an examinee to mark option D on the answer sheet when option A was being chosen. Therefore, because the answer is option A, both options A and D should be considered as correct answers.

Based on the interim answer key, this question was answered correctly by 74/84 examinees and yielded a high positive discrimination index of +0.39. Six of the ten examinees who answered incorrectly selected option D on form B. The answer key for form B has been changed to accept both options A and D as correct answers.

FACILITY--BEAVER VALLEY
EXAM—PWR FORM A/B
QUESTION: 44/72

While remotely investigating the condition of a typical normally-open motor control center (MCC) feeder breaker, an operator observes the following indications:

Green breaker position indicating light is lit.
Red breaker position indicating light is out.
MCC voltmeter indicates zero volts.
MCC ammeter indicates zero amperes.

Based on these indications, the operator can accurately report that the breaker is open and racked to _____ position.

A. the OUT

B. the IN

C. the TEST

D. an unknown

ANSWER: D.

COMMENT:

Two problems:

Defining "Remotely"
Defining "Typical"

If "Remotely" means control room, the MCC voltmeter and ammeter readings are not available. For some Breakers the test position will remove all remote (control room) light indication and others the light indication is maintained. Depending on assumptions concerning "Typical" and "Remote" choice "C" or "D" could be correct.

Recommendation: Drop from this exam. Recommend this question be removed from the examination bank, due to difficulty in defining "Typical". Various breaker types and control power schemes in use throughout the industry provide operators a widely varied view of what "Typical" means.

RESPONSE:

Partially concur. The term "remotely" is commonly used in the nuclear industry when referring to a location some distance away from the equipment being operated (such as the control room or another remote location). Regardless of where the remote station(s) is/are, the question provided the necessary information and should not have caused any confusion as to the source of the information.

Also, it is not clear how option C might be considered a correct answer. The comment states that, for some breakers, remote position indication will not be available if the breaker is racked to the TEST position. This would seem to indicate that the breaker must be racked IN. This would cause option B to be correct if the typical MCC feeder breaker at Beaver Valley loses remote breaker position indication when in the TEST position.

Upon request, the facility submitted electrical drawings for a sample MCC feeder breaker. These drawings show that remote breaker position indication is removed when the breaker is in the TEST position. Upon follow-up, the facility representative stated that the sample breaker position indication circuit is representative of most MCC feeder breakers at Beaver Valley. In addition, the facility provided follow-up documentation to clarify their position on this question. The follow-up documentation states "For Beaver Valley choice 'B' appears to be the correct answer."

Based on comments from this and another facility, it appears that this question is facility-specific. Therefore, because Beaver Valley has shown that their MCC feeder breakers lose remote position indication when in the TEST position, the answer to this question for Beaver Valley should be B.

Finally, the term "typical" is used to direct the examinee away from rare or uncommon examples of this breaker application. Without its occasional use, there could be cases where an exception would prevent testing some important knowledges.

Based on the interim answer key, this question was answered correctly by 36/84 examinees and yielded a very high positive discrimination index of +0.43. The answer key will be changed to accept option B as the correct answer for Beaver Valley.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—BEAVER VALLEY
EXAM—PWR FORM A/B
QUESTION: 60/88

A reactor had been operating at 50% power for two weeks when power was increased to 100% over a 3-hour period. In order to maintain reactor power stable during the next 24 hours, which one of the following incremental control rod manipulations will be required?

A. Withdraw rods slowly during the entire period

B. Withdraw rods slowly at first, then insert rods slowly

C. Insert rods slowly during the entire period

D. Insert rods slowly at first, then withdraw rods slowly

ANSWER: D.

COMMENT:

Although the candidates answers this question by considering the required control rod manipulations, most later stated that no rod motion was required to maintain reactor power stable.

Above the POAH steam demand controls reactor power and rods or boron control reactor coolant temperature.

Recommendation: Modify question for future use. Modify question to read, "In order to maintain Reactor Coolant temperature stable ...."

Reference(s): Westinghouse, Reactor Core for Large Pressurized Water Reactors, Pages 9-21 through 9-23.

RESPONSE:

Concur. The facility recommendation for question revision is valid.

Based in the interim answer key, this question was answered correctly by 82/84 examinees and yielded a small positive discrimination index of +0.12. No answer key change is required.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—BEAVER VALLEY
EXAM—PWR FORM A/B
QUESTION: 64/92

Why are burnable poisons installed in a reactor core?

A. To shield reactor fuel from thermal neutron flux until later in core life

B. To compensate for control rod depletion that occurs over core life

C. To flatten the axial thermal neutron flux distribution early in core life

D. To ensure that the control rods will be above the rod insertion limit when the reactor is critical

ANSWER: A

COMMENT:

Choice D appears to be an equally correct answer.

Purpose of Burnable poisons, includes offset of K_{excess}, when considered with soluble boron (limited by positive MTC) and control rods (limited by SDM and hot channel factor limits). Without poisons the reactor could be made critical with rods below insertion limit.

Recommendation: Accept 'A' or 'D' on this examination and replace distracter D for future examinations.

Reference(s): Westinghouse, Reactor Core Control for Large Pressurized Water Reactors, Pages 8-9 through 8-13.

RESPONSE:

Do not concur. It is true that burnable poisons permit a lower RCS boron concentration and consequently a more negative MTC. However, placing burnable poisons in the core does not ***ensure*** control rods will be above the rod insertion limit (RIL) when the reactor is critical. Rod position at criticality is a function of many core parameters, e.g., RCS boron concentration, core fission product poisoning, core burnup, residual burnable poisons, and others. The critical position of the control rods is estimated prior to a startup. If criticality is predicted with control rods below the RIL, then the operator will adjust RCS boron concentration to attain a higher estimated critical rod position. Therefore, RCS boron concentration is the only parameter that can be controlled to ensure that control rods are above the RIL when the reactor is critical.

Based on the interim answer key, this question was answered correctly by 20/84 examinees and yielded a moderate positive discrimination index of +0.24. No answer key change is required.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—BEAVER VALLEY
EXAM—PWR FOP.M A/B
QUESTION: 74/2

Consider a pressurizer containing a saturated water/steam mixture at 636 °F with a quality of 50%. If an outsurge removes 10% of the liquid volume from the pressurizer, the temperature of the mixture will _____ and the quality of the mixture will ____ _____. (Assume the mixture remains saturated.)

A. decrease; decrease

B. decrease; increase

C. remain the same; decrease

D. remain the same; increase

ANSWER: B.

COMMENT:

Not clear to all candidates that the pressurizer was half full of water and half full of steam, both at saturation. Some attempted to apply the stated 50% Quality to the steam space alone.

After assuming the steam had significant moisture content, they logically concluded that pressure would not drop until all moisture was gone (100% stm quality). Others that focused on just the quality of the steam space overlooked the obvious, "Outsurge removes liquid from the pressurizer."

Recommendation: Modify question for future use. See attached rewrite of question that avoids using the term Quality which is most commonly used to describe the amount of moisture (or lack of) in steam.

Proposed Replacement

Consider a pressurizer containing a steam bubble and a volume of water. Both the water and steam are uniformly at saturated conditions for 636°F. The volume of steam and water in the pressurizer are equal. If an outsurge removes 10% of the liquid volume from the pressurizer, the temperature of the pressurizer will _____, and the volume of the steam will be _____ than the volume of the water.

A. decrease, less

B. decrease, greater

C. remain the same, less

D. remain the same, greater

RESPONSE:

Concur. The proposed replacement contains some enhancements that will improve the clarity of the question.

Based on the interim answer key, this question was answered correctly by 52/84 examinees and yielded a small positive discrimination index of +0.11. No answer key change is required.

FACILITY—BEAVER VALLEY
EXAM—PWR FORM A/B
QUESTION: 84/12

Which one of the following pairs of fluids undergoing heat transfer through a heat exchanger will yield the greatest heat exchanger overall heat transfer coefficient?

A. Oil to water

B. Air to water

C. Steam to water

D. Water to water

ANSWER: C.

COMMENT:

This question asks about "Overall Heat Transfer Coefficient" but the "Preliminary Key" answer, "C" seems to indicate the question is asking about the heat transfer rate.

Overall Heat Transfer Coefficient for a Tube:

$$U = \frac{1}{h_1\left(\frac{r_1}{r_2}\right) + K\left(\frac{rm}{r_2}\right) + \frac{1}{h_2}}$$

Note that all components of this formula for heat transfer coefficient describes the physical arrangement and the film coefficients of the two fluids involved. This formula does not consider

whether the latent heat of evaporation is involved. Air-to-water could be the correct answer considering the very small film layer on the air side of the heat transfer boundary.

Recommendation: Modify question for future use. Modify the question lead to read. "..... yield the greatest heat exchanger heat transfer rate?"

Reference(s): BV, Nuclear Operator Training, Thermodynamics, Chap. 3 - Connection and Fluid Flow, Page 3.

RESPONSE:

Partially concur. While it would be correct to use the wording suggested in the comment, it is not necessary to change the current wording. The comment uses the equation for the overall heat transfer coefficient for a heat exchanger tube as justification. However, the question refers to the entire heat exchanger. The thermal conductivity of the heat exchanger tubes is not the only factor to be considered when evaluating the overall heat transfer coefficient of a heat exchanger in which fluids are flowing. More importantly, the convective heat transfer coefficients for the flowing fluids must be considered.

Based on the interim answer key, this question was answered correctly by 30/84 examinees and yielded a small positive discrimination index of +0.12. No answer key change is required.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—COMANCHE PEAK
EXAM—PWR FORM B ONLY
QUESTION: 37

A centrifugal pump is taking suction from the bottom of a vented cylindrical storage tank that contains 100,000 gallons of water at 60 °F. A pressure gauge at the inlet to the pump indicates 40 psig. Over the next several days storage tank temperature increases to 90 °F with no change in tank water level and no change in head loss in the pump suction line.

Which one of the following is the current approximate pressure at the inlet to the pump?

D. 39.8 psig

B. 37.4 psig

C. 34.6 psig

D. 31.2 psig

ANSWER: A.

COMMENT:

Question number 37 of the Form B examination has a typographical error.

Three of the four candidates using Form B correctly answered the question, circling the letter D by the correct response. When they copied their answer to the answer sheet, they filled in "D." See the attached pages from the student's examination.

Recommend that you accept both A and D for the Form B of this examination.

RESPONSE:

Concur. This typographical error occurred only on the form B exam. It had the potential of causing an examinee to mark option D on the answer sheet when option A was being chosen. Therefore, because the answer is option A, both options A and D should be considered as correct answers.

Based on the interim answer key, this question was answered correctly by 74/84 examinees and yielded a high positive discrimination index of +0.39. Six of the ten examinees who answered incorrectly selected option D on form B. The answer key for form B has been changed to accept both options A and D as correct answers.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—COMANCHE PEAK
EXAM—PWR FORM A/B
QUESTION: 84/12

Which one of the following pairs of fluids undergoing heat transfer through a heat exchanger will yield the greatest heat exchanger overall heat transfer coefficient?

A. Oil to water

B. Air to water

C. Steam to water

D. Water to water

ANSWER: C.

COMMENT:

This question asks for the combination of heat exchanger fluids yielding the greatest overall heat transfer coefficient. The key answer is **steam to water**.

The thermal conductivity of water is superior to steam, oil, and air (see attached Figure E.1). If the fluid on one side of the heat exchanger is steam, the overall thermal conductivity will have decreased from the same heat exchanger with water for the fluid. The drastic reduction of the heat transfer coefficient when Departure from Nucleate Boiling occurs is based on this phenomenon.

If the water on one side of the heat exchanger is undergoing a phase change (nucleate boiling, for example), then the overall heat transfer coefficient improves. The question does not give enough information for a candidate to consistently make that interpretation for **steam to water**.

Recommend that you change the answer key to accept either "**C. Steam to water**" or "**D. Water to water**."

RESPONSE:

Do not concur. The thermal conductivity values for steam and water are not the principle factors to be considered when evaluating the overall heat transfer coefficient of a heat exchanger in which fluids are flowing. More importantly, the thermal conductivity of the heat exchanger tubes and the convective heat transfer coefficients for steam and water should be considered.

It has been documented (Granet, I., Thermodynamics and heat power, 1980, page 537) that steam-to-water heat exchangers can result in maximum coefficient values exceeding 600 Btu/hr-ft^2-°F, whereas water-to-water heat exchangers can result in maximum coefficient values of only about 275

Btu/hr-ft^2-°F. This is consistent with the text, Heat Transfer and Fluid Flow, General Electric, February 1985, page 7-27, which shows a typical overall heat transfer coefficient of 600 Btu/hr-ft^2-°F for a BWR condenser and feedwater heaters, both of which are steam-to-water heat exchangers. Therefore, based on the maximum coefficients listed for various steam-to-water and water-to-water applications, option C is the only correct answer.

A phase change is an atypical occurrence for most water-to-water heat exchangers (an exception being a PWR steam generator). Such an occurrence would then give the heat exchanger the characteristics of a water-to-steam heat exchanger and would not result in a valid comparison.

Based on the interim answer key, this question was answered correctly by 30/84 examinees and yielded a small positive discrimination index of +0.12. No answer key change is required.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—GINNA
EXAM—PWR FORM A/B
QUESTION: 49/77

After the first fuel cycle, subcritical multiplication can produce a visible neutron level indication on the source range nuclear instrumentation following a reactor shutdown without installed neutron sources. This is because a sufficient source of neutrons is being produced by:

A. spontaneous neutron emission from control rods.

B. photo-neutron reactions in the moderator.

C. low level thermal fission in the fuel.

D. alpha-neutron reactions in the fuel.

ANSWER: B.

COMMENT:

Recommend deleting question. Dealt with source neutrons without installed sources. Installed sources were removed at Ginna for our Cycle 20 refueling (currently on Cycle 25) and the Westinghouse WCAP 14290 for fuel analysis states that the primary source of neutrons is spontaneous fission of CM-242 and CM-244. Therefore, no correct answer is given.

RESPONSE:

Do not concur.

The question does not ask for the *primary* source of source neutrons. The knowledgeable examinee should know that option B (photo-neutron reactions) is the only significant source of neutrons listed in the answer options.

Based on the interim answer key, this question was answered correctly by 51/84 examinees and yielded a small positive discrimination index of +0.12. No answer key change is required.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—GINNA
EXAM—PWR FORM A/B
QUESTION: 64/92

Why are burnable poisons installed in a reactor core?

A. To shield reactor fuel from thermal neutron flux until later in core life

B. To compensate for control rod depletion that occurs over core life

C. To flatten the axial thermal neutron flux distribution early in core life

D. To ensure that the control rods will be above the rod insertion limit when the reactor is critical

ANSWER: A

COMMENT:

Recommend answer C as a correct choice. Dealt with reasons for installing burnable poisons. Choice A, given as correct answer, describes what they do, not why they are installed. Ginna's Cycle 25 WCAP 14290 for fuel reload states that one of the reasons is to reduce axial power peaking, which is answer C. We use integral fuel burnable assemblies and only the center section of the fuel rod is coated. The primary reason is to reduce the amount of soluble boron needed to account for K excess at BOL, which was not one of the choices.

RESPONSE:

Concur. The facility-specific reference provided by the facility supports the facility recommendation to accept option C as a correct answer. Because Ginna was the only facility to provide justification for accepting option C, it appears that this function of burnable poisons is unique to Ginna.

Based on the interim answer key, this question was answered correctly by 20/84 examinees and yielded a small positive discrimination index of +0.24. The answer key has been changed to accept both options A and C as correct answers for Ginna.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—SALEM
EXAM—PWR FORM B ONLY
QUESTION: 37

A centrifugal pump is taking suction from the bottom of a vented cylindrical storage tank that contains 100,000 gallons of water at 60 °F. A pressure gauge at the inlet to the pump indicates 40 psig. Over the next several days storage tank temperature increases to 90 °F with no change in tank water level and no change in head loss in the pump suction line.

Which one of the following is the current approximate pressure at the inlet to the pump?

D. 39.8 psig

B. 37.4 psig

C. 34.6 psig

D. 31.2 psig

ANSWER: A.

COMMENT:

"A" is not shown as a possible selection. There are two "D's". Our proctors told candidates (who pointed out this) that the first "D" should be marked as "A" on the SCANTRON Form. The proctors did not make a general announcement so that we would not indicate someone else had arrived at the answer. Candidates should have determined or asked if the first "D" should be "A."

RESPONSE:

Concur. This typographical error occurred only on the form B exam. It had the potential of causing an examinee to mark option D on the answer sheet when option A was being chosen. Therefore, because the answer is option A, both options A and D should be considered as correct answers.

Based on the interim answer key, this question was answered correctly by 74/84 examinees and yielded a high positive discrimination index of +0.39. Six of the ten examinees who answered incorrectly selected option D on form B. The answer key for form B has been changed to accept both options A and D as correct answers.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—SALEM
EXAM—PWR FORM A/B
QUESTION: 64/92

Why are burnable poisons installed in a reactor core?

A. To shield reactor fuel from thermal neutron flux until later in core life

B. To compensate for control rod depletion that occurs over core life

C. To flatten the axial thermal neutron flux distribution early in core life

D. To ensure that the control rods will be above the rod insertion limit when the reactor is critical

ANSWER: A

COMMENT:

The answer on the examination key is "A" and it is correct. Selection "D" is a correct statement when one considers the long-life cores and the technical specification for moderator temperature coefficient (MTC). Burnable poison compensates for Kexcess at BOL, allowing a soluble boron concentration that supports MTC within technical specification limits. Without burnable poison, the control rod height at criticality would have to be very low to allow a boron concentration that supports MTC within technical specifications. We suggest that this question be revised prior to next use.

RESPONSE:

Do not concur. Placing burnable poisons in the core does not ***ensure*** control rods will be above the rod insertion limit (RIL) when the reactor is critical. Rod position at criticality is a function of many core parameters, e.g., RCS boron concentration, core fission product poisoning, core burnup, residual burnable poisons, and others. The critical position of the control rods is estimated prior to a startup. If criticality is predicted with control rods below the RIL, then the operator will adjust RCS boron concentration to attain a higher estimated critical rod position. Therefore, RCS boron concentration is the only parameter that can be controlled to ensure that control rods are above the RIL when the reactor is critical.

Based on the interim answer key, this question was answered correctly by 20/84 examinees and yielded a moderate positive discrimination index of +0.24. No answer key change is required.

NRC RESPONSE FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—SALEM
EXAM—PWR FORM A/B
QUESTION: 84/12

Which one of the following pairs of fluids undergoing heat transfer through a heat exchanger will yield the greatest heat exchanger overall heat transfer coefficient?

A. Oil to water

B. Air to water

C. Steam to water

D. Water to water

ANSWER: C.

COMMENT:

When steam condenses to water the convection heat transfer coefficient is quite large. When no phase change occurs the convection heat transfer coefficient is very low. The answer on the examination key is "C" - steam to water. If the steam is saturated and rejects some of its latent heat of vaporization, "C" is correct. If the steam is superheated, which it must be if it is to reject energy and remain as steam, the correct answer is "D" - water to water. Selection "D" could be considered as an alternate correct answer.

RESPONSE:

Do not concur. The facility comment failed to provide an example of an application of a superheated steam-to-water heat exchanger. The typical PWR has several applications that use steam-to-water heat exchangers, e.g. feedwater heaters and the main condenser. In these applications, steam condensation causes a large overall heat transfer coefficient, thereby supporting option C as the correct answer. It would be unwarranted and incorrect for an examinee to assume a heat exchanger application that is foreign to a PWR plant when several relevant applications exist.

Based on the interim answer key, this question was answered correctly by 30/84 examinees and yielded a small positive discrimination index of +0.12. No answer key change is required.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—VOGTLE
EXAM—PWR FORM A/B
QUESTION: 22/50

A centrifugal pump is operating at rated conditions in an open system with all valves fully open. If the pump suction valve is throttled to 50% closed, pump discharge pressure will _____ and pump differential pressure will _____.

A. remain the same; remain the same

B. decrease; remain the same

C. remain the same; decrease

D. decrease; decrease

ANSWER: D.

COMMENT:

We duplicated these conditions on our loop flow training device, which consists of an open tank as the suction source and the discharge collection point. All valves in the discharge and suction flowpaths of the centrifugal pump were fully opened after the pump was started. Discharge pressure and pump D/P were measured under full flow conditions. These parameters were measured again as a suction valve was partially throttled shut. The results were that pump D/P remained constant and pump discharge pressure decreased.

Based on this data, we believe that the correct choice on this question to be "B."

RESPONSE:

Partially concur. Depending on the assumption by an examinee both options B and D could be correct. If the examinee assumed that partially closing the suction valve maintained pump net positive suction head (NPSH), then B is correct. If the examinee assumed that partially closing the suction valve reduced pump NPSH below required NPSH, then cavitation would occur and the answer would be D. Therefore, both B and D could be considered to be correct answers.

Based on the interim answer key, this question was answered correctly by 29/84 examinees and yielded a near zero discrimination index of +0.01. The answer key has been changed to accept both B and D as correct answers.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—VOGTLE
EXAM—PWR FORM B ONLY
QUESTION: 37

A centrifugal pump is taking suction from the bottom of a vented cylindrical storage tank that contains 100,000 gallons of water at 60 °F. A pressure gauge at the inlet to the pump indicates 40 psig. Over the next several days storage tank temperature increases to 90 °F with no change in tank water level and no change in head loss in the pump suction line.

Which one of the following is the current approximate pressure at the inlet to the pump?

D. 39.8 psig

B. 37.4 psig

C. 34.6 psig

D. 31.2 psig

ANSWER: A.

COMMENT:

The correct answer was the first choice (39.8 psig) which apparently was mislabeled as choice D.

RESPONSE:

Concur. This typographical error occurred only on the form B exam. It had the potential of causing an examinee to mark option D on the answer sheet when option A was being chosen. Therefore, because the answer is option A, both options A and D should be considered as correct answers.

Based on the interim answer key, this question was answered correctly by 74/84 examinees and yielded a high positive discrimination index of +0.39. Six of the ten examinees who answered incorrectly selected option D on form B. The answer key for form B has been changed to accept both options A and D as correct answers.

NRC RESPONSE TO FACILITY COMMENTS FOR THE APRIL 1997 NRC GENERIC FUNDAMENTALS EXAMINATION

FACILITY—VOGTLE
EXAM—PWR FORM A/B
QUESTION: 44/72

While remotely investigating the condition of a typical normally-open motor control center (MCC) feeder breaker, an operator observes the following indications:

Green breaker position indicating light is lit.
Red breaker position indicating light is out.
MCC voltmeter indicates zero volts.
MCC ammeter indicates zero amperes.

Based on these indications, the operator can accurately report that the breaker is open and racked to _____ position.

A. the OUT

B. the IN

C. the TEST

D. an unknown

ANSWER: D.

COMMENT:

Question states that an operator is "remotely" investigating the condition of a typically normally-open MCC feeder breaker. The operator observes the green light is lit, red light is out, and MCC voltmeter and ammeter both read zero.

This question has two correct answers depending on how you interpret what the term "remotely" means.

If remotely means at the breakers' remotely controlled station (i.e., control room) then that fact that you have a green indicating light shows that the breaker has control power to the remote control circuit. This indicates that the breaker must be racked to the "in" position, which would make choice B correct.

If you interpret "remotely" to be at the breaker in question then choice D would be correct since indicating lights on the breaker cubical function in both the "test" and "in" positions.

Accept choices B and D as correct on this examination.
Other factors that provide additional confusion are:

MCC voltage and ammeter indications—where are you reading these from, locally at the MCC or remotely somewhere else?

MCC feeder breaker—implies the supply breaker to that MCC which is typically a 480 Vac switchgear breaker, which may be in a different room compared to the MCC.

RESPONSE:

Partially concur. The term "remotely" is commonly used in the nuclear industry when referring to a location some distance away from the equipment being operated (such as the control room). The term "local" would have been used if the indications were located at the breaker cabinet.

The comment states that the lit green indicating light indicates that the breaker must be racked to the IN position. However, no documentation was provided to support the comment. Upon request, the facility submitted electrical drawings for a sample 480 Vac MCC feeder breaker. These drawings show that remote breaker position is removed when the breaker is in the TEST position. Upon follow-up, the facility representative stated that the sample breaker position indication circuit is representative of most MCC feeder breakers at plant Vogtle

Based on comments from this and another facility, it appears that this question is facility-specific. Therefore, because Vogtle has shown that their typical MCC feeder breakers lose remote position indication when in the TEST position, the answer to this question for Vogtle should be B.

Regarding the other factors which caused confusion, it is not certain that the examinee would benefit from additional wording to address these issues. On the contrary, additional wording actually may reduce the quality of this question by introducing information that some examinees would find distracting.

Based on the interim answer key, this question was answered correctly by 36/84 examinees and yielded a very high positive discrimination index of +0.43. The answer key has been changed to accept option B as the correct answer for Vogtle.

ATTACHMENT 2