UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

NOV 7 1996

Neil S. Carns, President and
Chief Executive Officer
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P.O. Box 411
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SUBJECT: GENERIC FUNDAMENTALS EXAMINATION RESULTS

This letter forwards the results of the Generic Fundamentals Examination Section (GFES) of the written operator licensing examination that was administered on October 9, 1996, to nominated employees of your facility. We are forwarding the following items:

- the examinations, including answer keys;
- the results for your nominated employees; and
- copies of the individual answer sheets completed by your nominated employees.

We request that your training department forward the individual answer sheets and results to the appropriate individuals. It should be noted that the examination was administered in two forms, which were identical except for the sequence of questions.

In accordance with the Commission's regulations, 10 CFR 2.790, a copy of this letter and the examination and answer key will be placed in the NRC's Public Document Room (PDR). The individual results and answer sheets are exempt from public disclosure and, therefore, will not be placed in the PDR.

Questions concerning this examination should be directed to Dr. George Usova at (301) 415-1064.

Sincerely,

Kenneth E. Brockman, Acting Director
Division of Reactor Safety

Docket: 50-482
License: NPF-42

Enclosures: As stated

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FACILITY COMMENTS AND NRC RESPONSES FOR THE OCTOBER 1996 GFE

FACILITY -- WOLF CREEK

EXAM -- PWR FORM A/B

QUESTION: 60/88

A reactor is initially operating at 50% power with equilibrium core xenon-135. Power is increased to 100% over a 2 hour period and average reactor coolant temperature is adjusted to 588°F using manual rod control. Rod control is left in Manual and no subsequent operator actions are taken.

Considering only the reactivity effects of core xenon-135 changes, which one of the following describes the average reactor coolant temperature 6 hours after the power change?

A. Greater than 588°F and decreasing slowly

B. Greater than 588°F and increasing slowly

C. Less than 588°F and decreasing slowly

D. Less than 588°F and increasing slowly

ANSWER: A.

COMMENT:

This question deals with understanding when xenon will dip on a power increase from 50% to 100%. Answer "A" is stated as the correct answer; answer "B" should also be accepted based on the following:

There are two thumbrules used to estimate the time of the xenon dip. Either:

$.8 \times \sqrt{\text{Power Change}}$ from the INPO GFES bank (attached), or

$\sqrt{\text{Power Change}}$ from Wolf Creek's training material (attached).

Using the first thumbrule has xenon dipping at 5.7 hours, and using the second thumbrule has xenon dipping at 7.1 hours. It is not reasonable to have examinees determine the time of the xenon dip at the accuracy required to answer this question using a thumbrule. Estimating the effects of xenon either 4 or 8 hours following the power change would be more reasonable.

RESPONSE:

Do not concur. Following a power increase minimum core xenon will occur after 4 to 6 hours. For a 50% increase the required time is approximately 5 hours based on graphs provided in Westinghouse (Reactor Core Control for Large PWRs, 1983, p. 4-26) and General Electric (BWR Academic Series, Reactor Theory, 1984, p. 6-10a) Therefore, for the power change listed in the question, only option A is correct.

The facility comment stated that 7.1 hours should be accepted as the time to minimum core xenon based on a thumbrule in the facility lesson plan. However, the thumbrules in the facility training material refer to the number of hours to reach peak xenon following a power decrease. In fact, these thumbrules are most accurate when estimating the time to peak xenon following a reactor shutdown (or trip). They are not accurate when used to estimate the time to minimum xenon following a power increase.

Based on the interim answer key, this question was answered correctly by 61/124 examinees and yielded a moderate positive discrimination index of +0.21. No answer key change is required.

FACILITY -- WOLF CREEK

EXAM -- PWR FORM A/B

QUESTION: 88/16

Which one of the following must be present to prevent departure from nucleate boiling from occurring in a reactor core following a pressurizer vapor space instrument line rupture if the leak rate is less than normal makeup capability?

A. Reactor coolant pump flow capability

B. Pressurizer level in the indicating range

C. Emergency core cooling injection capability

D. Steam generator steaming capability

ANSWER: D.

COMMENT:

This question deals with determining what is necessary to prevent departure from nucleate boiling following a Pressurizer vapor space leak within the normal makeup capacity. Distractor "A" should also be considered correct because having RCP flow will increase the margin to DNB.

The attached training material states how flow oscillations, due to no forced flow, can lower the Critical Heat Flux required for DNB by as much as 40%. Also, Technical Specifications tout flow as being important when considering DNB.

RESPONSE:

Concur. An examinee knowledgeable in heat transfer and thermal hydraulics should be able to readily eliminate options B and C. Option B can be eliminated because it does not directly affect heat transfer conditions in the core. Option C can be eliminated because the leak rate is less than the normal makeup capability. Option D is the correct answer. Option A is the only remaining option that directly affects heat transfer in the core. Therefore, options A and D will be accepted.

Based on the interim answer key, this question was answered correctly by 48/124 examinees and yielded a small positive discrimination index of +0.09. The answer key has been changed to accept either A or D for full credit.

FACILITY -- WOLF CREEK

EXAM -- PWR FORM A/B

QUESTION: 96/24

Which one of the ? owing will prevent brittle fracture failure of a reactor vessel?

A. Manufacturing the reactor vessel from low carbon steel

B. Maintaining reactor vessel heatup/cooldown rates within limits

C. Maintaining the number of reactor vessel heatup/cooldown cycles within limits

D. Operating above the reference temperature for nil-ductility transition (RT_{NDT})

ANSWER: D.

COMMENT:

This question deals with the prevention of brittle fracture. Answer "D" is stated as the correct answer; answer "B" should also be accepted as correct because the training material implies brittle fracture is prevented by maintaining pressure and temperature to the right of curves based on heatup and cooldown rates (attached).

RESPONSE:

Do not concur. Option B does not mention maintaining pressure and temperature in accordance with the pressure-temperature curves for various heatup and cooldown rates. It refers to only heatup and cooldown rates (°F/hr). Simply maintaining heatup and cooldown rates within limits will not prevent brittle fracture. That is why pressure-temperature curves and overpressure protection systems were developed.

Based on the interim answer key, this question was answered correctly by 92/124 examinees and yielded a small positive discrimination index of +0.13. No answer key change is required.