PE nuclear practice exam
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About NCEES
NCEES is a nonprofit organization made up of the U.S. engineering and surveying licensing boards in all 50 states, the U.S. territories, and the District of Columbia. We develop and score the exams used for engineering and surveying licensure in the United States. NCEES also promotes professional mobility through its services for licensees and its member boards.

Engineering licensure in the United States is regulated by licensing boards in each state and territory. These boards set and maintain the standards that protect the public they serve. As a result, licensing requirements and procedures vary by jurisdiction, so stay in touch with your board (ncees.org/licensing-boards).

Exam format
The PE Nuclear exam is computer-based. It contains 85 questions and is administered one day per year via computer at approved Pearson VUE test centers. A 9.5-hour appointment time includes a nondisclosure agreement, a tutorial, the exam, and a break. You have 8.5 hours to complete the actual exam.

In addition to traditional multiple-choice questions with one correct answer, the PE Nuclear exam uses common alternative item types such as

- Multiple correct options—allows multiple choices to be correct
- Point and click—requires examinees to click on part of a graphic to answer
- Drag and drop—requires examinees to click on and drag items to match, sort, rank, or label
- Fill in the blank—provides a space for examinees to enter a response to the question

To familiarize yourself with the format, style, and navigation of a computer-based exam, view the video tutorials on the NCEES website.

Examinee Guide
The NCEES Examinee Guide is the official guide to policies and procedures for all NCEES exams. During exam registration and again on exam day, examinees must agree to abide by the conditions in the Examinee Guide, which includes the CBT Examinee Rules and Agreement. You can download the Examinee Guide at ncees.org/exams. It is your responsibility to make sure you have the current version.

Scoring and reporting
Results for computer-based exams are typically available 7–10 days after you take the exam. You will receive an email notification from NCEES with instructions to view your results in your MyNCEES account. All results are reported as pass or fail.

Updates on exam content and procedures
Visit us at ncees.org/exams for updates on everything exam-related, including specifications, exam-day policies, scoring, and corrections to published exam preparation materials. This is also where you will register for the exam and find additional steps you should follow in your state to be approved for the exam.
PE NUCLEAR PRACTICE EXAM
8. Assume that a particular PWR and a particular BWR both output 950 MWe net. The average temperature of the coolant in the PWR core is 580°F, and the average temperature of the coolant in the BWR core is 545°F. The PWR pressure is 2,200 psig. The temperature of the fluid in the pressurizer is ______________°F.

Enter your response in the blank.

9. A small sample of gallium-68 is located inside a close-fitting concrete cylinder. Ga-68 emits a 1-MeV gamma ray as it decays. The wall thickness of the cylinder is 26 mm. The buildup factor for the concrete cylinder is most nearly:

- A. 1.02
- B. 1.49
- C. 2.92
- D. 6.36

10. A 2-MeV photon is incident on a lead shield that has a thickness equivalent to 4 mean free paths. The thickness (cm) of the lead shield is most nearly:

- A. 1.48
- B. 1.94
- C. 4.00
- D. 7.78
38. Match each waste type with its most appropriate waste item.

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39. U.S. NRC Regulatory Guide 1.52 has certain requirements regarding the allowable mass of radionuclide per mass of filter media. To assess compliance with this regulatory guide for a set of post-accident filters, you are requested to compute the mass that resides on the filter media at 1 hour. The radioactivity is homogeneously dispersed within a building with a volume of 10,000 m³. The following data apply:

Radionuclide in air stream is I-134 with a half-life of 52.6 min.
There are $1 \times 10^6$ curies of I-134 radioactivity contained inside the building.
Air stream flow rate is 100 cfm.
Filter retention factor is 0.99.

The mass (g) of I-134 on the filter media is most nearly:

- A. $1.0 \times 10^{-5}$
- B. $4.8 \times 10^{-5}$
- C. $2.9 \times 10^{-4}$
- D. $1.0 \times 10^{-1}$
61. A solid target is used in an accelerator-driven system to produce a spallation-based neutron source. What are the most important material properties of the solid target to ensure a large neutron source?

Select the three that apply.

☐ A. Low neutron cross-section ratio, \( \alpha \)
☐ B. High neutron cross-section ratio, \( \alpha \)
☐ C. Low density
☐ D. High density
☐ E. Low atomic number
☐ F. High atomic number

62. Which of the following is the primary reason a breeder reactor does not thermalize the neutrons?

☐ A. To maintain the fission cross section of the fissile isotopes as high as possible
☐ B. To provide a longer neutron generation time
☐ C. To increase \( \eta \)
☐ D. To decrease the mass of fissile material required for a critical configuration

63. An 1,800-MWt (full power) reactor operates for 14 months with a capacity factor of 0.87. The number of fissions is most nearly:

☐ A. \( 1.80 \times 10^{27} \)
☐ B. \( 2.07 \times 10^{27} \)
☐ C. \( 2.38 \times 10^{27} \)
☐ D. \( 5.45 \times 10^{27} \)
75. During a loss of coolant accident (LOCA) at a commercial nuclear generating station, radiative heat transfer from the fuel cladding can become significant. The following information is provided:

The station is a 3,300-MWt PWR with 157 fuel assemblies.
Each assembly consists of a $17 \times 17$ rod array.
The rod pitch (i.e., rod centerline-to-centerline spacing) is 1.26 cm.
The equivalent core diameter is 3.0 m.
The active fuel length is 4.2 m.
The emissivity of the fuel cladding is 0.7.

Shortly after a reactor scram from a LOCA, the reactor power drops to 3% of full power. Assume that the core can be approximated as a solid cylinder and that its outer surface is at a uniform temperature. If the environment surrounding the core is maintained at 300°C, the surface temperature of the core periphery, which is radiating to the surrounding environment, is most nearly:

- A. 2,000 K
- B. 2,400 K
- C. 2,600 K
- D. 2,800 K

76. Which of the following parameters establish limits for core protection setpoints due to their relative levels of uncertainty?

Select the two that apply.

- A. Effective multiplication factor ($k_{\text{eff}}$)
- B. Burnable poison rod enrichment
- C. Neutron cross section
- D. Local power density
- E. Departure from nucleate boiling ratio
85. The configuration of a system is shown along with its fault tree logic diagram.

Match the appropriate logic gates to the fault tree diagram so that the fault tree correctly models the system (i.e., such that the system fails).
7. **Step 1:** Determine the appropriate equation: 
\[ P_f(PGA) = \ln(PGA/PGA_{\text{median}})/\beta \]

**Step 2:** Solve the equation for \( \beta \): 
\[ \beta = \ln(PGA/PGA_{\text{median}})/P_f(PGA) \]

**Step 3:** Determine \( PGA_{\text{median}} \).

- **a.** The median is the 50th percentile value.
- **b.** Thus, \( PGA_{\text{median}} \) is the \( PGA \) value for which \( P_f(PGA) = 0.5 \).
- **c.** Take 0.5 on the \( y \)-axis of the included graph to the curve and read the \( x \)-axis value for \( PGA_{\text{median}} \), which is about 0.63.

**Step 4:** Plug in the values to estimate \( \beta \):
\[ \beta = \ln(PGA/PGA_{\text{median}})/P_f(PGA) = \frac{\ln(1.0/0.63)}{0.86} = 0.54 \]

**THE CORRECT ANSWER IS: C**

8. The pressurizer is the reactor component that maintains the pressure in the primary loop of the PWR reactor. The pressure in the PWR core and the pressurizer are assumed equal. The coolant in the PWR core is subcooled liquid, whereas the fluid in the pressurizer exists as a mixture of saturated water and steam at 2,214.7 psia (= 2,200 psig + 14.7 psia). The temperature of the fluid in the pressurizer is obtained from the steam tables for the saturated liquid at 2,214.7 psia. Therefore, the temperature of the saturated steam/water mixture at 2,215 psia is 650.4°F.

**THE CORRECT ANSWER IS: 645–655°F**
The buildup factor is a function of the attenuating medium, the energy of the radiation source, and the thickness of the medium (refer to the *PE Nuclear Reference Handbook*).

**Step 1:** Determine the equation for buildup factor. Use the Berger's formula from the *PE Nuclear Reference Handbook* since it is the simplest equation to calculate the buildup factor.

\[ B = 1 + a \mu x e^{b \mu x} \]

where \( \mu \) is the linear attenuation coefficient of the medium, \( x \) is the thickness of the material, and the parameters \( a \) and \( b \) are functions of the material and the energy of the gamma rays.

**Step 2:** Look up the \( a \) and \( b \) coefficients based on the given information (i.e., 1-MeV photon energy and a concrete cylinder). See the *PE Nuclear Reference Handbook* for a table showing parameters for the Berger Form of air-kerma buildup factors in various media.

For 1-MeV photons in concrete: \( a = 1.27 \quad b = 0.032 \)

**Step 3:** Look up the mass attenuation coefficient \( (\mu/\rho) \) for 1-MeV photons in concrete. See the *PE Nuclear Reference Handbook* for a table showing mass attenuation coefficients for ANSI/ANS-6.4.3 Standard Concrete of Density of 2.3 g/cm\(^3\).

For 1-MeV photons in concrete: \( (\mu/\rho) = 6.369 \times 10^{-2} \) cm\(^2\)/g

**Step 4:** Calculate the linear attenuation coefficient.

\[ \mu = \left( \frac{\mu}{\rho} \right) \rho \text{, where } \rho \text{ for concrete is } 2.3 \text{ g/cm}^3 \]

\[ \mu = \left( \frac{\mu}{\rho} \right) \rho = 6.369 \times 10^{-2} \text{ cm}^2/\text{g} \times 2.3 \text{ g/cm}^3 = 0.1464 \text{ cm}^{-1} \]

**Step 5:** Calculate the buildup factor using the Berger's formula, where \( x \) is the wall thickness of the concrete cylinder. Ensure to use consistent units.

\[ B = 1 + 1.25 \times 0.1464 \text{ cm}^{-1} \times 2.6 \text{ cm} \left( e^{0.032 \times 0.1464 \text{ cm}^{-1} \times 2.6 \text{ cm}} \right) = 1.49 \]

**THE CORRECT ANSWER IS:** B
10. The mean free path is the most probable distance a photon will travel in a medium before experiencing an interaction. The reciprocal of the linear attenuation coefficient $\mu$ has units of length and is often called the mean free path (refer to the PE Nuclear Reference Handbook).

**Step 1:** Look up the mass attenuation coefficient ($\mu/\rho$) for 2-MeV photons in lead. See the PE Nuclear Reference Handbook for a table showing mass attenuation coefficients for natural lead with density of 11.35 g/cm$^3$.

For 2-MeV photons in lead: $(\mu/\rho) = 4.530E-2$ cm$^2$/g

**Step 2:** Calculate the linear attenuation coefficient.

$$\mu = \left(\frac{\mu}{\rho}\right)\rho$$

where $\rho$ for lead is 11.35 g/cm$^3$

$$\mu = \left(\frac{\mu}{\rho}\right)\rho = 4.53 \times 10^{-2} \text{ cm}^2/\text{g} \times 11.35 \text{ g/cm}^3 = 0.514 \text{ cm}^{-1}$$

**Step 3:** Calculate the value of one mean free path by taking the reciprocal of the linear attenuation coefficient.

$$1/\mu = 1/(0.514 \text{ cm}^{-1}) = 1.94 \text{ cm}$$

**Step 4:** Calculate the thickness of the lead shield in centimeters, which is equivalent to 4 mean free paths.

Lead thickness = $4 \times 1.94 \text{ cm} = 7.78 \text{ cm}$

THE CORRECT ANSWER IS: D
### PE NUCLEAR SOLUTIONS

**38.**

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**THE CORRECT ANSWERS ARE SHOWN ABOVE.**
Since the material is homogeneously distributed in the building, the volumetric specific activity of the air in the building may be determined as follows:

\[ A_v = \left( \frac{1 \times 10^6 \text{Ci}}{10,000 \text{m}^3} \right) \left( \frac{1 \text{m}^3}{35.314 \text{ft}^3} \right) = 2.832 \frac{\text{Ci}}{\text{ft}^3} \]

Convert cubic meters to cubic feet since the volumetric flow rate across the filter is in cfm. The filter media is exposed to this concentration of air. The filter retains 99% of the I-134 that passes through it. The volume that passes through it in time \( t \) is the volumetric flow rate \( F \times t \). By solving a first-order differential equation (or by inspection), it can be demonstrated that the activity on the filter at time \( t \) is:

\[ A(t) = A_v \cdot F \cdot t \cdot f \cdot e^{-\lambda t} \]

where:
- \( A_v \) = volumetric specific activity
- \( F \) = volumetric flow rate
- \( t \) = time
- \( f \) = filter retention factor
- \( \lambda \) = decay constant \( = \frac{\ln 2}{52.6 \text{ min}} = 0.013178 \text{ min}^{-1} \)

The activity on the filter at 1 hr is then:

\[ A(t) = \left( 2.832 \frac{\text{Ci}}{\text{ft}^3} \right) \left( 100 \frac{\text{ft}^3}{\text{min}} \right) (60 \text{ min}) (0.99) \exp \left( -0.013178 \text{ min}^{-1} \times 60 \text{ min} \right) = 7,629.4 \text{ Ci} \]

By definition, activity = (number of atoms) \( \times \) (decay constant) or \( A = \lambda N \). The activity in curies must be converted to disintegrations per min (dpm). Then solve for the number of atoms present, \( N \).

\[ 1 \text{ Ci} = 2.22 \times 10^{12} \text{ dpm} \]

\[ N = \frac{A}{\lambda} = \frac{7,629.4 \text{ Ci} \left( 2.22 \times 10^{12} \text{ dpm} \right)}{0.013178 \text{ min}^{-1}} = 1.2853 \times 10^{18} \text{ atoms} \]

Mass is computed using Avogadro's number and the molar mass of the nuclide, which for I-134 is 134 g/mol.

\[ \text{mass} = \frac{1.2853 \times 10^{18} \text{ atoms}}{6.022 \times 10^{23} \text{ atoms/mol}} (134 \text{ g/mol}) = 0.000286 \text{ g} = 2.9 \times 10^{-4} \text{ g} \]

THE CORRECT ANSWER IS: C
60. After the prompt drop in power when the reactor scrammed and after the shorter-lived delayed neutron precursors have decayed, the reactor power will decrease with a stable period of –80 sec such that

\[ P(t_2) = P(t_1) \exp \left( \frac{t_2 - t_1}{-80 \text{ sec}} \right) \]

where \( t_2 > t_1 \).

The time required for the reactor power to decrease by an additional factor of 300 is then calculated as

\[
\begin{align*}
  t &= t_2 - t_1 \\
  &= (-80 \text{ sec}) \ln \left[ \frac{P(t_2)}{P(t_1)} \right] \\
  &= (-80 \text{ sec}) \ln \left[ \frac{1}{300} \right] \\
  &= 456 \text{ sec} \\
  &\approx 460 \text{ sec}
\end{align*}
\]

THE CORRECT ANSWER IS: D

61. Spallation is a process in which fragments of material are ejected from a body due to impact. The spallation process, in contrast to fission, is not an exothermal process. Energetic particles (i.e., protons) are required to drive it. The neutron yield increases with the mass of the target nucleus; therefore, high-density, high-atomic-number targets will produce a prominent neutron source near term.

The neutron capture to fission ratio, \( \alpha = \frac{\sigma_c}{\sigma_f} \), should be kept low to avoid neutron capture in the target. A material like U-238 may absorb many neutrons while other uranium nuclides have much better ratios.

THE CORRECT ANSWERS ARE: A, D, F
A breeder reactor does not thermalize neutrons in order to increase $\eta$ (the number of neutrons released in fission per neutron absorbed by a fissile nucleus) so that on average more than one fissile atom is produced per fissile atom consumed. When $\eta > 2$, the possibility exists for breeding, in which more fissile nuclei are created than are consumed by the chain reaction. As shown in the highlighted portion below for the principal fissile nuclides, this condition exists only for fast neutrons (energy $> 0.1$ MeV), so breeder reactors are intentionally designed to not thermalize neutrons.

THE CORRECT ANSWER IS: C
63. The number of fissions that occur over a time period $T$ in a reactor operating at a thermal power (i.e., fission rate) $P(t)$ is given by

$$N(T) = \frac{1}{a} \int_0^T P(t) dt$$

$$= \frac{PT}{a}$$

where $a$ is a constant that gives the energy released per fission and $P(t)$ for this problem is a constant $P$. The number of fissions that occur during one year of constant operation at full power is

$$N(1 \text{ yr}) = \left( \frac{1,800 \times 10^6 \text{ J}}{\text{sec}} \right) \left( 365.25 \text{ d} \right) \left( 24 \text{ hr/d} \right) \left( 3,600 \text{ sec/hr} \right)$$

$$\div \left( 3.2 \times 10^{-11} \text{ J/fission} \right)$$

$$= 1.77 \times 10^{27} \text{ fissions}$$

This result must then be adjusted for operation at a capacity factor of 0.87 for 14 months.

$$N(14 \text{ months}) = (1.77 \times 10^{27} \text{ fissions}) (0.87) \left( \frac{14 \text{ months}}{12 \text{ months}} \right)$$

$$= 1.80 \times 10^{27} \text{ fissions}$$

**THE CORRECT ANSWER IS: A**
75. **Step 1:** Determine the equation for radiation emitted by a body (refer to the PE Nuclear Reference Handbook).

\[
\dot{Q} = (\varepsilon)(\sigma)(A)(T_{\text{surface}}^4 - T_{\text{env}}^4)
\]

- \( \dot{Q} \) = rate of heat transfer (typically units of W or kW)
- \( \varepsilon \) = emissivity of the body
- \( \sigma \) = Stefan-Boltzmann constant (5.67 × 10\(^{-8}\) \(\text{W}\text{m}^2\text{K}^{-4}\))
- \( A \) = surface area of the body
- \( T_{\text{surface}} \) = temperature of the surface of the body (in Kelvin)
- \( T_{\text{env}} \) = temperature of the surrounding environment (in Kelvin)

**Step 2:** Solve the equation for \( T_{\text{surface}} \).

\[
T_{\text{surface}} = \sqrt[4]{\frac{\dot{Q}}{\varepsilon \sigma A}} + T_{\text{env}}^4
\]

**Step 3:** Prior to calculating \( T_{\text{surface}} \), determine \( \dot{Q} \) and \( A \) from the information provided.

- \( \dot{Q} \) is determined from the power level after the scram (i.e., 3% of full power)

\[
\dot{Q} = (3,300 \text{ MWt}) \left( \frac{1 \text{E}6 \text{ W}}{1 \text{ MW}} \right)(0.03) = 9.9\text{E}7 \text{ W}
\]

- \( A \) is the surface area of a cylinder

\[
A = (\pi DL) + (2)\pi r^2 \quad \text{(Remember that the cylinder has a top and bottom.)}
\]

\[
A = (3.14)(3.0 \text{ m})(4.2 \text{ m}) + (2)(3.14)\left( \frac{3.0 \text{ m}}{2} \right)^2 = 53.72 \text{ m}^2
\]

**Step 4:** Calculate \( T_{\text{surface}} \).

\[
T_{\text{surface}} = \sqrt[4]{\frac{9.9\text{E}7 \text{ W}}{\left(53.72 \text{ m}^2\right)(0.7)(5.67\times10^{-8} \frac{\text{W}}{\text{m}^2\text{K}^4})}} + (300 + 273.15)^4
\]

\[ T_{\text{surface}} = 2,613 \text{ K} \]

**THE CORRECT ANSWER IS: C**
The local power density and the departure from nucleate boiling ratio bound the core protection setpoint analysis due to their relative levels of uncertainty. Both of these parameters involve comparing the actual peak power to the calculated or predicted peak power.

There is uncertainty in measuring the magnitude and precise location of peak local power density because the measured flux shape produced by the in-core detectors and the computer programs is based on the assumption that all fuel rods are identical. Variations from rod to rod along with eccentricity of the pellet-to-clad gap could make the actual peak power density greater than the measured.

The ratio between departure from nucleate boiling (DNB) and the actual operating local heat flux is called the departure from nucleate boiling ratio (DNBR). It may change axially and radially in the reactor. Limiting factors or safety factors are employed in core protection setpoint analysis as an allowance for calculational uncertainties from actual radial and axial peak power values.

THE CORRECT ANSWERS ARE: D, E

77. Step 1: Determine the appropriate equation. See the PE Nuclear Reference Handbook for discussion on limiting factors and peaking factor correction terms.

\[ F_{Q\text{-limit}} = \frac{\text{peak power}}{\text{average power}} \]

Step 2: Solve the equation for the peak linear heat generation rate.

\[ \text{peak power} = (F_{Q\text{-limit}})(\text{average power}) \]

Step 3: Prior to determining the peak linear heat generation rate, calculate the average linear heat generation rate by using the information provided.

\[ \text{average power} = \frac{(1,500 \text{ MW})}{(121 \text{ assemblies})} \left( \frac{1\text{E}6 \text{ W}}{1 \text{ MW}} \right) \left( \frac{1 \text{ kW}}{1,000 \text{ W}} \right) \left( \frac{11.5 \text{ ft}}{176 \text{ fuel rods}} \right) = 6.12 \text{ kW/ft} \]

Step 4: Calculate the peak linear heat generation rate. Remember that the hot channel factor limit of 2.00 was provided in the problem statement.

\[ \text{peak power} = (2.00) \left( \frac{6.12 \text{ kW}}{\text{ft}} \right) = 12.24 \text{ kW/ft} \]

THE CORRECT ANSWER IS: 12.24
For Train A to fail (Pump A/Valve A1), either the pump or the valve can fail (because they are in series); therefore, Gate 4 should be an OR gate.

For Train B to fail (Pump B/Valve B1), either the pump or the valve can fail (because they are in series); therefore, Gate 5 should be an OR gate.

For Train A/Train B to fail, both Train A and Train B must fail (because they are in parallel); therefore, Gate 3 should be an AND gate.

For the valve system C1/D1 to fail, both valve C1 and valve D1 must fail (because they are in parallel); therefore, Gate 2 should be an AND gate.

For the top event (System Fails/No Flow), either the right side or the left side of the system fails (because they are in series), which causes no flow; therefore, Gate 1 should be an OR gate.

THE CORRECT ANSWERS ARE SHOWN ABOVE.