PROBLEM 1-1N QUESTION

Worldwide Utilization Of Power Reactor Technology

Thermal Reactor Moderator-Coolant Matrix (attached page 2)

For each position, identify either:

1. The principal Technical reason that this moderator-coolant combination cannot be exploited, or

2. The name of one (or more) reactor plants that have been built using this combination.

References For Thermal Reactor Types

"List of operational Nuclear Power Plants, " *Nuclear News*, August 1992.

Dietrich and Zinn, *Solid Fuel Reactors*, Addison-Wesley Pub. Co., Reading MA, 1958.

Directory of Nuclear Reactors, International Atomic Energy Commission, Vienna, published annually.

Kuljian, *Nuclear Power Plant Design*, A.S. Barnes & Co., Cranbury NJ, 1968.

Safety Issues at Defense Production Reactors, National Academy Press, 1987.

Zinn, Pittman, Hogerton, *Nuclear Power*, USA, McGraw-Hill.

This is only a partial list of the resources available for this assignment. If there are any sources that you found to be particularly helpful, please give the name of the reference to the TA for future use.

WORLDWIDE UTLIZATION OF POWER REACTOR TECHNOLOGY
THERMAL REACTOR TYPES

<table>
<tr><td colspan="2" rowspan="2">Coolant
Moderators
↓</td><td colspan="2">Light Water</td><td colspan="2">Heavy Water</td><td rowspan="2">Organic
HB-40
Santowax-OM</td><td rowspan="2">Gas
Hydrogen,Nitrogen,
CO_2, Helium</td><td rowspan="2">Liquid Metal
NaK
Na</td></tr>
<tr><td>Pressurized</td><td>Boiling</td><td>Pressurized</td><td>Boiling</td></tr>
<tr><td colspan="2">Light Water</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td rowspan="2">Heavy Water</td><td>VESSEL</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>TUBE</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td colspan="2">Graphite</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td colspan="2">Beryllium</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td colspan="2">Organic</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
</table>

PROBLEM 10-10N QUESTION

Fully Developed Laminar Heat Transfer in a Circular Duct

For fully developed laminar flow in a circular duct with constant wall heat flux the Nusselt number value is 4.364

Identify all the assumptions made in arriving at this result.

PROBLEM 10-11N QUESTION

Thermal Analysis of a Lead-Cooled Reactor Fuel Assembly

An innovative fast reactor concept uses molten lead as the coolant with the small hexagonal fuel-assembly design shown in Figure 1. The geometry and operating conditions of the fuel assembly are described in Table 1. Each fuel pin consists of a cylindrical slug made of U-Zr with a stainless steel cladding. Since U-Zr swells significantly under irradiation, a relatively large gap must be provided for between the fuel slug and the cladding (Figure 1). The gap is filled with a "thermal bond" to prevent excessive temperatures in the fuel, when the reactor is at power. The thermal bond material is molten sodium. Useful properties for all materials in the fuel assembly are reported in Table 2 at the end of the problem statement.

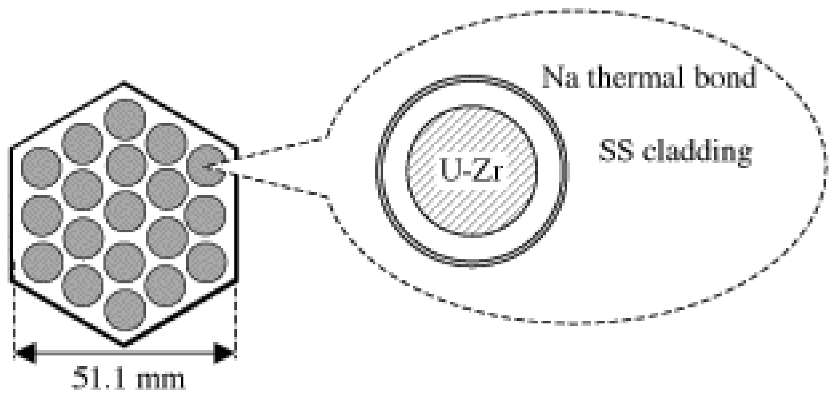


Figure 1. Cross Sectional View of the Fuel Assembly.

Table 1. Operating Conditions and Geometry of the Fuel Assembly.

Parameter	**Value**
Fuel assembly power (thermal)	456 kW
Inlet / outlet temperature	400°C / 550°C
Local / axial peaking factor	1.0 / 1.0
Fuel assembly inner width	51.1 (see Figure 1)
Number of fuel pins	19
Fuel pin pitch	11.0 mm
Fuel pin outer diameter	9.0 mm
Cladding thickness	0.6 mm
Fuel slug diameter	6.8 mm
Active fuel length	1.2 m

QUESTIONS

a. Select a suitable heat transfer correlation from your text book. (Assume fully-developed velocity and temperature profiles)

b. Evaluate the length of the entry region for the fuel assembly, and comment on the accuracy of the fully-developed velocity and temperature profiles assumption used in answering the previous question. Will the actual heat transfer coefficient be over- or under-estimated if a correlation for fully-developed flow is used? Explain.

c. Assuming a uniform axial power profile, sketch the coolant bulk temperature and the cladding outer temperature as a function of the axial coordinate. (Assume constant coolant properties)

d. Calculate the peak outer cladding temperature and the fuel centerline temperature. (In calculating the temperature drop across the gap, consider only heat conduction).

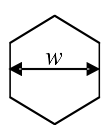
e. Suppose the plant operator increases the reactor power by 10% without changing the coolant mass flow rate and the inlet temperature. How do the peak cladding temperature and fuel centerline temperature change at these new operating conditions?

f. Wire wrapping is often used for fuel pin spacing in liquid-metal-cooled fast reactors. If this approach were used for the fuel assembly in Figure 1, would the coolant velocity, bulk temperature, heat transfer coefficient and pressure drop increase, decrease or remain the same? Why? (Assume that power, mass flow rate, inlet temperature and fuel pin geometry remain the same)

Table 2. Properties (all properties constant with temperature)

Material	**ρ (kg/m^3)**	**k (W/m·K)**	**μ (Pa·s)**	**c_p (J/kg·K)**
Molten Pb	10,400	16	1.9×10^{-3}	155
Stainless steel	8,000	14	/	470
Molten Na	780	60	1.7×10^{-4}	1,300
U-Zr	16,000	20	/	120

Hexagon area: $A = \dfrac{\sqrt{3}}{2}w^2$



Hexagon perimeter: $p = 2\sqrt{3}w$

PROBLEM 10-7N QUESTION

Reynolds Analogy And Equivalent Diameter Problem

Consider a uniformly heated tube (constant heat flux) of diameter 0.025m with fluid flowing at an average velocity of 0.5m/s. Find the fully developed heat transfer coefficient for two different fluids (Fluid A and Fluid B, whose properties are given in Table I) by the following two procedures:

Procedure #1 – Use only friction factor data. If you find this procedure not valid, state the reason.

Procedure #2 – Select the relevant heat transfer correlation.

In summary, you are asked to provided four answers, i.e.,

	FLUID A	FLUID B
PROCEDURE #1	h = ?	h = ?
PROCEDURE #2	h = ?	h = ?

TABLE I

FLUID PROPERTIES	FLUID A	FLUID B
k W/m˚C	0.5	63
ρ kg/m^3	700	818
μ kg/ms	$8.7\text{x}10^{-5}$	$2.3\text{x}10^{-4}$
C_p J/kg˚C	6,250	1,250

PROBLEM 10-8N QUESTION

Turbulent Heat Transfer Coefficient Calculations

The friction factor for a flow channel at a velocity of 10 m/s under fully developed turbulent conditions is 0.014. Find the approximate heat transfer coefficient under these same conditions assuming the fluid is: (1) water at 315˚C and (2) sodium at 538˚C (see *Nuclear Systems*, Vol. 1, p. 455 for properties).

PROBLEM 10-9N QUESTION

Equivalent Diameter And Reynolds Analogy Problem Involving A Fuel Element

A liquid sodium test reactor fuel element configuration of equivalent diameter = 0.01m with a unique spacer is proposed for application in an innovative light water core to be designed for long fuel cycle. Hence, the tightly packed, uniquely spaced fuel configuration of the sodium reactor is to be used in this water reactor core.

Friction factor test results in sodium are available at Re = 10^5 indicate that f = 0.08. It is desired to find the turbulent heat transfer coefficient for this fuel element configuration at Re = 10^5 in the water reactor core. Relevant properties are given in Table I. Is it possible to achieve the desired prediction? If so, make the prediction. If not, explain why you think it cannot be done with the information given.

Table I. Fluid Properties at Operating Conditions

Fluid Properties		Sodium	Water
k	W/m˚C	62.6	0.57
ρ	kg/m^3	818	740
μ	kg/ms	2.3×10^{-4}	9.6×10^{-5}
C_p	J/kg˚C	1250	5.4×10^3

PROBLEM 11-9N QUESTION

Wall Friction Components For Liquid And Vapor Only

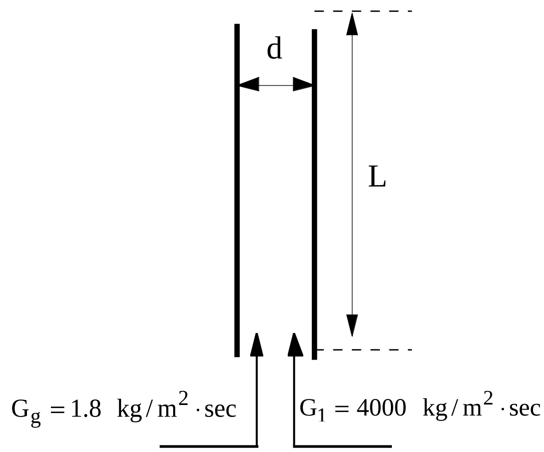
Propose an approach to compute the wall friction components for liquid and vapor only.

PROBLEM 11-11N QUESTION

Calculating Void Fraction In Adiabatic Steam-Water Flow

Consider an adiabatic tube in steady state steam-water upflow under the following conditions:

p = 1 atm
d = 0.01 m
L = 1 m
ρ_ℓ = 1,000 kg/m^3
ρ_g = 1,000 kg/m^3
h_{fg} = 2,246 kJ/kg
k_s = 0.025 W/m·K
μ_g = 12x10^{-6} N·sec/m^2



QUESTIONS:

A. Find the void fraction. Use at least two different methods to calculate your answer, and compare the results.

B. Now calculate the void fraction for the case of liquid downflow and vapor upflow. If your calculations indicate that the tube is flooding, verify this by applying an appropriate flooding correlation.

PROBLEM 11-12N QUESTION

Critical Flow During a Small-Break LOCA in a BWR

A small break (10 cm^2) occurs at a certain location on the coolant recirculation line of a BWR. Calculate the mass flow rate at which the coolant is discharged through the break into the containment. Use the following three models:

1) Non-equilibrium model for an orifice (L/D~0).
2) Non-equilibrium model for a short discharge nozzle (L/D~2).
3) Equilibrium model with Fauske's assumption for the slip ratio.

Assumptions:

- The coolant inside the primary system can be modeled as saturated liquid water at 6.9 MPa (1,000 psi).
- Assume that the containment pressure remains constant at 0.1 MPa.

PROBLEM 12-7N QUESTION

Computation Of The Axial Distribution Of Thermal And Hydraulic Characteristics Of A Horizontal Steam Generator

1) Operating conditions: see Table 1
2) Properties: determine using the given operating conditions
3) Material and geometry: see Table 1
4) Questions: compute the axial distribution of the following parameters on the secondary side:
   a) Temperature
   b) Enthalpy
   c) Quality
   d) Void fraction
   e) Mass flux (liquid, vapor, total)
   f) Volume flux (liquid, vapor, total)
5) Assumptions:
   - 1 dimension flow
   - Thermodynamic equilibrium
   - Slip ratio = 1.5
   - Once-through steam generator

Table 1

	Geometry	Thermal	Hydrodynamic
Primary: Horizontal U-Tube (Full-Power Conditions)	Tube O.D. = 0.687 in Tube thickness = 0.050 in Average tube length = 23.78 ft Number of tubes = 13856 Heat transfer area = 59,260 ft^2	Inlet temp. = 619.2˚F Outlet temp. = 555.0˚F Power = 900 MW h = 950 Btu/hr·ft^2˚F	Flow rate = 34.1×10^6 lbm/hr Pressure = 2250 psia
Secondary		Steam temp. = 540.2˚F Feedwater temp. = 440.0˚F	Steam pressure = 964.2 psia Flow rate = 3.96×10^6 lbm/hr
Overall	Tube bundle height = 12.25 ft Tube bundle cross-sectional area (for axial secondary flow) = 471.75 ft^2 Shell I.D. = 19.5 ft Shell length = 39.0 ft Collector I.D. = 48 in.		

PROBLEM 12-8N QUESTION

Shell And Tube Horizontal Evaporator

A shell-and-tube horizontal evaporator is to be designed with 30 tubes 1 cm diameter. Inside the tubes water at 100 psia (690 kN/m^2) enters at one end at 130ºC and leaves at the other end at 120ºC. The water velocity (V) in the tubes is 3 m/sec. In the shell, atmospheric pressure steam is generated at 100ºC.

Calculate:

1. The length of the tubes
2. Rate of evaporation, kg/sec
3. Rate of flow of the water, kg/sec
4. Pressure drop in the tubes on the water side.

   (Assume fully developed flow and neglect entrance and exit losses).

For the boiling side take C_{sf} = 0.013 in Eq. 14-22, *Boiling Condensation and Gas-Liquid Flow*, P.B. Whalley.

Make your calculations for heat flux at mid-point where the liquid is at 125ºC. Neglect thermal resistance of the thin tube wall.

H_2O: P = 1 atm T_{sat} = 100ºC

	Liquid		Vapor
ρ (kg/m^3)	960		0.60
c_p (kJ/kg ºC)	4.2		1.88
μ (kg/m s)	0.0003		0.000013
K (W/m ºC)	0.68		0.025
σ (N/m)		0.06	
$i_{\ell g}$ (kJ/kg)		2280	
Pr	1.9		0.97

Assume properties of <u>liquid</u> inside the tubes at 690 kPa in the range 120-130ºC are the same as the above at 1 atm and 100ºC.

PROBLEM 12-10N QUESTION

Calculation Of MCHFR And MCPR For A BWR Channel

Consider a BWR channel operating at 100% power at the conditions noted below. Using the Hench-Levy limit lines (Eqs. 12-69) and assuming the critical condition occurs at all powers at a position three quarters of the channel length from the inlet, i.e., $z = 0.75$ L, determine:

- MCHFR at 100% power, and
- MCPR at 100% power.

Operating Conditions

$q'(z) = 8\text{x}10^4 \exp(-1.96\, z/L) \sin \pi z/L$
where $z = 0$ is defined at the channel inlet
and q' is in units of BTU/hr·ft

$G = 1\text{x}10^6$ lb/hr·ft^2

$p = 1000$ psia

$h_{IN} = 475$ BTU/lb

Channel Conditions

L = 12 feet

P = 0.638 inches

D = 0.483 inches

NOTE: English units have been used because *Nuclear Systems*, Vol. 1, Eqs. 12-69 are written in these units.

$$\int_0^{nL} \exp\left(-\frac{\alpha z}{L}\right) \sin\frac{\pi z}{L}\, dz = \frac{\frac{L}{\pi}(1 - \exp(-n\alpha)\cos n\pi) - \frac{\alpha L}{\pi^2}\left[\exp(-n\alpha)\sin n\pi\right]}{1 + \left(\frac{\alpha}{\pi}\right)^2} =$$

2.08 ft. for $\alpha = 1.96$ and $nL = 0.25$ L

2.42 ft. for $\alpha = 1.96$ and $nL = 0.5$ L

3.00 ft. for $\alpha = 1.96$ and $nL = 0.75$ L

PROBLEM 12-12N QUESTION

Nucleate Boiling On A Tube Wall

A Tokamak is cooled by a single phase water flow which exits at a temperature which you will determine, and this water enters a steam generator through a large number of tubes in which parallel flow occurs (only one tube is shown in Fig. 1). Assume each tube has diameter 0.01m and all tubes together have an outside surface area of $10^6 m^2$. Take the flow in each tube at $Re=10^7$.

a) What outside wall temperature (see Fig. 1) would you desire to achieve on steam generator tubes for the length portion just entering the storage device? Use the boiling curve provided in Fig. 2.

b) What bulk coolant temperature, T_B, within a tube entering the storage device (see Fig. 1) is necessary to achieve the wall temperature you specify in Part 2.a?

You may neglect the thermal resistance of the tube wall. In the temperature range of interest, the relevant water properties are:

$Pr = 1.11$, $k = 0.503$ W/m K, $\mu = 83.5 \times 10^{-6}$ Ns/m^2, $c_p = 6.604$ kJ/kg K.

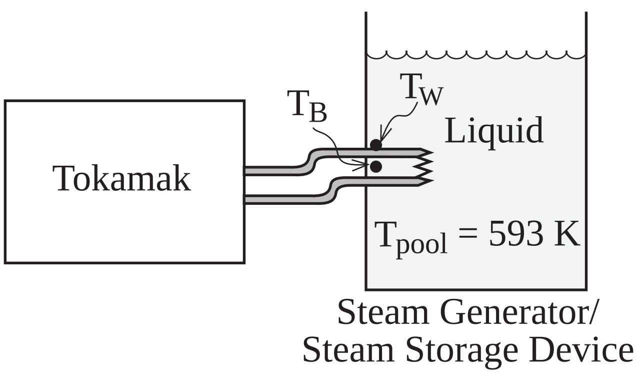


Figure 1

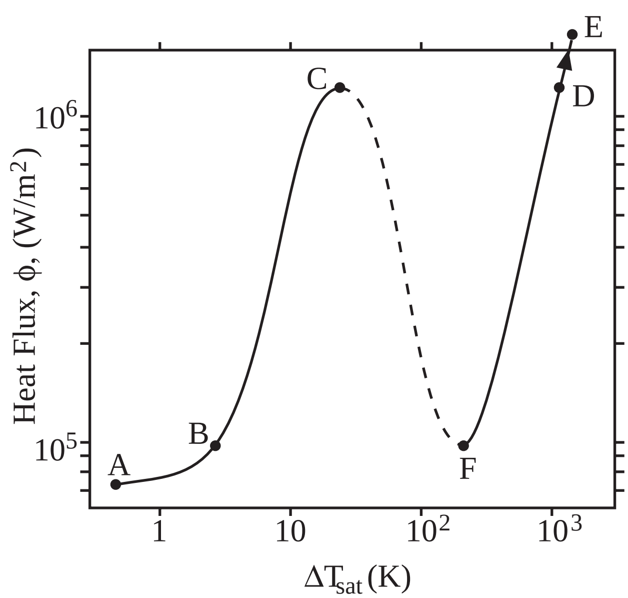


Figure 2. Boiling Curve.

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PROBLEM 12-13N QUESTION

Nucleate Boiling Initiation And Termination On A Heat Exchanger Tube

A heat exchanger tube is immersed in a water cooling tank at 290K, as illustrated in Fig. 1. Hot water (single phase, 550K) enters the tube inlet and is cooled as it flows at 2 kg/s through the 316 grade stainless steel tube (19 mm outside diameter and 15.8 mm inside diameter). Neglect entrance effects.

a) Compute the length along the horizontal inlet length of the tube where nucleate boiling on the tube O.D. is initiated.

b) Compute the length where nucleate boiling on the tube O.D. is terminated.

The heat transfer coefficient between the outer tube wall and the water cooling tank is 500 for single phase conditions and 5000 for nucleate boiling conditions. The wall superheat for incipient nucleation is 15˚C for this configuration. Estimate and justify any additional information you need to execute the solution.

Fluid Properties of Inlet Water
(assume they stay constant)

K	≡	Thermal Conductivity = 0.5 W/m˚C
ρ	≡	Density = 704 kg/m^3
μ	≡	Viscosity = 8.69x10^{-5} kg/ms
C_p	≡	Heat Capacity = 6270 J/kg˚C

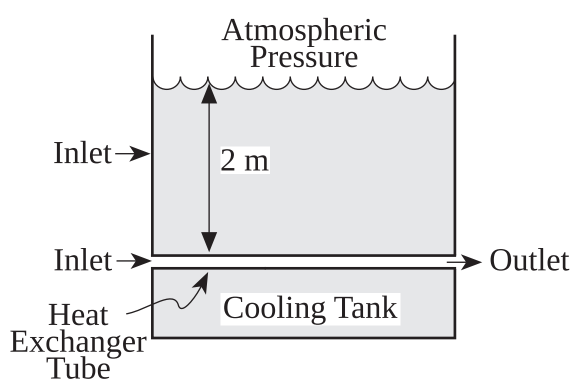


Figure 1

PROBLEM 12-14N QUESTION

Nucleation in Pool and Flow Boiling

A heat surface has conical cavities of uniform size, R, of 10 microns.

A) If the surface is used to heat water at 1 atmosphere in pool boiling, what is the value of the wall superheat required to initiate nucleation?

B) If the same surface is now used to heat water at 1 atmosphere in forced circulation, what is the value of the wall superheat required to initiate nuclear boiling? What is the surface heat flux required to initiate nucleation?

PROBLEM 13-7N QUESTION

Two Phase Flow Pressure Drop Calculation In BWR

Consider a hypothetical BWR fuel assembly with following characteristics: All coolant channels are identical and all fuel rods are operating at the same uniform axial heat flux of 0.8 MW/m^2.

Calculate the <u>friction</u> pressure drop across the fuel assembly assuming the HEM condition are valid.

DATA:

Operating Conditions:

Subchannel coolant mass flow rate	= 0.199815 kg/sec
Reactor coolant pressure	= 6.89 MPa
Inlet water temperature	= 276.7°C
Density of saturated liquid	= 741.65 kg/m^3
Density of saturated vapor	= 35.93 kg/m^3
Enthalpy of saturated liquid	= 1260.4 kJ/kg
Enthalpy of saturated vapor	= 2770.8 kJ/kg
Slip ratio from Bankoff's correlation	

Geometry:

Pitch	= 16.2 mm
Pin Diameter	= 12.27 mm
Active fuel length	= 3.6576 m
Number of fuel rods	= 54

PROBLEM 13-8N QUESTION

Thermal Hydraulic Analysis Of A Pressure Tube Reactor

Consider the light water cooled and moderated pressure tube reactor shown in Figure 1. The fuel and coolant in the pressure tube are within a graphite matrix. Each pressure tube consists of a graphite matrix that has 24 fuel holes and 12 coolant holes. Part of the graphite matrix and a unit cell are also shown in Figure 1.

An equivalent annuli model for thermal analysis is shown in Figure 2. Consider the fuel (although composed of fuel particles in each fuel hole) as operating at a uniform volumetric heat generation rate in the r, Θ plane. Operating conditions and some useful parameters are in Table 1.

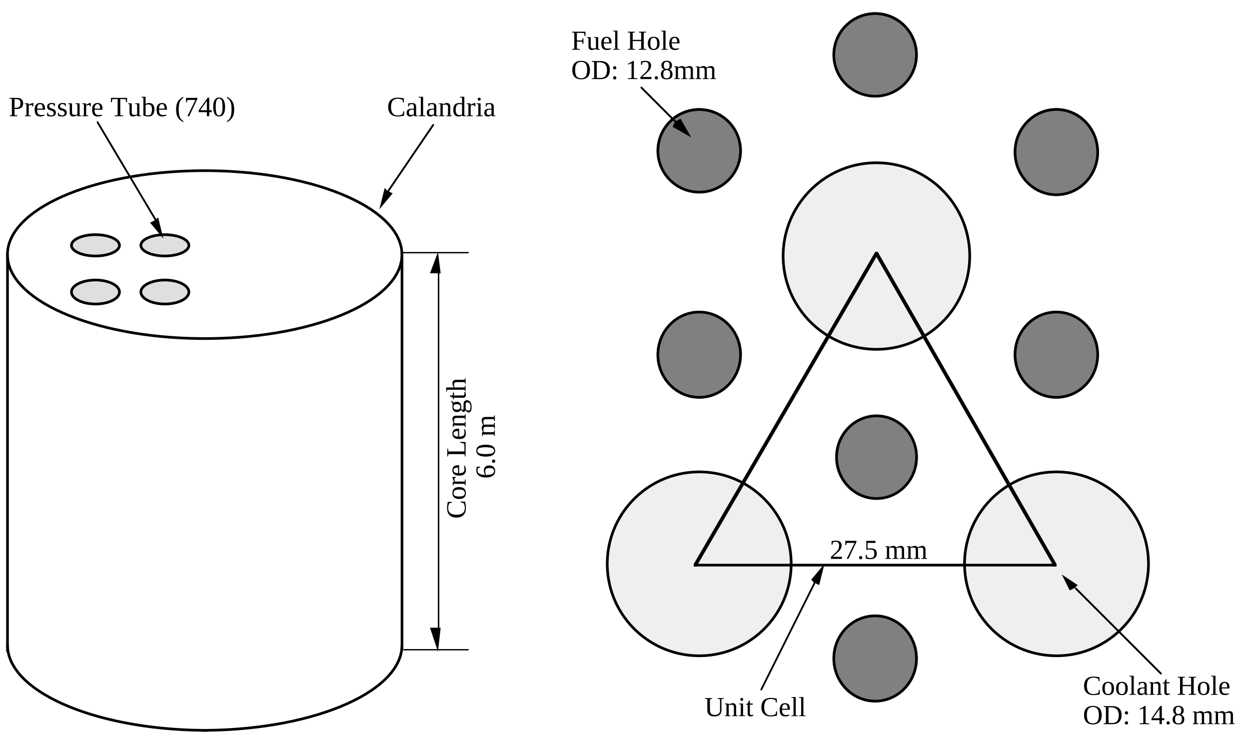


Figure 1 Calandria with Pressure Tubes and Unit Cell in the Pressure Tube

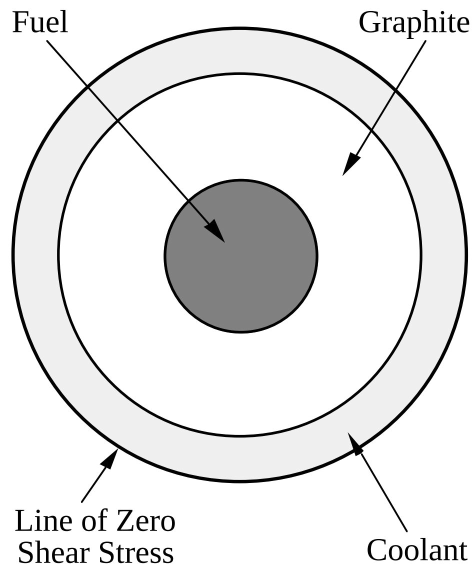


Figure 2 Equivalent Annuli Model (not to scale)

Table 1. Operating Data

	Units	Data		Units	Data
Reactor System			Fuel Hole		
Core thermal power	MWth	2000	Fuel hole diameter	mm	12.8
Number of pressure tubes		740	Mass of UC	kg/hole	2.3
Core radius	m	8.5			
Core length	m	6.0	Coolant Hole		
			Coolant hole diameter	mm	14.8
Primary System			Coolant flowrate	kg/s	1.4
Pressure	MPa	6.89			
Inlet coolant temperature	˚C	245	Unit cell pitch	mm	27.5

Assumptions:

- HEM (Homogenized Equilibrium Model) for two phase flow analysis is valid.
- Cosine axial heat flux (neglect extrapolation).

Useful Data:

Parameters	Fuel	Graphite	Coolant
Thermal conductivity, k (W/m·˚K)	7	23	0.59
Dynamic viscosity, μ (Pa·s)			101×10^{-6}
Specific heat, C_p (J/kg·˚K)			5.0×10^{3}
Coolant inlet enthalpy, h_{in} (kJ/kg)			1062.3
Single phase density, ρ (kg/m^3)			776.3

Saturated Coolant Data @ 6.89 MPa

Density: $\rho_f = 742.0$ kg/m^3, $\rho_g = 35.94$ kg/m^3

Enthalpy: $h_f = 1261.6$ kJ/kg, $h_{fg} = 1511.9$ kJ/kg

Saturated temperature @ P = 6.89 MPa, $T_{sat} = 284.86$˚C

QUESTIONS

1. What is the radial peaking factor assuming an axial cosine and radial Bessel function flux shape (neglect extrapolation length)?

For the following questions, assume that the total power of the fuel hole is 260 (kW/fuel hole) in the hot channel.

2. What is the coolant exit temperature in the hot channel?
3. What is the coolant exit enthalpy in the hot channel?
4. What is the exit void fraction in the hot channel?
5. What is the non-boiling length in the hot channel?
6. What is the fuel centerline temperature at the position where bulk boiling starts in the hot channel?
7. What is the pressure drop in the hot channel? To simplify your calculation, assume a uniform heat flux value that provides total power equivalent to the cosine shape heat flux distribution.

PROBLEM 13-9N QUESTION

Heat Transfer Problems for a BWR Channel

Consider a channel operating at BWR pressure conditions with a cosine heat flux distribution. Relevant conditions are as follows:

Geometry	Operating Conditions
$D = 17$ mm	$\rho = 7.5$ MPa
$L = 3.8$ m	$T_{in} = 270^\circ$C
$L_e = L$	$G = 1700$ kg/m^{2}s
	$q''_{max} = 1050$ kW/m^2
	$Pr = 1.0$
	$\mu = 8.7\text{x}10^{-5}$ kg/m·s

A) Find the axial position where the equilibrium quality, x_e, is zero.

B) What is the axial extent of the channel where the actual quality is zero? i.e., this requires finding the axial location of boiling incipience. (*It is sufficient to provide a final equation with all parameters expressed numerically to determine this answer without solving for the final result*.)

C) Find the axial location of maximum wall temperature assuming the heat transfer coefficient and given by the Thom, et al., correlation for nuclear boiling heat transfer (Eq. 12-28b).

D) Find the axial location of maximum wall temperature assuming the heat transfer coefficient is not constant but varies as is calculated by relevant correlations. Here you are not asked for the exact location, but whether the location is upstream or downstream from the value from Part C.

PROBLEM 13-10N QUESTION

Two Phase Problem Involving A Nuclear Power Plant

Consider a steam-cooled 170 MWe (electrical power output) power plant as shown in Fig. 1. The plant is heating the coolant (water) in two stages: the first stage is from waste heat supplied by an external boiler (located at the same site) through a heat exchanger; the second is through a nuclear reactor. The plant states are defined as follows:

Position	Quality (x)	Pressure (p)	State
1	•	5.0 MPa	saturated liquid
2	•	15.0 MPa	h = 1168.86 kJ/kg
2	30%	15.0 MPa	•
3	75%	15.0 MPa	•
4	•	5.0 MPa	Two-phase mixture

Additional Information:

$\dot{Q}_{HX}$	=	1,000 MWt
Number of fuel pins	=	25,000
Number of channels	=	25,000
Square array with P/D	=	1.15
Diameter of rod	=	1.0 cm
Active core length	=	2 m

Assumptions:

Uniform flat radial power distribution over entire core

Cosine power shape axially with zero extrapolation length

HEM flow throughout

For the pressure drop calculation assume:

- Entrance and exit losses negligible
- Liquid and vapor compressibility negligible
- Gravitational losses negligible
- $f_{2\phi} = f_{\ell o}$

QUESTIONS

A. Draw the cycle T-s diagram.

B. Compute reactor power and mass flowrate needed.

C. Compute p_{acc} across the core.

D. Indicate completely how to evaluate p_{fric} (but **do not** perform the integration).

E. Compute the exit void fraction.

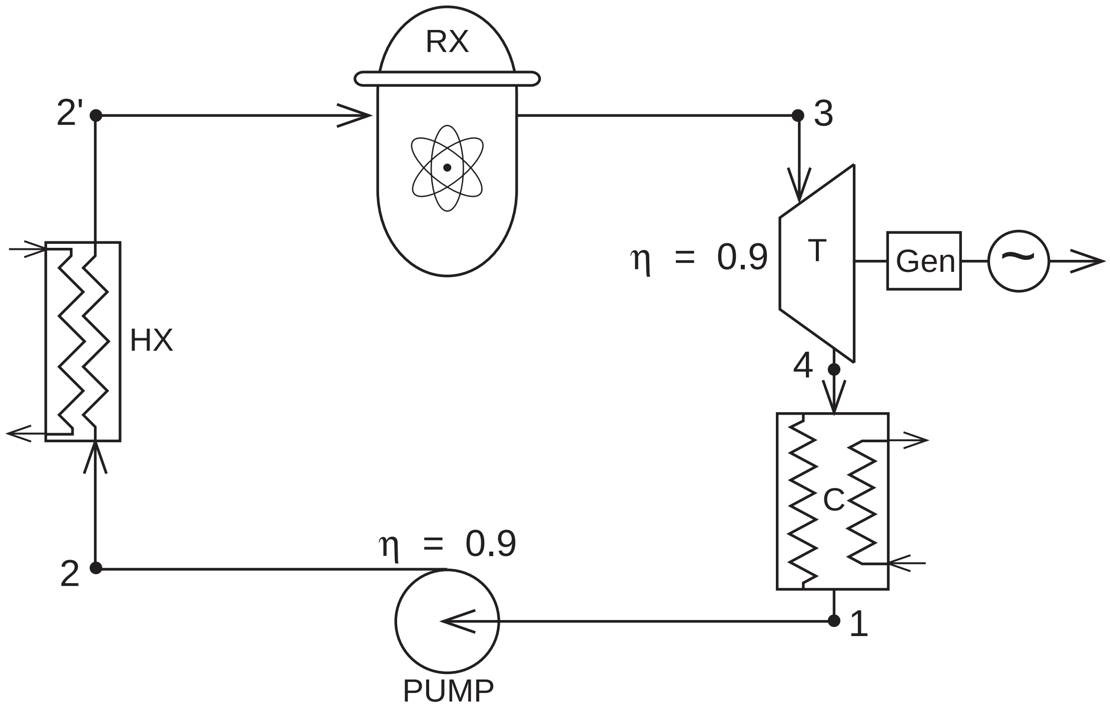
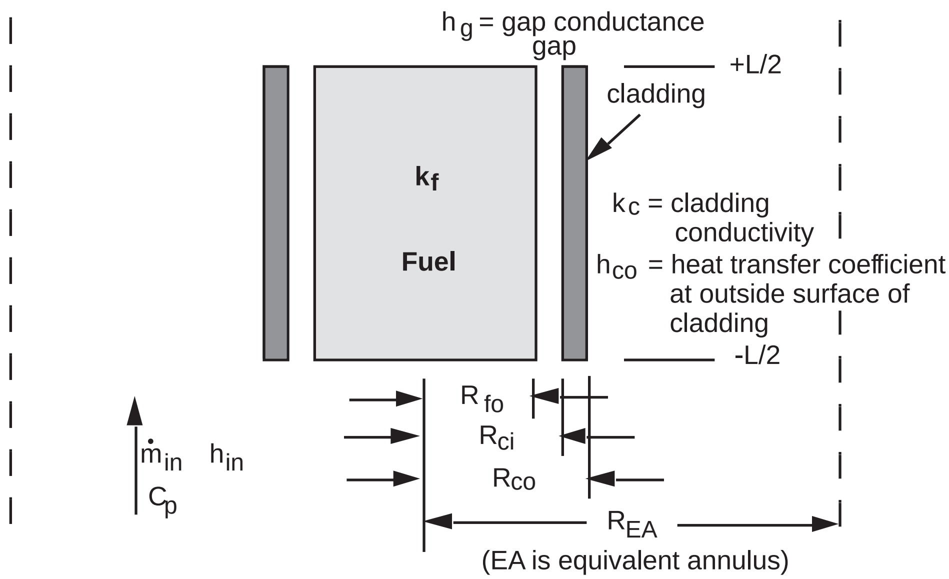


Figure 1

PROBLEM 13-11N QUESTION

Location Of Maximum Clad And Fuel Temperature For A Uniform Axial q'''

Consider a fuel rod in a channel (an equivalent annulus) cooled by a single phase flow over its entire axial length L. In this arrangement the heat transfer coefficient is constant over the axial length L. Take the fuel rod volumetric energy generation rate, q''', as uniform both axially and radially. Identify the axial locations of the maximum clad outside temperature and the fuel centerline temperature. See figure for nomenclature to be used. Be sure to present the basis for your answer.



PROBLEM 13-12N QUESTION

Thermal Behavior Of A Plate Fuel Element Following A Loss Of Coolant

A reactor fuel assembly of the MIT research reactor is made up of plate elements as shown in Fig. 1 (only 4 of 13 elements are shown). Suppose the flow channel between plates 2 and 3 is blocked at the inlet (Fig. 2) What is the axial location of the maximum fuel temperature in plate 3? Solve this in the following steps: (Steps A and B can be solved independently of each other).

A) Find $T_w(z)$ where T_w is the element 3 surface temperature on the cooled side (RHS).

B) Find $T_{Fuel_{LHS}}(z) - T_w(z)$ where $T_{Fuel_{LHS}}(z)$ is the element 3 surface temperature on the insulated side (LHS).

C) Find the axial location of the maximum $T_{Fuel_{LHS}}(z)$.

In solving this problem you can make the following assumptions:

- All heat transfer through the fuel element is radial, i.e. there is not axial heat transfer within the fuel element.
- All of the energy generated in plate 3 flow radially to the right to the coolant channel between elements 3 and 4, i.e., the left side of element 3 has an insulated boundary (see Fig. 3).
- For simplicity, we neglect the clad and take the elements as only composed of fuel - a metallic fuel.
- Assume the flow is fully developed.

Operating Conditions:

P = 55 psi	(0.379 MPa)
T_{inlet} = 123.8 F	(51°C)
$\dot{m}$ = 0.32 kg	
$q'''(z) = 8.54E5\cos(\pi z/L)$ kW/m^3	

Geometry:

L = 23 inches	(58.42 cm)
s = 0.098 inches	(0.249 cm)
t = 0.030 inches	(0.0762 cm)
w = 2.082 inches	(5.288 cm)

Properties:

Water: c_p = 4.181 kJ/kg-K
ρ = 987.2 kg/m^3
k = 0.644 W/m-K
μ = 544E-6 kg/m-s
Pr = 3.597

Fuel: k = 41.2 W/m-K

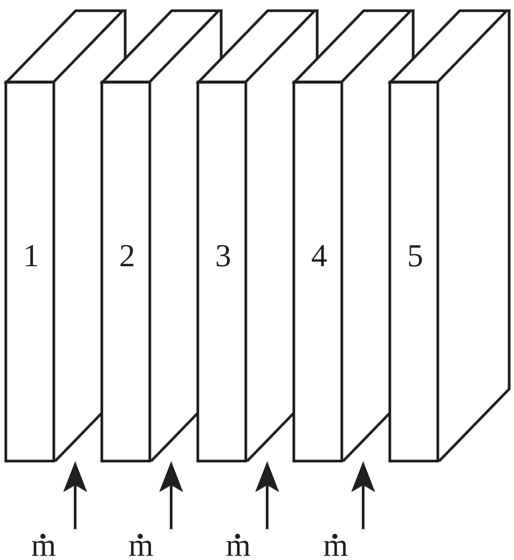


Figure 1

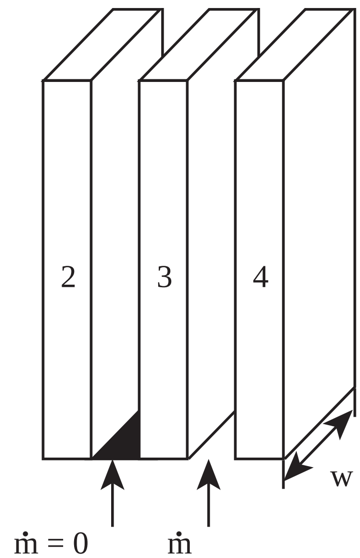


Figure 2

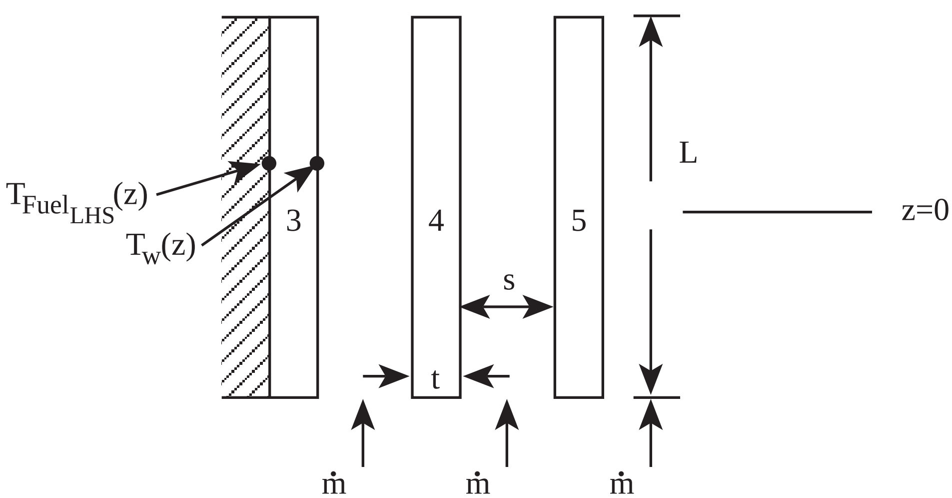


Figure 3

PROBLEM 13-13N QUESTION

Maximum Clad Temperature For LMFBR Reactor

Derive the relationship between the physical and extrapolated axial lengths for a LMFBR core such that the maximum clad temperature occurs at the core outlet during steady-state operating conditions. This relationship describes the truncation of the assumed sinusoidal thermal flux variation along the core axis.

Ignore the reactor blankets and assume the following remain constant along the axial length of the core:

(i) heated perimeter of channels
(ii) mass flux of coolant
(iii) coolant specific heat
(iv) film heat transfer coefficient

PROBLEM 2-6N QUESTION

Analysis Of Reactor Types

NOTE: The use of MathCAD is not required for this problem set; however, now would be a good time to familiarize yourself with MathCAD since it may significantly reduce the work involved with future problems.

This problem set involves calculations that show some typical differences between reactor types.

1) Compare average heat deposition rates for seven reactor types: (BWR, PWR(W), PHWR, HTGR, AGR, LMFBR core region (C), and LMFBR axial and radial blanket regions (BA & BR)). Details are:

a) Use information in Tables 1-2, 1-3, and 2-3 to find core average values of the linear heat generation rate, q' (kW/m). Do not use the "linear heat rate" row in Table 2-3. Note: 90% of the LMFBR power is deposited in the C region.

b) After calculating part a), why are the reactor types listed in the order found in 1)?

c) Use your calculate q' to compute the surface heat flux, q'' (kW/m^2), and the volumetric heat generation rate, q''' (kW/m^3) for the same reactor types.

PROBLEM 3-8N QUESTION

Decay Power Calculations Of A 3-Batch PWR Core

A PWR core has been operated on a three-batch fuel management scheme on an 18-month refueling cycle, e.g., at every 18 months, one third the core loading is replaced with fresh fuel. A new batch is first loaded into the core in a distributed fashion such that it generates 43% of the core power. After 18 months of operation, it is shuffled to other core locations where it generates 33% of the core power. After another 18 months, it is moved to other core locations where it generates 24% of the core power.

QUESTION

The plant rating is a 3400 MWth. Assume it is shutdown after an 18-month operating cycle. What is the decay power of the plant one hour after shutdown if it has operated continuously at 100% power during each of the preceding three 18-month operating cycles and the shutdown periods for refueling were each of 35 days duration.

Solve this problem in two ways.

a) Consider the explicit operating history of each of the three batches to the core decay power.

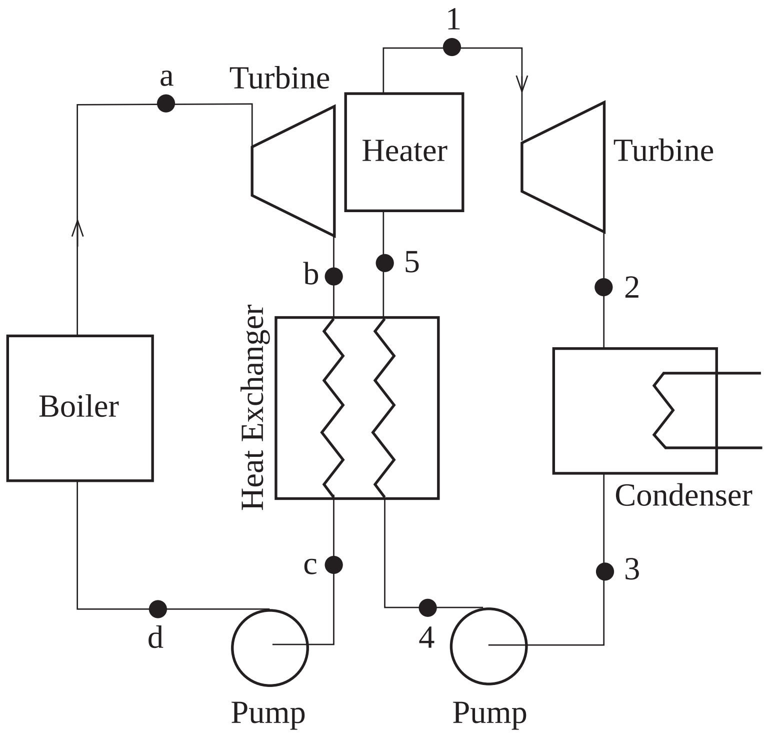
b) Assume the whole core had been operating for a infinite period before shutdown.

PROBLEM 6-8N QUESTION

Thermodynamics Of Binary Cycle Involving Sodium And Steam Water

Consider the binary cycle using the sodium and steam/water in the sketch below and operating at conditions in Table 1. Sodium properties are in the text Appendix E.

1. Draw T-S diagram of the cycle.
2. Compute the cycle thermal efficiency.



Schematic Diagram of Binary Cycle

Table 1. Operating Conditions

Points	Pressure (psia)	Condition
a	105.2	Saturated Vapor of Sodium
b	1.317	
c	1.317	Saturated Liquid of Sodium
d	105.2	
1	600.0	Superheated Steam at T = 680˚F
2	1.0	
3	1.0	Saturated Liquid
4	600.0	
5	600.0	Saturated Vapor

Turbine Isentropic Efficiency = 90%

Pump Isentropic Efficiency = 85%

PROBLEM 6-9N QUESTION

Proofs Involving The Brayton Cycle

A. The Brayton Cycle identified in Figure 1 utilizes intercooling. The replacement of the compressor process (a-b) with a two step compression process (a-c and d-e), an intermediate cooling phase (c-d) which defines the intercooling process subject to the constaint $T_d \geq T_a$ and a heating phase (e-b) to return to state b presumably creates a benefit. This benefit could be either:

(1) reduction in work required for the compression process, or

(2) reduction in the irreversibility of the required compression process.

Provide proofs which demonstrate whether benefits (1) and (2) are true or false. You may treat the cooling phase c-d and the heating phase e-b as unreversible.

B. You must select the intermediate pressure, P_c ($P_c = P_{cs} = P_d$) at which to perform the intercooling phase.

(1) Which pressure should be selected in order to maximize the reduction in work for the compression process? Prove your answer.

(2) Which pressure should be selected in order to minimize the irreversibility of the compression process? Prove your answer.

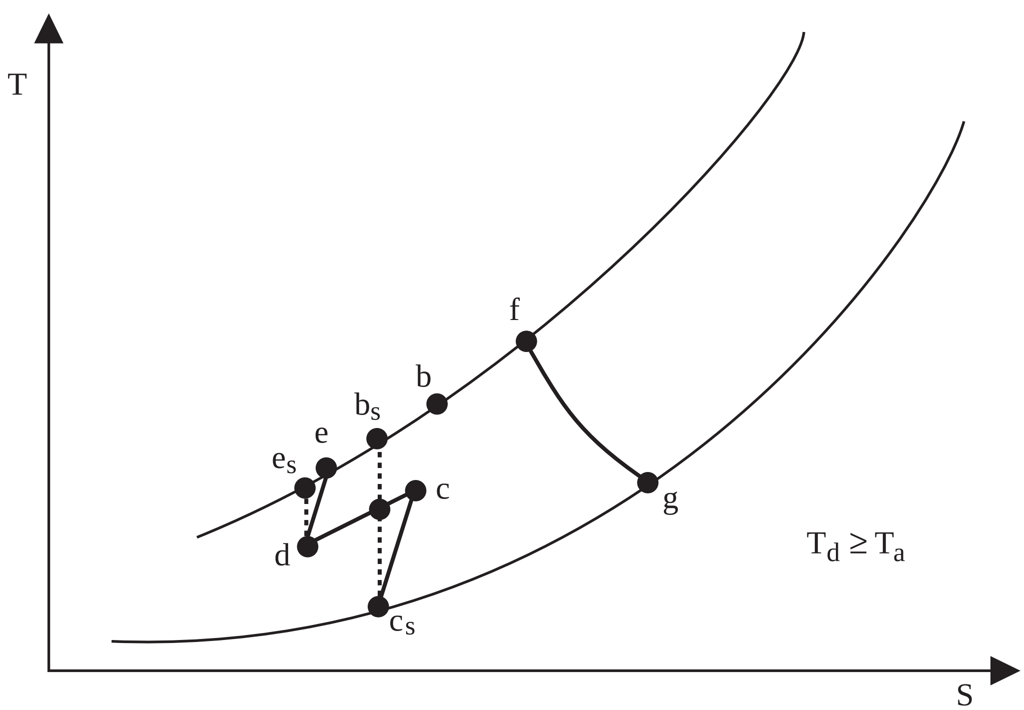


Figure 1

PROBLEM 6-10N QUESTION

Replacement Of A Steam Generator In A PWR With A Flash Tank

Consider a "direct" cycle plant with a pressurized water-cooled reactor. This proposed design consists of using most of the typical PWR plant components except the steam generator. In place of the steam generator, a large "flash tank" is incorporated with the capability to take the primary coolant and reduce the pressure to the typical secondary side pressure. The resulting steam is separated, dried, and taken to the balance of plant, the feedwater from the condenser return to this flash tank. The primary water from the flash tank is repressurized and circulated back to the core.

Make a schematic drawing of this direct cycle plant, and discuss the benefits and/or problems with this design. Also, compare a typical PWR plant design with this direct cycle design with respect to:

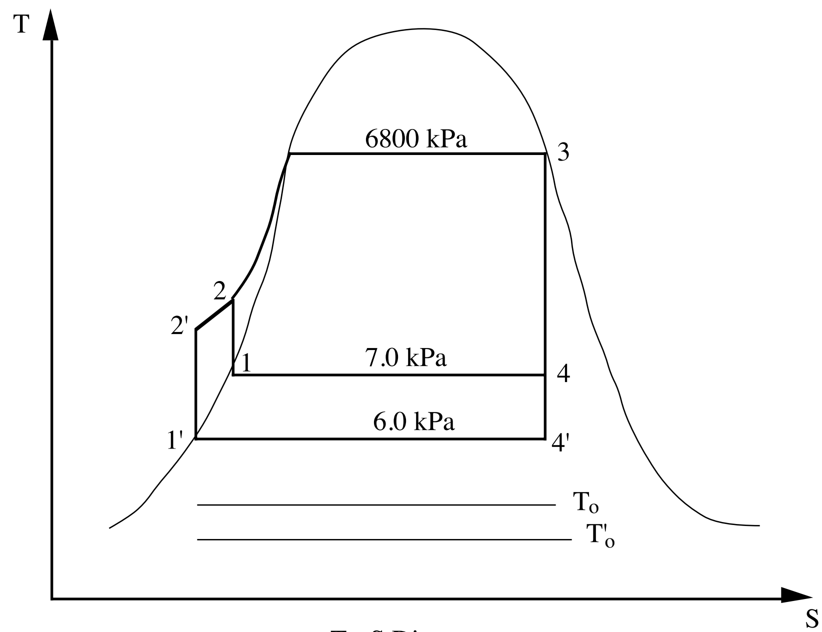
- Plant thermal efficiencies (perform a numerical comparison and explain your results), and
- Nuclear plant safety (perform a qualitative comparison).

PROBLEM 6-11N QUESTION

Irreversibility Problems Involving The Rankine Cycle

Consider the Rankine cycles given in the T-S diagram and defined by operating conditions of Table 1. The cycles differ in the temperature and pressure of the condensation process. What are the differences in cycle irreversibilities between the two cases for irreversibilities defined as:

1. Irreversibility per unit mass flowrate of working fluid, $\dot{I}/\dot{m}_s$, and
2. Irreversibility per unit mass flowrate of working fluid and energy input, i.e., $\dot{I}/\dot{Q}_{in}\ \dot{m}_s$.



T - S Diagram

Table 1. Operating Conditions

Points	Pressure (kPa)	Condition
1	7.0	Saturated Liquid
2	6800.0	
3	6800.0	Saturated Vapor
4	7.0	
1'	6.0	Saturated Liquid
2	6800.0	
3	6800.0	Saturated Vapor
4'	6.0	

PROBLEM 6-13N QUESTION

Optimizing The Pressure Ratio Of The Brayton Cycle In A Brayton-Rankine Combined Cycle

A. A Brayton cycle is presented in Figure 1. The Brayton cycle uses helium as the working fluid. The constraints for this cycle are that the highest temperature achievable, T_4, is 972K and the atmospheric reservoir is 290K. The temperature of State 1 is 10K above the atmospheric reservoir temperature, e.g., T_1 = 300K Cycle parameters are given in Table 1.

- Draw the T-S diagram for this cycle and the temperature distribution diagram within the regenerative heat exchanger.
- Demonstrate how to find the compression ratio that will result in the maximum value of cycle thermal efficiency in terms of only the given temperatures T_4 and T_1 and other given constants.

B. The Brayton cycle above has a Rankine bottoming cycle added as shown in Figure 2. The Rankine cycle uses H_2O. The mass flow rates in each cycle are equal. The constraints for this combined cycle are analogous to those above, e.g., T_4, is 972K and T_7 is 300K. Cycle parameters are given in Tables 1 and 2.

- Draw the T-S diagram for this cycle and the temperature distribution diagrams within the regenerative heat exchangers.
- Demonstrate how to find the compression ratio that will result in the maximum value of cycle thermal efficiency in terms of only the given temperatures T_4 and T_7 and other given constants.

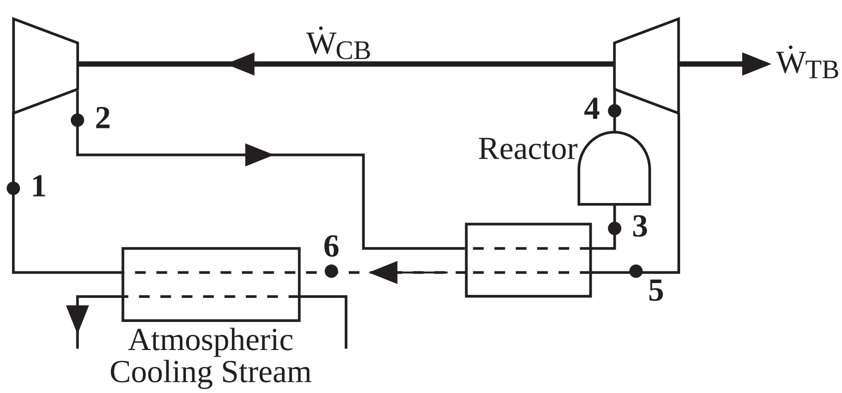


Figure 1 Brayton Cycle

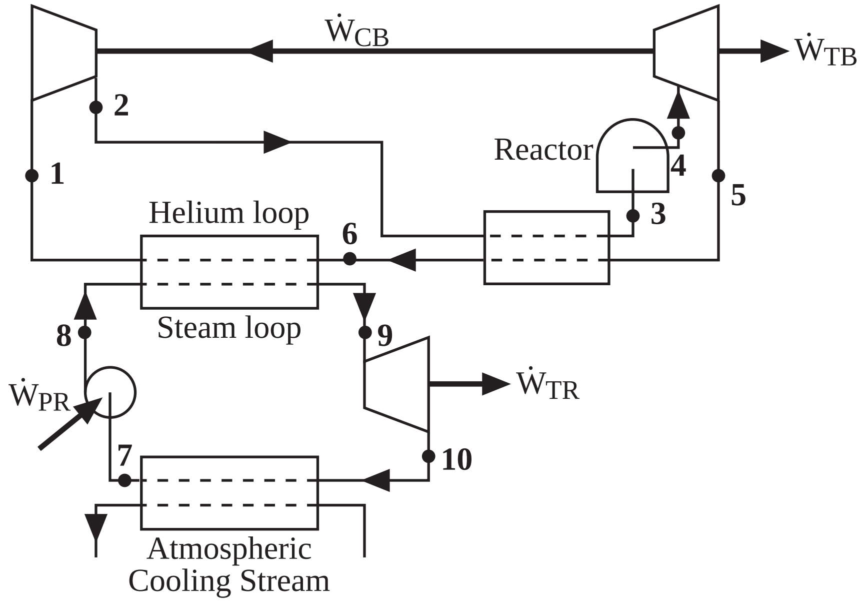


Figure 2 Combined Brayton-Rankine Cycle

Table 1

γ for Helium = 1.658 c_p for Helium = 5,230 J/kg˚K

Efficiency of the regenerative heat exchanger = 0.75
The pump and turbine are ideal, e.g., η_s = 100%
Both heat exchange processes are conducted at constant pressure

Table 2

Efficiency of both regenerative heat exchangers = 0.75
All pumps and turbines are ideal, e.g., η_s = 100%
All heat exchange processes are conducted at constant pressure
The pinch point of the helium-steam heat exchanger = 10 K

State 7 is saturated liquid state State 9 is a saturated vapor state

PROBLEM 6-14N QUESTION

BWR Operation At Supercritical Conditions

The BWR of Problem 6-3 is to be redesigned so as to operate at supercritical steam conditions. Hence, the cycle is modified as shown in Figure 1 by:

- the reactor core is redesigned to produce supercritical coolant at 23 MPa and 450^{o}C.
- a second pump is added in series after the feed water pump to boost the reactor inlet to supercritical pressure conditions of 23 MPa. The isentropic efficiency of this pump is 90%.
- an additional turbine is added at the reactor outlet through which the steam passes and is returned exactly to State 1 of Problem 6-3.

See Table 1 for properties to avoid having to interpolate in the steam tables.

QUESTIONS

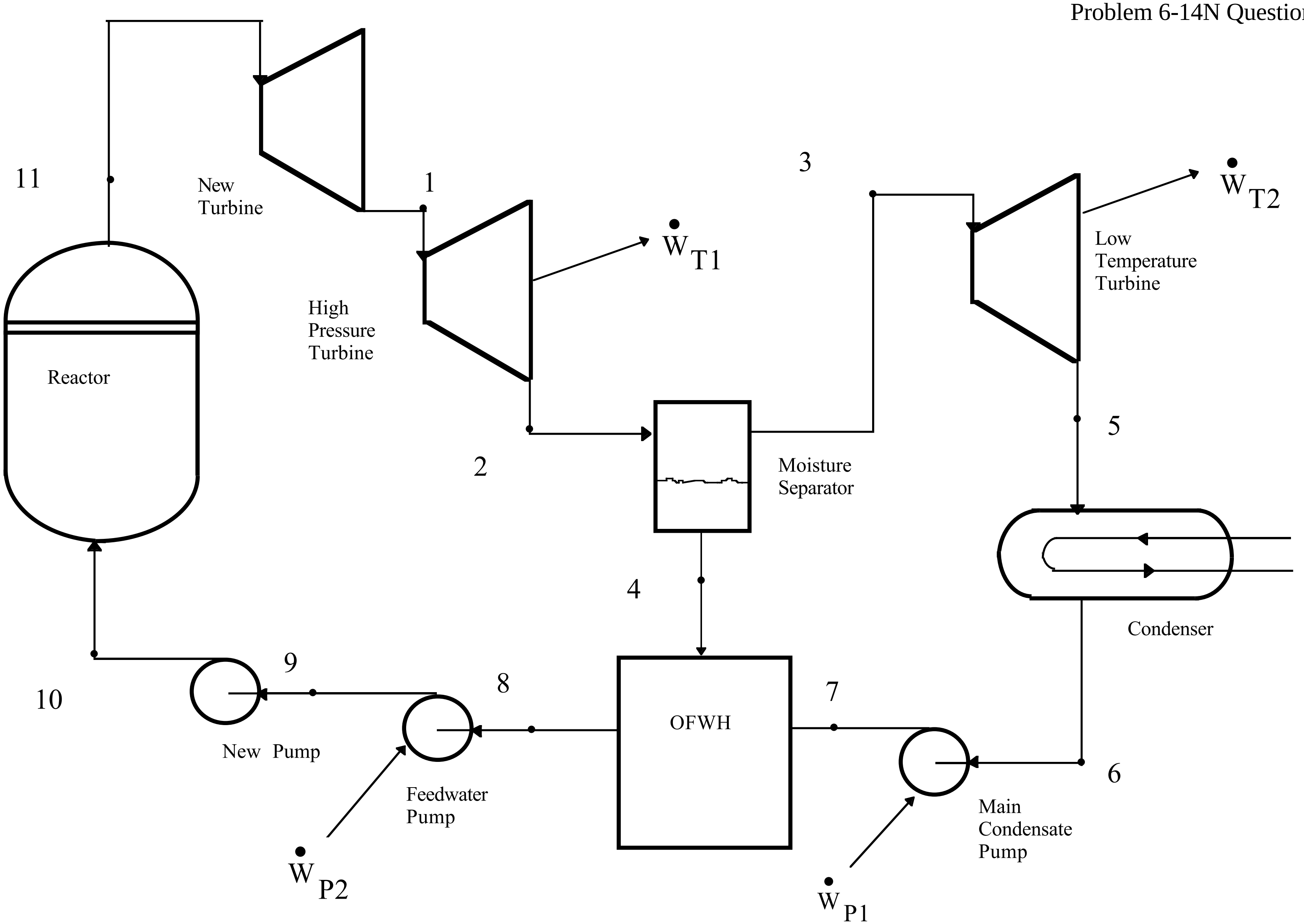
A. What is the new cycle thermodynamic effectiveness?

B. What is the new irreversibility of this reactor plant?

C. Identify what you think are the three most significant safety implications (positive or negative effects) of this new design.

Table 1

	kJ/kg K			kJ/kg		
	s_f	s_{fg}	s_g	h_f	h_{fg}	h_g
6.89 MPa	3.1111	2.7102	5.8213	1311.33	1462.20	2773.53
Specific volume of Stages 9 and 10s = 1.0089 x 10^{-3} m^3/kg						



11
New
Turbine
1
High
Pressure
Turbine
$\dot{W}_{T1}$
3
$\dot{W}_{T2}$
Low
Temperature
Turbine
Reactor
2
Moisture
Separator
5
Condenser
4
9
8
7
10
OFWH
6
New Pump
Feedwater
Pump
Main
Condensate
Pump
$\dot{W}_{P2}$
$\dot{W}_{P1}$

Figure 1

PROBLEM 6-16N Question

Tokamak Power Generation Problem

A problem associated with Tokamak fusion reactors is that power generation is intermittent and some type of energy storage device is required to maintain a constant electrical generation rate. One suggested solution to this problem is a combination steam generator/steam storage unit which expands in volume (at constant pressure) during the reactor "burn," and contracts to its original volume during the reactor down time. The Tokamak and the power cycle are given in Figure 1. Relevant physical properties and conversion factors are given in Table 1.

The Tokamak reactor burn cycle is 1000 seconds at 6680 MWt. The down time at zero power is for 100 seconds after which the power cycle is repeated. This reactor power cycle is illustrated in Figure 2.

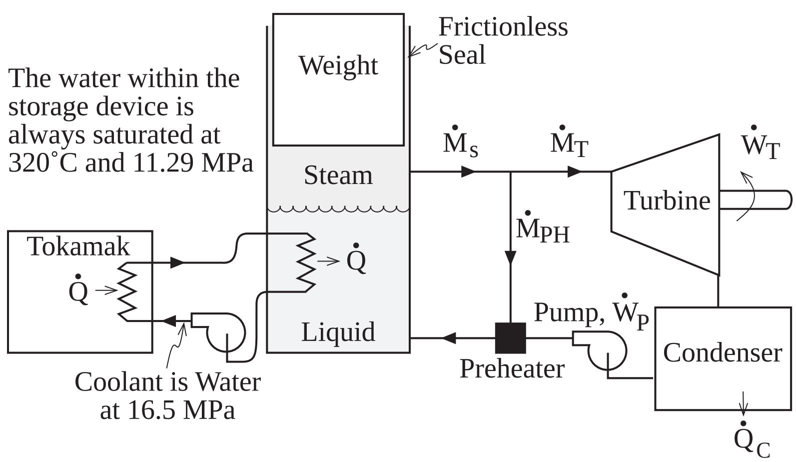


Figure 1 Tokamak and Power Cycle

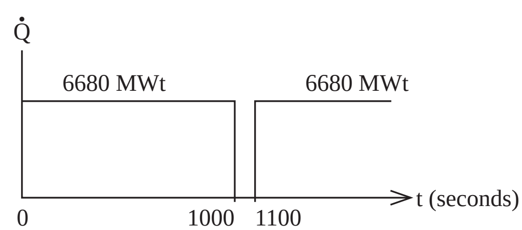


Figure 2. Reactor Power Cycle

TABLE 1. Property Data for 320˚C, 11.27 MPa

		Water, f	Steam, g	Water To Steam, fg
v	m^3/kg	1.5×10^{-3}	0.015	0.0135
h	kJ/kg	1462	2700	1238
u	kJ/kg	1445	2526	1081
c_p	kJ/kg˚K	6.604	8.060	
σ	N/m	9.89×10^{-3}	--	
μ	Ns/m^2	83.5×10^{-6}	20.95×10^{-6}	
k	W/m˚K	0.503	87.8×10^{-3}	
Pr		1.11	1.92	

Unit Conversions:

$$J = Nm = Ws$$

$$Pa = N/m^2$$

QUESTION

The liquid mass in the steam generator/storage unit necessary to cover the heat exchanger tubes is 3.6×10^6kg and the quality at the start of a reactor burn is 15%.

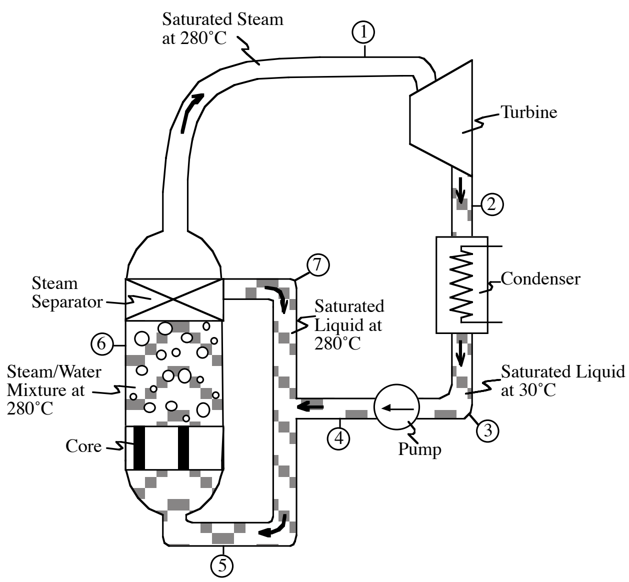
a) Sketch a graph of steam mass stored in the steam/generator/storage unit versus time, and of the liquid mass stored in the steam generator/storage unit versus time. Explain the basis for your sketches.

b) Calculate the required total volume of the steam generator/storage unit.

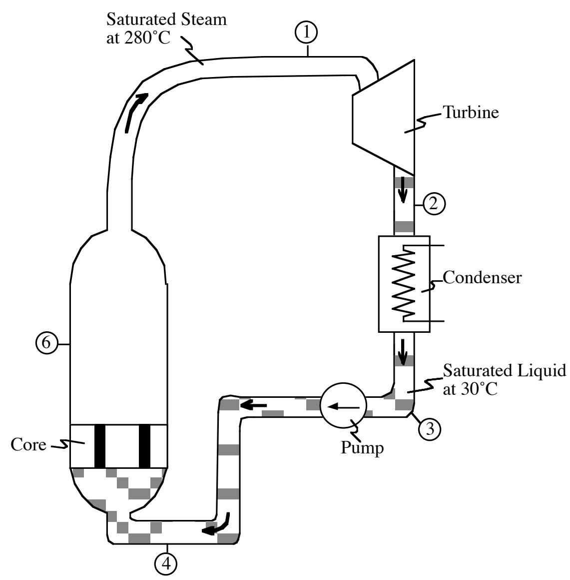
PROBLEM 6-17N QUESTION

Power Cycle for a Simplified BWR

The power cycle of a simplified BWR is shown in Figure 1a. Steam at 10% quality exits the core. The steam is separated from the water in a steam separator, and then is directed to the turbine, then completely condensed and pumped back to the reactor. The separated water is mixed with the feedwater coming from the pump, and recirculated to the core inlet.



a) With Recirculation



b) Without Recirculation

Figure 1. Schematic of a Simplified BWR Plant.

<u>QUESTIONS</u>

a. Sketch the T-s diagram for the cycle of Figure 1a. Make sure to include the effect of recirculation.

b. Using the data below, calculate the cycle thermal efficiency.

c. Consider now the same cycle but **without recirculation**, i.e., the feedwater from the pump goes directly to the core, the steam quality at the core outlet is 100%, and there is no steam separator (Figure 1b). Sketch the T-s diagram and calculate the thermal efficiency for this cycle. How does the thermal efficiency compare to that of the cycle with recirculation?

d. Given the results in 1.c., what are the advantages/disadvantages of using the cycle with recirculation?

Rev March 3, 2005

Assumptions:

- Assume perfect steam/water separation in the steam separator.
- Assume ideal turbine and pump.
- Assume constant water density in the pump.
- Neglect kinetic and gravitational terms.

Data for Saturated Water:

T (°C)	P (bar)	v_f (m^3/kg)	v_g (m^3/kg)	h_f (kJ/kg)	h_g (kJ/kg)	s_f (kJ/kg·K)	s_g (kJ/kg·K)
30	0.04	1.0×10^{-3}	32.9	126	2556	0.4	8.4
280	64	1.3×10^{-3}	0.03	1236	2780	3.1	5.9

PROBLEM 7-10N QUESTION

Containment Problem Involving a LOCA

Upon a loss of primary coolant accident (LOCA) the primary system flashes as it discharges into the containment. At the resulting final equilibrium condition, the containment and primary system are filled with a mixture of steam and liquid. A containment is being designed as shown in Figure1 which directs the liquid portion of this mixture to flood into a reactor cavity in which primary system is located. The condensate which passes back into the core through the break can satisfactorily cool the core if it can submerge it, i.e., if the condensate level is high enough.

Find the containment volume which will yield a final equilibrium pressure following primary system rupture sufficient to create the 125 m^3 of liquid required to fill the cavity and submerge the core.

The pressure and volume of the primary system are 15.5 MPa and 354 m^3, respectively.

- Neglect the initial relative humidity and the air in the containment.
- Neglect $\dot{Q}_{c\text{-}st}$ and $\dot{Q}_{c\text{-}atm}$.

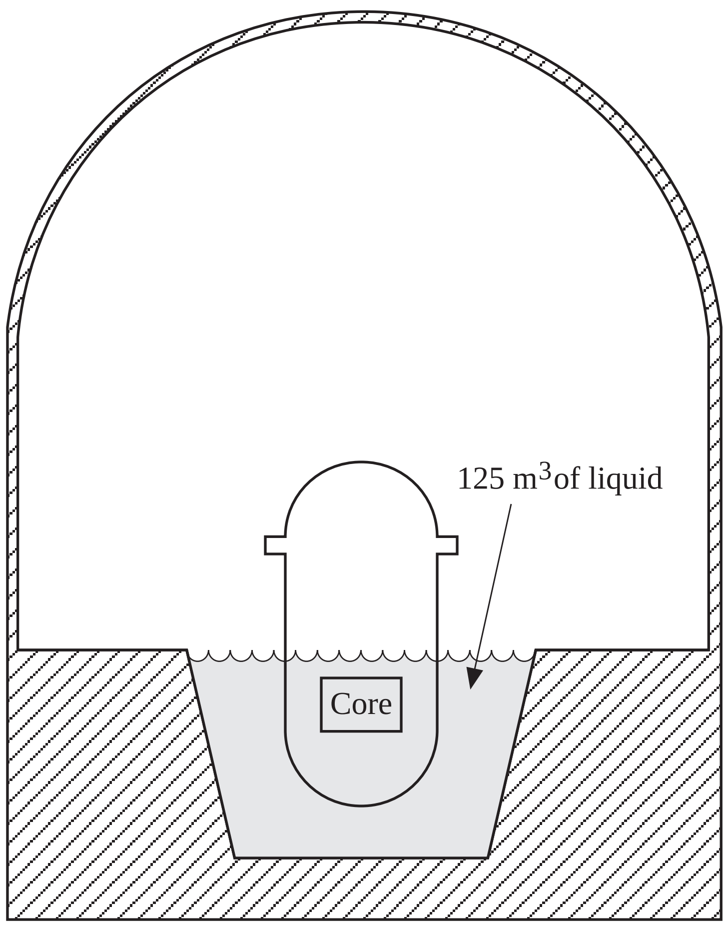


Figure 1

PROBLEM 7-11N QUESTION

Pressurizer Transient Problem

A pressurizer is to be designed in such a way that it can accommodate anticipated pressure fluctuations. Suppose the pressure range which the pressurizer should control is from 14.0 MPa to 16.6 MPa with 15.5 MPa as the nominal condition. During depressurization transients, because of rainout and flashing the water in the pressurizer is assumed at saturation conditions. However, for the overpressurization transient, the steam and liquid in the pressurizer is assumed at non-equilibrium conditions, i.e., steam is saturated because of the spray, but the liquid is subcooled (see Figure 1). Using the single region approach, answer the following questions with the information given in Table 1.

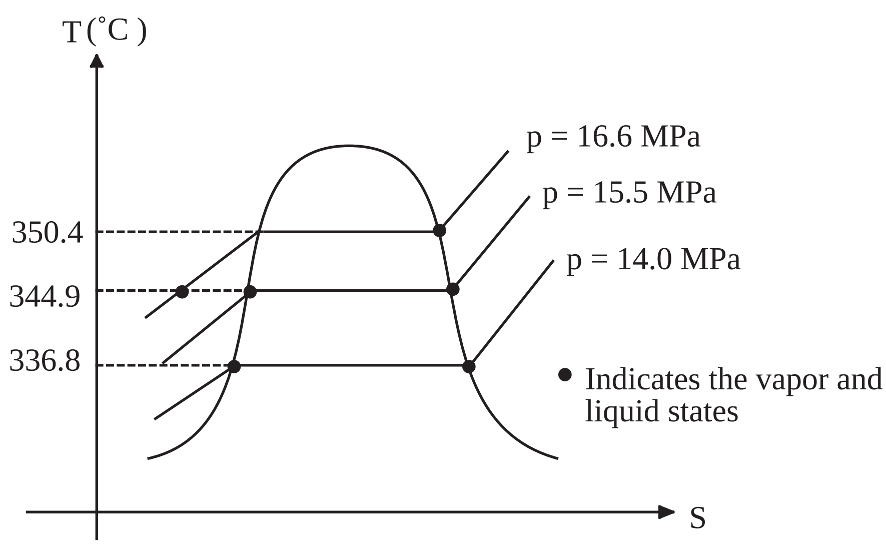


Figure 1

a. Determine the pressurizer volume which accommodates the overpressurization transient assuming the end state of this transient is at p = 16.6 MPa. What is the vapor volume fraction at the end of this transient?

b. Determine the heater size which accommodates the depressurization transient assuming the end state of this transient is 14.0 MPa with the pressurizer volume obtained in (a) above. What is the liquid volume fraction at the end of transient? Is it large enough to cover the heaters, if the minimum liquid volume to cover the heaters is 13% of the pressurizer volume?

TABLE 1

- Vapor and Liquid Conditions in the Pressurizer during Transients:

Pressure (MPa)	$\mathbf{v_f}$ **($10^{-3}m^3/kg$)**	$\mathbf{v_g}$ **($10^{-3}m^3/kg$)**	$\mathbf{u_f}$ **(kJ/kg)**	$\mathbf{u_g}$ **(kJ/kg)**
14.0	1.61	11.5	1549	2478
15.5	1.68	9.81	1600	2444
16.6	1.68*	8.73	1600*	2416

* Subcooled Condition

- Initial Conditions:

The pressurizer is initially 60% full of liquid water, and at 15.5 MPa.

- Overpressurization Transient:

$\dot{m}_{insurge} = 9200$ kg $\qquad$ $h_{insurge} = 1442$ kJ/kg

$\dfrac{\dot{m}_{spray}}{\dot{m}_{insurge}} = 0.09$ $\qquad$ $h_{spray} = 1285$ kJ/kg

During Overpressurization, Heaters are Off

- Depressurization Transient:

$\dot{m}_{outsurge} = 14{,}166$ kg $\qquad$ $h_{outsurge} = 1600$ kJ/kg

During Depressurization, Sprays are Turned Off

- Assumptions:

$h_{insurge}$, $h_{outsurge}$, h_{spray} are constant during transients.

For the depressurization transient, heaters are operating at the full capacity, and the transient is over after 15 minutes.

PROBLEM 7-12N QUESTION

Pressurizer Insurge Problem

For insurge case, why isn't latent heat of vaporization of vapor which is condensed sufficient to heat insurge mass to saturation?

PROBLEM 7-13N QUESTION

Pressurizer Sizing Analysis

The size of a pressurizer is determined by the criteria that the vapor volume must be capable of accommodating the largest insurge and the liquid volume must handle the outsurge. The important limitations of the design are that the pressurizer should not be totally liquid filled or the immersion heaters should not be uncovered after possible transients. To size the vapor volume, a maximum insurge is assumed to completely fill the pressurizer with liquid with some of the insurge being diverted to the spray to condense the vapor. Treating the entire pressurizer volume, V_t, as the control volume, find the vapor volume, V_{g1}, which will accommodate the insurge given below.

DATA:

Initial Pressurizer Conditions

Saturation at 2250 psia and 653˚F

Initial liquid mass = 1827 kg

Maximum Insurge (includes spray)

Mass = 2740 kg.

Enthalpy = 1.2×10^6 J/kg

Final Pressurizer Condition

Assume completely filled with liquid at 2250 psia

PROBLEM 7-14N QUESTION

Containment Pressurization Reactor Thermodynamics

What should the containment volume for a 3000 MWt PWR be to prevent a primary coolant pipe rupture from resulting in an overpressurization of 45 psi, assuming all heat removal systems fail and the fuel may add up to 180 Gw-sec. of energy before mitigating procedures begin? What overpressurization may occur with this design if a steamline ruptures and the steam generator provides the same amount of energy before any safety system intervention? Assume thermal equilibrium within the containment and no heat losses to structures.

DATA

Initial Containment Atmosphere

Pressure	p_{ao} = 14.7 psia
Temperature	T_{ao} = 90˚F
Relative humidity	ϕ = 95%

Specific Heat at Constant Volume

Air	c_{va} = 0.172 B./lb.F
Steam	c_{vw} = 0.379 B./lb.F

PWR – NSSS Operating Conditions and Design Parameters

Primary coolant volume	V_p = 12,500 ft^3
Secondary coolant volume	V_s = 3,145 ft^3
Primary system pressure	p_{po} = 2,250 psia
Secondary system pressure	p_{so} = 1,000 psia
Primary system temperature	T_{po} = 650˚F
Secondary system temperature	T_{so} = 550˚F

PROBLEM 7-15N QUESTION

Drain Tank Pressurization Problem

A drain tank is used to temporarily store water discharged from the pressurizer through the PORV (Figure 1). The drain tank has a burst disk on it which ruptures if the pressure inside the drain tank becomes too large. For this problem, assume that the PORV at the top of the pressurizer is stuck open, and saturated water at 15.4 MPa leaves the presssurizer at a constant flowrate of 3kg/sec and enters a perfectly insulated drain tank of total volume 12 m^3. In addition, assume that the initial conditions (before the water due to the stuck open PORV has entered the drain tank) in the drain tank are:

No air present	Initial vapor volume = 10 m^3
Initial pressure = 3 MPa	Initial liquid volume = 2 m^3

Also assume that the liquid and the water vapor are in thermal equilibrium at all times in the drain tank, and that the burst disk on the drain tank ruptures at 10 MPa.

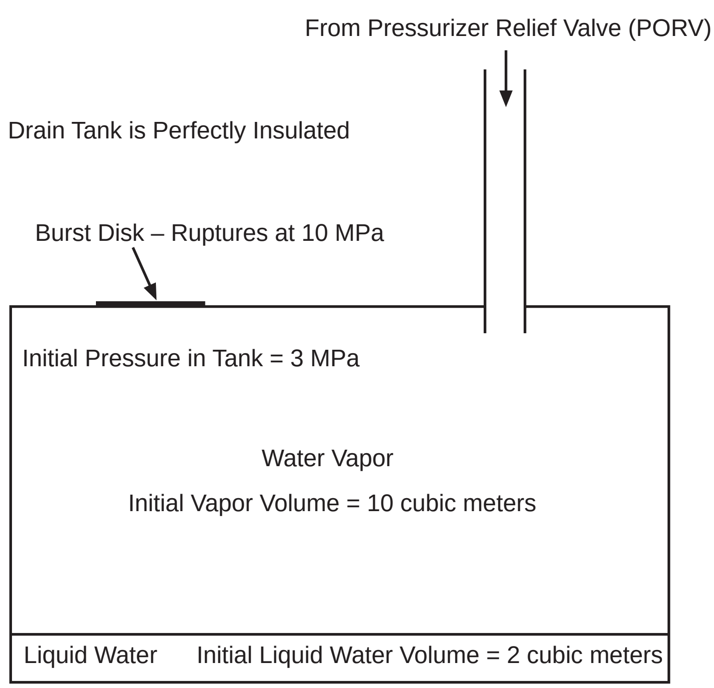


Figure 1

QUESTIONS

A . Define the control mass or control volume you will use and the equation set you will develop.

B . Solve for the elapsed time to burst disk rupture.

C . Now assume 11.93 kg of air are present in the drain tank along with the liquid water and water vapor, $(P_{1w})_{initial} = 3$ MPa and that the change in volume of the liquid water from the initial state to the final state is large. What is the new time to rupture?

PROBLEM 7-16N QUESTION

Containment Pressurization Following Zircaloy-Hydrogen Reaction

Consider a LOCA in a typical PWR in which the emergency cooling system is insufficient to prevent metal-water reaction of 75% of the Zircaloy clad and the hydrogen produced subsequently combusts. Using the results of Problem 3.6, this sequence of events yields the following material changes and energy releases relevant to the containment pressurization:

Primary coolant released	= 2.1×10^5 kg
Zr reacted	= 0.75 x 24,000 kg
Energy released from Zr-H_2O reaction	= 1.195×10^{11} J
H_2 produced and reacted	= 394.7 kg mol
Energy released from H_2 combustion	= 9.47×10^{10} J
O_2 consumed	= *you must determine*
Net H_2O change	= *you must determine*

Take the initial primary coolant and containment vessel geometry and conditions the same as Table 7-2. Also, assume that nitrogen has the same properties as air.

QUESTION

For the sequence described (e.g., LOCA, 75% Zircaloy clad reaction and subsequent complete combustion of the hydrogen produced):

(a) Demonstrate that the final equilibrium temperature is 449 K, neglecting containment heat sinks using the initial conditions of Table 7-2, and

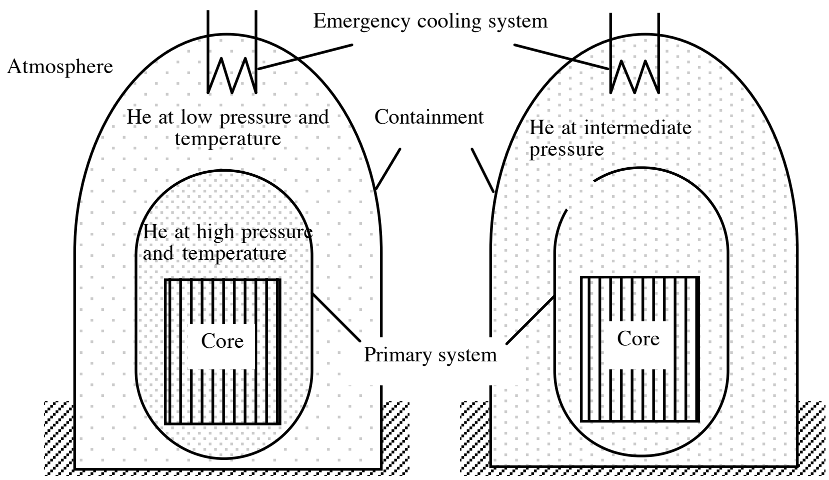
(b) Find the final equilibrium pressure.

HINT: Is the final state likely saturated water or superheated steam in equilibrium with the air? Consider the energy releases compared to those of Example 7.2.

PROBLEM 7-17N QUESTION

Containment Sizing for a Gas-Cooled Reactor with Passive Emergency Cooling

An advanced helium-cooled graphite-moderated reactor generates a nominal thermal power of 300 MW. To prevent air ingress in the core during a Loss Of Coolant Accident (LOCA), the reactor containment is filled with helium at atmospheric pressure and room temperature (Figure 1a). The reactor also features an emergency cooling system to remove the decay heat from the containment during a LOCA. To function properly, this system, which is passive and based on natural circulation of helium inside the containment, requires a minimum containment pressure of 1.3 MPa.



a) Normal Operation Conditions b) Post-LOCA situation

Figure 1. Helium-Cooled Reactor with Helium-Filled Containment.

QUESTIONS

a. Find the containment volume, so that the pressure in the containment is 1.3 MPa immediately after a large-break LOCA occurs (Figure 1b). (Assume that thermodynamic equilibrium within the containment is achieved instantaneously after the break)

b. Assuming that the emergency cooling system removes 2% of the nominal reactor thermal power, calculate at what time the pressure in the containment reaches its peak value after the LOCA. (Calculate the decay heat rate assuming infinite operation time)

c. To reduce the peak pressure in the containment, a nuclear engineer suggests venting the containment gas to the atmosphere through a filter. What would be the advantages and disadvantages of this approach?

Rev March 3, 2005

Assumptions:

- Treat helium as an ideal gas.
- Neglect the heat contribution from fission and chemical reactions.
- Neglect the thermal capacity of the structures.

Data:

Gas volume in the primary system: 200 m^3
Initial primary system temperature and pressure: 673 K, 7.0 MPa
Initial containment temperature and pressure: 300 K, 0.1 MPa
Helium specific heat at constant volume: c_v=12.5 J/(mol·K)
Helium atomic weight: A=0.004 kg/mol
Gas constant: R=8.31 J/(mol·K)

PROBLEM 8-8N QUESTION

'Conductivity Integral'

Describe an experiment by which you would obtain the results of Figure 8-2, i.e., the value of the conductivity integral. Be sure to explicitly state what measurements and observations are to be made and how the conductivity integral is to be determined from them.

PROBLEM 8-9N QUESTION

Thermal Conduction Problem Involving Design Of A BWR Core

A core design is proposed which locates BWR type UO_2 pins in holes within graphite hexagonal blocks (Fig. 1). These blocks then form a core of radius R_o. The achievable linear heat power of the core (MW/m) is desired as a function of core radius, R_o(m) for constraints of Table 1. Present the result as a plot. Constants and terms are defined in Figure 2 and Table 1.

Basically these constraints exist under decay power conditions where the outside of the core radiates its energy to a passive air chimney. However , the outside of the core which is in touch with a vessel at the same temperature is limited to 500˚C. The clad outside temperature, T_{co}, which radiates to the graphite, T_{gi}, is also constrained, here to a temperature 649˚C.

Constants and Constraints for Homogenized Core Power Analysis

Table 1

Constraints	Constants
$T_{co} < 649˚C$	$A_{cell} = 7.30 \times 10^{-4}\ m^2$
$T_{go} < 500˚C$	$d_1 = 12.5$ mm
	$d_2 = 19.8$ mm
	$\epsilon_1 = 0.6$
	$\epsilon_2 = 0.7$
	$k_g = 60$ W/m
	$\sigma = 5.669 \times 10^{-8}\ W/m^2 \cdot K^4$
	$\frac{1}{\epsilon_1} + \frac{d_1}{d_2}\left(\frac{1}{\epsilon_2} - 1\right) = \epsilon$ $\epsilon = 1.937$

Unit Cell Dimensions
Part 1

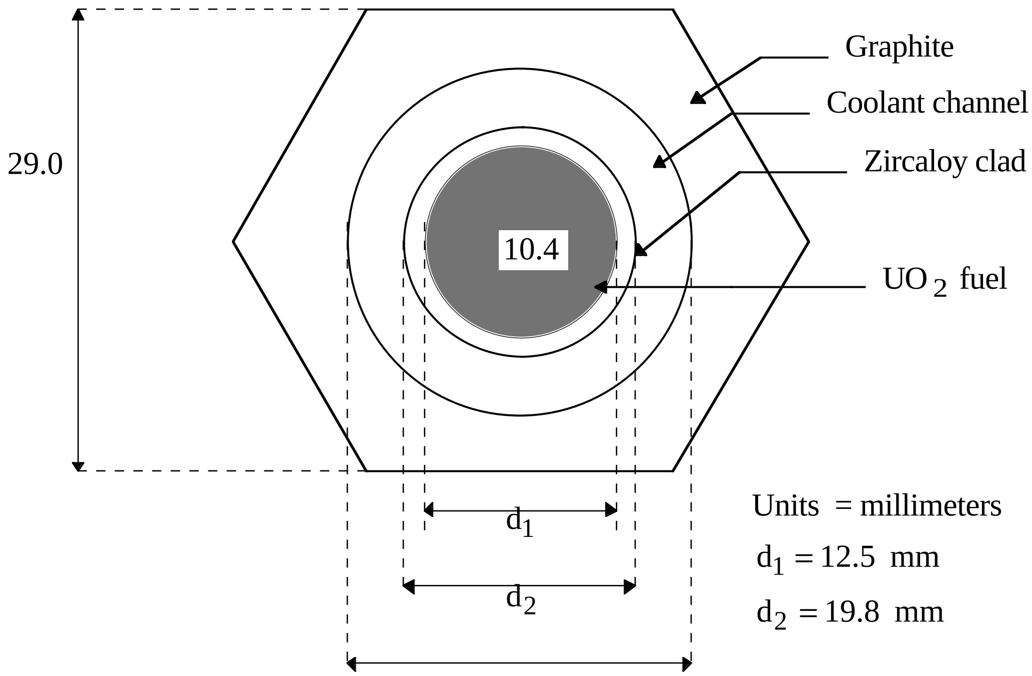


Figure 1

Unit cell using BWR fuel pin in MHTGR prismatic block holding the ratio of fuel to graphite constant.

Coolant channel size established by taking the area of water normally associated with a pin in conventional BWR.

Configuration of Solid Core and Variables

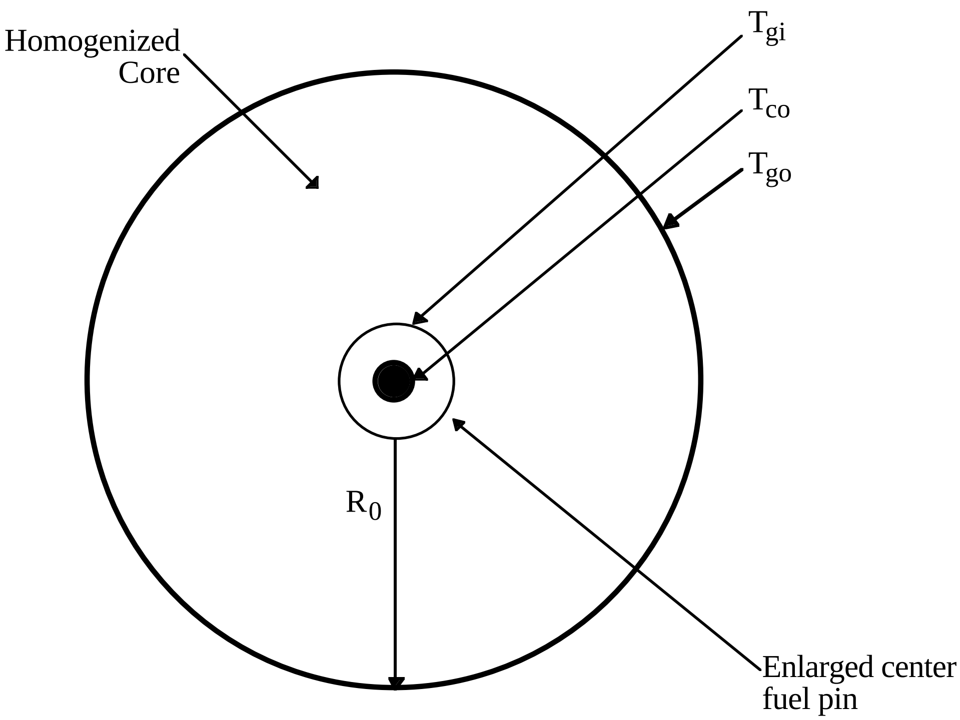


Figure 2

Notes:

T_{gi} is the temperature at the inner surface of the matrix graphite surrounding the fuel pin.
T_{go} is the temperature of the matrix graphite at the core outer surface.
T_{co} is the temperature at the clad outer surface.
d_1 and d_2 are shown in the Fig. 1.

PROBLEM 8-10N QUESTION

Important Features Of Fuel Element Temperature Calculation

Note: You may use the MathCAD program to solve this problem set but the use of MathCAD is not mandatory.

This problem set illustrates some important features of fuel element temperature calculations.

Consider an LMR fuel rod with the dimensions, thermal calculations characteristics, and operating conditions as follows:

- The clad is an austenitic stainless steel with outside diameter = 8.5 mm; thickness = 0.7 mm; and thermal conductivity = 23 $W/(m \cdot k)$.
- The fuel is enriched UO_2 contained in hollow pellets with outside diameter = 6.9 mm; inside diameter = 0.8 mm; as-fabricated density = 88% TD (theoretical density); and a thermal conductivity versus temperature as defined in a section "fuel conductivity" at the end of this problem statement.
- The gap conductance = 14 $kW/(m^2 \cdot K)$; and the heat transfer coefficient at the clad outer surface = 170 $kW/(m^2 \cdot K)$.
- The linear heat deposition rate = 57 kW/m; and the coolant temperature = 540°C.

Treat porosity as spherical using the Biancharia relation of the text Eq. 8-21. Neglect cracking and relocation effects.

1) Consider first that there is no restructuring.

   What are the temperatures at the following locations: clad outer radius (R_{co}); clad inner radius (R_{ci}); fuel outer radius (R_{fo}); and fuel inner radius (R_{fi})? Prepare a sketch (to scale and similar to text fig. 8-17 showing temperature versus radius.

2) Consider the same fuel rod and the same operating conditions but consider also that restructuring has occurred. The sintering temperatures and densities are those of Westinghouse in text Table 8-5.

   What are (T_{fi}), (R_1), (R_2), and (R_{fi})? Add a "restructured" temperature distribution to the sketch of part 1.

FUEL CONDUCTIVITY

This conductivity information applies to 95% TD UO_2. It is based on Lyon and is taken from the B&W polynomial of text Eq. 8-16c. The plot of text Fig. 8-1 is also applicable. SI units are adopted here (text information is non-SI).

Temperature (˚C)	Thermal Conductivity (W/m·k)	Conductivity Integral (kW/m)
0	9.32	0
200	6.38	1.54
400	4.78	2.64
600	3.85	3.50
800	3.26	4.20
1000	2.86	4.81
1200	2.58	5.35
1400	2.41	5.85
1600	2.29	6.32
1800	2.29	6.78
2000	2.31	7.24
2200	2.39	7.71
2400	2.53	8.20
2600	2.74	8.73
2800	2.94	9.30

PROBLEM 8-12N QUESTION

Transient Fuel Pin Analysis

A PWR fuel pin experiences DNB followed by SCRAM with the resulting reduction of power. The transient begins with full power steady-state operation at time t = 0.0, followed by DNB with a decrease in the fluid film heat transfer coefficient at the outside surface of the fuel rod. The reactor power drops to 10% of initial power with SCRAM at t = 4.0 sec., which reflects both the decay heat and some delayed fission. This situation is assumed to remain constant during the time of interest for this calculation (less than 15 seconds).

The data to be used to describe the fuel rod follows:

<u>dimensions (ft.);</u>

r_f = 0.016958

r_{ci} = 0.017708

r_{co} = 0.020542

<u>fuel linear heat rate (B/hr.ft);</u>

45734.0 at t = 0.0 to 4.0 sec.

4573.4 at t = 4.0 to end of calculation

<u>thermal properties;</u>

k_f = 1.15 B/hr.ft.F = 1.25 (sintered)	p_f = 635.0 lbm/ft^3 = 677.4 (sintered)	c_f = 0.08 B/lbm.F = 0.08 (sintered)
k_c = 10.0	p_c = 409.0	c_c = 0.08
h_{gap} = 1000 B/hr.ft.^{2}F	Sintering temperature, T_s = 3,092.0˚F	

<u>boundary conditions;</u>

T_b = 536.0 F

h_f = 10000 B/hr.ft.^{2}F at t= 0.0

= 100 from t = 0.0 to end of calculation

<u>computational parameters;</u>

number of iterations between plots (NP) = 100

maximum number of iterations (MAX) = 2500

number of fuel nodes (N) = 25

total number of nodes (M) = 29 (4 clad nodes)

time step in hours (DELT) = 0.000001388

node of sintered radius (NS) = 13

a. Provide plots of the following for the first 12.5 sec. of the transient and for sintered and unsintered fuel:
   (i) maximum fuel temperatures
   (ii) maximum clad temperatures (do not read from plots)
   (iii) heat flux at the outer surface
   (iv) stored energy in the fuel
   (v) temperature drop across the gap (do not read from plots)

b. Stainless Steel may be used as an alternative to Zircaloy for clad material. Using the solid (unsintered) fuel model, provide plots for the parameters listed in part a. where the clad has the following properties:

k_c = 15.0 B/hr.ft.F
c_c = 0.12 B/lbm.F
p_c = 500.0 lbm/ft.^{3}

c. Swelling of the fuel pellet and creep-down of the clad can result in pellet-clad interaction which can be harmful to the clad integrity. Another result of this phenomenon is the enhancement of heat transfer between the fuel and clad. Provide plots for the parameters listed in part a. where the solid fuel has interacted with the Zircaloy clad and the heat transfer coefficient at the gap has increased to:

h_{gap} = 5000 B/hr.ft.^{2}F

d. Comment on the following:
   (i) the effect on the results of including sintering in the model
   (ii) the effect of changing the clad material to stainless steel
   (iii) the effect of pellet-clad interaction on the thermal results
   (iv) discuss the reasons for maximum values in some of the parameters after time t = 0.0

Note: plot the same parameter for the various conditions on one graph to facilitate comparisons

Information for computer usage:

Submit card decks at IPC counter on second floor of building 39 with the following JCL;

```
//AAAAAAAA JOB     BBBBBB,                AAAAAAAA is the jobname
//  PROFILE = 'LOW, MEMORY = 256', TIME = 1      BBBBBB is the group user identification no.
//*PASSWORD CCCCCC                               CCCCCC is the group user password
// EXEC FTG1XEQ,PROG = 'SAWDYE.DDDDDDDD(NELIB)'
                                                 DDDDDDDD is the program name;
//GO.SYSIN DD *                                      TRANROD for the solid fuel
                                                     pellet model
         (data cards)                                TRANRODS for the sintered
                                                     fuel model
/*
```

Output will be placed in BIN no. 422 at the user counter with the group identification printed on the cover sheet

student groups	A	B	C	D
identification	NE313A	NE313B	NE313C	NE313D
passwsord	A313	B313	C313	D313

data format from listings on the following pages

PROBLEM 8-14N QUESTION

Radially Averaged Fuel Temperature And Stored Energy In Solid And Annular Pellet

Consider a solid pellet of radius b and an annular pellet of inside radius a, and outside radius b, each operating at the same linear power rate, q'.

Define $\Delta T(r) \equiv T(r) - T_b$

and $\overline{\Delta T(r)} \equiv \overline{T(r)} - T_b$.

- Find across each pellet, the value of $\overline{\Delta T} / \Delta T$. Use the subscript "s" for solid and the subscript "a" for annular.
- What is the ratio of the stored energy in the solid to the annular pellet?

PROBLEM 8-15N QUESTION

Two-Zone Sintering Of An Annular Fuel Pellet

An initially annular UO_2 fuel pellet is put under operation at 600 W/cm with an outside surface temperature of 700˚C. Assume sintering occurs reconfiguring the pellet with the linear power and outside surface temperature maintained constant.

<u>QUESTIONS</u>

a) Draw a temperature-radius plot for the pellet before and after sintering (numerical values not required, but do illustrate relative magnitudes, i.e., a figure analogous to Fig. 8-16 showing T_{max}, T_s, T_{f_o}, R_{v_i}, R_{v_f}, R_{s_i}, R_{s_f}, and R_{f_o}, where state i is at power but before sintering, and state f is at power but after sintering).

b) What is the maximum pellet temperature achieved after sintering occurs?

You may assume a two-zone sintering model where:

$$T_{sintering} \equiv T_s = 1{,}800\text{˚C} ,$$

$$\rho_{sintered} \equiv \rho_s = 98\% \, \rho_{TD} .$$

Take the initial annual pellet, $\rho_i = 88\% \, \rho_{TD}$, and of dimensions:

$$D_{fo} \text{ (outside fuel diameter)} = 9.5 \text{ mm} ,$$

$$D_v \text{ (void diameter or inside fuel diameter)} = 4.5 \text{ mm} .$$

PROBLEM S1-2 QUESTION

Stress Field Determination Using Mohr's Circle

A thin plate lying in the xy plane (small thickness in the z direction) is acted upon by forces so that the stress components with respect to the xy axes are as given in Fig. 1.

QUESTIONS

A. Find the stress components (σ_a, σ_b, τ_{ab}) with respect to the ab axes, which are inclined at 45° to the xy axes.

B. Display results from A above on the inclined element below in Fig. 2.

C. Display all results from Fig. 1 and part A above on a Mohr's circle. Also compute and include the magnitude of the radius of the circle and the magnitude of the angle between the stresses on the x plane and the plane of the principle stresses.

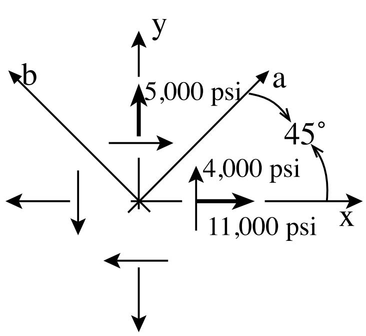


Figure 1

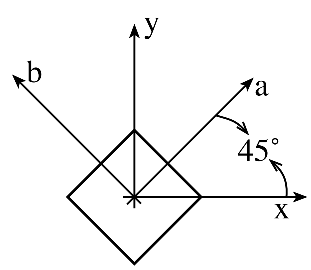


Figure 2

PROBLEM S4-2 QUESTION

Interior Pressure to Fail Cylindrical and Spherical Vessel by Several Failure Theories

Determine the internal pressure required to yield the inner surface of each vessel (assure plane stress for the cylinder) for the maximum normal stress theory, the maximum shear stress theory, and the maximum distortion energy theory. Hence, you are asked to provide six answers to fill in the following matrix.

Internal Pressure to Cause Yielding at Inside Surface	Cylinder	Sphere
Maximum normal stress theory	?	?
Maximum shear stress theory	?	?
Maximum distortion energy theory	?	?

Assume that the cylindrical and spherical vessels illustrated in Fig. 1 are made of steel with properties S_y = 100 ksi and ν = 0.3

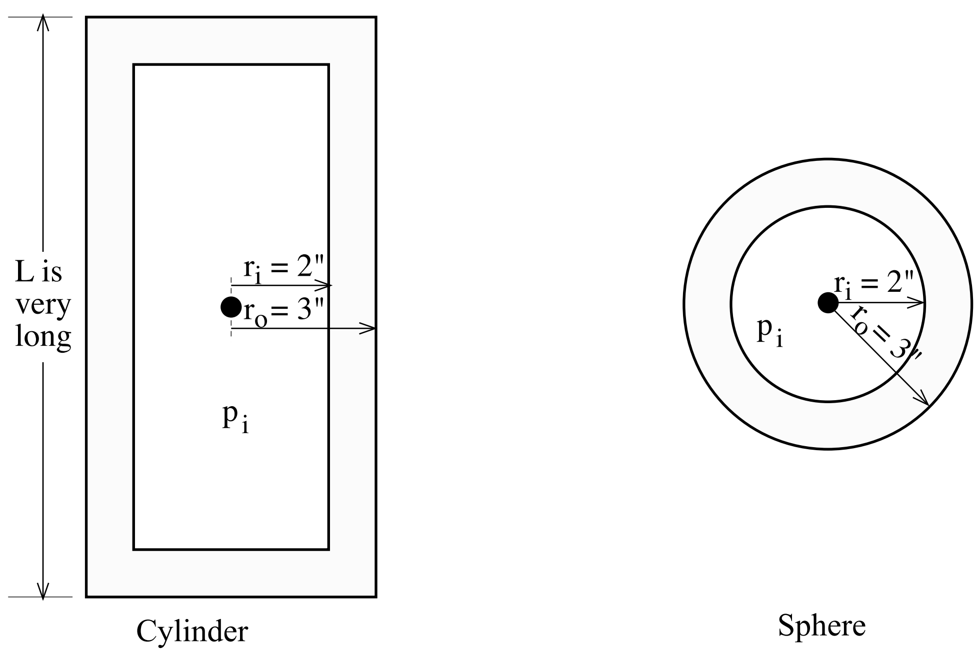


Figure 1

PROBLEM S5-1 QUESTION

Determining Allowable Pressure by ASME Criteria

Consider the reactor vessel geometry, material properties and operating conditions given below. Based on these results, evaluate the design pressure, P_d, at which the following ASME code limits will be violated:

1. The limit on General Primary Membrane Stress, P_m,
2. The limit on Primary Membrane plus Bending plus Secondary Stresses, and
3. The limit on Fatigue Usage Factor considering pressurization cycles only (i.e., use Fig. N.415(A) of ASME Code, pp. 5, 6 and 17-26, "Rules for Construction of Nuclear Vessels," ASME Boiler and Pressure Vessel Code Section III, ASME, 1968).

The steel used for construction has the following properties:

Young's Modulus	= E	= 1.91×10^{11} Pa
Poisson's Ratio	= ν	= 0.286
Yield Stress	= s_y	= 345 MPa = 2.45×10^8 Pa
Ultimate Stress	= S_u	= 552 MPa = 5.52×10^8 Pa
Design Fatigue Curve	= Dashed curve of Fig. N-415(A) of ASME Code, pp. 5, 6 and 17-26, "Rules for Construction of Nuclear Vessels," ASME Boiler and Pressure Vessel Code Section III, ASME, 1968	

The maximum values of stress occurring at the inside surface of the sphere are:

$$\sigma_t\big|_{max} = 10.53\ p$$

$$\sigma_\ell\big|_{max} = 9.85\ p$$

$$\sigma_r\big|_{max} = -\ p$$