ATOMIC PHYSICS

I. Structure of Matter

A. Element: A substance which cannot be seperated into any other substance different than itself by ordinary means.

B. Atom: Smallest Particle of a subdivided element having all the properties of that element.

C. Molecule: Smallest quantity of a substance which can exist by itself.

D. Compound: A material that consists of two or more different kinds of elements.

II. Fundamental Particles

A. Proton: A positive charged particle having a mass of 1.67243×10^{-24} grams and a charge of +1.

B. Electron: A negative charged particle having a mass of 9.1085×10^{-28} grams and a charge of -1.

C. Neutron: A neutrally charged particle having a mass of 1.67474×10^{-24} grams.

III. Structure Of Atoms

A. Atom Structure

1. An atom consists of a hard core nucleus containing protons and neutrons surrounded by orbiting electrons.
2. Diameter of nucleus = 10^{-12} cm

III. Structure of Atoms (Con't)

A. Atom Structure

3. Diameter of atom = 1×10^{-8} cm

B. Chemical and Physical Properties

1. Element identification by the number of protons within the nucleus.

2. Isotopes: One of several nuclides having the same number of protons in their nucleus but containing a different number of neutrons.

C. Identification of Atoms

1. By a abbreviated name of the element.

2. By the number of protons contained in the nucleus (Z) called the "atomic number".

3. By the total number of protons and neutrons within the nucleus (A) called the "atomic mass number".

$A - Z = N$ where N equals the number of neutrons within nucleus.

4. Example: Element Uranium 238 is symbolized by $_{92}U^{238}$.

D. Atomic Weight

$1\ u = 1.6604 \times 10^{-24}$ Kg = 1/12 weight of C^{12} atom

1. Atomic weight of an atom is equal to the actual weight of the atom expressed in atomic mass units where one atomic mass unit (AMU) is equivalent to 1.6605×10^{-27} Kg.

Example:

	Atomic Weight (AMU)
Proton	1.00758
Electron	0.00055
Neutron	1.00897

E. Avogadros number

1. Avogadros number (6.02×10^{23}) is a number which equals the number of atoms contained in one gram atom of any substance or the number of molecules contained in one gram mole of any substance.

2. A second form of Avogadros Law applies to gases. At standard temperature and pressure (STP), one gram molecular weight of any gas occupies 22.4 litters.

   (a) STP is equivalent to $0^{\circ}C$ ($32^{\circ}F$) and 14.7 Psi.

IV. Physical Chemistry

A. Electronic Structure of an atom

(a) Electrons are arranged in a series of circular orbits.

(b) Each orbit has a definite maximum number of electrons $2(N)^2$

(c) Orbit or shell identification from innermost shell outward: K,L,M,N,...etc.

B. Chemical Properties of an Atom

1. Determined by electron configuration of the atom.

2. Most important electrons - ones contained in outer most shell.

3. Atoms containing 8 electrons in outer shell are the most stable atoms, and therefore, have no tendency to undergo chemical reactions. These elements are found to be gases and are called inert gases. Ex. Helium, Argon etc.

4. Those elements with < eight electrons in their orbit shell tend

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4. HEAT TRANSFER AND HEAT GENERATION

A. General

(1) The study of heat transfer is concerned with the details of the rate of flow of heat energy between bodies. It has been stated that heat will flow between two bodies if one is at a higher temperature, but for design purposes it is also important to know how fast the process will occur. The three basic categories of heat transfer are:

- conduction
- convection
- radiation

(2) Conduction is direct transference of heat by molecular impact. When a part of a metal bar is heated, molecules at the point being heated vibrate more and more rapidly, collide more vigorously with their neighbors, and transmit some of their energy to them. A good example of conduction is the flow of heat up a spoon from the coffee in your cup to your fingers.

(3) Heat transfer by convection is the transfer of thermal energy by the motion of a fluid that is being heated. There are two general categories of convection, called natural convection and forced convection. Natural convection takes place because the density of the heated fluid is less than that surrounding it and as a result it rises. This can be seen on an automobile on a hot summer day. The air touching the metal surface is heated by conduction and convection currents are set up, resulting in a transfer of heat away from the surface. Forced convection occurs when the fluid motion is caused by some factor other than the density difference, such as pumps or fans. For forced convection the effect of the different fluid densities usually has little significance. A good example of convection is the operation of a coffee percolator, which runs on natural circulation.

(4) For radiation heat transfer, physical contact between two bodies is not required. This is the method by which the sun warms the earth or by which heat is felt from an open flame. Thermal radiation is not a significant area of concern for power reactors except in certain accident situations, and thus will not be discussed further.

(5) All the laws of heat transfer give relations between temperatures and the rate at which heat energy flows from place to place. This rate of flow of heat energy is usually expressed in terms of heat flux:

$$\text{Heat flux} = \frac{\text{Heat energy that flows through a surface (Btu)}}{\text{Time it took (hr.) to flow} \times \text{Surface area (ft}^2\text{)}} \qquad (1)$$

Thus heat flux has the units:

$$\frac{\text{Btu}}{\text{hr-ft}^2}$$

When the surface encloses a steady source of energy, we can also say that:

$$\begin{array}{l}\text{Power produced}\\ \text{by energy source}\\ \text{inside}\end{array} = \frac{\text{Heat energy that flows through surface}}{\text{Time it took to flow}} \qquad (2)$$

so that we can re-write equation (1) as

$$\text{Heat flux} = \frac{\text{Power inside}\left(\frac{\text{Btu}}{\text{hr.}}\right)}{\text{Surface area (ft}^2\text{)}} \qquad (3)$$

One must be careful not to confuse the concept of a flux as applied to heat transfer with the concept that the reactor physicists use. As used here, flux is the flow of something (heat energy in this case) per unit area and per unit time.

(6) If there is a use of knowing it, one can define the rate at which people pass through a door as a "people flux." For instance, if a door is 2 ft x 6 ft and 100 people go through it in 2 hours, then the people flux is:

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$$\text{People flux} = \frac{100 \text{ people}}{2 \text{ hr} \times 12 \text{ ft}^2}$$

$$= 4.16 \text{ people/hr-ft.}^2$$

(7) Example:

One fuel rod produces a power of 253,000 Btu/hr. Its clad outside diameter is 0.422 inches and its length is 12 ft. Find the average heat flux at the surface of the clad.

$$\text{Heat flux} = \frac{\text{Power (Btu/hr)}}{\text{Area (ft}^2\text{)}}$$

$$\text{Power} = 253{,}000 \text{ Btu/hr}$$

$$\text{Area} = \pi \times \text{diameter} \times \text{height}$$

$$= 3.14 \times 0.422 \text{ in.} \times \frac{1 \text{ ft}}{12 \text{ in}} \times 12 \text{ ft.}$$

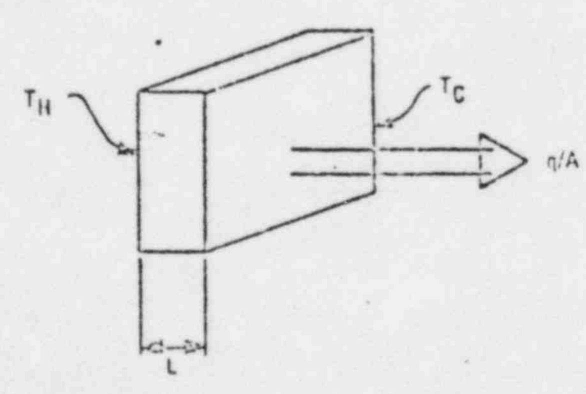
$$= 1.325 \text{ ft.}^2$$

$$\text{Heat flux} = \frac{253{,}000 \text{ Btu/hr}}{1.325 \text{ ft}^2} = 191{,}000 \frac{\text{Btu}}{\text{hr-ft}^2}$$

(8) The symbol q/A is often used to designate heat flux.

B. Conduction

(1) The basic law of conduction heat transfer is best illustrated for a slab of material



If heat is being transferred from temperature T_H to T_c, the rate of heat flow per unit area (that is, the heat flux q/A) is :

$$q/A = \frac{k\ (T_H - T_c)}{L} \tag{4}$$

where

q/A = heat flux through the slab (Btu / hr - ft^2);

T_H, T_c = temperatures (^oF) on the hotter and colder sides of the slab;

L = slab thickness (ft);

k = thermal conductivity (Btu/hr-ft-^oF)

(2) In Equation (4), $(T_H - T_c)/L$ can be thought of as a "hill" down which the heat "rolls:" Figure 4A(1) illustrates this interpretation by plotting temperature vertically and distance horizontally, the hot end of the slab being at the left and the cold end at the right. Since heat flows from hot to cold, it can be thought of as "rolling down the hill." We can make the hill steeper by decreasing L Figure 4A(2)), by increasing T_H (Figure 4A(3)), or by decreasing T_c(Figure 4A(4)). In any of these cases, the heat flow (that is, the heat flux) will increase as we would expect.

(3) The thermal conductivity (k) depends on the particular material of which the slab is made. Some expamples are:

Material	k (Btu/hr-ft-^oF)
Copper	232
Steel or Zirc	8-10
Concrete	0.5 to 0.8
Asbestos	0.1
UO_2	1.5 to 3

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Figure 4A. HEAT CONDUCTION

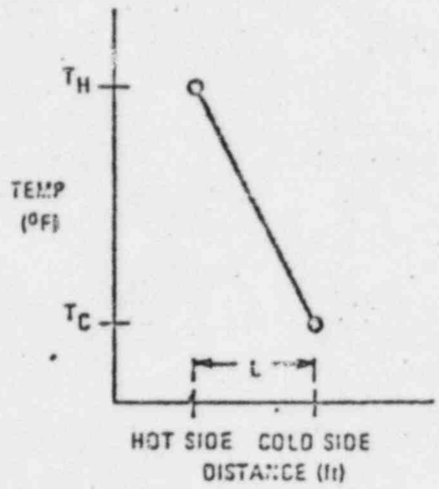

Figure 4A (1) HEAT CONDUCTION DOWN THE HILL

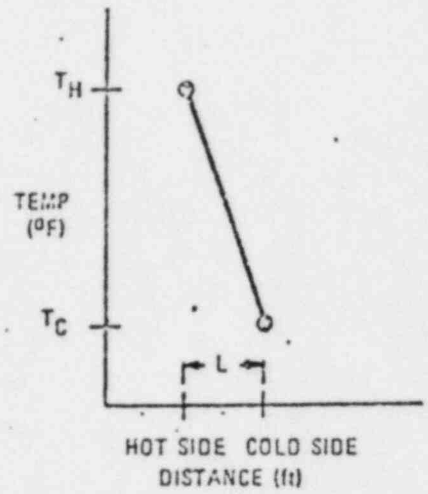

Figure 4A (2) INCREASING HEAT CONDUCTION BY DECREASING L

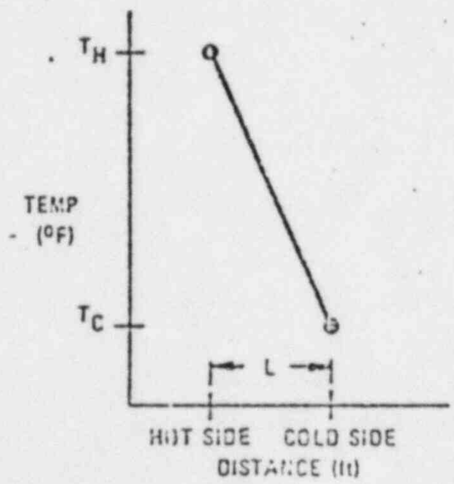

Figure 4A (3) INCREASING HEAT CONDUCTION BY INCREASING T_H

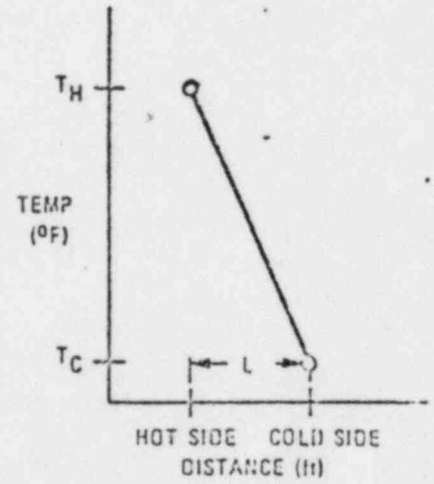

Figure 4A (4) INCREASING HEAT CONDUCTION BY DECREASING T_C

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You can see why asbestos is a good thermal insulator and copper is a good conductor. We have shown a range of values for the conductivity of UO_2 because it depends on the temperature of the UO_2.

(4) Example

A steel tank has a 2-inch thick wall with an inner surface temperature of 300°F. The outer surface temperature is 100°F. How much heat is lost through each square foot of wall?

$$\frac{q}{A} = k\frac{(T_H - T_c)}{L}$$

$$= 8\,\frac{\text{Btu}}{\text{hr-ft-}^{\circ}\text{F}}\;(200^{\circ}\text{F})\;\frac{1}{2\text{ inches}}\;12\text{ inches/ft.}$$

$$= 9600\text{ Btu/hr-ft}^2$$

If 2-inch asbestos insulation is put on the tank such that the outer steel surface is 300°F and the outer asbestos surface is now 100°F, how much heat will be lost per square foot?

$$\frac{q}{A} = k\frac{(T_H - T_c)}{L}$$

$$= 0.1\,\frac{(200)}{2}\,12$$

$$= 120\text{ Btu/hr-ft}^2$$

(5) Example

The temperature of the clad on a PWR fuel rod is 613°F at the outer surface and 661°F at the inner surface. The clad thickness is 0.024 inches. What

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is the heat flux through the clad?

$$T_H = 661^\circ F$$

$$T_c = 613^\circ F$$

$$k = 8\ \text{Btu/hr-ft-}^\circ F$$

$$L = 0.024\ \text{in} \times \frac{1\text{ft}}{12\text{in}} = 0.0020\ \text{ft.}$$

$$\frac{q}{A} = \frac{8\text{Btu}}{\text{hr-ft-}^\circ F} \times \frac{(661-613)^\circ F}{.0020\ \text{ft}}$$

$$= 192,000\ \text{Btu/hr-ft}^2$$

(6) Equation (4), which describes heat conduction, is written so that it is convenient to calculate the heat flux q/A if the values of k, T_H, T_c, and L are known. This equation can be re-arranged for cases in which other combinations of the variables are known. Multiplying both sides of equation (4) by L/k gives:

$$T_H - T_c = \frac{L}{k}\left(\frac{q}{A}\right) \qquad (5)$$

Equation (5) expresses the temperature difference across a heat conducting slab in terms of its thickness L, its conductivity k, and the heat flux q/A.

Adding T_c to both sides of equation (5) gives:

$$T_H = T_c + \frac{L}{k}\left(\frac{q}{A}\right) \qquad (6)$$

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Equation (6) expresses the temperature T_H on the hot side of the slab in terms of the temperature T_c on the cold side, its thickness L, its conductivity k, and the heat flux q/A.

Subtracting (L/k) (q/A) from both sides of equation (6) gives:

$$T_c = T_H - \frac{L}{k}\left(\frac{q}{A}\right) \tag{7}$$

Equation (7) expresses the temperature T_c on the cold side of the slab in terms of the temperature T_H on the hot side, its thickness L, its conductivity k, and the heat flux q/A.

(7) Some examples of heat conduction in nuclear plants, and the typical temperatures involved, are shown below:

Location	Hot Side	Cold Side
Fuel Pellet	Centerline (3000°F)	Surface (750°F)
Fuel Clad	Inner (650°F)	Outer (600°F)
Reactor Vessel	Inner (500°F)	Outer (400°F)
Steam Generator	Tube Inside (600°F)	Tube Outside (530°F)

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C. Convection

(1) In heat transfer by convection, heat energy is transferred away from a surface by the motion of fluid on one side of it. The equation which describes convection is a relation between the heat flux, the surface temperature, and the average fluid temperature:

Heat flux = constant x (surface temp. - fluid temp.)

$$q/A = h\,(T_H - T_c) \tag{8}$$

where

q/A = heat flux (Btu/hr-ft^2)

T_H = temperature ($^\circ$F) of the hotter material - that is, of the solid surface;

T_c = temperature ($^\circ$F) of the cooler material - that is, of the fluid such as water;

h = heat transfer coefficient (Btu/hr-ft^2 $^\circ$F)

(2) Equation (8) for convection is similar though not identical to equation(4) for conduction. Just as for conduction, increasing T_H or decreasing T_c will increase the heat flux by convection.

(3) The value of the heat transfer coefficient h depends both on the properties of the surface and on the properties of the fluid which removes the heat. Varying the density, viscosity, velocity, or temperature of the fluid will change the value of h. Since an

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increase in heat flux itself changes the properties of the fluid by increasing its temperature, the value of h may also be thought of as depending on the heat flux.

(4) The temperature of the inner clad surface on a PWR fuel rod is 661°F and the temperature of the fuel pellet surface is 914°F. The heat transfer coefficient of the gap is 1000 Btu/hr-ft²-°F. What heat flux is passing across the gap?

$$T_H = 661\ ^\circ F$$

$$T_c = 914\ ^\circ F$$

$$h = 1000\ \text{Btu/hr-ft}^2\ ^\circ F$$

$$q/A = \frac{1000\ \text{Btu}}{\text{hr-ft}^2\ ^\circ F} \times (914 - 661)^\circ F$$

$$q/A = 253{,}000\ \text{Btu/hr-ft}^2$$

(5) Some examples of convection in nuclear power plants are given below:

Location	Hot Material	Cool Material
Clad-pellet gap	Pellet surface (750°F)	Clad inner surface (650°F)
Clad-water	Clad outer surface (560°F)	Water (540°F)

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D. Boiling Heat Transfer

One of the most effective heat transfer mechanisms is boiling heat transfer. Why does boiling increase the heat transfer rate? Because the steam bubbles leaving the surface carry a greater amount of energy than the same amount of liquid and also cause a greater amount of coolant motion adjacent to the surface.

There are two kinds of boiling, nucleate and film.

(1) Nucleate Boiling

The word nucleus is closely related to the word nucleate. In fact the word nucleate means to form a nucleus - to cluster. In nucleate boiling we are generating small spheres or bubbles of steam, distinct and separate. These bubbles then, break away from the heated surface and are carried into the coolant where they may join or cluster with other bubbles.

(2) Film Boiling

In this kind of boiling, a thin film of vapor completely covers the heated surface. The primary mode of heat transfer is radiation through the vapor. Surface temperature increases greatly since the vapor film acts as a barrier to conduction or convection heat transfer.

For either of these two kinds of boiling, there are two conditions at which they can occur. These conditions are called subcooled boiling and bulk boiling.

(3) Subcooled Boiling

When either nucleate or film boiling is occurring and the average fluid temperature in which the boiling is taking place is less than the saturation, the condition is called subcooled boiling. There is no net steam generation since all the steam is condensed in the cooler water when it breaks away from the heated surface.

(4) Bulk Boiling

When either nucleate or film boiling occurs in a fluid that is at saturation temperature, the condition is called bulk boiling. In this condition, net steam generation is realized.

In a PWR steam generator and a BWR core where there is net production of steam, the heat transfer mechanism is nucleate boiling under both subcooled and bulk boiling conditions. In a PWR core, the primary heat transfer mechanism is subcooled nucleate boiling.

As an aid in further understanding the boiling phenomenon, let us imagine the following experiment. A wire is immersed in a tank of water at room conditions. The ends of the wire are connected to a source of electric current. Figure 4B shows a schematic of this system. Instruments for measuring water and wire temperature and electric power are also provided.

o Phase I - Natural Convection

The power is turned on and the wire becomes heated. Water adjacent to the wire becomes warmer and rises, because of its lower density; and is replaced by cooler water. This is called natural convection heat transfer. Now take note of the slope of the curve of this phase in Figure 4C

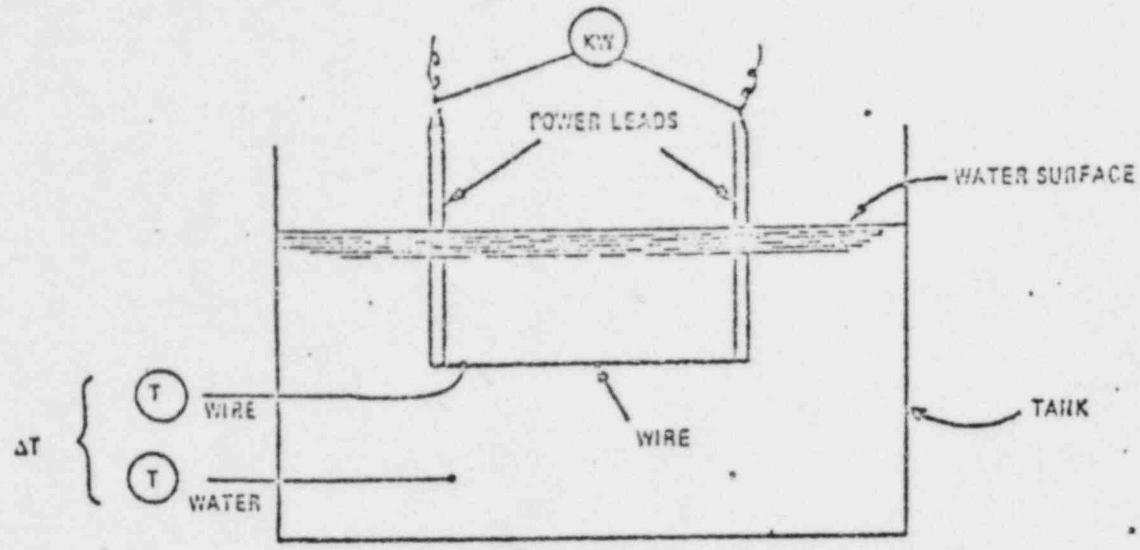

FIGURE 4B
HEATING ELEMENT IN WATER

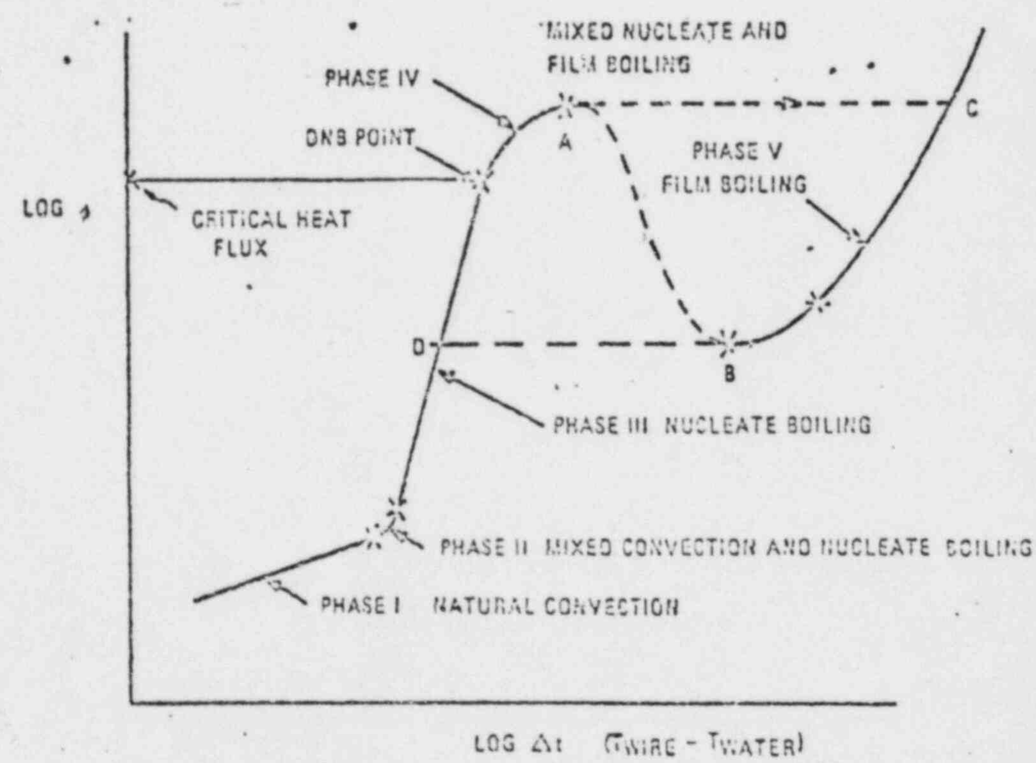

FIGURE 4C
BOILING CURVE

- Phase II - Transition from Convection to Nucleate Boiling

Power is increased until a few small bubbles begin to form on the wire. The situation now looks like Figure 4D. Note that there are still areas of the wire where no bubbles form and natural convection persists. Figure 4C shows how the bubble generation increases the heat transfer rate. Note how the slope of the curve increases in this phase. The bubbles of vapor eventually collapse due to condensing of the bubbles in the room-temperature water.

- Phase III - Nucleate Boiling

Increasing power still further increases the number of bubbles generated until the entire wire seems entirely covered with bubbles, which break away with increased rapidity. Some of the bubbles coalesce as they rise through the cooler water. The vapor in the bubbles eventually condenses and no vapor arises from the water surface. Note that if the wire were positioned near the surface, insufficient time would be available for the bubbles to completely collapse, and we would see vapor coming off the water. This type of boiling is, however, still known as subcooled nucleate boiling. If the power is held and all the water reaches 212°F, then the type and conditions become bulk nucleate boiling.

- Phase IV - Transition from Nucleate to Film Boiling

Instead of waiting for the water to reach saturation, power is increased further and we pass into a situation where bubble action is so vigorous that some of them coalesce before breaking away from the wire, causing part of the wire to be completely covered with a film of vapor. The flux where this starts to occur is called the critical heat flux, and, at this point,

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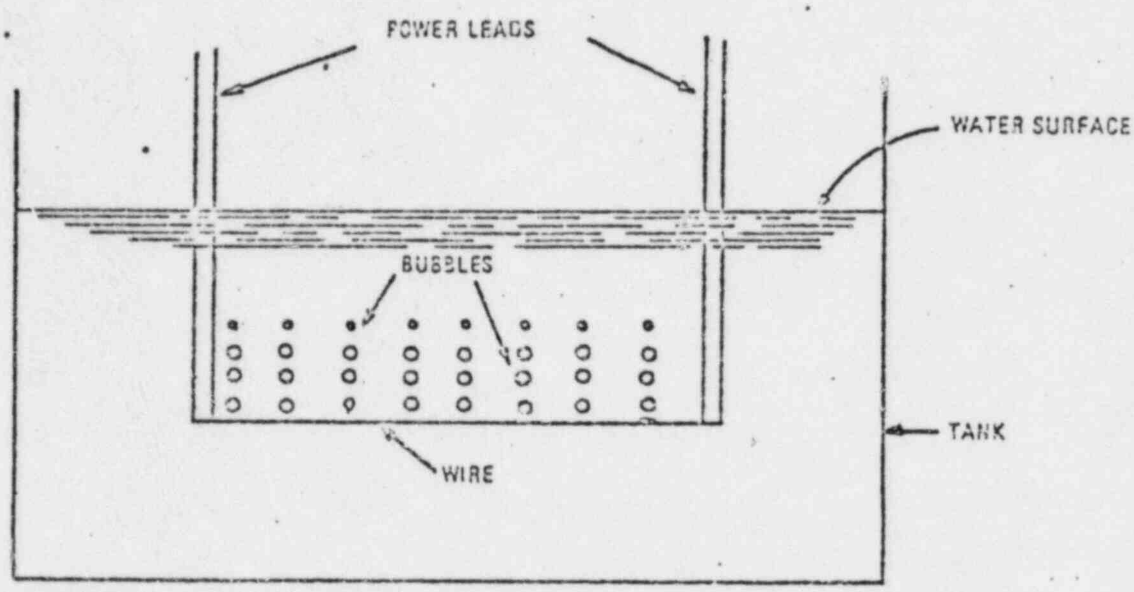


TRANSITION FROM CONVECTION TO NUCLEATE BOILING

FIGURE 4D

we depart from nucleate boiling and pass into film boiling. In fact, we call this point DNB, departure from nucleate boiling. If we try to increase flux even further, more of the wire becomes covered with vapor. If the wire does not melt, to sustain the rate of heat transfer, the temperature difference between wire and water has to increase considerably, and we find our system operating at point "C" in the film boiling regime. The transition from "A" to "C" is very rapid and its path is unknown; therefore, it is shown as a dotted line in Figure 4C. Note that if we decreased flux from point "A" we would start back toward the nucleate boiling region.

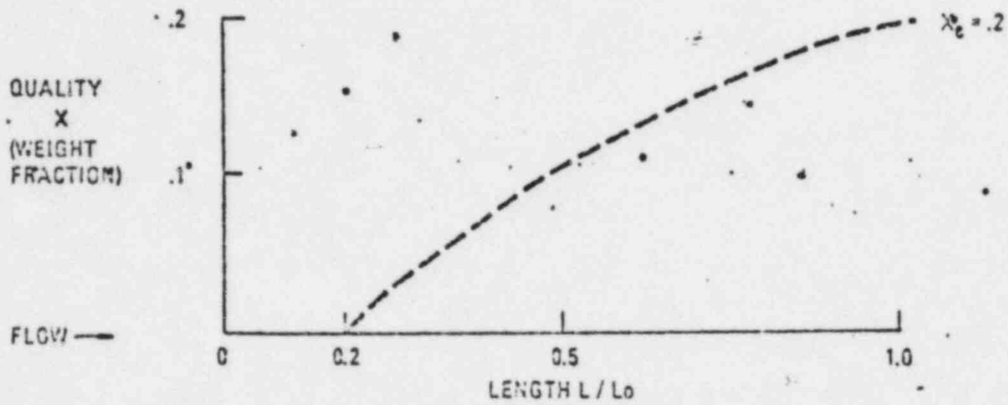
- Phase V - Film Boiling

The wire is now completely covered with a nearly cylindrical film of vapor which stays in place. The chief mode of heat transfer is by radiation from the wire to the water surrounding the vapor barrier along with a little conduction through the vapor. As pictured here, the boiling is considered subcooled. The usual material used for wire cannot withstand the high temperature and would melt if we did not quickly reduce power, thereby sliding along the curve of Figure 4C toward point "B". Continued lowering of the heat flux below point "B" would cause a sudden transition back into nucleate boiling at point "D".

The portion of the boiling curve from "A" to "B" is not well known and is impossible to obtain with the pool boiling experiment described above. The reason for this is that we can only control flux and not temperature difference

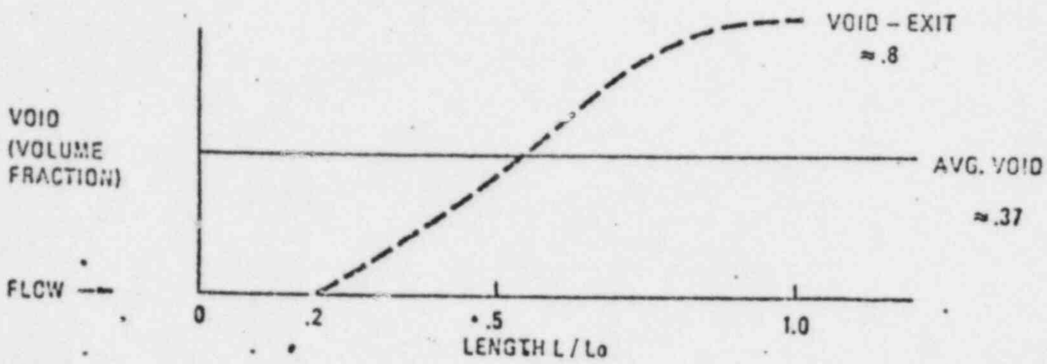
One of the important thermal design criteria for the design of a reactor core is that unanticipated transients must not result in reaching the critical heat flux point or, restated, must not cause departure from nucleate boiling.

Let's examine how varying some conditions affects the boiling phenomenon and how these effects are displayed to you. First, we can look at a profile of quality as a function of heat surface length, assuming that there is net steam generation.



In this instance, the first 20% of the heated surface is used to raise the temperature of the fluid from its subcooled condition to a saturated condition. In this region our heat transfer mechanism will be both convection and subcooled nucleate boiling. From 20% on, our heat transfer mechanism is bulk nucleate boiling.

Knowing the quality, we can compute the void fraction. The void fraction is the volume fraction of steam in a steam-water mixture. Its profile would look like this:



We can see that the void fraction is a very large number when compared to the weight fraction. This is the result of steam having a large specific volume-ft^3/lb.

Babcock and Wilcox supplies a once-through steam generator that provides superheated steam. In the case of this steam generator (shown in Figure 1L, page 1-19), the void fraction at the exit will be 1.0, since there is no moisture carryover.

The motivation for generating superheated steam is that more energy per pound of steam flow is supplied to the turbine. This improves the primary plant heat rate and the overall plant efficiency. The enthalpy of saturated steam at 925 psig is 1194.6 BTu/lb, while the enthalpy of 50°F superheated steam at 925 psig is 1244.5 BTu/lb.

The B & W steam generator transfers heat from the reactor coolant system to the feedwater, employing all of the heat transfer modes discussed in this chapter. The subcooled feedwater enters the steam generator and is mixed with some steam from the tube bundle area through nozzles in the lower baffle plate. This section of the steam generator is commonly referred to as either the downcomer or the economizer section. The steam next passes under the

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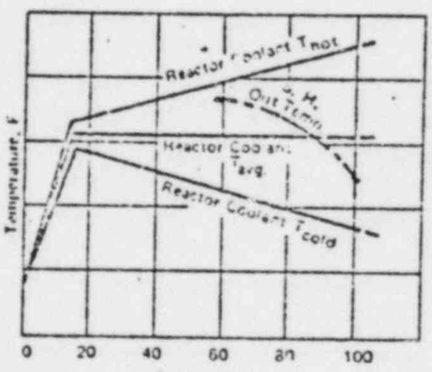
lower baffle plate and starts up along the tubes. It reaches saturated conditions in the lower part of the steam generator. The void content of the mixture increases almost uniformly in a region of nucleate boiling. At some level in the steam generator, the DNB point is reached and film boiling occurs. The upper region in the steam generator is where superheat occurs. This superheated steam then travels outside of the upper baffle plate before exiting to provide a blanket of superheated steam around the upper portion of the steam generator.

The amount of superheat obtained from the steam generator is a function of the plant load (or, more precisely, main steam-feedwater flow). The figure below shows the typical performance characteristics of the B & W steam generator.



At power levels above 15% load, the reactor coolant system is maintained at constant flow and T_{avg}. The secondary system is maintained at constant pressure, but the flow varies as a function of load.

Let's look at what happens if one feedwater heater is taken out of service. The feedwater inlet temperature will be lower. This will increase the subcooling load on the heat source, which means that a larger percentage of the source will have to be used to heat the feedwater up to saturation. This

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results in a decrease of steam generator volume for making steam and, finally, superheating. The result is that less superheat will occur and the steam outlet temperature will be lower.

- Decreasing feedwater temperature causes steam generator outlet temperature to drop.

What happens to the steam outlet temperature if the plant load is reduced in a controlled manner from 100% to 50%? When load is reduced, the feedwater flow will decrease significantly. The effect of this is that the same heat source volume is available to heat considerably less secondary process fluid. This results in more space being available for superheating and a resulting higher steam outlet temperature.

- Load or feedwater flow reductions result in an increase in the steam generator outlet temperature, except at low power levels.

The reactor coolant in a PWR system is kept under pressure to prevent bulk boiling in the core. In the case of an abnormal transient, where this pressure is lost and some steam is generated in the core, how will we know it? We will see a large increase in level in the pressurizer until pressure is built back up above the saturation value corresponding to the temperature in the core. The steam bubbles will then condense, and the level will drop back down close to its normal value.

Problem

The secondary side of a once-through steam generator has a net value of 3500 ft^3. At the time of a plant trip, the following operational characteristics applied to the steam generator:

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Subcooled water volume	10%
Boiling region volume with average void fraction of 0.7	70%
Superheat region	20%

If one auxiliary feedwater pump (assume constant capacity of 700 gpm) is started, how long will it take to fill the steam generator?

(5) Departure from Nucleate Boiling (DNB)

We stated earlier that when we reached the critical heat flux (CHF), we departed from nucleate boiling and entered a region where conditions were unstable. In this region, our heated surface temperature has to increase in order to maintain the heat rate. This, obviously, is an operating condition we would like to avoid, so we will, therefore, design our reactor so that the critical heat flux is not exceeded during normal operation and predictable transients. This will assure us that we stay out of the region where the fuel clad or wall temperatures exhibit large changes.

Operation in a region where we have departed from nucleate boiling does not necessarily mean that there will be fuel damage. We certainly increase our chances of having damage, however, as long as we are above the critical heat flux. We can define critical heat flux in a more positive way as the limit below which fuel damage will not occur. That definitely is a good region in which to operate our reactor, as long as we can adequately define and measure the critical heat flux.

Analytical expressions for critical heat flux have been generated, but, primarily, a correlation based on test results is used to predict the flux at which DNB occurs. The DNB correlation is a function of system pressure,

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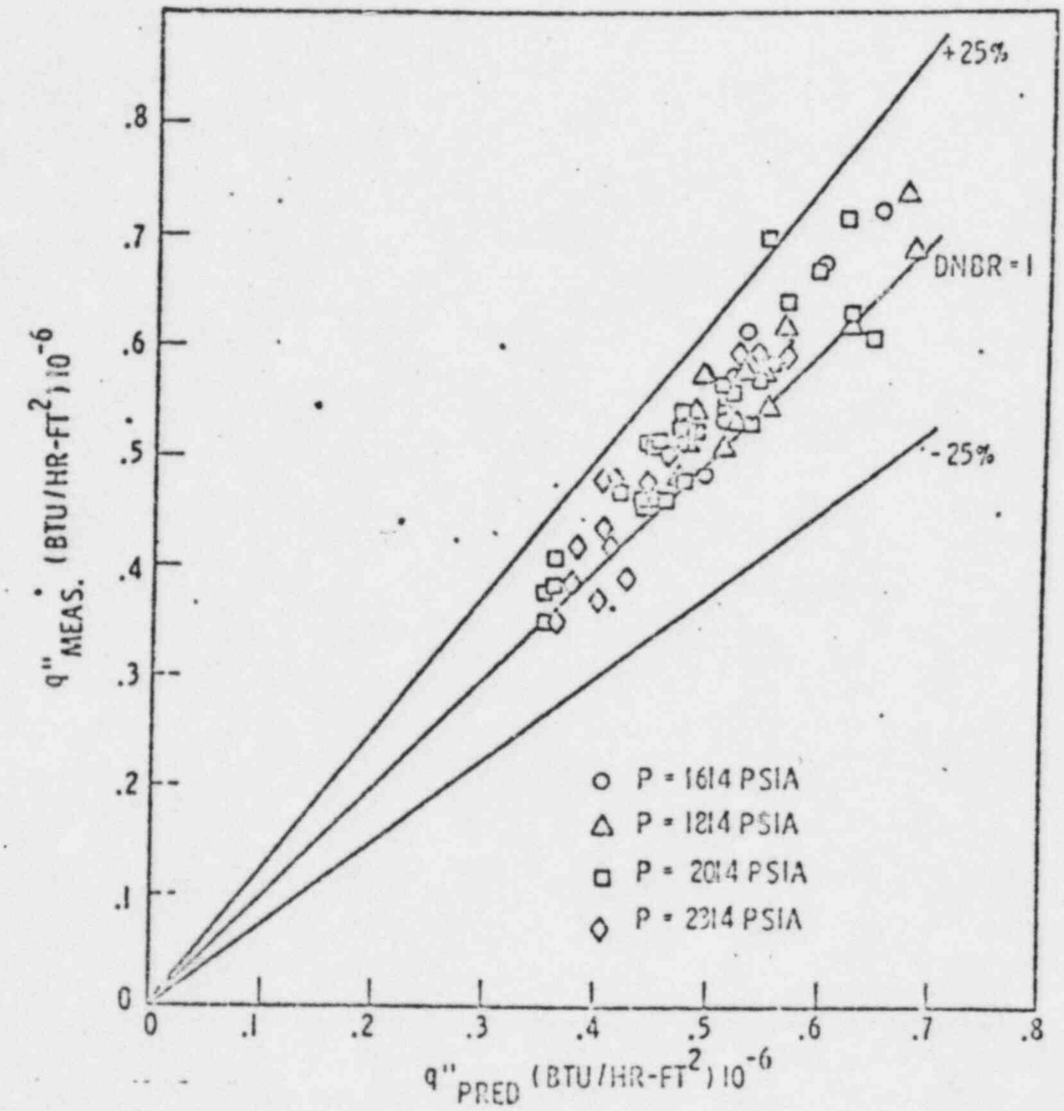
inlet enthalpy, flow velocity, and fuel assembly geometry. This correlation is such that the predicted CHF covers the data points within some error spread, such as ±25%.

A comparison of measured DNB flux (or CHF) with those predicted by the correlation is shown in Figure 4E. Work continues on the determination and measurement of DNB flux. A better understanding of DNB flux and how it occurs will lead to better reactor designs. In a later section we will examine DNB ratio, but first we have to look at how we determine the actual flux in core.

FIGURE 4E. COMPARISON OF MEASURED DNB FLUXES WITH THOSE PREDICTED BY THE CORRELATION



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Problem Set - Chapter 4

1. A particular PWR fuel rod is 12 ft. long and has a clad outside diameter of 0.422 in. Find the average heat flux at the surface of the rod if the power produced inside the rod is:

   (a) 500,000 Btu/lb

   (b) 400,000 Btu/lb

   (c) 300,000 Btu/lb

   (d) 200,000 Btu/lb

   (e) 100,000 Btu/lb

   (f) 0

2. The zirc-clad PWR fuel rod produces 226,000 Btu/hr-ft^2. The average temperature at the outside surface of the clad is 628°F. What is the average temperature at the clad inner surface? (clad thickness = .025 in.)

3. The heat transfer coefficient between the pellet surface and the inner clad for the fuel rod described in problem 2 is

   $$h = 1000 \text{ Btu/hr} - \text{ft}^2 - {}^\circ\text{F}$$

   What is the temperature at the pellet surface?

4. A once-through steam generator with a tube surface area of 60,000 ft^2 has an average heat flux across the tubes of 7.51×10^4 Btu/hr ft^2. Feedwater enters the steam generator at 455°F at the rate of 5.58×10^6 lb/hr and exits as superheated steam at 900 psia. Find the temperature of the steam at the exit.

5. CORE THERMAL PERFORMANCE

The manner in which heat is transferred within the core and the margin that is maintained from damage regions during that transfer are a measure of thermal performance. One term that has been identified and is used to determine thermal performance is heat flux. Another term that can be used to calculate performance is linear heat rate, which is a measure of the thermal output of the core or a fuel rod.

A. Linear Heat Rate

The "linear heat rate" (sometimes called the "thermal output") is defined by considering a length of fuel rod. This length may be an entire fuel rod, a part of one rod, or a combination of several fuel rods. The linear heat rate for this length of rod is defined as follows:

$$\text{Linear Heat Rate} = \frac{\text{Power Developed in that Length}}{\text{Length}}$$

The power in the numerator of this definition must always refer to the same length of fuel used in the denominator. The power is usually expressed in kw and the length in feet, so the units of linear heat rate are:

$$\text{Linear Heat Rate} = \frac{\text{kw}}{\text{ft}}$$

As an example, we will calculate the <u>average</u> linear heat rate in a typical PWR core:

Core power is 1650 MWt

The core consists of 121 fuel assemblies, each with an active length of 12 ft., and each containing 179 fuel rods.

$$\text{Power in kw} = 1650 \text{ MW} \times \frac{10^3 \text{ kw}}{\text{MW}}$$

$$= 1.650 \times 10^6 \text{ kw}$$

Total length of fuel rods =

$$\frac{179 \text{ rods}}{\text{assy}} \times 121 \text{ assy} \times \frac{12 \text{ ft.}}{\text{rod}} = 2.6 \times 10^5 \text{ ft.}$$

$$\text{Linear Heat Rate} = \frac{16.5 \times 10^5 \text{ kw}}{2.6 \times 10^5 \text{ ft}}$$

$$= 6.35 \frac{\text{kw}}{\text{ft}}$$

Notice that we were careful to use the length of fuel rod associated with the specified power production rate - the entire core in this case.

Let's assume that a particular fuel assembly in the core used in the previous example is generating 20 MW. What is the average linear heat rate in this assembly?

$$\text{Power} = 20 \text{ MW}$$

$$= 20 \text{ MW} \times \frac{10^3 \text{ kw}}{\text{MW}}$$

$$= 2 \times 10^4 \text{ kw}$$

$$\text{Length} = 179 \text{ rods} \times \frac{12 \text{ ft}}{\text{rod}}$$

$$= 2.14 \times 10^3 \text{ ft}$$

$$\text{Average Linear Heat Rate in Assy} = \frac{20 \times 10^3 \text{ kw}}{2.14 \times 10^3 \text{ ft}}$$

$$= 9.35 \text{ kw/ft}$$

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The average linear heat rate in this one assembly is higher than the average for the entire core. This is not unexpected, since some fuel assemblies generate more than average power and some generate less. In this respect, linear heat rate is like a team batting average. Some players on a team will be batting above the team average and others below it. Similarly, some fuel assemblies have a linear heat rate higher than average and others have a linear heat rate lower than average.

The average linear heat rate in any particular core is proportional to the thermal power of the core. This means that the graph of linear heat rate vs. core power is a straight line, as shown in Figure 5A. In this figure, the percents refer to percent of rated conditions, and 100% power means 100% of rated power. 100% linear heat rate means 100% of the average linear heat rate at rated power. The figure shows that you get 100% linear heat rate at 100% rated power; you get 50% linear heat rate at 50% power; you get 120% linear heat rate at 120% power. So, we can make a rule:

- Whatever you do to the total power in the core, you do to the average linear heat rate in the core. That is, if you double the core power, you double the average linear heat rate; if you halve the total core power, you halve the average linear heat rate.

Linear heat rate and flux are directly related. If you know one, you can calculate the other. It is necessary only to know the square feet of heat transfer area per foot length of fuel rod. To convert linear heat rate to heat flux, the following conversion is used.

$$\underset{(\text{kw/ft})}{\text{Linear Heat Rate}} \longrightarrow \underset{\text{Btu/hr - ft}^2}{\text{Heat Flux}}$$

$$\frac{\text{kw}}{\text{ft}} \times \frac{3413\ \text{Btu}}{\text{hr-kw}} \times \frac{12\ \text{ft}}{1.3\ \text{ft}^2} = \text{Heat Flux}$$

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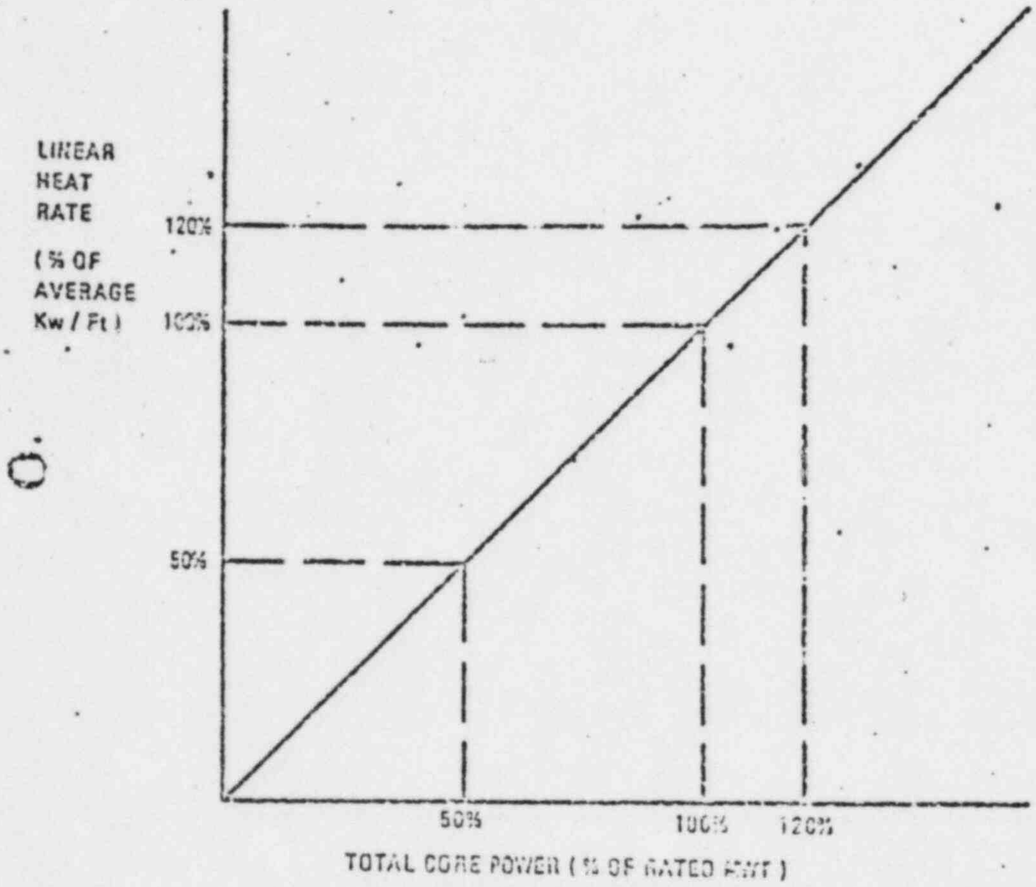

FIGURE 5A
VARIATION OF LINEAR HEAT RATE
WITH CORE POWER

For the core average linear heat rate, the equivalent average heat flux is:

$$\text{Average Core Heat Flux} = \frac{6.35 \times 3413 \times 12}{1.3} = 200{,}000 \ \frac{\text{Btu}}{\text{hr-ft}^2}$$

In a similar manner, the fuel assembly (F/A) equivalent flux is:

$$\text{Average F/A Heat Flux} = \frac{9.35 \times 3413 \times 12}{1.3} = 295{,}000 \ \frac{\text{Btu}}{\text{hr-ft}^2}$$

The ratio of the F/A linear heat rate to the core average linear heat rate is:

$$\frac{\text{F/A}}{\text{Core}} = \frac{9.35 \text{ kw/ft}}{6.35 \text{ kw/ft}} = 1.475$$

The ratio of the F/A average heat flux to the core average heat flux is:

$$\frac{\text{F/A}}{\text{Core}} = \frac{295{,}000 \ \text{Btu/hr - ft}^2}{200{,}000 \ \text{Btu/hr - ft}^2} = 1.475$$

This ratio relates the heat flux or linear heat rate of the F/A to the heat flux or linear heat rate of the core. The ratio has a specific name, but it is also part of a general group of ratios called peaking factors.

B. Peaking Factors

The nonuniformity of the power or heat flux distribution in the core can be expressed by the use of peaking factors. These ratios relate the amount of power or average flux of a fuel rod or fuel assembly to the core average values. They also relate the maximum heat flux or power in a fuel rod to the rod average values.

Power is not uniformly distributed throughout the core, because of neutron leakage out of the top, bottom, and sides of the core. When a neutron leaks out, it cannot cause a fission in the fuel. There is less fissioning and, thus, less power generated in those parts of the core where the neutrons can escape.

Along the length of a fuel rod, the heat generation rate will vary as shown in Figure 5B. The ratio of the maximum heat generation rate to the rod average is called the axial peaking factor. The design value in your core for this factor will be in the range of 1.5 to 1.7.

Across the horizontal plane through the core, there is a variation in total power produced in the fuel rods and the fuel assemblies. The local rod factor is defined as the flux in the local fuel rod divided by the average flux in all the rods in the assembly. The local rod power factor can be written as:

$$\frac{\text{Power Produced in One Fuel Rod}}{\text{Assembly Average Rod Power}}$$

The term in the denominator is found by dividing the total assembly power by the total number of fuel rods in the assembly. This factor will be displayed to you in the form shown on Figure 5C, where it is called the nuclear peaking factor; the enthalpy rise factors shown on Figure 5C are the ratio of the outlet enthalpy in a given channel in the fuel rods to the average outlet enthalpy for the assembly.

The radial distribution can also be expressed in terms of the fuel assembly radial peaking factor. This ratio is defined as:

$$\frac{\text{Power Produced in One Fuel Assembly}}{\text{Average Power per Assembly in Core}}$$

Figure 5D shows a typical display of the assembly radial peaking factor (P/P).

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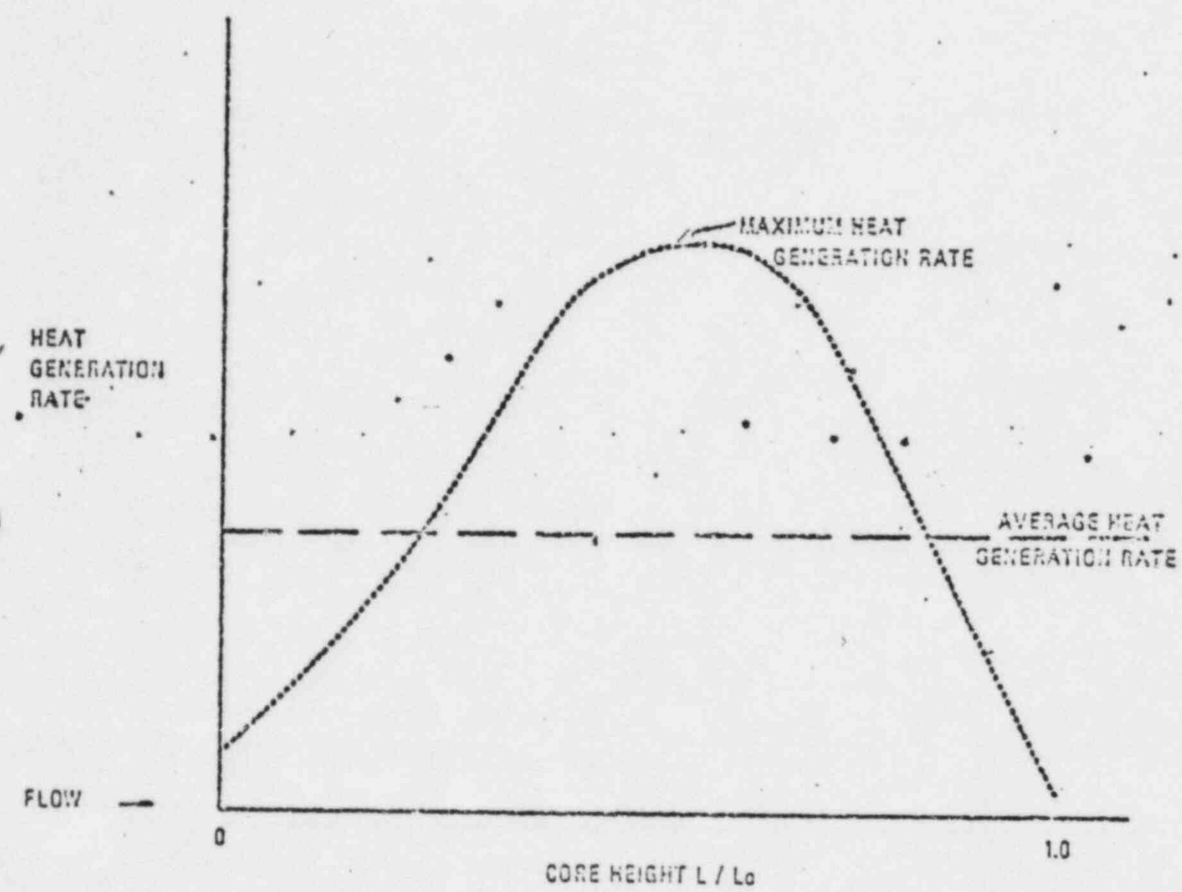

FIGURE 5B
AXIAL FLUX DISTRIBUTION ALONG A FUEL ROD

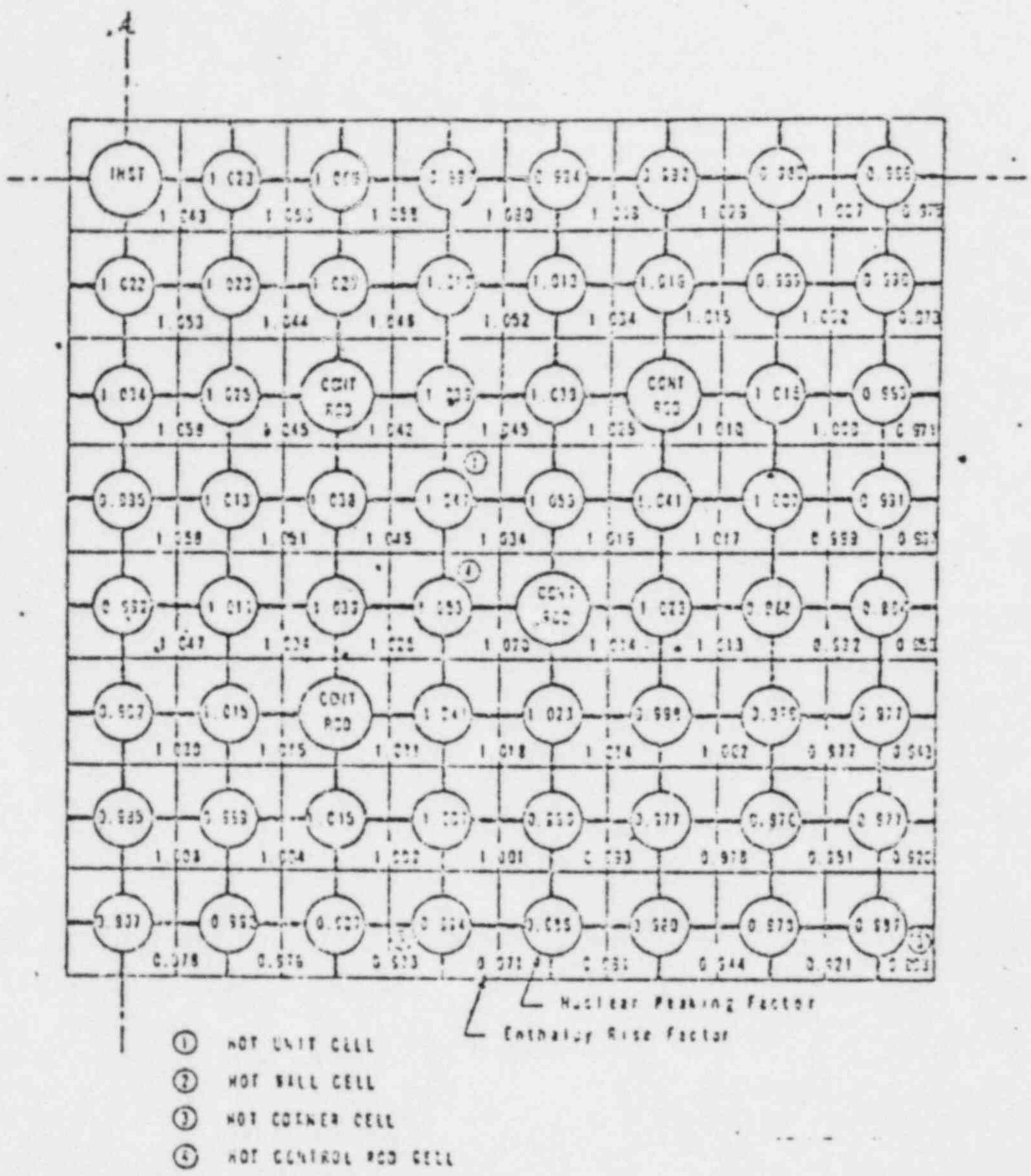

FIGURE 5C

NOMINAL FUEL ROD POWER PEAKS AND CELL EXIT ENTHALPY RISE RATIOS

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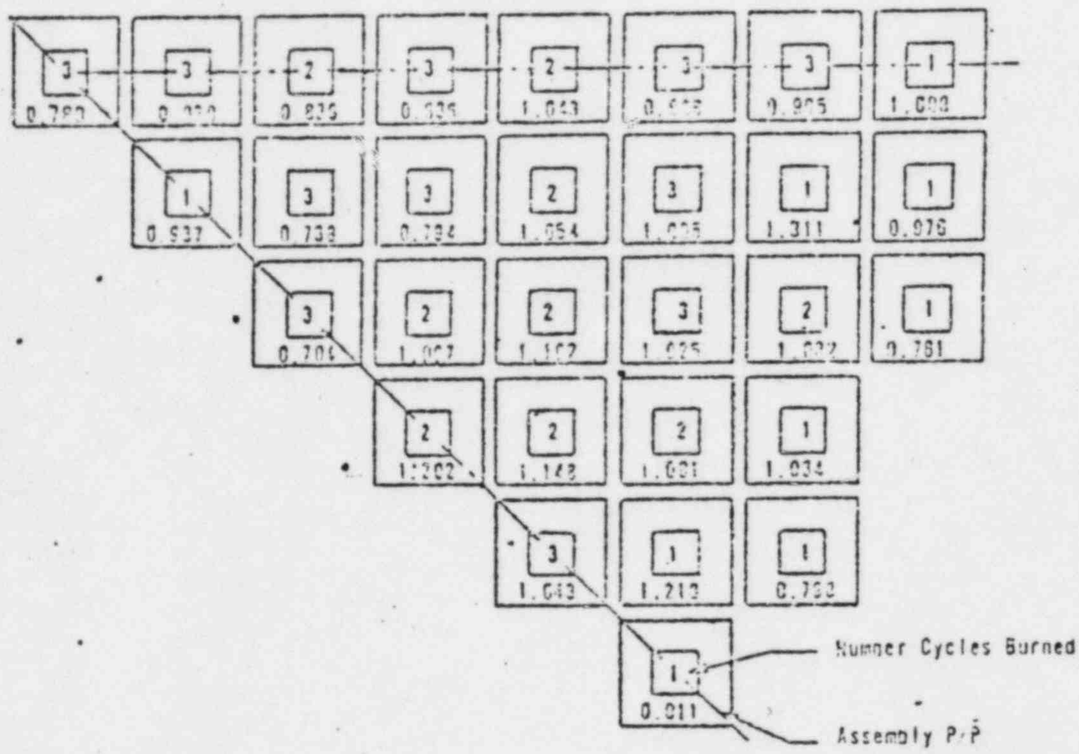

FIGURE 5D
TYPICAL REACTOR FUEL ASSEMBLY
POWER DISTRIBUTION FOR 1/8 CORE

The total rod radial power factor and the axial power peaking factor can be combined to give the total nuclear peaking factor (F_q).

$$F_q = F_{\Delta h} \times F_z$$

Using the maximum design values for the axial power peaking factor and the total rod radial power factor, the total nuclear peaking factor is

$$F_q = 1.7 \times 1.7 = 2.89$$

The point of maximum flux in the core is 2.89 times the core average flux.

The total radial power factor ($F_{\Delta h}$) can be calculated by multiplying the local rod power factor by the fuel assembly radial peaking factor. The maximum value for $F_{\Delta h}$ will also range between 1.5 and 1.7.

Peaking factors are defined for nominal conditions and for design conditions. Nominal peaking factors are those expected to exist when the reactor is operating at normal conditions. When measurements are made, they should closely approximate the nominal values.

The design factor is the highest value at which safe operation of the core is assured. It is based on the worst conditions that could occur and on the more conservative design calculations relating to core flow, temperatures, and the like. If measurements of peaking factors in the core yield design values, something is probably out of spec somewhere.

The peaking factors defined to this point relate to fuel assemblies or fuel rods. There is another factor that is used to define the maximum values of flux in a channel. This factor is called the nuclear enthalpy rise factor and is the following ratio:

$$\frac{\text{Average Heat Flux of Four Rods in a Channel}}{\text{Core Average Heat Flux}}$$

A channel is defined as the space between four fuel rods. The average heat flux in the channel is calculated by assuming that each of the four rods contributes 25% of its total heat flux.

All of the factors that have been defined are nuclear factors. They describe the heat flux or power distribution resulting from neutron behavior characteristics. There are other considerations involved that influence the heat flux and power distribution, and these are non-nuclear in nature.

C. Hot Channel Factors

In the preceding section, we examined several peaking factors that can be multiplied to produce an overall peaking factor. These factors were the max/avg axial power ratio factor (F_z), the max/avg total radial power ratio factor ($F_{\Delta h}$), and the total nuclear peaking factor (F_q). Since these factors are all "physics factor," we can define a name for this product. The name chosen is Nuclear Hot Channel Factor. There is another factor that must be applied to the nuclear factor, and we call that factor the Engineering (or Mechanical) Hot Channel Factor.

Engineering Hot Channel Factors are used to describe variations in fuel loading, fuel and cladding dimensions, and flow channel geometry from nominal physical quantities and dimensions. These tolerances, when assumed to yield the most adverse effect, have an impact on the nuclear characteristics of the core.

The Engineering Hot Channel Factor is broken down into the following two sub-factors:

F_Q: Heat Input Factor

F_Q": Local Heat Flux Factor at a hot spot

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These factors take into account uncertainties in enrichment, density, rod and pellet diameter, and length.

A new hot channel factor, F_q, is obtained by multiplying the nuclear factors by these engineering factors:

$$F_q \text{ (nuclear)} = F_z \times F_{\Delta h}$$

$$F_q \text{(nuclear + mechanical)} = F_q \text{ (nuclear)} \times F_Q \times F_Q"$$

There is another factor that limits the power capability of the core; this factor deals with the enthalpy rise of the coolant as it passes through the hot channel. The enthalpy rise is dependent upon "physics or nuclear factors" as well as variations from design flow conditions. The nuclear factor for the enthalpy rise can be represented in the following terms:

$$\frac{\text{Average Power of Four Rods in a Channel}}{\text{Average Core Power}}$$

There is an engineering factor for enthalpy rise accounting for the same variations from the design as discussed for the F_q hot channel factor. The variations are evaluated as they affect power, and the value of the engineering factor for the enthalpy rise is different.

The enthalpy rise factor is also affected by a Core Flow Distribution Factor (F_A). This factor is expressed as the ratio of each fuel assembly flow to the average fuel assembly flow. The finite values of the ratio are more or less than 1.0, depending on the position of the assembly being evaluated. The flow in the central fuel assemblies is generally larger than that in the outermost assemblies. The enthalpy rise hot channel factor is the product of the nuclear enthalpy rise factor, the engineering enthalpy rise factor, and the core flow distribution factor.

The following are typical values of the various hot channel factors for a PWR core:

Hot Channel Peaking Factors:

F_z = 1.70

$F_{\Delta h}$ = 1.78

F_q (nuclear) = 3.03

F_Q = 1.011

F_Q'' = 1.014

F_q (nuclear + mechanical) = 3.12

Enthalpy Rise:

F (nuclear enthalpy rise)	=	1.65
F (engineering enthalpy rise)	=	1.10
F_A (interior cells)	=	0.98
F_A (wall cells)	=	0.97
F_{EN} (total)	=	1.83

In a table of design parameters for a 2500 Mwt, we can look up values for the average flux and linear heat rate as follows:

Average q/A : 165,000 BTu/hr-ft^2

Average Linear Heat Rate : 5.40 kw/ft

To obtain maximum values, all we need to do is multiply these values by F_q, our hot channel factor, to obtain the maximum flux and the linear heat rate:

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Max q/A: $165{,}000 \times 3.12 = 515{,}000$ BTu/hr-ft^2

Max heat rate: 16.85 kw/ft

The enthalpy rise factor is not as readily apparent. From our parameter table, we can compute the average core enthalpy rise as follows:

$$\text{Average} \quad \Delta H = \frac{\text{Core Power}}{\text{Core Flow}}$$

$$\Delta H = \frac{2500 \text{ Mwt}}{131 \times 10^6 \text{ lb/hr}}$$

$$\Delta H = \frac{(2.50 \times 10^6 \text{ kw})(3413 \text{ BTu/hr}/_{\text{kw}})}{131 \times 10^6 \text{ lb}/_{\text{hr}}}$$

$$\Delta H = 65.2 \text{ BTu/lb}$$

The enthalpy rise through the hot channel is:

$$\Delta H_{max} = H_{max} \times F_{EN}$$

$$\Delta H_{max} = 65.2 \text{ BTu/lb} \times 1.88$$

$$\Delta H_{max} = 122.3 \text{ BTu/lb}$$

If we add this enthalpy rise to the coolant entering the core, we find that the maximum core exit enthalpy is 678 BTu/lb. The saturation pressure for this enthalpy is 2050 psia, which is well below the reactor coolant system operating pressure. This indicates that boiling will not occur in the core.

D. Departure from Nuclear Boiling (DNBR)

Departure from nucleate boiling is predicted from a combination of hydraulic and heat transfer phenomena and is affected by local and upstream conditions,

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including the flux distribution. In reactor design, the heat flux associated with DNB and the location of DNB are important. We are now in a position, as a result of our discussion of peaking factors, to calculate our local heat flux. From our experimental correlation, we can predict the DNB flux and its location. With these two fluxes available to us, we can define a new term, called DNBR, as the ratio of the DNB heat flux to the local heat flux:

$$\text{DNBR} = \frac{\text{DNB Flux}}{\text{Local Flux}}$$

We have previously stated that we did not want to exceed the DNB heat flux anywhere in the core. Ideally, this would mean:

$$\text{DNBR} \geq 1.0$$

However, the correlation that we use to predict the DNB heat flux is not 100% accurate, as Figure 4E (page 4-23) shows. To account for this uncertainty in the correlation, we require the actual heat flux to be considerably less than the DNB flux obtained from the correlation, that is, we require a DNBR considerably larger than 1.0. The usual restriction is:

$$\text{DNBR} \geq 1.30$$

This constitutes our safety limit. For DNB ratios larger than 1.30, we can essentially guarantee that DNB will not occur. For DNB ratios less than 1.30, DNB _may_ occur.

When we talk of maintaining DNBR ≥ 1.30, we actually are talking about one location along a fuel rod, a "hot spot." Some illustrations may help. From the core power distribution, we can get a profile of the heat flux along the length of the fuel rod in the hot channel. This flux may look like this:

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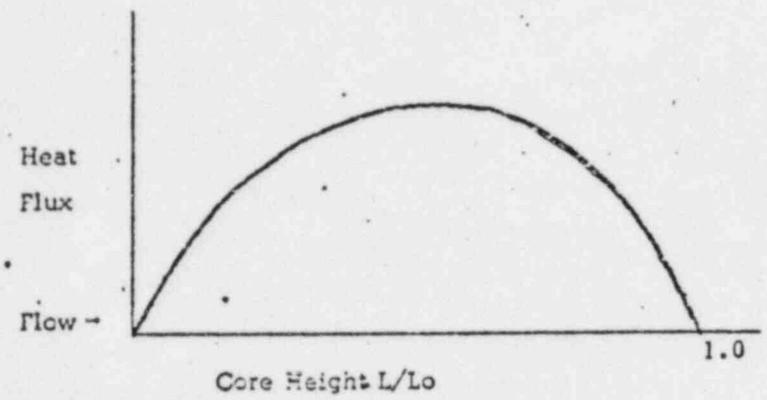
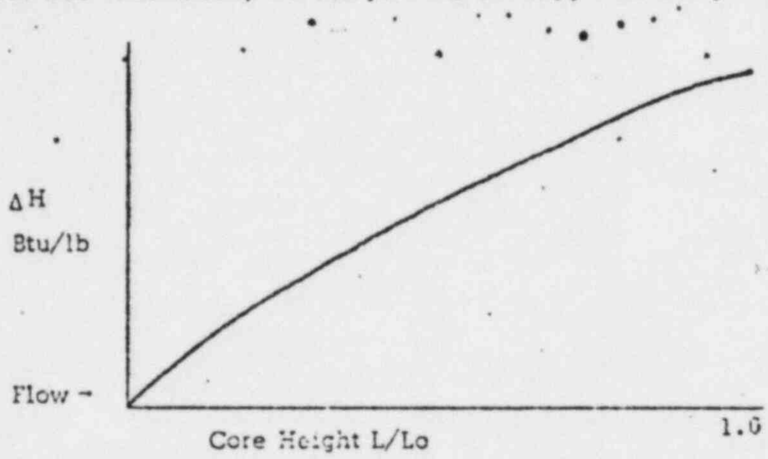
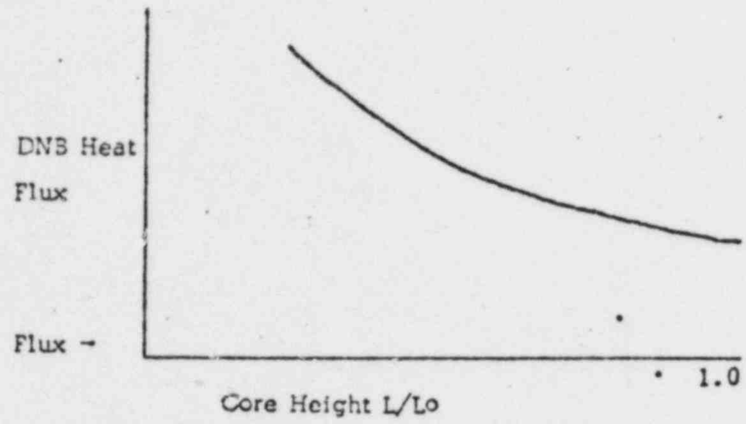


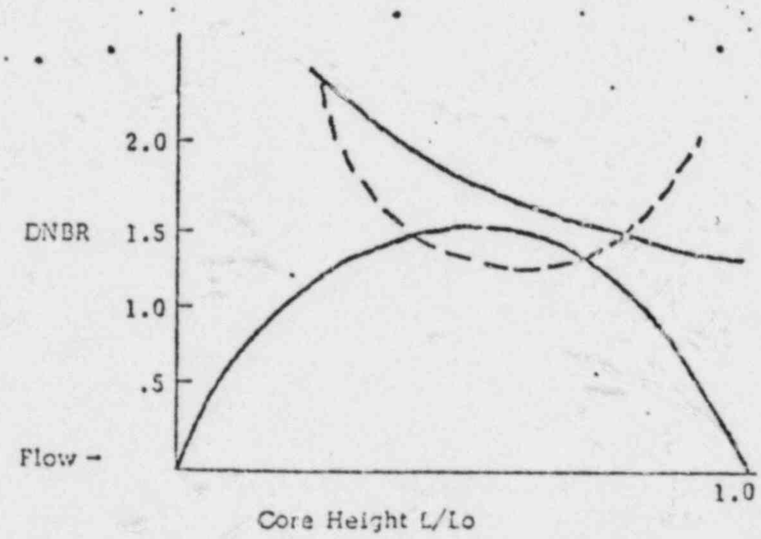
With this information, we can plot the enthalpy rise along the length of the rod.



Using calculational methods derived from the DNB flux correlation, and having values for inlet enthalpy, enthalpy rise, pressure, and flow, we can plot the DNB heat flux.



It we combine the graphs showing local heat flux and DNB heat flux, we can then compute and plot DNBR.



Remember that DNB heat flux is a function of many variables, so it is quite possible that the minimum DNBR may not occur at the point of highest local flux.

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$$\text{Heat flux} = \frac{\text{Power inside}}{\text{Surface Area}}$$

$$\text{Heat flux} = \overset{\text{heat transfer coefficient}}{h}\,(T_H - T_C)$$

E. Fuel Cladding Integrity

(1) Importance

The cladding that surrounds the UO_2 fuel is the first barrier that prevents the release of radioactivity. (Remember that the fuel becomes radioactive because of the fission products that result from fissioning the uranium to obtain energy.) One of the primary aims of AEC licensing is to make sure that the plant is designed and run in such a way that the cladding remains intact. To design and run the plant properly, we have to know exactly what it is that we are protecting against, i.e., what could cause the cladding to fail.

There are two factors which could cause the clad to fail and allow the release of radioactivity:

- If the temperature of the clad becomes too high at some particular point, it could melt at that point.
- If the pressure exerted on the clad from the inside becomes too great, the clad could burst.

(2) Clad Failure by Melting

Under ordinary conditions, the temperature of the clad is between about 600°F and 800°F. The melting point of the clad is in the 2000°-3000°F range, so no minor change in core conditions can bring the clad anywhere near its melting point. The only phenomenon which could do this is a sudden drop in heat transfer coefficient, h, at the clad surface. Only if we reach the critical heat flux; i.e., if we reach DNB, does this drop in h occur.

(3) Clad Failure by Bursting

For properly designed fuel, the only thing which could cause the clad to burst is a sudden increase in the volume of the UO_2 inside the clad. Such an increase occurs only if the UO_2 starts melting. Like most materials (except water), UO_2 expands when it melts, so if we avoid melting anywhere in the UO_2, we know that the clad will not fail by bursting.

The hottest position in any UO_2 pellet is at its centerline. If we keep the centerline UO_2 temperatures below about 5000°F everywhere (5000°F is the approximate melting point of UO_2), we will avoid melting it.

The centerline temperatures are not directly measurable, but they can be correlated with the linear heat rate. Since centerline melting of the UO_2 begins at a linear heat rate between 21 and 23 kw/ft, if we design and operate the plant so that the linear heat rate is always below 21 kw/ft everywhere in the core, we will avoid clad bursting.

F. Step Back Design and Safety Limits

(1) Step Back Design

We stated previously that (for properly designed fuel) the fuel clad will remain intact as long as we always operate so that the minimum DNBR is larger than 1.30, and the maximum linear heat rate is less than 21 kw/ft. Obviously, we would not want to operate the plant so that the minimum DNBR is 1.301 and the maximum linear heat rate is 20.99 kw/ft. Instead, we "step back." We design and operate the plant so that under normal conditions the minimum DNBR is considerably above 1.3 (usually in the range from 1.7 to 2.00) and the maximum linear heat rate is considerably below 21 kw/ft (usually in the range 17-20 kw/ft). This gives us a safety margin so that, during a transient, we can scram the plant before exceeding the safety limit.

The way we operate the plant affects the minimum DNBR and the maximum linear heat rate, because we can change the peaking factors by moving the control rods. If there were no alarms or trips, an operator could conceivably cross the DNBR and heat rate limits by moving the control rods incorrectly. We prevent the operator from passing these limits by:

- Establishing administrative procedures which restrict control rod motion;

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- Providing nuclear instrumentation to obtain information on the actual peaking factors;
- Building in alarms to alert the operator to potentially dangerous situations;
- Building in scrams which automatically shut the plant down if certain limits are exceeded.

(2) Safety Limits

Safety limits have been established for your plant so that you will not exceed maximum linear heat rate or have any fuel rods at DNB flux. You might expect to find many safety limits in your Tech Specs, but, for your PWR plant, you will find that there are usually only two. The first places a limit on maximum reactor system pressure. This limit is established to avoid radiation release, but it is aimed at the secondary fission product barrier, i.e., the reactor coolant system. A high pressure limit has little bearing on the DNB flux; in fact, the DNB flux will be greater at the higher pressure, so the limit placed on pressure is derived from other considerations. The pressure limit is the highest pressure allowed by reactor vessel and piping design codes. An increase in pressure to the safety limit will not appreciably change conditions in the core. Since the safety limit is well below the pressure at which the reactor coolant system might rupture, core performance is assured.

If our first limit has no direct bearing on minimum DNBR or maximum linear heat rate, certainly the second one must place a limit on these variables. It does and it doesn't. Limits on DNBR and linear heat rate are not directly called, but rather are used as the bases for safety limits on other parameters. This is done because these items can not be directly observed. The onset of fuel damage, by clad melting or bursting, can not be monitored by any instrumentation presently in use in nuclear plants. Instead, limits are placed on directly measurable parameters such that at their maximum or minimum values, DNBR and linear heat rate limits are not exceeded. It is in this manner that minimum DNBR and maximum linear heat rate serve as the basis for establishing limits on observable parameters.

Three other parameters that can be measured are:

- Pressure
- Flow (function of number of pumps operating)
- Reactor coolant temperature

These limits are displayed as a series of curves in your Tech Specs, similar to those shown in Figure 5E. The first two curves in Figure 5E represent combinations of the three variables for which DNBR is never less than 1.3. As long as you operate above and to the left of the appropriate curve, core integrity is ensured.

There are many other variables that have an effect on core performance. There are many opportunities and methods to cause these parameters to vary, perhaps in a manner that could lead us to exceed our minimum DNBR or maximum linear heat rate. How, then, can we get by with so few safety limits? Here is where we get help from the reactor plant designer. We will see how protection systems are established to head off the approach of conditions that could be unsafe. In another course, you will study systems that have been designed to prevent many of the actions that might cause unsafe conditions. You will also see how operating limits established by your Tech Specs assist you in avoiding the causes of unsatisfactory core performance.

(3) Operating Limits

An operating limit is one that is placed on the value of a key parameter that is important to the operability of an essential system. There are two classes of operating limits: built-in and non-built-in.

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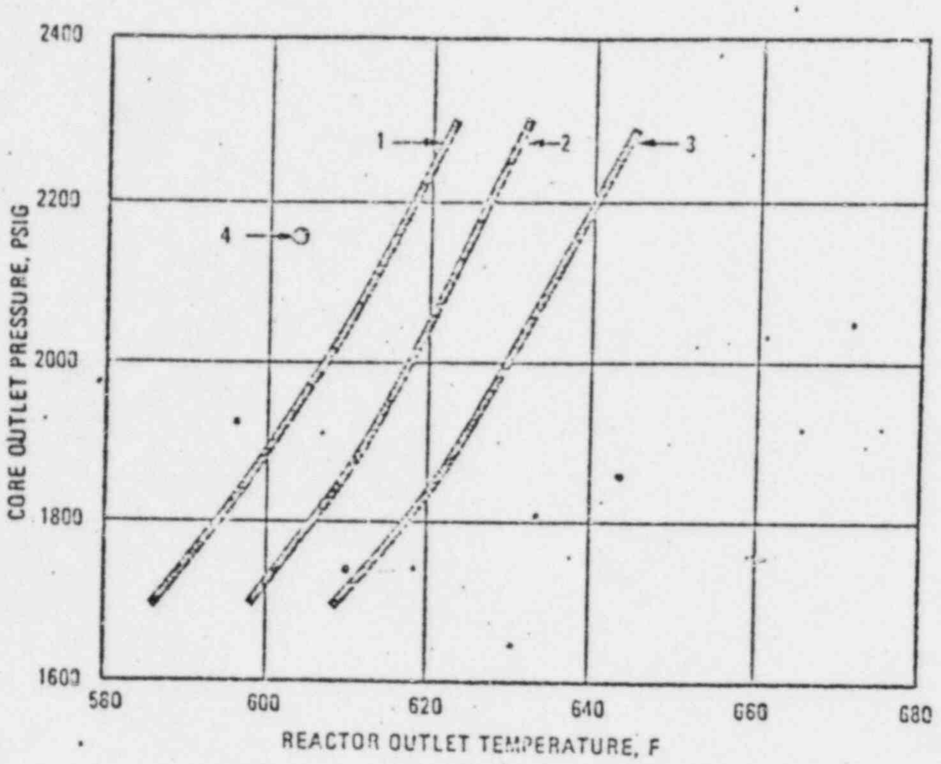


CURVE	REACTOR COOLANT FLOW (LBS/HR)	POWER	PUMPS OPERATING
1	137.8×10^6 (100%)	112%	FOUR PUMPS
2	102.5×10^6 (74.4%)	86%	THREE PUMPS
3	63.8×10^6 (48.5%)	58%	ONE PUMP IN EACH LOOP
POINT 4	NORMAL FOUR PUMP OPERATING POINT		

FIGURE 5E

CORE PROTECTION SAFETY LIMITS

A built-in operating limit is a limit on a system characteristic that is built into the design of a system or component. The following are examples of built-in operating limits:

- Safety injection system flow rate
- Diesel generator voltage
- Containment spray pump discharge pressure
- Main steam line trip valve closing time
- Auxiliary steam generator feedwater pump back pressure

Because such limits are built in, the Tech Specs generally do not itemize them. The required values of the limits were determined from portions of the Safety Analysis Report, from equipment specifications, and from operating procedures. Although these limits are not normally called out in the Tech Specs, the capability of the various systems or components to operate within the built-in limits must be checked. When they are to be checked and the intervals between checks are detailed in your Tech Specs under Surveillance Requirements. Make a note that you can find operating limits in that part of your Tech Specs.

A non-built-in limit is a limit on a parameter associated with the operability of a system that does not have the limit built into it. Some examples include the following:

- Reactor coolant system heatup and cooldown rates
- Minimum temperature for criticality
- Boric acid injection paths
- Borated water storage tank concentration
- Decay heat removal system operation
- Control rod bank insertion limits

These limits are determined analytically and are monitored separately. Because of their diversity, they can not be lumped together as a series of curves, so you will find that they are specified in the Tech Specs under the heading of Limiting Conditions for Operation. Together with those listed under Surveillance Requirements, these constitute your operating limits. A typical set of Tech Specs for a PWR plant may have as many as 25 to 30 operating limits included in these sections. A majority of those are directed at the task of limiting radiation release by assuring the integrity of the primary fission product barrier, which is the clad.

The performance of the core, and, in particular, the integrity of the clad is the single most important item of which you must be aware. Operation of your plant following the requirements of the Tech Specs, which is your legal document, will assure you that you meet your responsibilities as an operator. The designer has furnished you with systems, components, and procedures, and has assisted you in preparing the Tech Specs which put it all together. Above all else, know your Tech Specs.

CORE PERFORMANCE
5. Core Thermal Performance (cont.)

Problem Set - Chapter 5

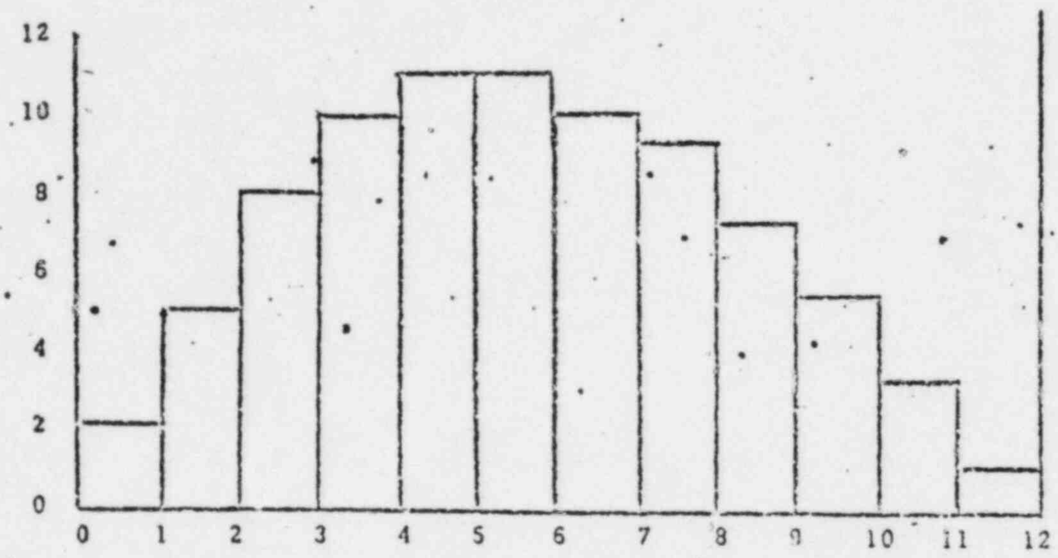
Linear Heat Rate = Power / Length

Length = 12 ft x 179 = 2148 ft

LHR = 7000 kW / 2148 ft = 3.25 kW/ft

1. A particular PWR fuel assembly generates 7MW. It has 179 fuel rods, and it is 12 ft. long. What is the average linear heat rate in this assembly?

2. The bar graph below shows the average linear heat rate for each foot of a particular PWR fuel rod when the plant is running at 100% power.



avg L.H.R. = Total Pwr / T. Length = 82K / 12 = 6.83K

(a) What is the average linear heat rate for the whole rod (one number)?

11 kw/ft (b) What is the maximum linear heat rate along this rod?

6 kw/ft (c) What is the average linear heat rate from 0 to 3 feet above the bottom?

10.3 (d) What is the average linear heat rate from 3 to 6 feet?

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(e) What is the average linear heat rate from 6 to 9 feet?

(f) What is the average linear heat rate from 9 to 12 feet?

(g) Plot your answers to parts (c), (d), (e), and (f) of this problem on a bar graph similar to the one given in the problem. (Your graph will be only 4 bars on it instead of 12.)

3. Suppose the power level in the plant used for problem 2 is reduced from 100% to 50% of full power. Consider the same rod, and re-plot the graph to represent the new power level. (Assume that the power reduction is accomplished so that control rod and void distributions are not changed, and the relative power distribution is also unchanged.) Answer, for the new power level, all of the questions in problem 2.

4. Calculate the ratio of maximum linear heat rate to average linear heat rate for problem 2.

5. Calculate the ratio of maximum linear heat rate to average linear heat rate for problem 3.

6. A core contains fuel rods having a 0.440 inch clad outside diameter and active length of 12 feet. The maximum linear heat rate of fuel rod is 17.8 kw/ft. The core average heat flux is 176,000 BTu/hr-ft^2. What is the total nuclear plus mechanical hot channel factor?

7. In problem 6, if the axial power ratio factor is 1.5 and the engineering hot channel factor is 1.035, what is the nuclear radial power ratio factor?

8. If the minimum operating pressure is 2200 psia for a core producing 2560 Mw(t), and having a total flow of 129.8 × 10^6 lb/hr, what is the core inlet enthalpy that would result in the start of bulk boiling at the hot channel outlet? Assume the F_{EN} (nuclear plus mechanical) enthalpy rise factor for the hot channel is 2.06.

9. Using Figure 5E, classify the following combinations of reactor coolant outlet temperature, reactor coolant pressure, and number of reactor coolant pumps operating as within the safety limits (DNBR ≥ 1.3) or outside the safety limits (DNBR < 1.3).

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5. Core Thermal Performance (Con't)

	No. of Pumps Running	Outlet Temp (°F)	Pressure (psia)
(a)	4	610	2200
(b)	4	600	1600
(c)	3	620	2000
(d)	3	600	1800
(e)	2	620	1800
(f)	2	630	2100

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Mikes notes

$$\text{Hot Channel Factor} = \frac{\text{Max Heat Flux}}{\text{Ave Heat Flux}} = \text{DNB}$$

Look Back in notes to DNB

Hot channel = nuclear x eng

F_q = Total Radial x axial

$= F_{\Delta h} \times F_z$

Δh = Local x radial   axial

TPF = Local x radial x axial

TPF x Ave Heat Flux = Max Heat Flux

assemblies = ave thermal per assembly

$$\text{Radial Peaking} = \frac{\text{individual assembly power}}{\text{ave assembly power}}$$

POWER DISTRIBUTIONS

I. Ideal

A. All portions of core producing equal amounts of power.

B. Uniform Fuel Burn-Up

1. core size minimized

2. lowest fuel cost per MW_t.

C. Unavoidable phenonena makes it impossible

II. Homogeneous Reactor

A. Bare

1. No reflector

2. Power density drops near edges due to neutron leakage.

3. Power drops abruptly to zero, while the thermal flux gradually falls to 0.

B. Reflected

1. Scatters some neutrons which leakout back into the fuel.

2. Moderates fast neutrons.

III. Heterogeneous Reactor

A. The power distribution would be of a modified cosine shape. This is due to the discontinuities caused by the moderator and fuel separation.

B. The thermal flux is depressed towards the center of each fuel rod, while fast neutrons are thermalized in the moderator causing peaks outside each rod.

C. Consists of discrete fuel bundles, 177, which is made of fuel rods, 208. The fuel bundle remains intact throughout its useful life.

D. The power distribution between bundles is found by the bundle peaking factor:

$$\text{APF} = \frac{\text{Total power generated in the fuel bundle (assembly) of interest}}{\text{Total power generated in "average" bundle (assemblies)}}$$

(sometimes called radial peaking factor)

E. Bundles near the edge of the core usually have lower BPF's than those near center.

F. Increase in bundle power causes a corresponding increase in the void content, which limits the power increase. This assists flattening effect.

G. In order to try and flatten the power distribution, PWR's usually use different enrichments. The highest enrichment in the outside annulus, while the lowest enrichment is in the center.

H. These enrichments tend to smooth out the neutron flux, however, the sharp breaks in the power generation still exists. This is because the $P = \emptyset \Sigma_f$, if the $\emptyset$ constant, f or enrichment cause Σf to change.

I. The differing zone enrichment is not desirable for large reactor cores.

1. Zones larger than average crow flight path.
2. Zones tend to act independent of each other.
3. Constant boron distribution makes the center zone K_∞ <1 or subcritical while the other zones are keff = 1.0.

J. Burnable poison located in appropriate areas can flatten the power distribution.

IV. Axial Power

A. The power distribution from the bottom to the top of the core.

B. Slice a fuel bundle into 10 equivalent sized slices, refered to as nodes.

C. Each node is generating 1/10 of the BPF.

D. The power production by this node is found by the axial power factor:

$$\text{APF} = \frac{\text{power generated by node of interest}}{\text{average nodal power for that bundle}}$$

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E. In BWR the peak axial power occurs somewhere below the centerline.

F. Uniform control results in very high axial peaking in the bottom of the core, while bottom entry (BWR) results in lower peaking.

G. In PWR the higher moderator temp. at the top of the core tends to push the axial flux peak slightly below the centerline. Top entry (PWR) exaggerates flux peaking in lower half, therefore, it is advantageous to use uniform axial control (chem. shim).

V. Local Peaking

A. There is reflector peaking in the water gaps which increase the power generation. This is most strongly felt by the fuel near the gap left by the removal of a control rod.

B. Contrary to this, when the rod is in place the fuel operates at a much lower power.

C. Local Peaking Factor:

$$LPF = \frac{\text{power in fuel rod in a given node}}{\text{average power in rod for given nodes}}$$

D. Fuel rods in the vicinity of the control rod are usually loaded with slightly lower enrichment, which reduces maximum local peaking by $\approx 10\%$.

E. Each fuel rod consists of a length of pellets followed by and end connector causing a thermal neutron pileup in the end connector unless the connector is made of a special material.

(TPF) Total Peaking Factor = Radial (RPF) $\overset{1.5}{}$ x Axial (APF) $\overset{1.2}{}$ x (LPF) Local $\overset{1.4}{}$

TPF = 2.52

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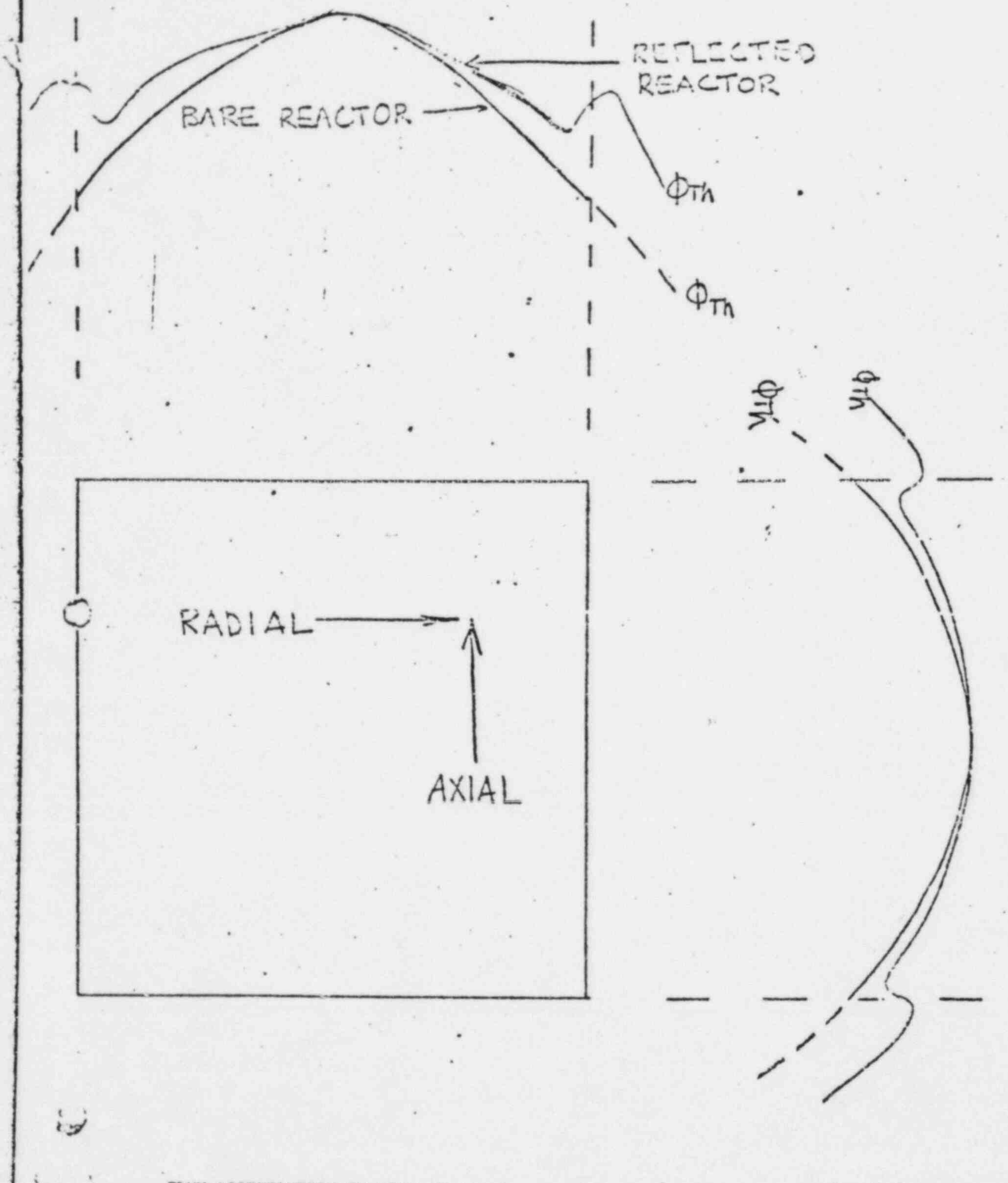
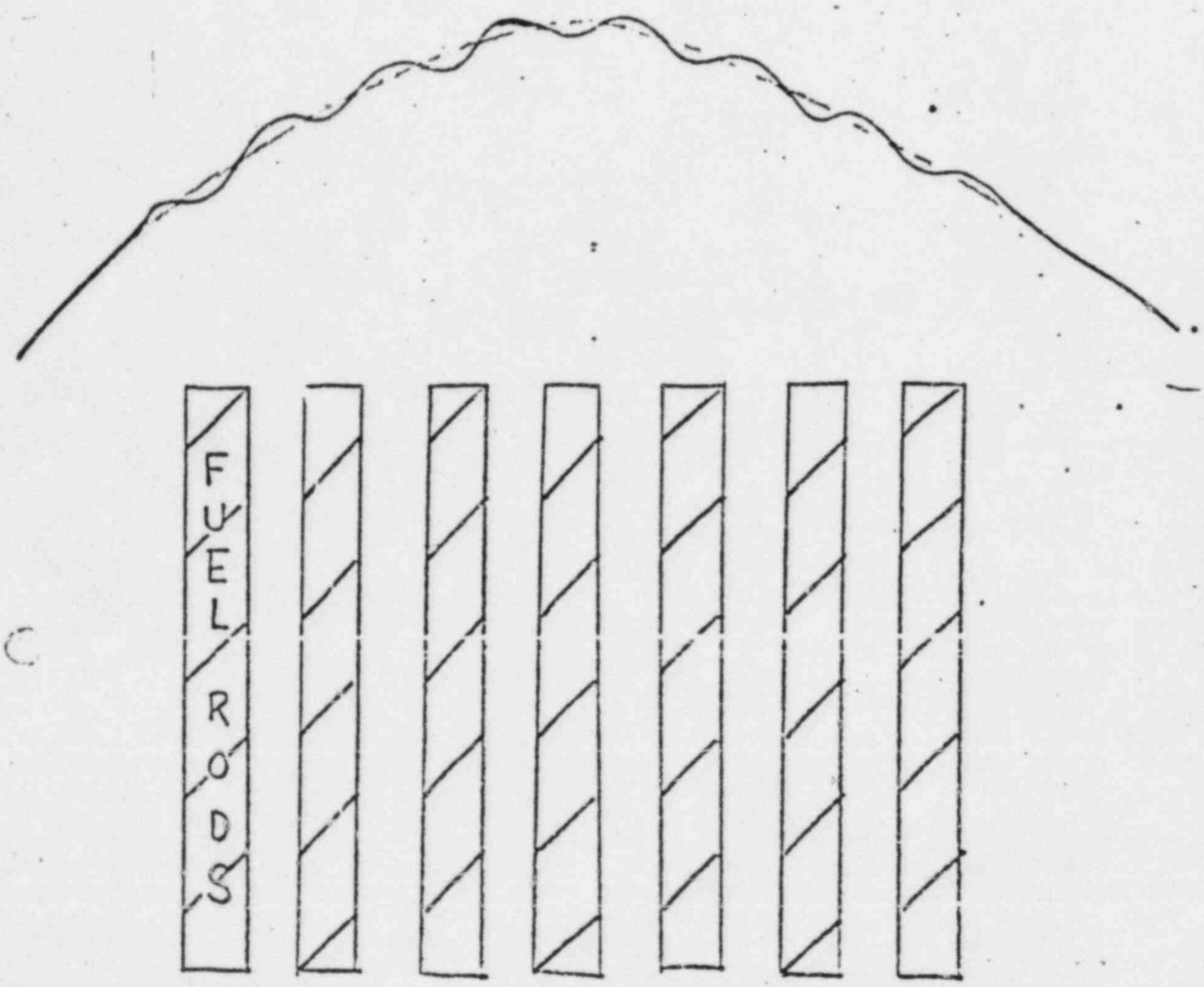


FLUX DISTRIBUTION IN HOMOGENEOUS CORE



FLUX DISTRIBUTION IN HETEROGENEOUS CORE

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POWER DISTRIBUTION WITH RELATION TO THERMAL-HYDRAULIC LIMITS

I. The reactor must operate within the limits of CHFR and fuel center temperature.

A. Physical geometry of the bundle

1. Influences the heat transfer coefficient (temperature of cladding and fuel).

2. Pressure drop and coolant flow in core.

$CHFR = \frac{CHFlux (DNB)}{AHFlux}$

B. Power density (heat flux)

1. numerator of the CHFR

2. needed fuel center temp.

3. DNB occurs at a specific point, y:

$$\left(\frac{\dot{Q}}{A}\right)_y = (BPF)_y \times (APF)_y \times (LPF)_y \times \left(\frac{Q}{A}\right)_{core\ ave.}$$

4. The product $(BPF)(APF)(LPF)_y$ = total peaking factor (TPF) so

$$\left(\frac{Q}{A}\right)_y = (TPF)_y \times \left(\frac{Q}{A}\right)_{ave.}$$

C. Condition of the coolant

1. heat content per weight, enthalpy H $\left(\frac{BTU}{lb}\right)$.

2. pressure tells if coolant is subcooled or saturated.

3. Enthalpy rise peaking factor:

$$F_{\Delta H} = \frac{\text{enthalpy rise of coolant in flow channel}}{\text{average enthalpy rise of coolant in core}}$$

4. BWR - a flow channel refers to the entire coolant volume contained within the limits of a single fuel bundle, because of turbulence by voids, spacers, etc. produced uniform mixing of the coolant.

5. PWR - uniform mixing is not assumed, therefore a coolant channel is the region between four fuel rods.

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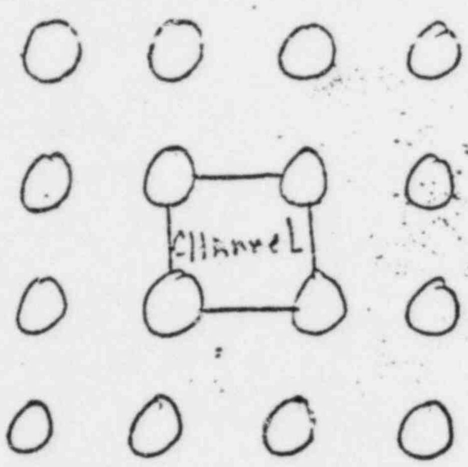

A. (PWR) We are dealing with subcooled water and all heat added to the coolant goes into raising the temperature, $\dot{Q} = \dot{m}\ T\ Cp$.

B. To find enthalpy; the temperature increases is divided by the flow rate:

$$H \text{ subcooled} = \frac{\dot{Q}}{\dot{M}} = \frac{\dot{M}\ Cp\ \Delta T}{\dot{M}} = Cp\,\Delta T$$

C. Any flow will reflect itself in the channel temperature rise.

D. Example:

The peak bundle is a particular PWR operates at a BPF = 1.45. Within this bundle, the peak rods operate at a LPF = 1.08. Assume that flow is uniform throughout the core. What is FΔH?

$$\dot{Q} \text{ peak} = \left(\frac{Q}{A}\right)_{ave} \times (BPF) \times (LPF) \times (A \text{ Channel})$$

$$\dot{Q} \text{ peak} = \left(\frac{Q}{A}\right)_{ave} \times (1.45) \times (1.08) \times (A \text{ Channel})$$

The enthalpy rise in the channel is simply the heat addition divided by the flow rate.

$$\Delta H \text{ peak} = \frac{(\dot{Q}/A)\text{Ave} \times (1.45) \times (1.08) \times (A \text{ Channel})}{\dot{M} \text{ ave Channel}}$$

The expression for the enthalpy rise in the average channel will be the same as that for the peak except all peaking factors are 1.0.

H ave =

$$\frac{\left(\frac{Q}{A}\right) \text{ave} \times (1.0) \times (1.0) \times (A \text{ Channel})}{\dot{M} \text{ Ave Channel}}$$

thus,

$$F\Delta H = \frac{\Delta H \text{ peak}}{\Delta H \text{ Ave}} = \frac{\dfrac{\cancel{\left(\frac{\dot{Q}}{A}\right)\text{ave}} \times (1.45) \times (1.08) \times \cancel{(A \text{ Channel})}}{\text{In Ave } \cancel{\text{Channel}}}}{\dfrac{\cancel{\left(\frac{\dot{Q}}{A}\right)\text{ave}} \times (1.0) \times (1.0) \times \cancel{(A \text{ Channel})}}{\dot{M} \cancel{\text{ Ave Channel}}}}$$

$$F\Delta H = 1.45 \times 1.08 = \underline{\underline{1.57}}$$

E. Therefore, for a PWR, $F\Delta H$ = (BPF) x (LPF)

II. The total peaking factor between BWR's and PWR's do not differ greatly, although the BWR has higher local peaking due to the control rod water gaps, but has lower radial and axial peaking because of the flux flattening effects of the voids coupled with the bottom entry of the control rods.

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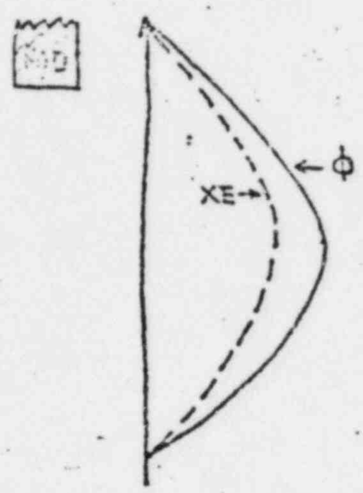
SPATIAL XENON CONCENTRATION FLUCTUATIONS

I. PWR operating at full power and all rods are essentially withdrawn.



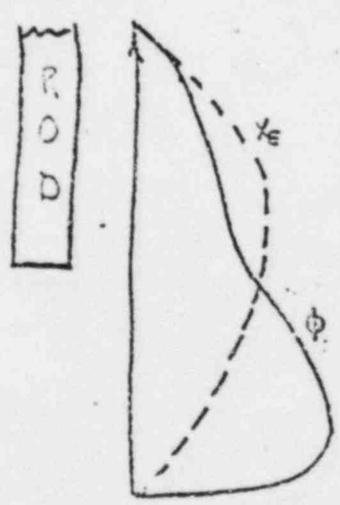
A. The axial flux will be approximately cosine.

B. The xenon concentration will follow basic cosine by following the flux.

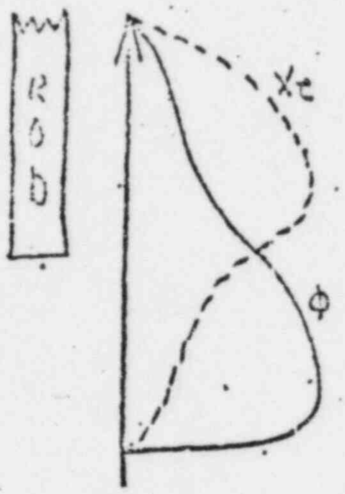
II. Boron dilution and control rods are driven in to compensate for boron removal.

A. Power distribution goes to bottom of core

B. Xenon shape will remain the same because xenon delay time lags flux change.



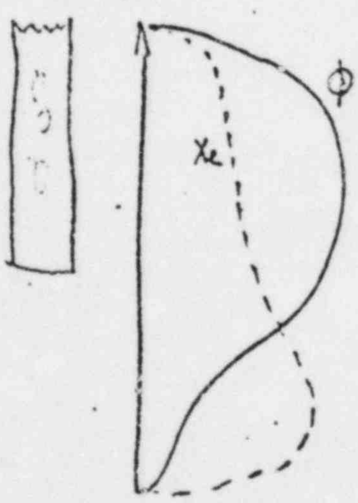
III. After a few hours the higher flux at the bottom of the core causing Xe to burn-out quicker there, while the lower Ø in the top allows the xenon to build up as it decays.



A. The increased xenon at the top enhances the flux in the bottom.

B. This could cause the core to approach DNB and/or center melting limits.

IV. Eventually the xenon delay will catch the flux which will cause the xenon distribution to shift towards the bottom. As the xenon shifts toward the bottom the flux will shift towards the top.



V. These fluctuations will continue constantly with time.

VI. Shifts of xenon

A. Shift power in core

B. Even though entire core reactivity may be 0, low xenon region may be on + reactivity while high xenon region may be on - reactivity.

VII. Rate of change of regional power with respect to time, $\Delta P/\Delta T$.

A. Directly related to the net reactivity of the region.

B. $\frac{\Delta P}{\Delta T} = \Delta K_\infty + \Delta K_{leakage} - P_{xe} C_{xe}$

1. ΔK_∞ = excess reactivity the region would possess if no leakage. Region at 0 power, 0 xenon

2. α = power coefficient of region in ΔK/unit power.

3. P = regional power

4. ΔK leakage = reactivity effect of leakage

5. pxe = reactivity effect of an atom of xenon

6. Cxe = number of xenon atoms in region

C. Leakage always goes from regions of higher neutron density to lower density regions.

D. If $\frac{\Delta P}{\Delta t} = 0$, the power in region is stable.

1. Ideally one would want one of the terms to be balanced by the other terms to keep the power in region to 0.

2. This will keep region reasonably stable.

6. CORE PERFORMANCE AND CONTROL

A. Measurement of Local Heat Generation

After a reactor core has been designed, manufactured, and installed, it is important to have some way of measuring the performance during operation to insure that the limits are not being exceeded. We have seen how to calculate overall thermal power using a heat balance, but it is not sufficient to know merely that the total power is at the rated value. We must also be sure that the power in all parts of the core does not exceed the limits imposed on the fuel rods. In all reactor cores, some method is provided for measuring the relative neutron flux at various locations within the core. This will indicate the relative power generation at these points.

It is important to note the word relative when referring to flux measurements and power generation. The relative flux means that the flux can be determined in one part compared with another; if the absolute flux could be measured, we would know exactly how many fissions were occurring. Once the relative flux levels are measured, and the overall thermal power is known, then the heat generation can be determined at various points in the core.

The results of the flux measurements will give us our power distribution in the form of radial and axial flux shapes. From these flux shapes, heat generation peaking factors can be determined and applied to the average heat generation and heat flux values obtained from our reactor heat balance. This will give us the maximum values of heat generation, heat flux, and enthalpy rise in the core.

In a PWR core, the local flux-levels and temperatures within the reactor are measured by in-core instruments. A number of the fuel elements are supplied with a column for instrumentation in place of a fuel pin. A thermocouple is positioned at the exit of these instrumented fuel elements. A typical 2500 Mwt reactor has 52 positions for in-core flux and temperature monitoring. The detector assembly consists of seven local flux detectors, located at seven axial positions in the core, and referenced to the background detector output so that the resulting differential signal is a true measure of neutron flux.

The in-core instruments provide information that can be used in conjunction with previously determined analytical information to determine the power distribution. The instrument readings can be used to calculate reactor coolant enthalpy distribution and fuel burnup distribution and to estimate the coolant flow distribution.

The flux monitors are miniature solid-state neutron detectors. The leads for the detector column penetrate the bottom of the reactor vessel. When the reactor is de-pressurized, the in-core detector columns can be retracted through guide tubes, which are on the bottom of the reactor vessel. These detector columns are withdrawn from the core during refueling and later are re-inserted in the new fuel elements. It is not necessary to re-calibrate the detectors during plant life, because the computer output has a program to compensate for burnup of the neutron-sensitive material.

The data gathered from the in-core instrumentation is used to calculate core performance, both thermal and hydraulic, and to determine other physics parameters of the core. External or out-of-core instrumentation is provided to measure average core power, detect flux tilts, and provide signals to the reactor control and protection system.

The in-core instrumentation is not anticipated to be placed in operation on a scheduled basis for the collection of data to calculate power distribution. It will be used during approach to full power for the first time to verify that control rod patterns do not cause larger than calculated peaking factors and

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that the out-of-core instrumentation is sensitive to flux tilts. After the plant is in normal operation, power distribution calculations will be made to assess the effects of xenon on peaking factor location and to evaluate any changes in the control rod patterns. The reactor engineer will utilize data from the in-core instrumentation to measure fuel burnup. Additionally, the data can be used to confirm that DNBR is above the design limit. The in-core instrumentation is an information system; it is not involved directly in controlling and protecting the core. This latter function is accomplished by the out-of-core instrumentation, which, when combined with other plant parameter measurements, operates to limit core operation and maintain conditions within the design safety limits established in the Technical Specifications.

B. Reactor Control

The reactivity control of the reactor is provided by neutron-absorbing control rods and by a soluble chemical neutron absorber in the reactor coolant. The reactivity control system functions to maintain fuel rod integrity and prevent fission product release. As discussed in an earlier section, one of the limiting thermal criteria is that the minimum allowable DNBR during operation, which includes anticipated transients, is 1.30. The control rods provide sufficient reactivity to terminate any normal power transient prior to reaching a DNBR of 1.30.

How do we know when to insert the control rods in order to prevent a DNBR of less than 1.30 or a linear heat rate of less than 21 kw/ft? Certain conditions in the primary system will cause an immediate and rapid trend to exceed the design limits. Two such conditions are high reactor coolant outlet temperature and low reactor coolant system pressure. The conditions are measured and used as inputs into the reactor protection system. High

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IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

reactor outlet temperature or loss of system pressure will cause immediate reactor shutdown and prevent a DNBR of less than 1.30.

Another monitored parameter that can result in loss of cooling for the core is primary system flow. The flow reduction input is part of the power/imbalance/flow trip input to the reactor. The trip logic for this setpoint is that the power level trip setpoint is decreased as a function of the decreased flow and corrected for any power imbalance between the upper and lower portions of the core. Figure 6A shows typical power imbalance boundaries for this reactor protection function. Some plants have an additional trip setpoint for loss of reactor coolant pumps. Typically, a reactor trip is initiated on loss of one pump for power levels above approximately 90% and trip of two pumps at any power level.

Other trip functions are calculated from various plant parameters for protection against slowly increasing conditions that might lead to exceeding design limits. One of these trips is the overpower trip. When the neutron flux level (which is proportional to power) reaches a certain setpoint on two out of four detectors, the reactor is tripped. Figure 6B shows typical pressure-temperature setpoint boundaries. When the reactor conditions are determined to be outside the boundaries, the reactor is tripped. High reactor coolant system pressure will also initiate a reactor trip. This trip insures that the design pressure of the reactor coolant system is not exceeded; this could result in a lack of integrity of the primary system boundary.

In the preceding discussion, we examined how the reactor protection system functions to keep DNBR and linear heat rate within limits. During normal operation of the reactor, there is some control available to the operator in shaping the core axial power distribution. This is accomplished by the use of axial power shaping rod assemblies (APSRA). Some vendors call these part length rods, because only 3 feet or so of the control rod has a neutron

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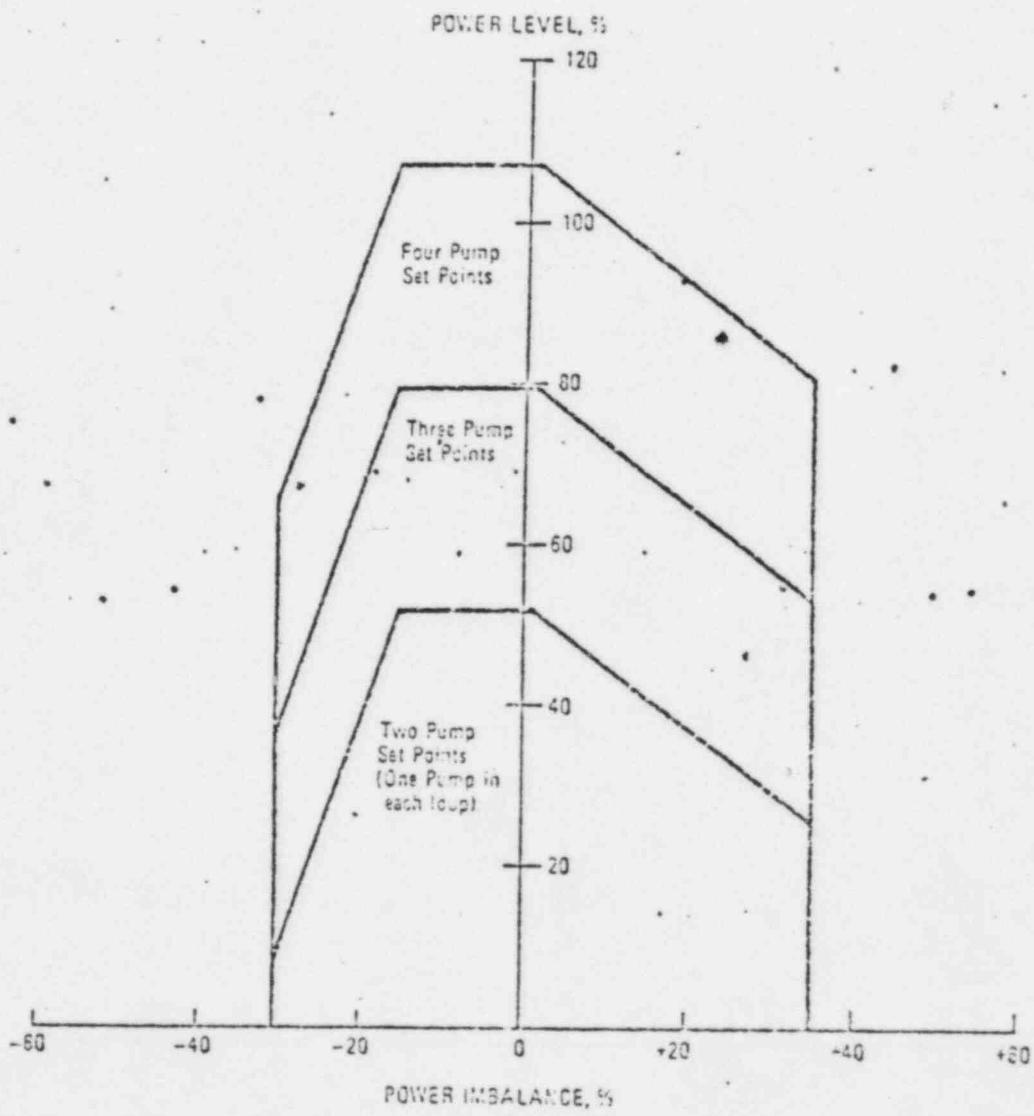

FIGURE 6A
POWER IMBALANCE BOUNDARIES

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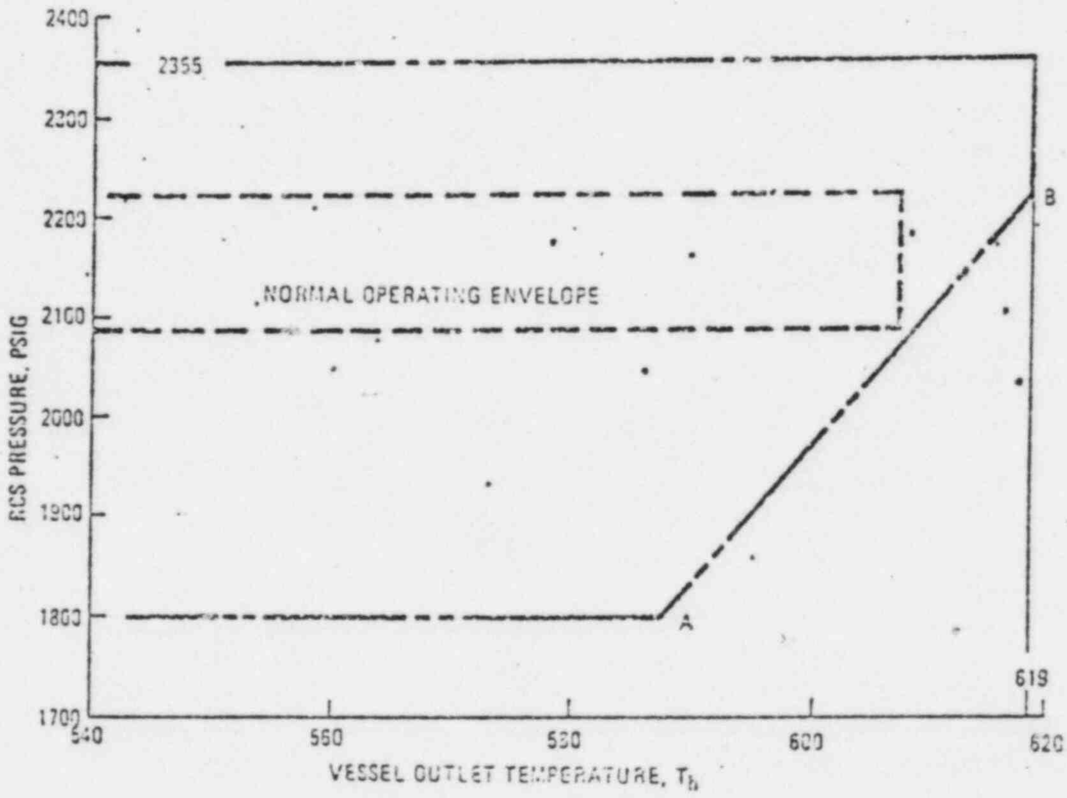

FIGURE 6B
PRESSURE TEMPERATURE BOUNDARIES

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absorber in it. These APSRA's are manually operated and are not scrammed. They are used to improve the capability of maintaining desired core conditions while providing the means to damp out xenon oscillations.

The main reason that APSRA's can be used as an effective control is that they represent the largest percentage of control rods that are still in the core. Most of the full length control rods are out of the core. At full power, only one control group of rods is partially inserted. The reason that a majority of the control rods are out of the core is that the soluble chemical neutron absorber (boric acid) concentration has been established to allow the rods to be withdrawn from the core. This allows more uniform power distribution and evens fuel burnup between assemblies. It also increases the control rod relative shutdown capability. The control group of rods still partially inserted in the core is used primarily to restore T_{avg} after a load change.

These rods are controlled automatically and are capable of reactor control over the power range of 15-100% of rated power. The control system will also compensate for some small amount of fuel burnup. Final compensation for burnup is made by adjusting the boric acid concentration in the reactor coolant; in this instance, the concentration would be reduced.

The automatic reactor control system functions to allow the reactor to be a load following unit. At power levels above 15%, the plant is operated at constant main steam pressure and constant primary T_{avg}. The control system responds to changes in load by altering feedwater flow. As an example, if the unit load demand increases, the turbine will require more steam to maintain speed. Feedwater flow increases to provide more steam. This increases the temperature difference across the tubes and allows more heat to be transferred from the reactor coolant to the steam/water mixture in the steam generator. As more heat is taken from the reactor coolant, its temperature is decreased as it leaves the steam generator. This colder water is sensed by the reactor inlet temperature sensor. The T_{avg} computer gets its signal from the reactor inlet and outlet temperature sensors and computes T_{avg} from the following relationship:

$$T_{avg} = \frac{T_{out} + T_{in}}{2}$$

A signal is then sent to the reactor control system. This signal is compared to the setpoint and, if it is found to be less than the setpoint, the control rods are moved in a direction to restore the T_{avg} to the desired value. In this example, the rods would be withdrawn to increase the core power to match the secondary system demand.

Problem Set - Chapter 6

1. Refer to Figures 5E and 6B to determine whether or not this plant will trip for the conditions given below:

(a) $T_h = 600^{\circ}F$

$P_{RCS} = 1900$ psig

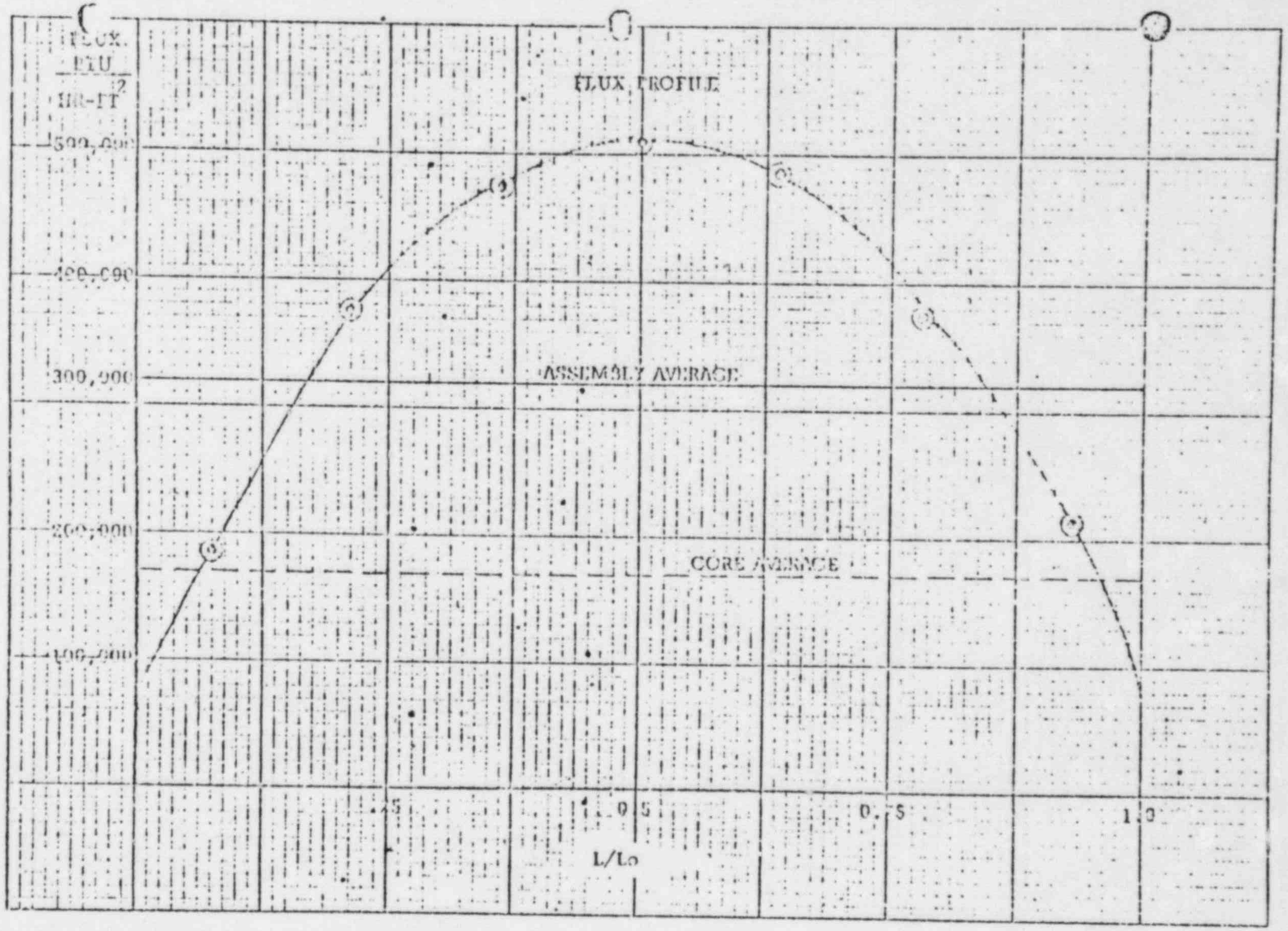
4 pumps running

(b) $T_h = 600^{\circ}F$

$P_{RCS} = 2100$ psia

3 pumps running

2. Using the accompanying flux profile, what is the axial peaking factor? What is the relative assembly for? If the local peaking factor is 1.15, what is the maximum flux somewhere in this assembly?



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7. REACTOR MATERIALS

Reactor core materials can be conveniently categorized with regard to their functions in the reactor. These categories are: fissionable fuel, fuel clad, neutron absorber (control rod-poison section), coolant moderator, and reactor internals.

A. Fissionable Fuel

(1) The desirable properties of a nuclear fuel material are:

- It must be fissionable.
- Nonfissionable material must be minimized or must have low absorption cross section.
- The material should be physically stable; that is, it must neither disintegrate nor expand too much.
- It should not corrode easily in hot water in case of clad leaks.
- It must be economically competitive with respect to fabrication, enrichment, and reprocessing costs.
- The thermal properties of conductivity and melting temperature should be such that it will not melt in operation.

(2) Uranium is a metal, and consideration was once given to using it in metallic form. Unfortunately, uranium corrodes rapidly in water and also exhibits undesirable physical behavior when irradiated. Unless stringent manufacturing controls are exercised to insure that a certain metallurgical crystalline structure exists, uranium metal tends to grow under irradiation. Another problem is that while the melting point (2070°F) is satisfactorily high for its thermal conductivity (about 1/10 that of aluminum),

it undergoes several metallurgical phase transformations between room temperature and its melting point. A phase transformation involves a re-arrangement of the molecular structure, and, as a result, the volume will change if not constrained. If it is constrained, as in a fuel element, high stresses can develop within the cladding. The only extensive use of uranium metal as a fuel has been in reactors designed for the production of plutonium.

(3) The above considerations led to the investigation of ceramic materials for use as fuel materials. The two ceramics of principal interest are uranium carbide (UC) and uranium dioxide (UO_2). Thermal conductivity and nuclear properties of UC are better than those of UO_2 but the melting point of UC is somewhat lower (UC, 4280°F; UO_2 4980°F). One property predominates, however, in the exclusion of UC from water reactors: it decomposes readily when exposed to hot water. One of the products of decomposition is acetylene, which could create problems if generated.

Uranium dioxide has turned out to be the best material for water reactors when considering all the factors of safety: thermal, nuclear, and chemical properties; and the economics of production and utilization. Its major disadvantage is a relatively low thermal conductivity, but that is compensated by a high melting point and no phase transformations. Uranium dioxide retains fission products well, does not drastically expand in volume until high burnups are reached, and is not a neutron poison. Some enrichment (3 to 6 percent) is required because of the dilution of uranium atoms by the oxygen atoms.

B. Fuel Clad

(1) The desirable properties of the clad material are:

- Good strength at operating temperatures
- Not susceptible to corrosion
- Low neutron absorption cross section
- High thermal conductivity

For water cooled power reactors, two metals have been used for clad materials: stainless steel, and zirconium alloys.

(2) Stainless steel was used extensively in the original PWR and BWR designs. It has given way to Zircaloy, however, in both types of reactors. The only significant problem of stainless steel in PWR's is that it has a higher neutron absorption cross section than Zircaloy, and thus a fuel with a somewhat higher enrichment is required. In BWR's, stainless steel is inadequate for another reason: it is susceptible to stress-corrosion cracking. Small microcracks form, starting at the surface, and weaken the material. Since the clad is thin to start with, cracking is unacceptable.

(3) Zircaloy-4 (1.5 percent tin, 0.18 percent iron, and 0.10 percent chromium by weight) is now the preferred fuel clad material in water cooled power reactors. It is not as strong as stainless steel, but its low neutron cross section and immunity to stress-corrosion cracking outweigh the lower strength. The lower strength is compensated for by using a thicker tube wall.

One possible source of trouble with Zircaloy clad is hydriding. This refers to the tendency of zirconium and its alloys to combine with any available hydrogen, forming zirc hydride platelets which decrease the strength of the Zircaloy. Hydriding is avoided or minimized by:

- Careful manufacturing procedures which prevent hydrogen from getting inside the fuel rods in the form of water vapor.

• Control of water chemistry in the plant to avoid corrosion of the Zircaloy from the outside.

C. Nuclear Fuel Performance

(1) Fuel Damage Criteria

The fundamental criterion is that no fuel element damage is to occur during steady state operation, operational transients, or reasonably probable accidental transients. Examples of operational transients are reactor startup or change of load. Examples of reasonably probable accidental transients are loss of electrical power to a circulating pump, continuous control rod withdrawal, or loss of turbine load. For at least some of these accidental transients, the reactor would be protected by scram. The criterion of no fuel element damage does not by definition apply to the highly improbable MCA (maximum credible accident).

The consequences of fuel failure range from substantially unnoticeable pinholes in the cladding to severe activity release caused by gross failure of cladding. Normally encountered fuel failures allow unrestricted normal operations and are not a hazard to the surrounding population, but severe cases may require restrictions on power level to reduce release of radioactivity to the environment.

(2) Fuel Burnup Units

The most often used units of fuel burnup are megawatt days per metric ton of uranium (MWD/MtU). One MWD/MtU is equivalent to operating with 1000 kilograms (one metric ton) of contained uranium at a power of one megawatt for one day. The uranium referred to here is total uranium — not

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just U-235, and not UO_2. We can talk about the burnup of any amount of uranium that we please — the uranium in a fuel pellet, in a fuel rod, in a fuel assembly, in a whole batch of fuel, or in the whole core.

When referring to a whole core of uranium, burnup becomes another way of measuring full power days. In most large nuclear plants, one full-power month is equivalent to 1000 to 1300 MWD/MtU for a PWR. Considering that a plant is not operated at full power all the time, one finds that the typical PWR core accumulates a burnup of 10,000-11,000 MWD/MtU every year. Fuel is typically used for about 3 years, so that the average burnup of fuel which is discharged for reprocessing is somewhat above 30,000 MWD/MtU. Particular fuel rods and particular pellets in this fuel will have much higher burnups.

For a BWR, one full-power month is equivalent to 500 to 700 MWD/MtU, so that the typical BWR core accumulates a burnup of 5,000-6,000 MWD/MtU each year. BWR fuel is typically used for 4-5 years, so that the average burnup of fuel discharged for reprocessing is somewhat above 25,000 MWD/MtU.

D. Operating Experience with Power Reactor Fuel

(1) Zircaloy-Clad Fuel

During the past few years, several reactors with Zircaloy-clad fuel have been placed in operation, and they are currently accumulating significant additional irradiation experience. Some of the accumulated experience will be discussed in the following paragraphs.

(2) Zircaloy Experience in BWRs

As of the completion of the 1967 Dresden I refueling outage, failed fuel location tests and underwater visual inspections have shown that between 22 and 30 out of 536 original Type I fuel assemblies had developed clad perforations in service. By the March-April 1968 refueling, at least ten other assemblies (Types III and IV) with Zircaloy-2 clad oxide fuel, including several with vibratory compacted UO_2 powder fuel, had defected. Approximately 20 assemblies with additional failed fuel were removed in the 1969 summer shutdown. Many of these failures were attributed to fuel being operated to approximately twice its design exposure, and to manufacturing defects.

Big Rock Point experience with Zircaloy clad UO_2 during the period 1966-1969 included failures in more than forty assemblies. These failures can be divided into two types:

(a) Failures in Dynapak Powder, vibratory compacted assemblies apparently caused by hydrogenous material contamination during manufacture.

(b) The remainder of the Zircaloy clad failures were characterized by gross craters in areas of spalled crud. Observations and analyses to date indicate that the fuel rod failures resulted from heavy buildup of crud scale that caused the cladding surfaces to operate at abnormally high temperatures and corrode at the high rates expected at those high temperatures at the peak power locations. The crud apparently resulted from the materials in the feedwater-condensate system.

The major fuel failure experienced with a European BWR has occurred in the KRB reactor in West Germany which has been operating since 1966. Serious

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failures of Zircaloy-2 clad UO_2 fuel rods have occurred in this General Electric fuel. By the time of the first refueling in the summer of 1969, approximately one-third of the core assemblies were defected. Many individual rods were found with identifiable defects. The defects have been attributed to: fretting of wire segments carried from a failed steam separator into the fuel assemblies, internal hydriding and tubing defects that occurred during manufacture.

BWR Zircaloy-clad UO_2 experience in most other reactors has been relatively good, although failures are known to have occurred in KWL (Lingen), KAHL and Dresden II. In Dresden II, 29 assemblies were identified as failed on the basis of sip signals or visual inspections and were replaced soon after startup (summer of 1970). Activity levels in other BWR's such as Tarapur, Tsuruga and Oyster Creek have been slightly higher than normal, which is probably indicative of a few fuel failures or defects. With the large number of BWRs currently operating or due to be started up in the near future, considerable additional experience with Zircaloy-clad UO_2 fuel should be accumulated rapidly. However, except for unusual circumstances, significant fuel performance experience is not available until after several years of operation and/or the first refuelings.

In summary, peak linear heat generation rates from approximately 10 to 16 kw/ft with burnup up to 33,000 MWD/MtU have been achieved with BWR Zircaloy clad oxide production fuel. Several failures have been experienced. However, when all General Electric production fuel types are considered, less than 0.5% have experienced failure due to cladding perforation. Most failures to date appear to have been caused by manufacturing defects, quality control or plant chemistry problems rather than the exceeding of any design or materials limitations.

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(3) Zircaloy Experience in PWRs

PWR experience in the United States with Zircaloy-clad oxide fuel is more limited than BWR experience, and few failures have been reported. A limited amount of information on fuel failures and operating experience is given below.

The Shippingport blanket represented the first commercial use of Zircaloy-clad UO_2 fuel elements. Of some 94,000 Zircaloy-clad fuel rods fabricated for Shippingport Core I, only three showed any evidence of leaks during or after irradiation. Two clad perforations were reported in 16,000 rods during the six-year life of the Core I blanket. These were attributed to transverse tubing cracks originating at the inner surface during fabrication. These defected rods were operated for long periods in the Shippingport reactor without any interference with plant operation. The Zircaloy-clad fuel in the blanket rods of Shippingport Core I attained maximum burnups of 37,000 MWD/MtU without significant changes in fuel rod dimensions or change in corrosion behavior as a result of irradiation.

Although CVTR was a heavy water cooled and moderated pressure tube thermal reactor, irradiation experience there is of interest. Most of the fuel for CVTR was Zircaloy-clad UO_2 manufactured by Westinghouse and similar in design to present PWR fuel. Approximately 1,49? Zircaloy-clad UO_2 rods were irradiated with failures in five assemblies over the period October 1966-January 1967. These failures included three R&D high power density assemblies which failed at peak ratings between approximately 16 and 21 kw/ft and two standard elements operating at approximately 12 to 14 kw/ft. Examination of the failed rods showed gross cracking and blistering had occurred with local areas of massive hydriding.

Zircaloy-4 clad oxide fuel is also utilized in the Zorita, Beznau and Obrighheim reactors in Europe and the Ginna plant in U.S. In each of these reactors, there has been some evidence of apparent fuel defects. Zircaloy-clad UO_2 fuel has been tested in Saxton and Yankee-Rowe with no known defects. All PWR plants that are currently in various stages of early operation or construction utilize Zircaloy cladding. Similar to the case for BWRs, significant additional fuel performance experience will be accumulated gradually over the next few years as more plants come on line and/or the above plants are refueled.

(4) Stainless Steel Experience in BWRs

Performance of stainless steel clad fuel in the BWR has not been good. Most of the failures reported for stainless steel elements have occurred in these reactors. Experience at VBWR showed SS-304 susceptible to intergranular cracking originating at the surface.

The same bad experience has occurred in Dresden, where 17 severe failures occurred (from one assembly about 111 inches of fuel rod were missing), in Big Rock and Humbolt Bay, where severe corrosion damage was observed, and in Elk River, where several fuel rods were bowed.

On the basis of the poor performance of free-standing SS-304 above 15,000 MWD/t, plus improved economics with high burnup Zircaloy-clad fuel, General Electric no longer uses stainless steel clad fuel in boiling water reactors.

(5) Stainless Steel Experience in PWR's

PWR irradiation constitutes the bulk of stainless steel experience. These PWR elements have been relatively free of failure. Most experience has been obtained with annealed SS-348 and SS-304 ≥ 0.02 inch thick. Local peak exposure obtained with such fuel is 46,000 MWD/t at Yankee.

To improve neutron economy, second generation PWR fuel elements have used cold worked SS-304. This permits a reduction in cladding thickness to 0.014-0.016 inch. Some failures have occurred with this fuel in San Onofre, Connecticut Yankee and Indian Point.

Because of improved economics, all new PWR's will have Zircaloy-clad fuel, and it is anticipated that, in existing PWR's, Zircaloy-clad fuel will gradually replace stainless steel clad fuel.

E. Neutron Absorber Materials

(1) The main requirements for a control rod absorber material are that it have a high neutron absorption cross section over as large a spectrum as possible, and that it last for a reasonably long time, that is, not burn up rapidly. Burnup means that the absorber atoms, after having absorbed a neutron, are no longer of use. Another desirable quality is corrosion resistance to hot water.

(2) The best single material, cost not considered, for thermal reactor control is hafnium, a metal. Hafnium has four major isotopes that have moderate thermal cross sections and good epithermal cross sections. Hafnium has a long lifetime, because, when it absorbs a neutron, the result is another isotope of hafnium which also has a good cross section.

(3) Cadmium, another metal, has an extremely high thermal absorption cross section, but in the epithermal range its cross section is nil. Because of this, and because of its low melting point, cadmium cannot be used alone.

(4) An alloy being successfully used for control rod materials is silver-indium-cadmium in the weight percent ratio 80-15-5. This material combines the good epithermal cross sections of silver and indium and the high thermal cross section of cadmium. This material is used for the control rods of many modern PWR's. It is extruded into rods which are then inserted into stainless steel tubes. The stainless steel tubes are the rod "fingers" which ride up and down within a fuel assembly, carrying the absorber material inside.

(5) Four rare earth elements: samarium, europium, gadolinium, and dysprosium, would make good control materials but their scarcity makes them too expensive for general use. They are usually used in the oxide form dispersed in stainless steel.

(6) The last material to be considered is boron, of which the isotope B-10 has a high thermal cross section and good epithermal cross section. B-10 occurs in an abundance of about 20 percent by weight in natural boron, which is readily available and cheap. It is used in the form of powdered ceramic material in stainless steel tubes. In General Electric BWR's a series of these tubes is arranged in a cruciform pattern. The ceramic material is boron carbide (B_4C). The large number of tubes eliminates the possibility of gross failure due to water inleakage and reaction with the carbide. The major problem in the use of boron is that the byproduct of the neutron absorption reaction is helium gas. This requires that the density of the B_4C be held to 65 to 75 percent of maximum possible to allow space for helium release. At one time a dispersion of boron in stainless steel was used, but the helium release resulted in brittle behavior of the stainless steel in which severe cracking could result. In PWR's the "finger" rods are sometimes made out of B_4C encased in stainless steel tubes.

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F. Coolants

(1) The choice of coolant for a nuclear reactor is governed by a number of factors. Some of the factors to be considered and their desired characteristics are:

- Heat transfer and transport properties: the coolant should have a good heat transfer coefficient and a good heat capacity on a volume basis.
- Nuclear properties: for thermal reactors, a low absorption cross section is necessary for good neutron economy, and good moderating properties are desirable.
- Vapor pressure: the coolant should have a low vapor pressure at the operating temperature in order to minimize the piping and container thickness required.
- Corrosion and chemical activity: there should be a minimum of corrosion or chemical reaction between the coolant and the material (fuel, structure, air, etc.) contacted by the coolant.
- Radioactivity: coolants that become activated and form hard γ emitters with long half-lives should be avoided. Activated coolants that are α and β emitters can easily be shielded, but must be contained.
- Decomposition: coolants should be stable under irradiation and at the operating temperatures of the reactor.
- Melting point: the melting point of coolants that are solid at room temperature should be easily achievable by compartment heating.

The choice of a coolant usually depends on a compromise, since no known coolant embodies all the desired characteristics. The coolant used in all large power reactors in the U.S.A. is water.

(2) Water is a fairly good heat-transfer agent, is easy to handle, provides some lubrication, offers no serious corrosion problems that are not surmountable by proper choice of materials, and is readily available and economical. For these reasons, it has found wide application as a reactor coolant.

(3) Although the (n,p) reaction of O-16 forms N-17, a very hard γ emitter, its half-life of 7.5 sec is short enough so that there is no serious activation problem. There is also a minor activation of O-18 with an 18-sec half-life of no particular consequence. The water must be continuously purified with mixed-bed ion exchangers or other means to keep down the activity of corrosion products. The most serious drawback with water is its high vapor pressure; even the modest coolant temperature of 550°F requires a primary system pressure of 1,500 psi.

G. Reactor Internals

(1) The reactor internals are designed to support and orient the fuel assemblies and control rod assemblies, absorb the forces generated by dropping the control rods, and transmit these and other loads to the reactor vessel flange, provide a passageway for the reactor coolant, and support in-core instrumentation. The internals are designed to withstand the combination of forces due to weight, preload of fuel assemblies, differential hydraulic pressure, control rod dynamic loading, vibration, and earthquake acceleration. The components of the reactor internals are divided into three parts, consisting of the lower core support structure including the entire core barrel and thermal shield, the upper core support structure, and the in-core instrumentation support structure.

(2) The major containment and support member of the reactor internals is the lower core support structure. This support structure assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the thermal shield, the intermediate diffuser plate and the bottom support plate which is welded to the core barrel. All the major material for this structure is Type 304 stainless. The lower core plate is perforated to allow water to flow through it, and it contains the lower locating pins for the fuel assemblies. The lower core support structure also provides passageways and control for the coolant flow. Inlet coolant flow from the vessel inlet nozzles will proceed down the annulus between the core barrel and the vessel wall, on both sides of the thermal shield, and into a plenum at the bottom of the vessel. It will then turn and flow up through the bottom support plate, pass through the intermediate diffuser plate and then through the lower core plate. The flow holes in the diffuser plate will be arranged to prevent gross inlet flow maldistribution to the core. After passing through the core the coolant will enter the area of the upper support structure and then generally flow radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles.

(3) The upper core support assembly consists of the top support plate, beam sections, upper core plate, support columns, and guide tube assemblies. The upper core support assembly is removed as a unit during refueling operation. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper core support assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies and control rods is assured by this guidance arrangement.

(4) The in-core instrumentation support structures consist of an upper system to convey and support thermocouples penetrating through the vessel head, and a lower system of flux thimbles penetrating the bottom of the vessel.

(5) For the most part, reactor internals are made of stainless steel to prevent corrosion and provide the required strength. Neutron economy is not a factor in the regions where these components are located.

H. Reactor Vessel

(1) From the operator's viewpoint, the most important property of the reactor vessel is its ductility. The more a material can deform under load before reaching the fracture point (or the yield point), the more ductile it is said to be. Good ductility is a desirable characteristic in pressure vessel materials.

(2) Ductility is often thought of as deformation under slowly applied loads or forces, but experience has shown that materials also differ in their ability to absorb rapidly applied or "shock" loads, and that this ability is not necessarily the same as the ductility determined for slowly applied loads. In other words, a material which deforms with high ductility under slowly applied loads may not show high ductility under rapidly applied loads. This requires an additional test of the material to determine ductility under rapidly applied loads. The Charpy test is one that is used frequently and consists of swinging a weighted pendulum against a standardized specimen. Because the presence of surface irregularities was found to affect the results of these tests, a controlled irregularity, or standard sized V-shaped notch is used in the Charpy tests.

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The energy of the impact required to break the specimen is an indication of its ductility — the larger the energy required, the greater the ductility.

(3) Ductility is important in the reactor vessel because the lack of it can cause a type of failure referred to as a "brittle fracture," in which the vessel splits open. Actual failures due to brittle fracture have been experienced in tanks, structures, and even ship hulls of certain types of steels employed in cold environments.

(4) It has been found that the energy required to break the specimen in a Charpy test depends upon its temperature. For every specimen, there is some temperature below which the energy required is negligible — that is, below this temperature the metal becomes almost non-ductile. This temperature is known as the NIL DUCTILITY TRANSITION TEMPERATURE, abbreviated (for obvious reasons) NDTT. This temperature varies, even among the same kind of material, due to minor differences in composition and in the methods and rates of heating, cooling, and fabricating the metal. Therefore, samples must be taken from the same metal used to fabricate each reactor vessel, and these samples must be tested to determine the NDTT for each vessel. The manufacturer can generally maintain the NDTT below 30°F at the time of fabrication. It is, therefore, important not to subject the pressure vessel to high pressure when the metal temperature is near the NDTT.

(5) Unfortunately, radiation with fast neutrons is known to increase the NDTT. Therefore, the vessel is manufactured and shielded in such a way that even at the end of plant life (after 30 or 40 years) the NDTT should be significantly below the operating temperatures.

(6) The change in NDTT can be correlated with the fast flux exposure experienced by the vessel. The fast flux exposure

(sometimes the symbol n v t is used for it) may be calculated as

$$n\,v\,t\ (n/cm^2) = \varphi_{Fast}\,t$$

where

φ_{Fast} = fast flux at vessel (assumed constant with time). $(n/cm^2 - sec)$

t = time in seconds

(7) Example

The fast flux at the surface of a reactor vessel is 4×10^{10} at full power. What is the fast flux exposure of this part of the vessel after 1 full-power year?

$$1 \text{ year} = \frac{3600 \text{ sec}}{\text{hr}} \times \frac{24 \text{ hr}}{\text{day}} \times 365 \text{ days}$$

$$= 3.154 \times 10^7 \text{ sec}$$

$$\begin{aligned} n\,v\,t &= \varphi t \\ &= 4 \times 10^{10} \times 3.15 \times 10^7 \\ &= 1.26 \times 10^{18}\ n/cm^2 \end{aligned}$$

(8) Since the NDTT is in the neighborhood of 30°F when the plant starts up originally, and since the NDTT increases with fast flux exposure, it is to be expected that <u>some</u> plant operations may be conducted in the neighborhood of the NDTT or below it. Power operation is, of course, restricted to occur far above the NDTT, but heat-ups and cool-downs will occur near the NDTT. Therefore, to avoid trouble, the approach taken is to avoid stressing the reactor vessel significantly until its temperature is 50-75°F above the NDTT. Stresses are avoided by:

- Limiting the maximum primary pressure to a very low value until the vessel temperature is high enough.

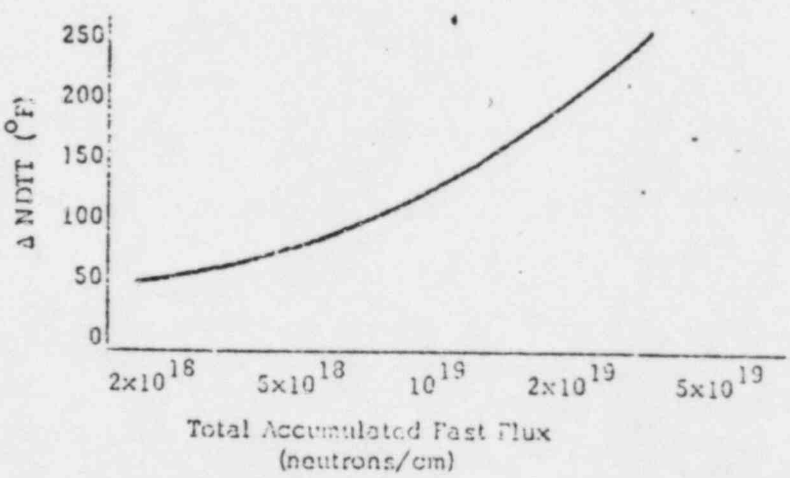
- Limiting the heat-up and cool-down rates ($^{o}F/hr$) to low values until the vessel temperature is high enough. This avoids the high thermal stresses which may occur when there are large temperature differences across the vessel.

Problem Set - Chapter 7

1. A typical reactor has 175 fuel elements, each of which contains 1000 lbs. of uranium (all isotopes) when it is fresh. The rated core power is 2700 Mw(t). What is the average fuel burnup of a fuel element if each element is exposed to the same integrated neutron flux during three years of full power operation? Express your answer in MWD/MtU.

2. (a) The critical section of a reactor vessel was exposed to a fast neutron flux of 4×10^{10} neutrons/cm^2 sec in the first ten years of plant operation, during which time the plant availability was 80%. What is the total accumulated fast flux exposure?

   (b) Given the graph below, what is the change in NDTT after ten years?



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8. TECHNICAL SPECIFICATIONS

A. Introduction

In this chapter, we will examine some typical Technical Specifications to see how they relate to the safety analysis and to the operating restrictions.

As we pointed out in Chapter 1, the tech specs consist of 5 parts:

1. Safety limits and maximum safety system settings;
2. Limiting conditions for operation;
3. Surveillance requirements;
4. Design features;
5. Administrative controls.

Parts 1 and 2 are those that relate especially to core performance. We will examine in detail one spec from Part One (Safety Limit - Reactor Core) and one from Part Two (Control Rod and Power Distribution Limits). We will reproduce each specification completely so that you can see what it looks like. The comments in the square brackets [ ] are not part of the spec, but are explanations added by NUS.

B. Examination of a Tech Spec

(1) Applicability:

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, coolant temperature, and flow when the reactor is critical. [The purpose of the "applicability" statement is merely to "tip off" the reader on what the spec is about.]

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(2) Objective:

To maintain the integrity of the fuel cladding. [This is what we've been concentrating on in the Core Performance course. You will find that all "objectives" in the tech specs relate to safety — not to economics or efficiency.]

(3) Specification:

[The "specification" section gives the exact legal rules which you must follow.]

a. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 2.1-1 when full flow from three reactor coolant pumps exists and shall not exceed the limits shown in Figure 2.1-2 when full flow from two reactor coolant pumps exists. [Rule "a" relates to operation in the vicinity of rated power. Notice that it doesn't give any explanation of "why," it just tells "what" the limits are. The "whys" will come later in the "Basis" section.]

b. When full flow from one reactor coolant pump exists, the thermal power level shall not exceed 20%, the coolant pressure shall remain between 1820 psig and 2400 psig and the Reactor Coolant System average temperature shall not exceed 590°F. [Rule "b" covers operation with only one coolant pump, which is restricted to low power levels.]

c. When natural circulation exists, the thermal power level shall not exceed 12%, the coolant pressure shall remain between 2135 psig and 2400 psig and the Reactor Coolant System average temperature shall not exceed 602°F. [Rule "c" covers operation

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with no coolant pumps, and is therefore restricted to very low power levels. Ordinarily you wouldn't operate with no pumps or with only one pump; the purpose of rules 2 and 3 is to make it legally clear that you can operate under these conditions if you want to or need to.]

d. The safety limit is exceeded if the combination of Reactor Coolant System average temperature and thermal power level is at any time above the appropriate pressure line in Figures 2.1-1 or 2.1-2 or if the thermal power level, coolant pressure, or Reactor Coolant System average temperature violates the limits specified above. [The only purpose of rule "d" is to avoid legal hassles by making it explicitly clear what regions of operation are allowed and what are not allowed.]

(4) Basis:

[The "specification" above the four rules a-d is the legally binding part of the tech spec. The "Basis" is background information which explains the logic used in arriving at the four rules, but statements in the "Basis" are not themselves legally binding.]

[The "Basis" starts with a paragraph that explains what the problem is and how it is attacked. The Core Performance course should enable you to understand most of this paragraph.]

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by maintaining the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the

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coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters --thermal power, reactor coolant temperature and pressure, have been related to DNB through the W-3 DNB correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNB ratio, DNBR, during normal operational transients and anticipated transients (those transients listed on page 14.1-1 of the FSAR) is limited to 1.30. A DNB ratio of 1.30 corresponds to a 95 % probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions. The DNB ratio limit of 1.30 is a conservative design limit which is used as the basis for setting core safety limits. Based on rod bundle DNB tests, no fuel rod damage is expected at this DNB ratio or greater.

[The following two paragraphs explain the implications of rule "a".]

The curves of Figure 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (three-loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which the DNB ratio is not less than 1.30. The area where clad integrity

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is assured is below these lines. In order to completely specify limits at all power levels, arbitrary constant upper limits of average temperature are shown for each pressure at powers lower than approximately 75%. The temperature limits at low power are considerably more conservative than would be required if they were based upon a minimum DNB ratio of 1.30 but are such that the plant conditions required to violate the limits are precluded by the self actuated safety valves on the steam generators. An arbitrary upper safety limit of 130% for thermal power is shown. The upper limit is well below the damage limit of 1.7% for maximum clad strain which is reached at 140% thermal power with design hot channel factors.

The curves of Figure 2.1-2 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two-loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the DNB ratio is equal to 1.30 or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNB ratio reaches 1.30 and, thus, this arbitrary limit is conservative with respect to maintaining clad integrity. In order to completely specify limits at all power levels, arbitrary constant upper limits of average temperatures are shown for each pressure at powers lower than approximately 40%. The limits at low power as well as the limits based on the average enthalpy at the exit of the core are considerably more conservative than would be required if they were based upon a minimum DNB ratio of 1.30. The plant conditions required to violate these limits are precluded by the protection system and the self-actuated safety valves on the steam generator. An upper limit of

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70% for power is shown to completely bound the area where clad integrity is assured. This latter limit is arbitrary but cannot be reached due to the Permissive 8 protection system setpoint which will trip the reactor on high nuclear flux when only two reactor coolant pumps are in service.

[The following one-sentence paragraph tells us that rules "b" and "c" give us as much margin at lower power as rule "a" gives us at high power.]

The limits specified for one-loop operation and natural circulation result in DNB ratios greater than 1.30.

[The following paragraph tells us how this spec relates to things you can measure or control from the operating room - hot channel factors and control rod insertion limits.]

The specified limits are based on the following nuclear hot channel factors:

$$F_q^N = 3.13; \; F_{\Delta H}^N = 1.75$$

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are covered by Specification 3.10. Somewhat worse hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits dictated by Figure 3.10-1 ensure that the DNB ratio is always greater at part power than at full power. [Recall that the hot channel factor is a ratio and that the actual max heat flux is:

Max heat flux = Average heat flux x hot channel factor.

When the power level is reduced, the average heat flux decreases. Therefore, for a fixed max heat flux, higher values of the hot channel factor are permissible at lower powers.

[The next paragraph merely points out that these limits include some extra margin to account for errors in positioning the part-length rods. Such errors could increase the hot channel factors - but not beyond the values quoted above.]

The hot channel factors are also sufficiently large to account for the degree of malpositioning of part-length rods that is allowed before the reactor trip set points are reduced and rod withdrawal block and load runback may be required. Rod withdrawal block and load runback occurs before reactor trip setpoints are reached.

[The next paragraph merely points out that this whole spec assumes the pumps are running normally, and does not cover a "loss of flow" event, in which flow is lost because of pump failure or loss of all electrical power.]

The safety limit curves given in Figures 2.1-1 and 2 are for constant flow conditions. These curves would not be applicable in the case of a loss of flow transient. The evaluation of such an event would be based upon the analysis presented in Section 14.1 of the FSAR.

[The following paragraph gives some of the assumptions which went into the analyses that determined these safety-limit curves. Note that maximum measurement errors are specified here. Therefore, you have to be able to show that your actual measurement errors are less than these. This is why we discussed heat balances early in our course.]

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure and thermal power level that would result in a DNB ratio of less than 1.30 based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 574.2°F and a steady state nominal operating pressure of 2235 psig. Allowances are made in initial conditions assumed for transient analyses for steady state errors of $\pm 2\%$ in power, $\pm 4^{\circ}$F in Reactor Coolant System average temperature and $\pm$ 30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions. The steady state nominal operating parameters and allowances for steady state errors given above are also applicable for two-loop operation except that the steady state nominal operating power level is less than or equal to 45%.

[The spec concludes with a promise by the utility to "keep its eyes open" for fuel defects.]

To provide the commission with added verification of the safety and reliability of pre-pressurized zircaloy clad nuclear fuel, a limited program of non-destruction fuel inspection will be conducted. The program shall consist of a visual inspection (e.g., underwater TV, periscope, or other) of the two lead burnup fuel assemblies during the second and third refueling outages. Any condition observed by this inspection which could lead to unacceptable fuel performance may be

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the object of an expanded effort. The visual inspection program and, if indicated, the expanded program will be conducted in addition to that being performed in the Saxton and Cabrera reactors. If another domestic plant which contains pre-pressurized fuel of the same design as that used for H.B. Robinson Unit No. 2 reaches the second and third refueling outages first, and if a limited inspection program is or has been performed there, then the program may not have to be performed at H.B. Robinson Unit No. 2. However, such action requires approval of the AEC.

(5) Explanation of Safety Limit Curves

(i) The average temperature in the core is plotted vertically. To relate this to measurable quantities, it is taken as the average of the inlet and outlet temperatures.

(ii) The percent of rated thermal power is plotted horizontally.

(iii) Each curve represents the safety limit of the nominal primary pressure associated with that curve.

(iv) Pick a point on one of the curves at the average core temperature and power which correspond to that point, and at the pressure corresponding to that curve, the minimum DNBR in the core is 1.30. For the same pressure, points <u>below</u> the curve will result in minimum DNBR's higher than 1.30. Points <u>above</u> the curve will result in minimum DNBR's less than 1.30 (Actually, in some places the curves are drawn so that you <u>could</u> go above them without falling below the DNBR limit of 1.30, but we needn't worry about this.)

1103 1-3-73

(6) Significance of the tech spec

This spec gives combined limits on temperature, pressure, and power level. In addition to not exceeding these limits, you must not do anything which would invalidate the assumptions used in deriving these limits. These assumpions relate to

- Measurement errors
- The reactor control and protection system

Thus, you must be sure that the instrumentation, both the automatic instrumentation in the control system, and the manual instrumentation used in heat balances, operates properly in order to keep plant operation within legal bounds.

C. Examination of a Tech Spec - Control Rod and Power Distribution Limits. [The layout of all tech specs is similar, so we will restrict ourselves to technical comments on this spec. In addition, because the "specification" is the only legally binding part, we will not include the "Basis".]

(1) Applicability:

Applies to the operation of the control rods and power distribution limits.

(2) Objective:

To ensure (1) core subcriticality after a reactor trip, (2) limited potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.

"Core subcriticality after a reactor trip" is the same thing as the shutdown margin discussed in Reactor Operations. It is important to limit the "potential reactivity insertions" because some maximum value was assumed in the Safety Analysis. The "core power distribution" has to be limited to avoid exceeding the hot channel factors used in the tech spec on Reactor Core Safety Limit.

(3) Specification:

1.0 Control Rod Insertion Limits

1.1 When the reactor is subcritical prior to startup the shutdown margin shall be at least that shown in Figure 3.10-2.

1.2 When the reactor is critical, except for physics tests and control rod exercises, the shutdown control rods shall be fully withdrawn.

1.3 When the reactor is critical, except for physics tests and control rod exercises, the control group rods shall be no farther inserted than the limits shown by the solid lines on Figure 3.10-1 for 3-loop or 2-loop operation.

1.4 After 70% of the second and subsequent cycles as defined by burnup, the limits shall be adjusted as a linear function of burnup toward the end-of-core life values as shown by the dotted lines on Figure 3.10-1.

1.5 During physics tests and control rod exercises, the insertion limits need not be observed, but the limits shown in Figure 3.10-2 must be observed.

[Note that items 1.2 to 1.4 relate to normal power operation. Items 1.1 and 1.5 relate to zero-power or test conditions. It is the reactor engineer's responsibility to decide how to set the rods to meet the shutdown required in items 1.1 and 1.5.]

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2.0 Power Distribution Limits

2.1 If the quadrant to average power tilt ratio exceeds 1.1 except for physics tests, or:

If a part length or full length control rod is more than 15 inches out of alignment with its bank, then within eight hours:

a. The situation shall be corrected, or

b. The hot channel factors shall be determined and maximum allowable power shall be reduced one percent for each percent the hot channel factor exceeds the design values of

$$F_q^N = 3.13;\ F_{\Delta H}^N = 1.75, \text{ or}$$

c. Power shall be limited to 75% of rated power for 3-loop operation or 45% of rated power for 2-loop operation.

2.2 If after a period of 24 hours, the power tilt ratio in 3.10.2.1 is not corrected to less than 1.1:

a. An evaluation of the cause of the discrepancy shall be made and reported to the Atomic Energy Commission, and

b. The nuclear overpower, overpower ΔT and overtemperature ΔT trips shall be reduced one percent for each percent the operating power level has been reduced.

2.3 If the quadrant to average power tilt ratio exceeds 1.25, the reactor shall be put in the hot shutdown condition utilizing normal operating procedures and the Atomic Energy Commission shall be notified.

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[The power distribution limits all refer to abnormal conditions. If everything is normal, one needn't worry about these specs. Although it is the reactor engineer's responsibility to calculate power tilts and hot channel factors, operators and instrument technicians are involved in keeping up the instruments and may assist in making measurements. Since certain conditions require reducing power and reporting to the AEC, it is important that the instruments be working properly and that the measurements be made carefully. Furthermore, it can be almost as costly to erroneously think you've exceeded a tech spec limited as to actually exceed it.]

3.0 Rod Drop Time

3.1 The drop time of each control rod shall be no greater than 1.8 seconds at full flow and operating temperature from the beginning of rod motion to dashpot entry.

4.0 Inoperable Control Rods

4.1 A control rod shall be deemed inoperable if (a) the rod is misaligned by more than 15 inches with its bank, (b) if the rod cannot be moved by its drive mechanism or (c) if its rod drop time is not met.

4.2 No more than one inoperable control rod shall be permitted during power operation.

4.3 If a control rod cannot be moved by its mechanism, boron concentration shall be changed to compensate for the withdrawn worth of the inoperable rod such that shutdown margin equal to or greater than shown on Figure 3.10-3 results.

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[The spec on rod drop time is included because a maximum drop time of 1.8 seconds was assured in the safety analysis. The spec on inoperable control rods relates to an abnormal condition. Normally, one need not worry about this.].

(4) Explanation of Figure 1, Control Group Insertion Limits.

(i) The percent withdrawal is plotted vertically. 100% means the rod bank is all the way out; 0% means it is all the way in.

(ii) The control rod banks are designated by the letters A through D. Group A is the shutdown bank, which must be completely withdrawn when the reactor is critical and is therefore not shown on this figure.

(iii) Horizontally, we plot the reactor power in percent of rated MWt.

(iv) Ignore the dotted lines for the moment. At any given power level, the position of a control rod bank must be <u>above</u> the value read from the corresponding curve. For example, at zero power, group B must be <u>at least</u> 91% withdrawn (it <u>may</u> be farther withdrawn.) Above 13% power, group B must be 100% (completely) withdrawn. Group C must be at least 33% withdrawn at zero power, at least 61% withdrawn at 40% power, at least 90% withdrawn at 80% power, and 100% withdrawn above 94% power. At 100% power, group D must be at least 47% withdrawn.

(v) The dotted lines are revised limits which apply after 70% of the <u>second</u> cycle's life. We won't worry about them here.

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(5) Significance of the tech spec

These limits on control rod insertion are important because you must always observe them in operating the plant. If you violate these limits you have violated the tech specs.

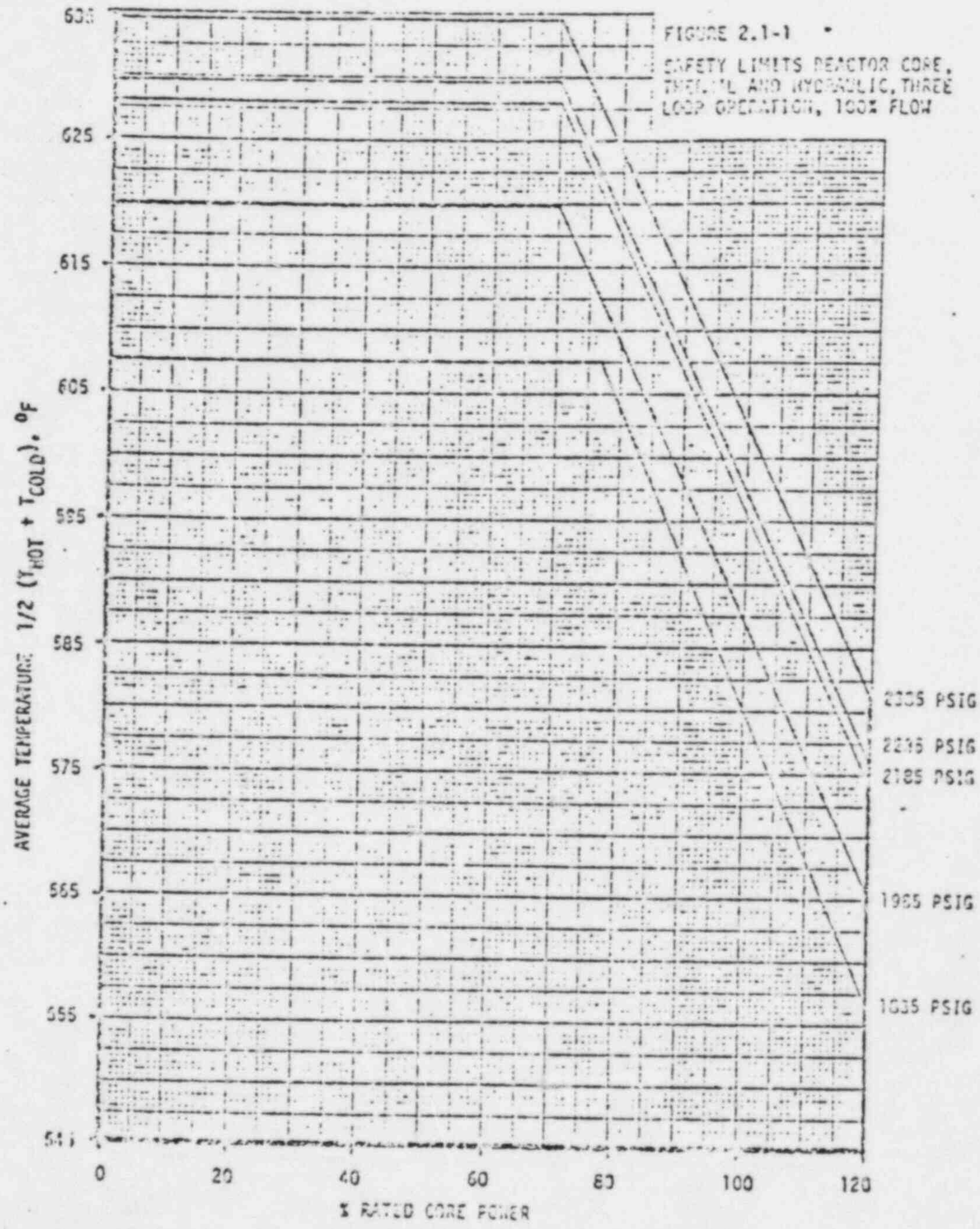
FIGURE 2.1-1
SAFETY LIMITS REACTOR CORE, THERMAL AND HYDRAULIC, THREE LOOP OPERATION, 100% FLOW
AVERAGE TEMPERATURE 1/2 (T HOT + T COLD), °F
625
615
605
595
585
575
565
555
0
20
40
60
80
100
120
% RATED CORE POWER
2235 PSIG
2185 PSIG
1985 PSIG
1835 PSIG

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CORE PERFORMANCE
8. Technical Specifications (cont.)

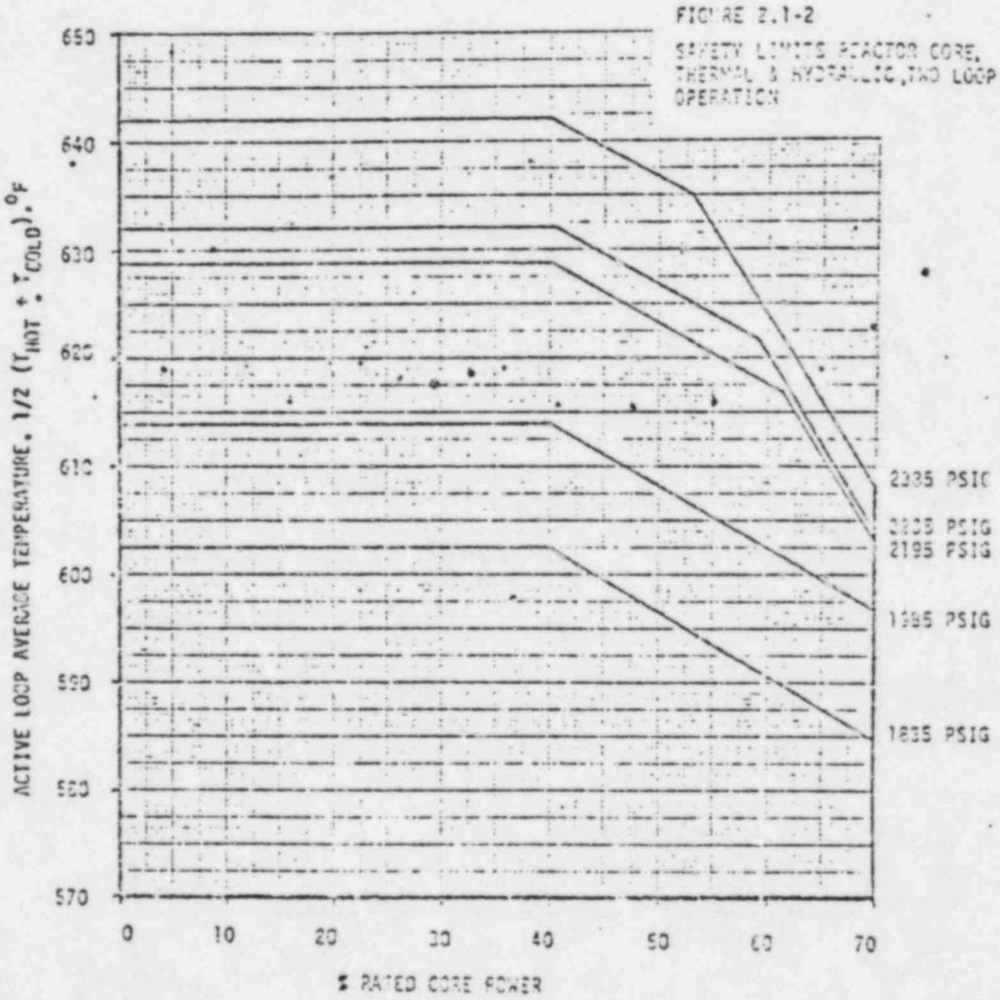

FIGURE 2.1-2
SAFETY LIMITS REACTOR CORE, THERMAL & HYDRAULIC, TWO LOOP OPERATION

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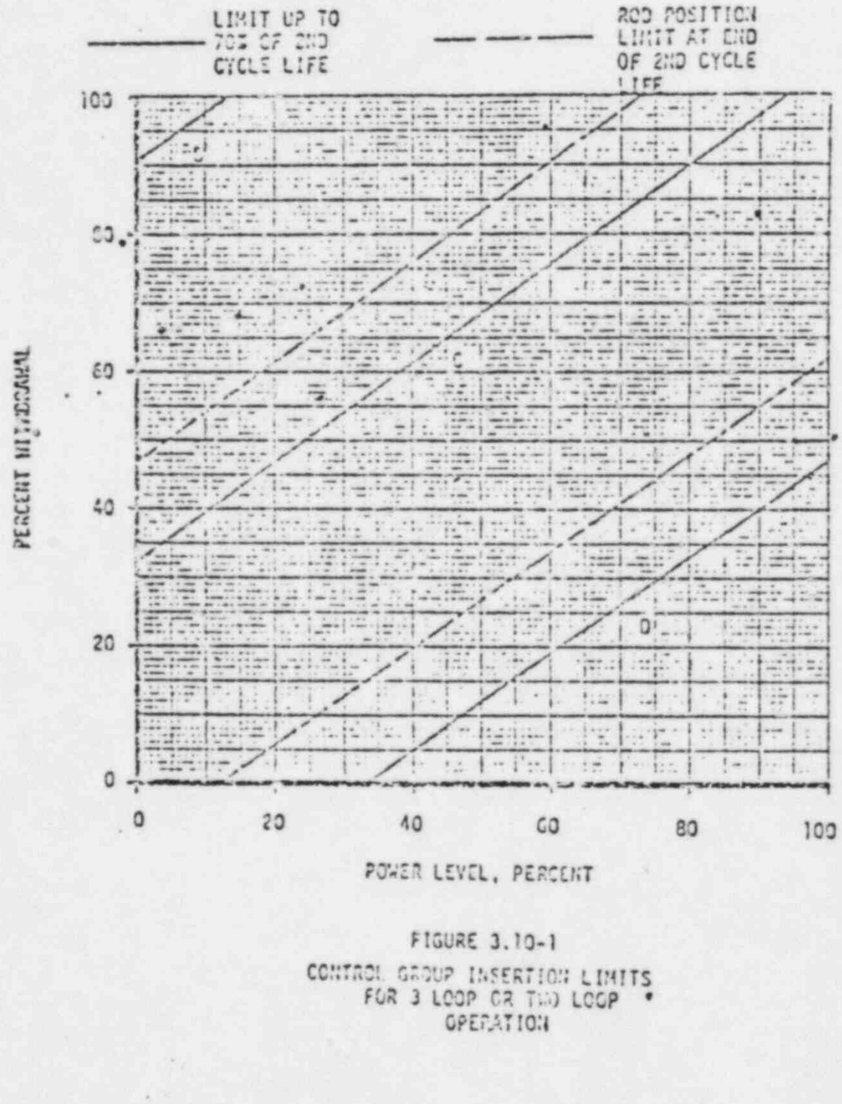

FIGURE 3.10-1
CONTROL GROUP INSERTION LIMITS FOR 3 LOOP OR TWO LOOP OPERATION

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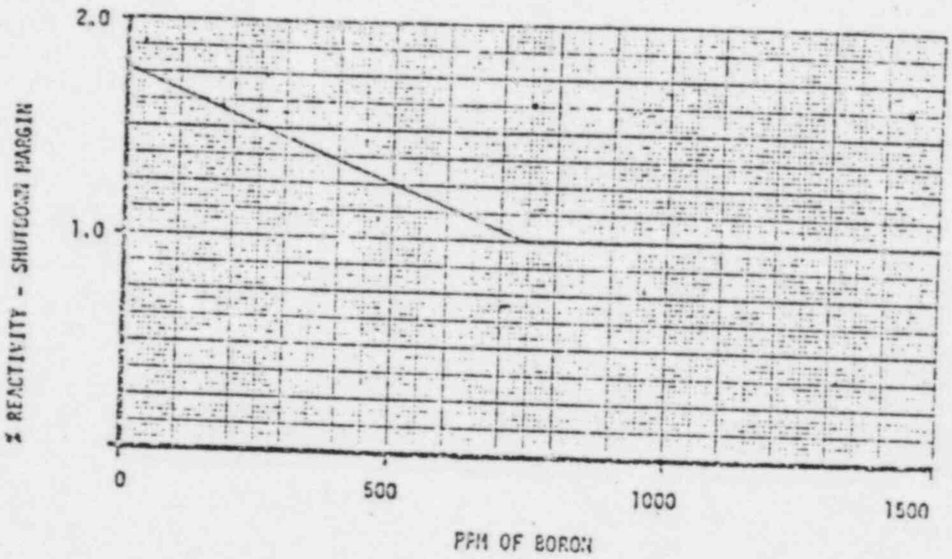

FIGURE 3.10-2
REQUIRED SHUTDOWN MARGIN
VS REACTOR COOLANT BORON CONCENTRATION

STEAM TABLES

APPENDIX A

STEAM TABLES

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NUCLEAR POWER PREPARATORY TRAINING

CORE PERFORMANCE

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Rockville, Maryland

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CORE PERFORMANCE - P-B

CONTENTS

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FIGURES

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CORE PERFORMANCE

1. BACKGROUND INFORMATION

A. Introduction and Scope

Core Performance is:

Getting the heat out of the fuel
Into the water
And to the turbine.

The object of learning about core performance is to operate the plant:

- Efficiently
- Safely
- Legally

The last two items are particularly important in a nuclear plant because an AEC license is required, and the AEC not only requires you to operate safely, but to be able to prove it! Your plant has a set of Technical Specifications approved by the AEC. These "Tech Specs" are a set of rules that limit how you can run the plant. These rules are concerned only with avoiding the uncontrolled release of radioactivity -- the AEC cares only that you operate safely. Efficiency is not of AEC concern.

To understand where these safety rules come from and what are the consequences of violating them, as well as how to operate efficiently, we need to know a little about:

- Thermodynamics
- Fluid Flow
- Heat Transfer

- Behavior of Materials in Core
- In-Core Instrumentation

This course will cover these topics, but only to the extent to which you must know them in order to understand the legal safety limits in the Technical Specifications.

B. General Description of Boiling Water Reactor

We will now look at some of the systems of a boiling water reactor (BWR) steam supply to point out some of the more important areas with which we will be concerned. The other general type of large power reactor that is being built in the United States today is the pressurized water reactor (PWR), in which the steam is generated in a different manner. All of our discussions will be related to your plant, though we will review the other type.

The reactor system is shown on Figure 1A and the reactor vessel and a simplified sketch of the recirculation system are shown in Figure 1B. A cutaway view of the reactor vessel and some internals are shown in Figures 1C, 1D, and 1E, and the core, fuel assemblies and fuel rods are shown in Figures 1F through 1H.

The half of the system bounded by the turbine and feed pump will not be described in detail because of the similarity to fossil-fired plant systems.

The reactor vessel is a large steel cylinder with a welded bottom head, a bolted upper head, and nozzles for penetrations. The vessel contains:

the core, which includes the fuel and the neutron absorber rods used for control

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- steam separation equipment which is similar in principle to that in conventional fossil-fired steam generators
- jet pumps (operating principle is similar to air ejectors)
- feedwater distribution sparger.

In addition, the vessel contains other items not directly connected with thermal performance. Some examples are shielding, instrumentation, irradiation specimens, support structure for all the internals, and sparger for safety cooling in the event of certain possible but improbable accidents.

The core is composed of many fuel assemblies positioned parallel to each other in a gridwork. Each fuel assembly is made up of a large number of <u>fuel rods</u>. Each fuel rod is a vertical sealed tube of Zircaloy* containing a stack of pellets of uranium dioxide (UO_2). The UO_2 is a ceramic material with a very high melting point (about 5000°F). A spring and insulating disc in the upper part of the fuel rod holds the pellets down, but allows them to expand as they heat up. The upper part of the fuel rod is a void that serves as a collection point for gases released by the fuel.

Heat is generated in the UO_2 pellets and most of it passes radially outward through the thin Zircaloy fuel <u>cladding</u>.

Water within the fuel assembly flows upward. The heat passing to the water causes it to boil, and a significant amount of boiling takes place in the upper two-thirds of the fuel assembly. A water-steam mixture leaves the fuel assembly with a quality of 10 to 15 percent (moisture content 85 to 90 percent). The water-steam mixture passes through the core

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* Zircaloy is an alloy of zirconium, a metal which is corrosion resistant in high temperature water and has strength characteristics between steel and aluminum.

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discharge plenum and into the steam separator assembly, which consists of a base into which is welded an array of standpipes. There is a steam separator at the top of each standpipe. Figure 1E schematically shows a separator. Like all cyclone-type separators, the "turning" vanes are fixed and impart a swirling motion to the fluid which enhances the separation by centrifugal action. From the steam separator assembly, the steam passes through another plenum to the steam dryer assembly (Figure 1D) which is composed of corrugated baffles that farther separate the entrained water. The design exit conditions from the dryer assembly are about 1020 psia, 545°F saturated steam with 0.3 percent moisture content.

The heated returning feedwater is introduced through a sparger assembly which encircles the steam separator and standpipe assembly. The feedwater mixes with the water from the separators and about 70 percent passes to the suction side of the jet pumps.

The jet pumps require a driving flow which is provided by the <u>recirculation system</u>, basically two closed loops, each with an electric motor-driven pump and pump isolation valves. The core coolant, feedwater, steam, and recirculation water are all the same fluid.

The water passes downward, some going through and some passing around the jet pumps. That which passes around the jet pumps is the suction water for the recirculation pumps. That which goes through the jet pumps passes into the core inlet plenum where the flow direction is reversed and the cycle starts over again through the fuel assemblies.

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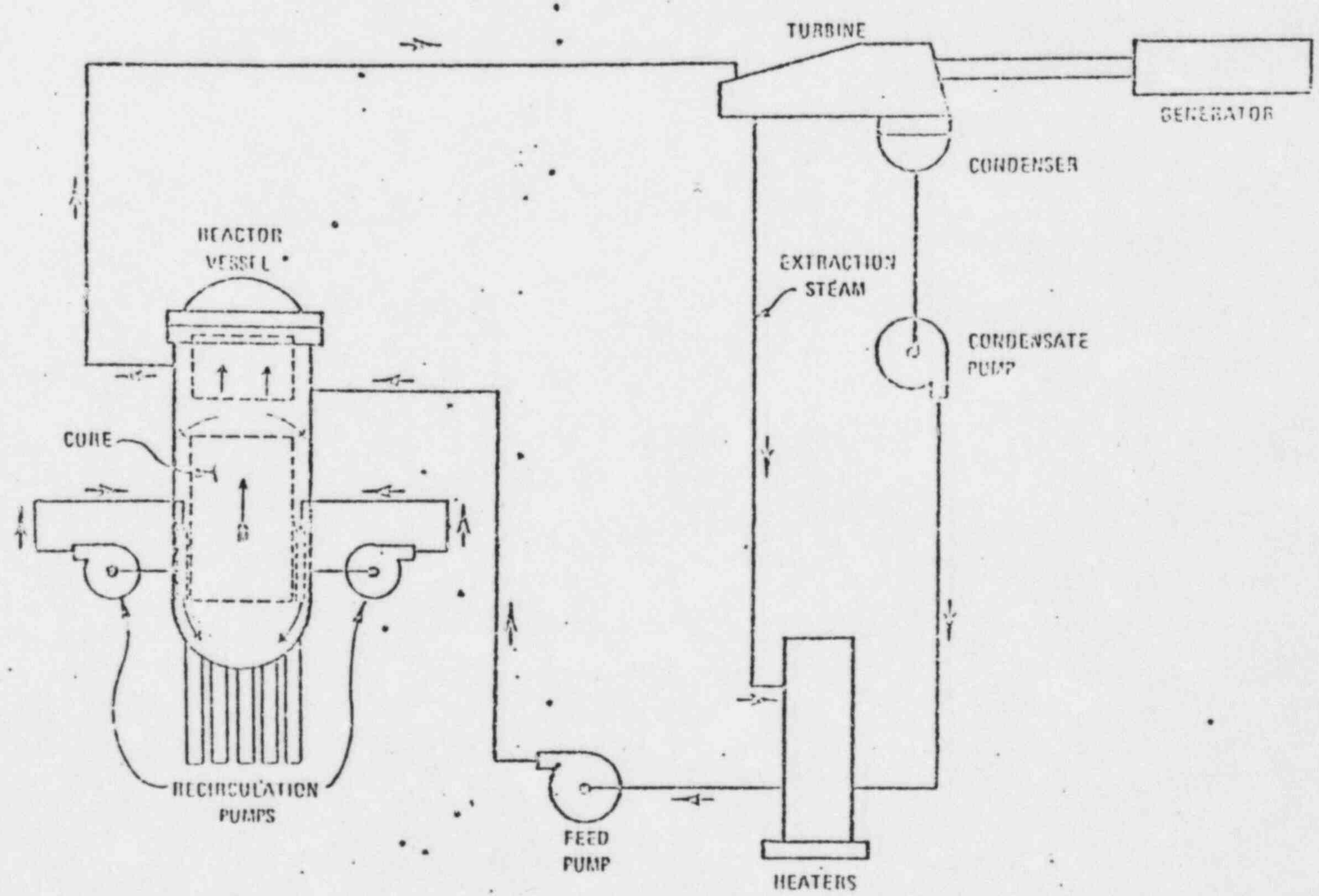

FIGURE 1A BWR REACTOR SYSTEM

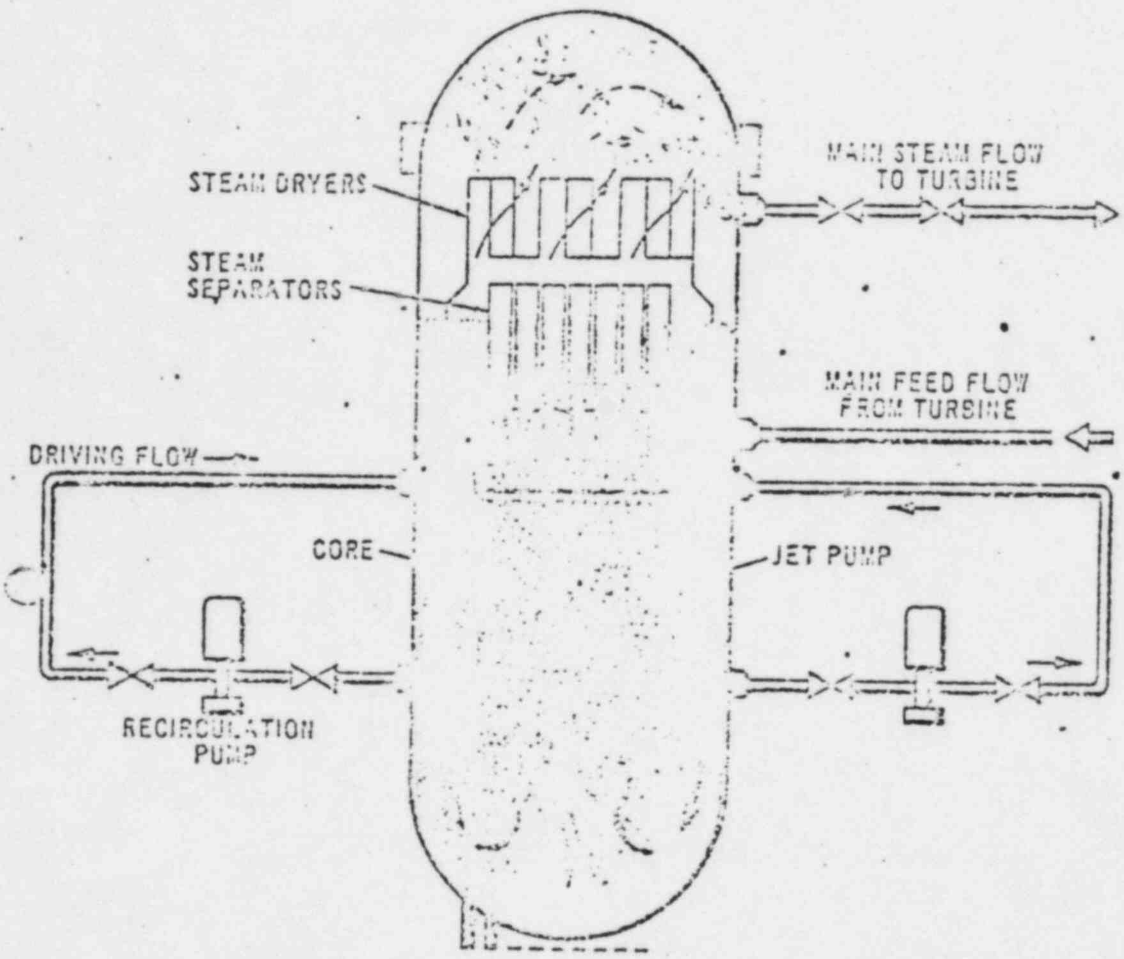

FIGURE 1B
STEAM AND RECIRCULATION WATER FLOW PATHS

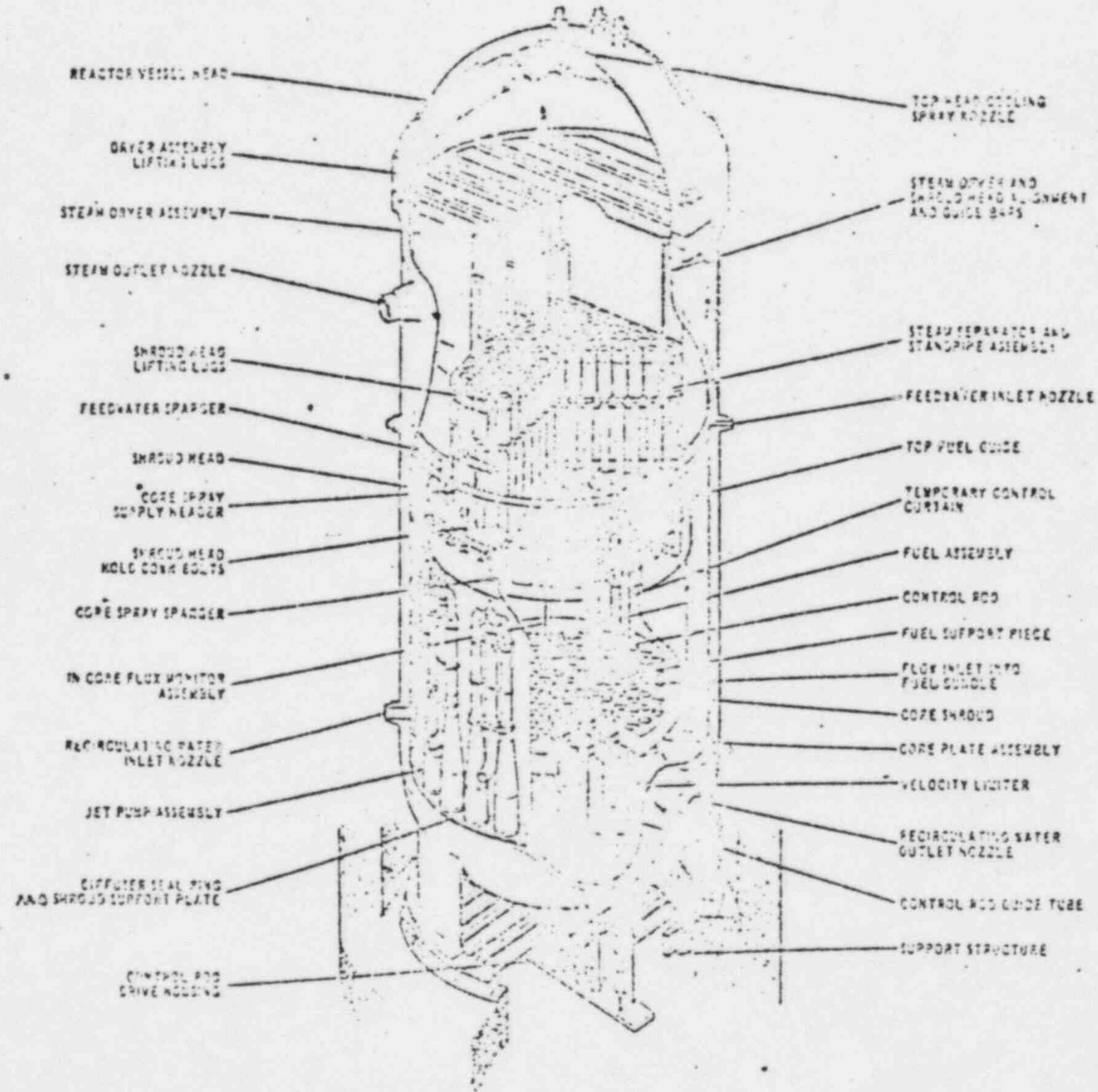

FIGURE 1C
REACTOR VESSEL CUTAWAY

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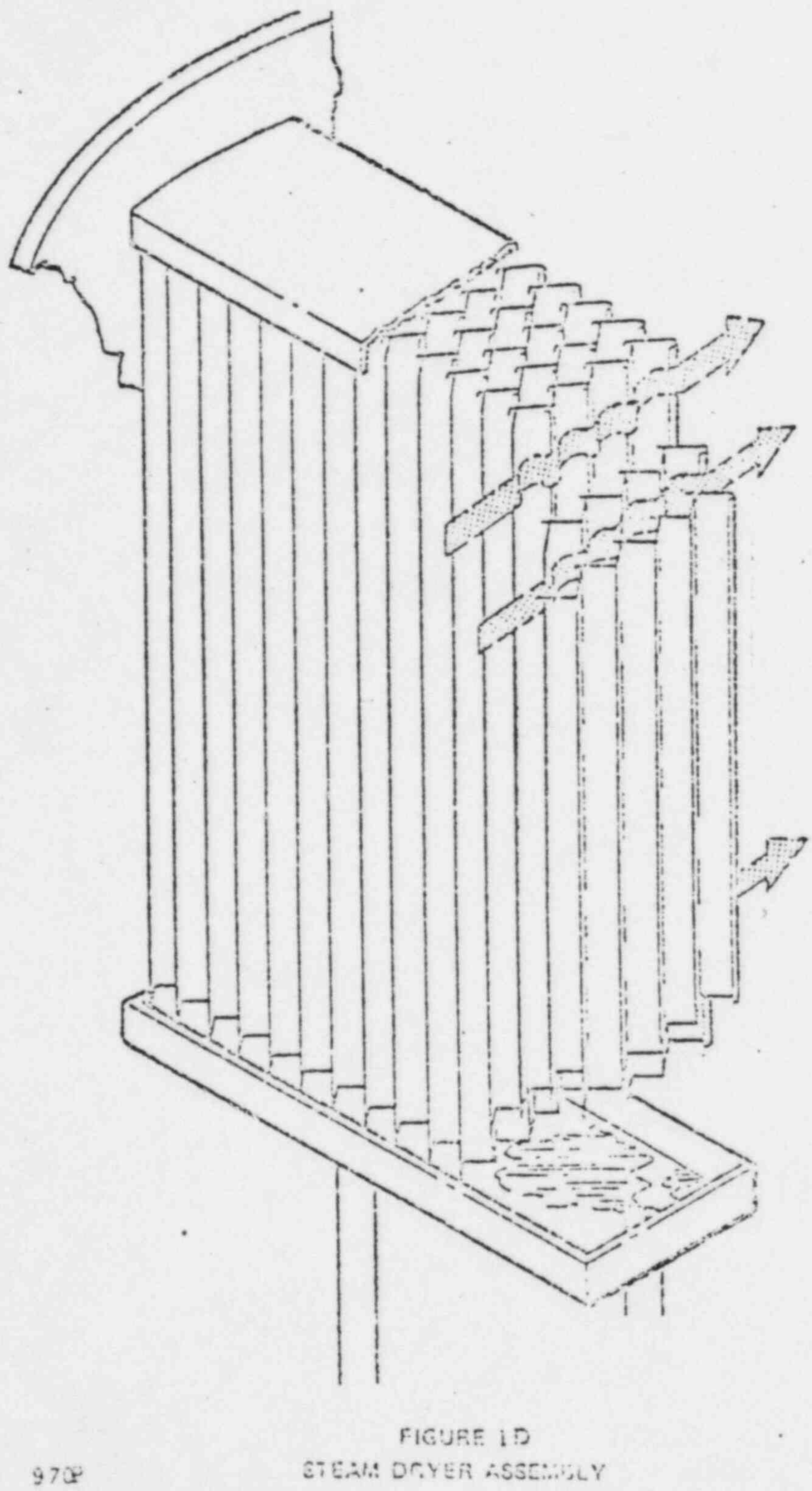

FIGURE 1D
STEAM DRYER ASSEMBLY

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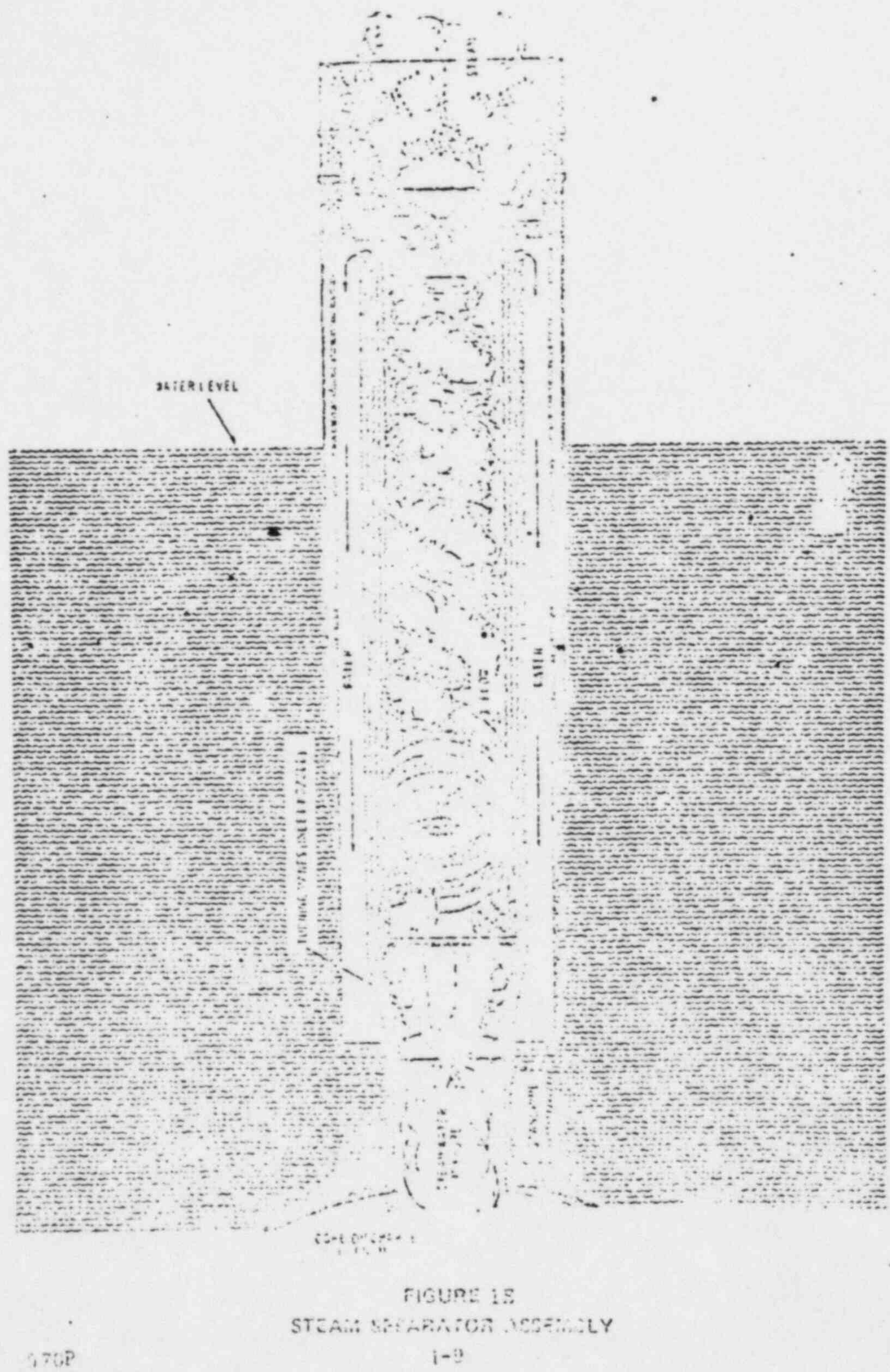

FIGURE 15
STEAM SEPARATOR ASSEMBLY

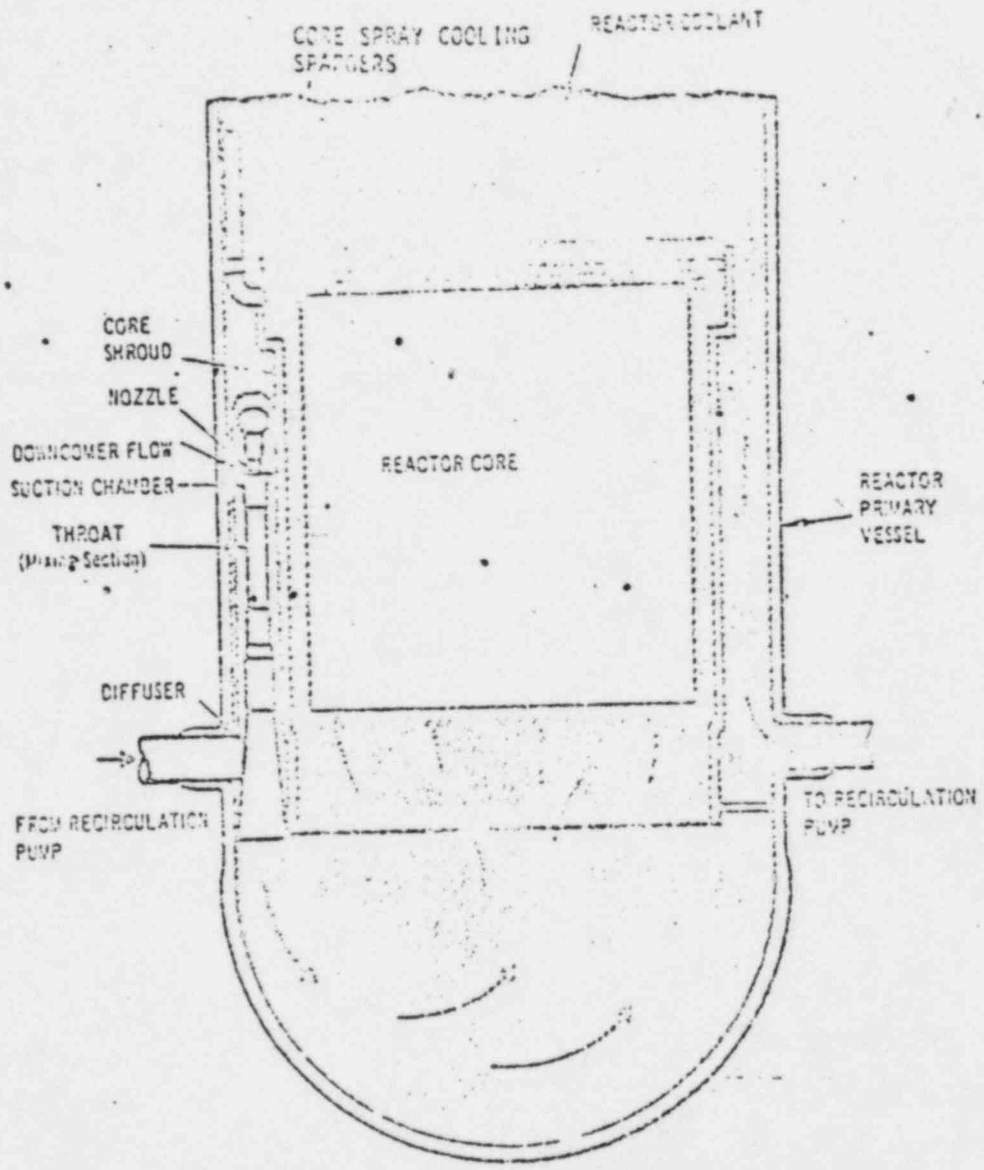

FIGURE 1F
RPV - INTERNAL LOOP

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FIGURE 1G
TYPICAL CORE ARRANGEMENT

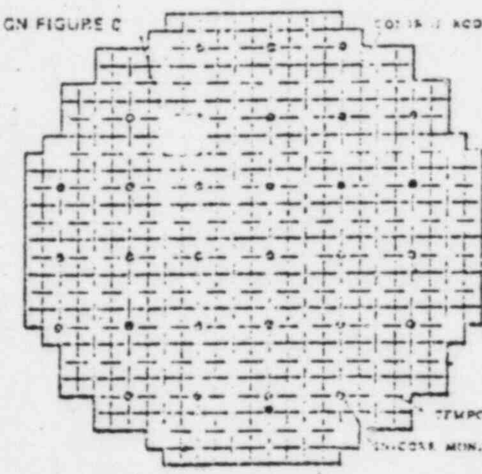
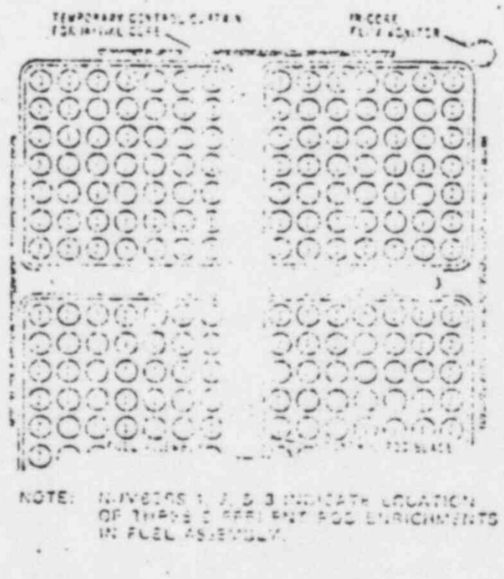


FIGURE 1H
CORE LATTICE



NOTE: NUMBERS 1, 2, & 3 INDICATE LOCATION OF THREE DIFFERENT ROD ENRICHMENTS IN FUEL ASSEMBLY.

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C. PWR Plant Review

The PWR system is shown in Figure 11. It differs from the BWR system in the following important ways:

- No steam is generated in the reactor vessel; the water which cools the core remains liquid at all times.
- A steam generator is provided in which heat from the water that cools the core is used to create steam in a "secondary" system; it is this steam which flows through the turbine.
- A pressurizer is attached to the core cooling system to maintain the proper pressure.

The core coolant system (often called the "primary") usually consists of three closed loops connected to the reactor vessel. As shown in Figure 11, each loop contains a steam generator, a pump, and piping. A pressurizer is connected to the hot and cold legs of one or more of the loops in order to maintain the pressure at about 2000 psia. This high pressure is needed to prevent boiling in the primary water. Up to 2500 ppm of boron is sometimes added to the primary water to aid in the nuclear control of the plant, but this does not affect its flow or heat transfer properties.

The reactor vessel, shown in Figure 11, is a large steel cylinder with a hemispherical bottom, and a gasketed and flanged upper head. The vessel contains:

- the core, which includes the fuel and the neutron absorber rods used for control;
- baffles to regulate the flow distribution and pressure drop above and below the core.

In addition, the vessel contains other items not directly related to thermal performance, such as shielding, instrumentation, irradiation specimens, support structure for all the internals, and a safety injection system for use in certain possible but improbable accidents.

Coolant enters the reactor vessel through three nozzles in its side and flows downward through the annular space between the vessel wall and the core barrel into a plenum at the bottom of the vessel, where it reverses direction. After passing upward through the reactor core, the coolant flows out of the vessel through three exit nozzles located on the same level as the inlet nozzles.

The core is composed of many fuel assemblies positioned parallel to each other in a gridwork. Each fuel assembly, shown in Figure 1K, is made up of a large number of fuel rods which are vertical sealed tubes of zircaloy, about one-half inch in diameter, containing a stack of pellets of uranium dioxide (UO_2). The UO_2 is a ceramic material with a very high melting point (about 5000°F). A spring and insulating disc in the upper part of the fuel rod holds the pellets down, but allows them to expand as they heat up. The upper part of the fuel rod is a void that serves as a collection point for gases released by the fuel. Heat is generated in the UO_2 pellets and most of it passes radially outward through the thin zircaloy fuel cladding. The maximum temperature at the outer surface of the cladding is only about 650°F, but the temperature at the center of the UO_2 can be as high as 4000°F.

Water flows upward through the fuel assemblies, removing the heat from the cladding as it passes. PWR assemblies, unlike BWR assemblies, are not surrounded by any shroud or can, and so a certain amount of crossflow can occur, with cooling water flowing out of one assembly and into its

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neighbors beside it. Several percent of the total flow consists of by-pass flow, which comes up between the outside of the core and the thermal shield or core barrel, and is, therefore, not directly available for removing heat from the core. Typically, water enters the bottom of the core at a temperature of about 540°F, and leaves the top of the core at about 600°F.

The primary water is driven through the core and the steam generators by centrifugal pumps in the cold leg of each primary loop. Each pump uses about a 5000 horsepower motor to circulate about 80,000 gpm of water.

Babcock and Wilcox supplies steam generators that produce superheated steam. This once-through steam generator, shown in Figure 1L, is a tube-in-shell-type, with primary water flowing through the tubes at primary pressures in excess of 2000 psia and secondary water on the shell side at about 1000 psia. The reactor coolant enters at the top and exits through two nozzles at the bottom. Secondary feedwater enters the steam generator through numerous nozzles. It is sprayed into the downcomer (also known as the economizer section), and it is preheated in the section by mixing with ste[illegible] which passes through small nozzles in the lower baffle plate. The feedwater then passes under the lower baffle plate and starts to travel up the tube bundle, where steam is formed. The upper portion of the tube bundle is used to superheat the steam before it exits to the turbine. Circulation of the feedwater through the steam generator and its conversion to steam follows the process of coolant flow through a BWR very closely, except that a BWR does not produce superheated steam. The other two PWR vendors provide vertical U-tube steam generators that produce saturated steam at the exit.

The pressurizer maintains the required reactor coolant pressure during steady-state operation, limits the pressure changes caused by coolant thermal expansion and contraction during normal load transients, and prevents the pressure in the Reactor Coolant System from exceeding the design pressure. Under normal full power operating conditions, water fills about one half of the pressurizer. The pressurizer vessel contains replaceable direct immersion heaters and is equipped with multiple safety and relief valves, a spray nozzle, interconnecting piping, valves and instrumentation. The electric heaters, located in the lower section of the vessel, are capable of raising the temperature of the pressurizer and its contents at the desired rate during startup of the reactor plant.

The pressurizer is designed to accommodate positive and negative surges caused by load transients. During a positive surge (caused by a decrease in plant load), power operated spray valves admit water from the cold leg of a coolant loop to condense steam in the pressurizer so that the pressure will remain below the value that would actuate the power operated relief valves. In addition, the spray valves can be operated manually by a switch in the central control room. A small continuous spray flow prevents excessive cooling of the spray piping. The resultant recirculation of reactor coolant through the pressurizer minimizes the difference between reactor coolant loop and pressurizer boron concentrations. During a negative surge (caused by an increase in plant load), flashing of water to steam and generation of steam by automatic actuation of heaters keep the pressure above the minimum allowable limit. Heaters are also energized on high water level during positive surges to heat the subcooled surge water entering the pressurizer from the reactor coolant loop.

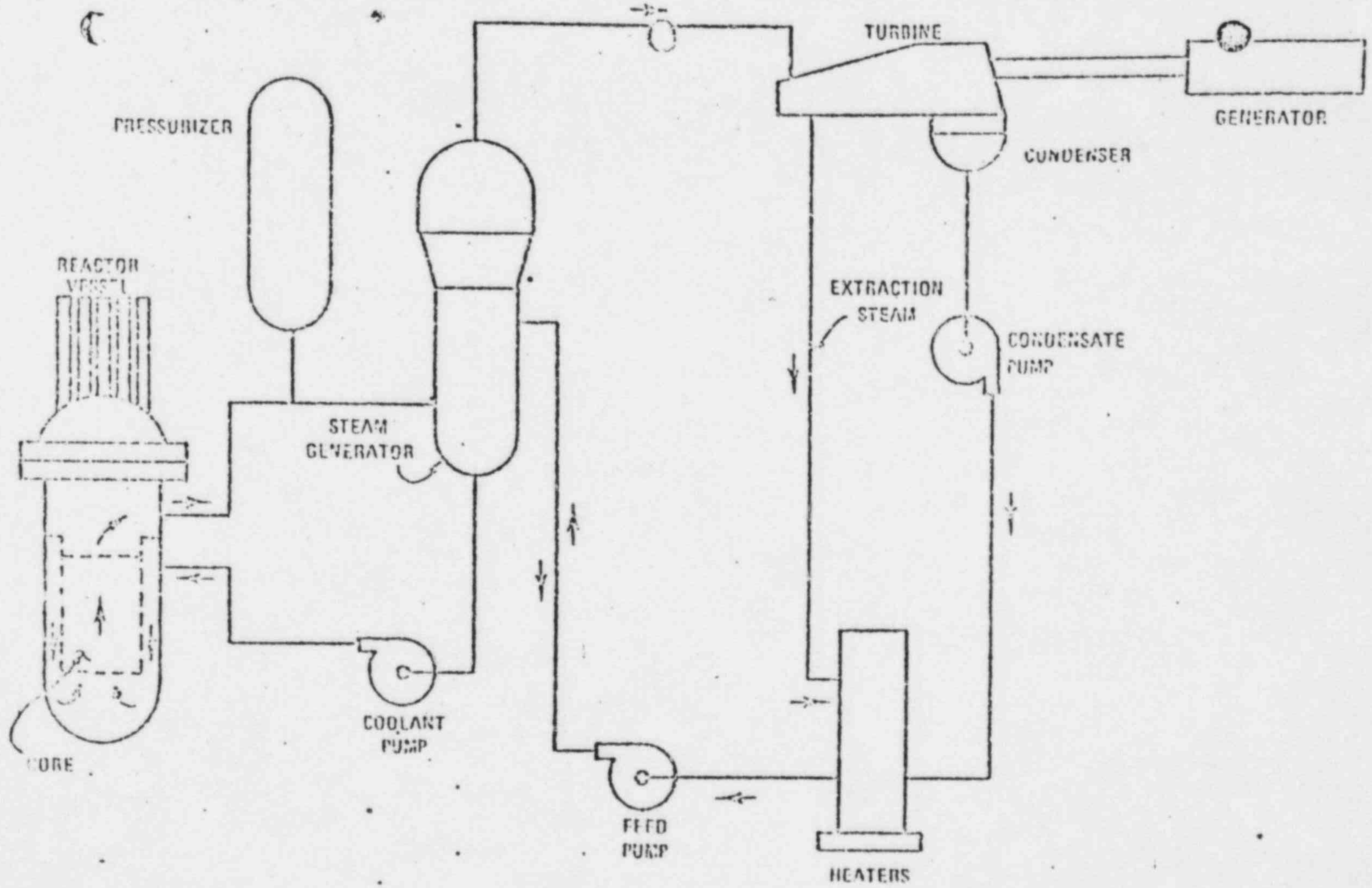

FIGURE II PWR REACTOR SYSTEM

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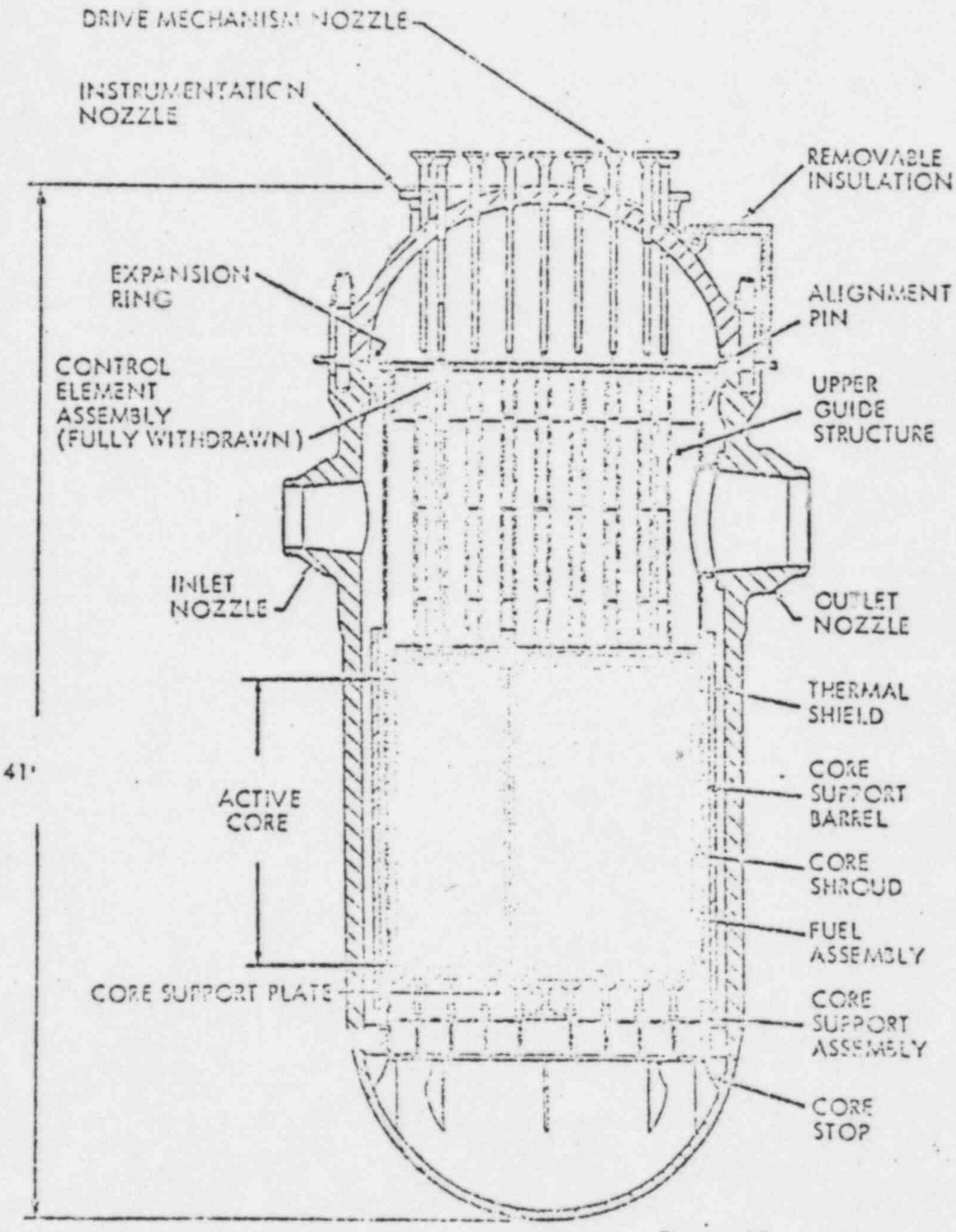

Figure 17

PWR REACTOR ARRANGEMENT

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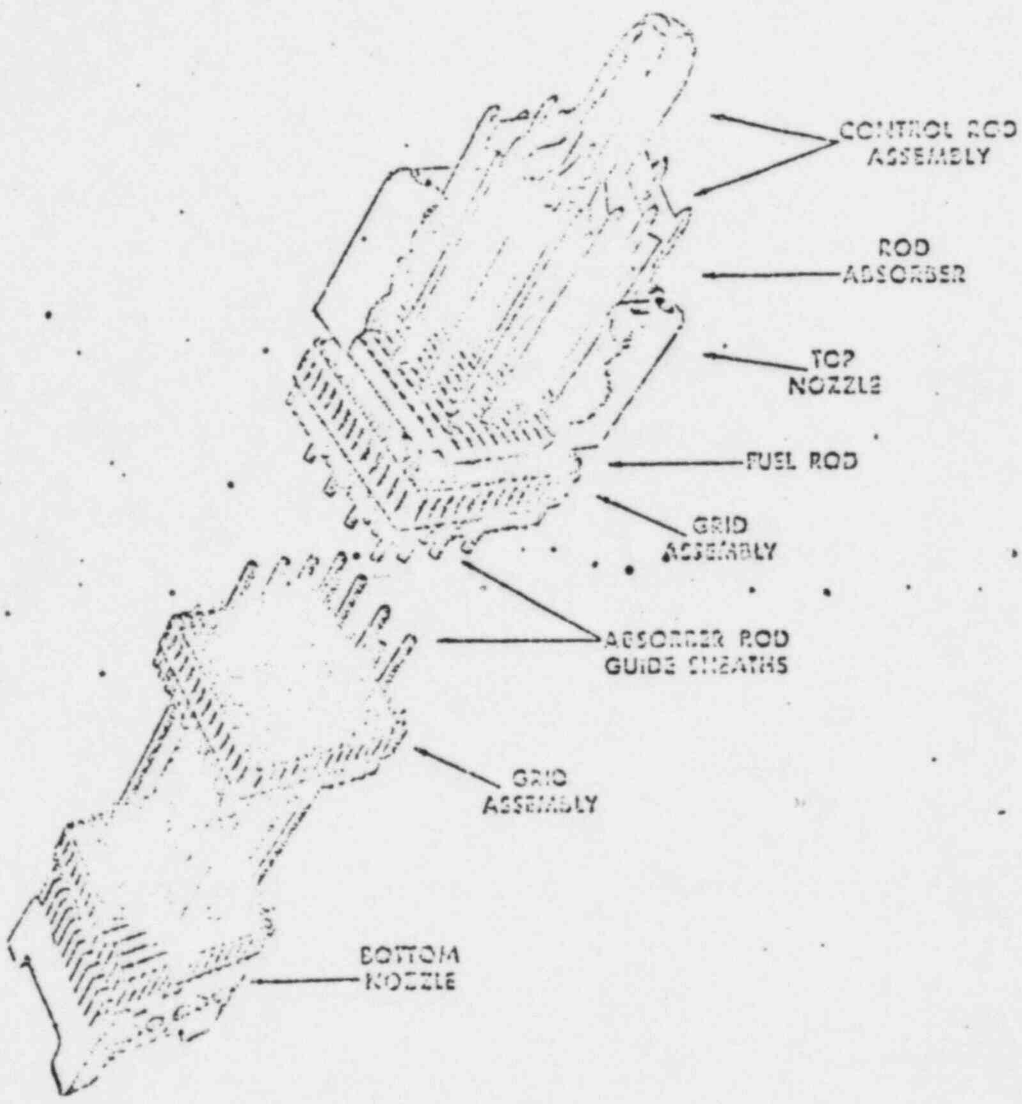

Figure 1K

PWR FUEL ASSEMBLY

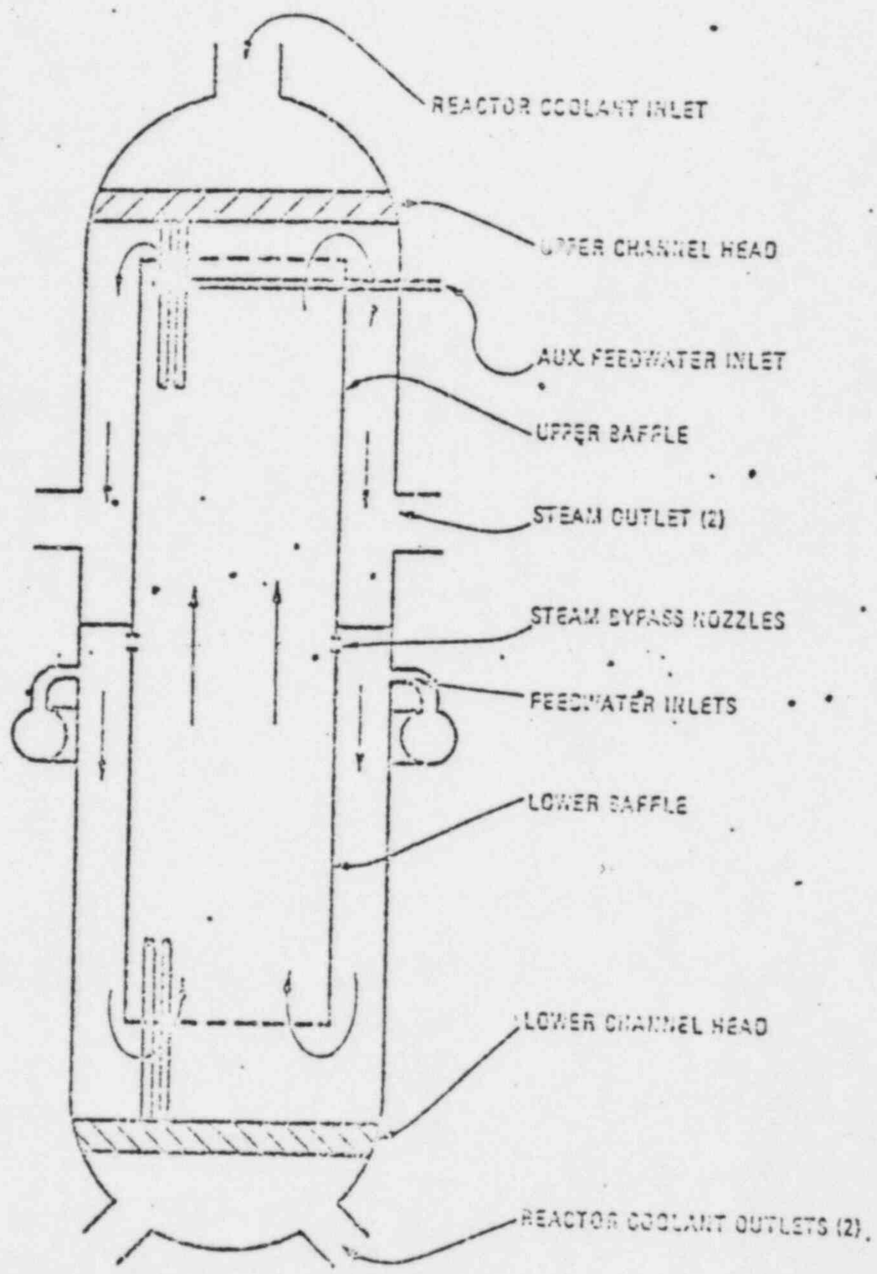

FIGURE 1L
SINGLE PASS SUPERHEAT STEAM GENERATOR

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D. Technical Specifications

In the interest of public safety, the AEC requires each nuclear power plant to submit a set of "Technical Specifications." The object of the Technical Specifications (Tech Specs, for short) is to prevent an uncontrolled release of radioactivity by limiting the manner in which you run the plant. The Tech Specs are a legal document and the AEC can shut down your plant if you violate them, so reactor operators and supervisors must be careful not to do anything which would violate them. You will not be responsible for verifying that your operating procedures meet Tech Spec limitations (this is done by management and by the reactor engineer), but you should know what the limits are, appreciate why they exist and understand both the engineering and legal consequences of violating them.

The actual Tech Specs consist of five kinds of rules:

(1) Safety limits and limiting safety system settings. (i) Safety limits are limits upon important process variables (such as power, flow rate, etc.) which are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor shall be shut down until the Commission authorizes resumption of operation. (ii) Limiting safety system settings are settings for automatic protective devices (such as trips, emergency cooling systems, etc.) related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be so chosen that automatic protective action will correct the most severe abnormal situation anticipated before a safety limit is exceeded. If, during operation, the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down

the reactor. He shall notify the Commission, review the matter and record the results of the review, including the basis for corrective measures taken.

(2) Limiting conditions for operation. Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met. The licensee shall notify the Commission, review the matter and record the results of the review, including the basis for corrective measures taken.

(3) Surveillance requirements. Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within the safety limits, and that the limiting conditions of operation will be met.

(4) Design features. Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in subparagraphs (1), (2), and (3) of this paragraph.

(5) Administrative controls. Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

We are concerned mostly with rule 1, relating to Safety limits and limiting safety system settings. Figure 1M shows the relationship between these and other settings. The safety limit is the level above which damage to the fuel may occur; below it damage will definitely not occur. The limiting safety system setting is the maximum point at which the scram or other automatic action may be set to occur. The space between these two levels is a safety margin. Normally, the scram is set below the limiting safety system setting and an alarm is set below the scram. The reason for having an alarm and for setting the scram point below the limiting value is to avoid exceeding the limiting value under any circumstances, since the rules (Item 1 above) specify that "if any safety limit is exceeded, the reactor shall be shut down."

The exact values of the safety limits are based on calculations summarized in the Safety Analysis Report. The object of this course is to teach you the safety significance of these limits.

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Problem Set - Chapter 1

1. What do the letters BWR and PWR stand for?

2. What components of a BWR plant are important to its core performance?

3. What components of a PWR plant are important to its core performance?

4. What is the object of having Technical Specifications?

2. THERMODYNAMICS AND HEAT BALANCES

In Chapter I, several terms were mentioned and it was indicated that we would give you a working feel as to what they meant. The terms that we'll examine are those used in defining thermodynamic properties and concepts.

A. Some Basic Definitions

(1) Thermodynamics is a word we have used quite often. It is made up of the word thermos which is of Greek origin and means heat and the word dynamics which defines something that is in motion. Literally then, thermodynamics means heat in motion. Put into more technical phraseology, "thermodynamics mean the processes involved in the transfer of energy." Other course material presented in Basic Nuclear Physics was concerned with the examination of the atom and its particles. Thermodynamics is of no value in predicting the behavior of atoms or molecules. It is useful only when applied to large amounts of matter. This will result in our examining processes using the black box approach, without really caring what goes on inside the box, but only what crosses its borders. Again, in Basic Nuclear Physics you were presented with certain laws used by the physicist. They were called the Laws of Conservation of Energy and Matter. Thermodynamics has laws also and they are very similar, but are restated for use by engineers. It is not necessary for us to explore them in any detail, since they are quite complicated. But rather than dismiss them, let's attempt a very loose translation, recognizing that we will be sacrificing accuracy. On this basis we can write the two laws as follows:

- Law #1 - You can't get something for nothing
- Law #2 - Not only that (Law #1), you can never break even, you always lose.

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Let's consider an idle power plant. If you close the generator breaker, you won't get anything out to the bus until you start the turbine. The turbine won't roll until you supply it with steam. You can't make steam until you have a source of heat and for that you need fuel. No fuel in, no power out. Let's assume you supply the plant with fuel and start it up. Now you are supplying power to the bus, but when you consider what's going in to what's coming out, you find there is a large discrepancy. You are putting much more energy in than you are getting out. The second law is affecting you. That's a glimpse of what thermodynamics is all about. We won't be mentioning "thermo" too much after this but remember, anytime you deal with the transfer of energy, you're working with thermodynamic processes.

(2) <u>Properties</u> was used previously and can be defined as those quantities necessary to describe the state of a substance. By state we mean how hot or cold it is, how dense, under what pressure, and how much volume does it occupy. Temperature, pressure, density, enthalpy, viscosity, and specific volume are all properties. Let's take an example to illustrate the use of properties to describe the state of a substance. The state of superheated steam can be established by specifying its pressure and temperature. When this is done, other properties of the steam, such as specific volume, are no longer free variables. Instead, they have a definite and fixed value. In that respect, the state of superheated steam could have been established by specifying the specific volume and the temperature. Now pressure becomes the fixed variable. It normally requires the establishment of two properties to define the state of a substance. Of course, the substance itself, such as superheated steam, must also be identified. It should be noted that the total volume or weight of a material is not necessary to specify its state.

(3) Pressure is an easy term to define. It's a force per unit area. The only thing to watch out for is the units. Engineers always work with absolute units such as pounds per square inch, absolute (PSIA) or inches of mercury, absolute (in.Hg abs.). Engineering tables listing properties as a function of pressure are indexed in absolute units. The units of pressure that you will obtain from the plant are read in psi gage (psig) or inches of mercury, vacuum (in. Hg vac.). Remember to convert your instrument readings to absolute units before entering engineering tables. How do you convert? For pressure readings from a gage (psig), add 15 to get absolute pressure psia.

- PSIA $\cong$ Pressure gage reading +15

To change inches of mercury, vacuum to absolute pressure, use the following formula and sketch:

(Vacuum)		(Absolute)
0 in. Hg	15 psia = 0 psig	
15 in.Hg	7.5 psia	
30 in. Hg	0 psia	

- PSIA $\cong$ 1/2 (30 - in.Hg, vac.)

If condenser pressure is normally 28.5 in.Hg, vac., the absolute pressure is:

$$P \cong 1/2\ (30 - 28.5)$$
$$P \cong 1/2\ (1.5)$$
$$P \cong 0.75 \text{ psia}$$

(4) Temperature is a measure of how much hotter or colder one body is when compared with others. It can also be thought of as the potential or driving force that causes heat to flow from one body to another. Fortunately, the unit of temperature measurement used almost universally throughout your plant, degrees Fahrenheit (^{o}F), is also the unit we'll use in a majority of our calculations. For the special case of determining theoretical efficiencies, we will use absolute temperature measured in ^{o}R (Rankine). In the General Science course, you may recall the following relationship:

- $^{o}R = {}^{o}F + 460$

Since your earlier coverage of temperature and its various scales was very complete, we'll not explore it any further.

(5) Specific Heat is the heat required to raise the temperature of one pound of material $1^{o}F$. The standard unit of heat is the British Thermal Unit. It is the amount of heat required to raise the temperature of one pound of water by $1^{o}F$. By definition, the specific heat of water is 1. All other specific heats are relative to this. The units for specific heat are $BTU/LB/^{o}F$ and it is represented by the symbol c. A typical application of specific heat can be found in this problem:

- How much heat must be supplied to 100 lb of a substance with a specific heat of 0.5 $BTU/LB/^{o}F$ to change its temperature from $90^{o}F$ to $212^{o}F$.

  Heat = lb x specific heat x temperature change

  Heat = 100 x 0.5 x (212 - 90)

  Heat = 50 x 122

  Heat = 6100 BTU

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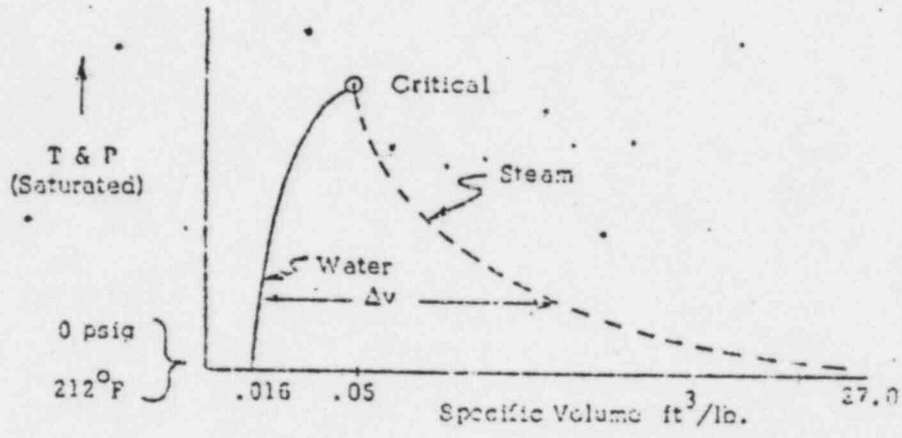
If we take the first two terms of the previous equation (lb and specific heat) and multiply them together we have a new term called heat capacity. Its units are BTU/$^{\circ}$F and it is the amount of heat required to raise the temperature of a substance by 1°F. The heat capacity is dependent upon the weight of the substance. For that reason, heat capacity is not a property describing the state of the substance. What is the heat capacity of your reactor vessel if it weighs 550 tons and has a c of 0.1 BTU/LB/$^{\circ}$F?

(6) Specific Volume is a measure of the number of cubic feet occupied by one pound of a material. It is the reciprocal of density and is represented by the letter v. As a material becomes more dense, its specific volume decreases. We will be interested primarily in the specific volume of saturated water and steam and how it changes as pressure and temperature are increased. We also want to know how specific volume changes when water is turned into steam. One useful aid to help us in remembering how specific volume changes is the fact that at a high temperature (705°F) and pressure (3206 psia), water and steam have the same properties. At this critical point, as it is called, the specific volumes of each are equal. At low pressures and temperature we know from experience that one pound of water does not occupy a large volume; therefore, its specific volume is small. As we add heat to the water, it expands and occupies more volume. As we put the water under pressure, what happens? Water is essentially incompressible so there is no change in volume due to increasing pressure. As the temperature and pressure of water are increased, the specific volume increases due to the temperature effect. Steam at low pressure and temperature has a large specific volume. That's why

the low pressure turbine exhaust casings are so large. As temperature and pressure are increased, the pressure becomes the dominant factor because steam is compressible.

So the specific volume of steam decreases with increasing temperature and pressure, just the opposite of water. On a graph, specific volume of water and steam looks like this:



The change in specific volume (shown as Δv) associated with the change in state as water becomes steam, or the reverse, can be determined from the graph. Δv decreases as temperature and pressure increase, the same behavior as that of the steam. The specific volume of steam is represented by v_g (g for gas), that of water by v_f (f for fluid) and Δv is v_{fg}.

(7) <u>Enthalpy</u> is a measure of the contained thermal energy in a substance. It is a relative property describing the state of a material in relation to a reference value. This reference is water at

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$32^{\circ}F$ and, at this condition, enthalpy is zero. We can write an equation for enthalpy as follows:

$$\text{enthalpy} = \text{specific heat} \times \text{temperature above } 32^{\circ}F$$
$$h = c \times (T-32)$$

Checking the units:

$$h = BTU/LB/^{\circ}F \times {}^{\circ}F$$
$$h = BTU/LB$$

- What is the enthalpy of water at $212^{\circ}F$?

$$h = 1 \times (212 - 32)$$
$$h = 180 \text{ BTU/LB}$$

If we look up the value of h for water at $212^{\circ}F$ in a water properties table, we'll find it is listed as 180.08 BTU/LB. The reason for this discrepancy is that the value of c, specific heat, is not precisely equal to 1.0 at all temperatures. However, for most of the problems we'll be solving, assuming a c = 1 for water is accurate enough up to a pressure of 1000 psia and $545^{\circ}F$. As in the case of specific volume, the subscripts f and g will be used in identifying the enthalpy of water and steam respectively.

(8) Latent Heat of Vaporization is the amount of energy needed to change a pound of water into a pound of steam with no change in temperature. The word latent means "hidden" and is used because there is no temperature change during the process. We use the symbol h_{fg} and the unit is BTU/LB. At atmospheric pressure h_{fg} is 970 BTU/LB. At critical pressure it is 0, so it can be seen that it behaves in the same manner as v_{fg}, that is, as pressure and temperature are increased, h_{fg} decreases.

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B. Steam Tables

The Steam Tables are nothing more than a set of tables which list the numerical values of the properties introduced in the last section. The values in these tables have been obtained from measurements made by engineers, starting back before the year 1900 and continuing right up to now. The set of steam tables published by the American Society of Mechanical Engineers (ASME for short) is the standard which you should always use. Because the ASME set is a large book, various organizations have published short tables consisting of selected values taken from the ASME tables. The short set which we will be using includes three tables, the first two of which are for "saturated" steam and the last for "superheated" steam.

(1) Saturated Steam is steam which is ready to start condensing into water if thermal energy is removed from it. Saturated steam is dry--it has no liquid water contained in it, but if we decrease its energy slightly some liquid water will be formed by condensation.

(2) Saturated Water is water which is ready to start evaporating into steam if thermal energy is added to it. Saturated water is completely liquid, but if we increase its energy slightly some steam will be formed by evaporation.

We have to make two choices in order to read the steam tables. That is, we can choose the numerical values of two and only two of the following:

- pressure
- temperature
- specific volume
- enthalpy.

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The steam tables will then give us the numerical values of the other two properties. The pressure and temperature are the properties whose numerical values we most often choose, but there will be times when we use a different pair. Sometimes instead of choosing the numerical values of two properties we choose only one numerical value, but also specify that the steam is saturated. In effect, specifying that the steam is saturated uses up one of our choices and so we can only choose one numerical value in this case.

The steam tables distributed with these notes are a "short" set taken from the 1967 ASME Steam Tables. (This is the latest edition of the ASME tables.) Three tables are included. We will describe the tables and then give some examples of how to use them.

(3) Table 1 - Saturated Steam; Temperature Table

This table gives the properties of saturated steam and saturated liquid. Because the table is for saturated H_2O, one of our choices is already made for us, and we can choose the numerical value of only one other property. Table 1 is laid out on the assumption that you will choose the value of the temperature t. Therefore, temperature (in degrees Fahrenheit) is listed in the left-most column, and — for convenience — again in the right-most column. The temperature values increase by jumps of either 2°F or 4°F as we move down the column.

The corresponding [illegible] absolute pressure p in lbs. per square inch are listed in column [illegible] have already said that the steam is saturated, once we pick the [illegible] all the other values are determined by nature. Therefore, the pressure values are not round numbers and are not in convenient jumps of 2 or 4 like the temperatures are, because these pressure values are properties of nature.

For each temperature, column 3 gives the specific volume v_f of the saturated liquid, and column 5 gives the specific volume v_g of the saturated vapor (that is, of steam.) Column 4 gives v_{fg}, the amount by which the volume increases when the saturated liquid evaporates completely to form saturated steam. You will find that the value in column 5 is the value in column 3 added to the value in column 4. All the specific volumes are given in cubic feet per lb.

Column 6 gives the enthalpy h_f of the saturated liquid, and column 8 gives the enthalpy h_g of the saturated vapor (that is, of steam.) Column 7 gives the amount h_{fg} by which the enthalpy increases when the saturated liquid evaporates to form saturated steam. You will find that the value in column 6 plus the value in column 7 gives the value in column 8. All the enthalpies are given in BTU per lb.

We will not discuss the entropy (columns 9, 10 11) in this course.

(4) Table 2 - Saturated Steam; Pressure Table

Like Table 1, this table gives the properties of saturated steam and saturated liquid. Because the table is for saturated H_2O, one of our choices is already made for us, and we can choose the numerical value of only <u>one</u> other property. Table 2 is laid out on the assumption that you will choose the value of the pressure p. Therefore, pressure (in lbs. per square inch absolute) is listed in the left-most column, and — for convenience — again in the right-most column. The pressure values increase by jumps of either 5, 10, 50, or 100 lbs. per square inch, depending on what part of the table we look at.

The corresponding values of temperature t in degrees fahrenheit are listed in column 2. These are not round numbers and are not in convenient jumps, because once we specify that the H_2O is saturated, and what its pressure is, all other values are determined by nature.

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For each pressure, column 3 gives the specific volume v_f of the saturated liquid and column 5 gives the specific volume v_g of the saturated vapor (that is, of saturated steam.) Column 4 gives v_{fg}, the amount by which the specific volume increases when the saturated liquid evaporates completely to form saturated steam. You will find that the value in column 3 added to the value in column 4 gives the value in column 5. All the specific volumes are given in cubic feet per lb.

Column 6 gives the enthalpy h_f of the saturated liquid, and column 8 gives the enthalpy h_g of the saturated vapor (that is, of saturated steam.) Column 7 gives the amount h_{fg} by which the enthalpy increases when the saturated liquid evaporates to form saturated steam. You will find that the value in column 6 plus the value in column 7 gives the value in column 8. All the enthalpies are given in BTU per lb.

We need not discuss the entropy (columns 9, 10 and 11) in this course.

(5) Table 3 - Superheated Steam

This table gives the properties of superheated steam and also repeats the properties of saturated liquid and saturated steam presented in Tables 1 and 2.

As usual, we have to choose the numerical values of two properties in order to read the table. The table is laid out on the assumption that you will choose the numerical values of the pressure and the temperature. Therefore, values of the pressure are listed in the left-most column and values of temperature are listed across the top. The properties of superheated steam at the pressure given in the left hand column and the temperature given at the top are found where that column and that row intersect.

The top number given at this intersection is Sh, the degrees of superheat in degrees Fahrenheit. The second number is the specific volume v of the superheated steam in cubic feet per lb. The third number is the enthalpy h of the superheated steam in BTU per lb. We will not discuss the entropy values.

For your convenience, this table also repeats the saturation values listed in Tables 1 and 2. For each pressure listed in the left-most colume, the saturation temperature is given beneath it in parentheses; the properties of saturated liquid are given in column 2, and the properties of saturated steam in column 3. These properties are listed in the same order, and in the same units as for superheated steam.

(6) Example A

What are the pressure p, specific volume, and enthalpy h of saturated steam at 532°F?

In Table 1, look down the left column (the temperature column) until you find 532°F (in the middle of page 10.) Reading across this line gives:

Absolute pressure (p) = 900.34 psia

Specific volume of saturated vapor (v) = 0.50070 ft.^{3}/lb.

Enthalpy of saturated vapor (h_g) = 1196.4 BTU/lb.

(7) Example B

What are the temperature t, the specific volume v_g, and the enthalpy h_g of saturated steam at 900 psia?

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Since the pressure of saturated steam is given, use Table 2. Look down the left hand column until you find 900 psia (about one quarter of the way from the top of page 13.) Reading across this line gives:

Absolute temperature (t) = 531.91°F

Specific volume of saturated vapor = 0.50091 ft. 3/lb.

Enthalpy of saturated vapor = 1196.4 BTU/lb.

(3) Example C

Solve example 2 using Table 3.

Look down the left-most column of Table 3 until you find 900 psia (at the bottom of page 21.) The saturation temperature is given just below the 900, and its value is 531.95°F. The other values for the saturated vapor are given in column 3:

Specific volume of saturated vapor = 0.5009 ft.^{3}/lb.

Enthalpy of saturated vapor = 1196.4 BTU/lb.

Notice that the steam properties are about the same for all three examples. This is because the value of the pressure that we chose to use in examples B and C is nearly the same as the pressure value that we obtained from the table in example 1. This shows that:

- The 3 tables give consistent values of saturation properties.
- For saturated steam, it is more convenient to use Table 1 if the temperature is given, and to use Table 2 or Table 3 if the pressure is given.

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C. Energy, Work and Power

By using the steam tables, we can calculate the amount of thermal (i.e., heat) energy in steam. But a power plant uses not only thermal energy, it also uses mechanical energy, and it changes energy from thermal to mechanical form (in the turbine, for example).

Mechanical energy is called "work" and is the energy used in moving a piece of hardware over some distance. The piece of hardware can be a cylinder in your automobile engine (which moves only a few inches), or it can be your car itself (which you might push several hundred feet to the garage if the car broke down), or it can be a turbine rotor (which is continuously "pushed" around by expanding steam).

The amount of work we do in moving the piece of hardware equals the distance the hardware travels times the average force we use to push it:

WORK = FORCE X DISTANCE

This agrees with our common, everyday idea of what work is: the farther we push our car, for example, the more work we do: and the harder we have to push to move it, the more work we do.

The usual unit we use to measure work is the ft-lb. The first half of this unit — the ft — is the unit in which we measure the distance that we move something; the second half of this unit — the lb — is the unit in which we measure the average force we have to exert to move it.

(1) Example A

Your car breaks down and you have to push it 100 ft to the garage. To do this, you must push with a force of 50 lb (it's a very small car). How much work did you do?

o Answer: WORK = FORCE X DISTANCE

Force = 50 lb

Distance = 100 ft

Work = 100 ft x 50 lb = 5000 ft-lb

The dimensions of work and energy are the same, but since thermal energy is expressed in Btu and work in ft-lb, a conversion factor is needed. It is

778 ft-lb = 1 Btu

(2) Example B

We showed in an earlier example that the amount of thermal energy needed to increase the temperature of a typical reactor vessel by $1^{o}F$ was 110,000 Btu. How many foot-lbs is this?

o Answer:

$$110{,}000 \frac{Btu}{^{o}F} \times 778 \frac{ft\text{-}lbs}{Btu} = 85.58 \times 10^{6} \frac{ft\text{-}lb}{^{o}F}$$

Did you notice that we have so far not said anything about the time required to move the hardware? Time enters only when we speak of power.

Power is an expression of the rate at which either:

- thermal energy is produced; or
- work (mechanical energy) is used or produced.

Power has the units of energy divided by time:

$$Power = \frac{Energy}{Time}$$

Thermal energy is usually measured in Btu, so that we measure thermal power as:

$$\text{Thermal power} = \frac{\text{Thermal Energy}}{\text{Time}} = \frac{\text{Btu}}{\text{hr}}$$

Mechanical energy is measured in ft-lbs, so that we measure mechanical power as:

$$\text{Mechanical Power} = \frac{\text{Mechanical Energy}}{\text{Time}} = \frac{\text{ft-lb}}{\text{sec}}$$

Not only are the thermal and mechanical energy units different here, but the units of time are also different in the two cases, since we used hours in speaking of thermal power and seconds in speaking of mechanical power. This difference in the units of time, however, is simply a matter of common practice. We could have used any time units.

(3) Example C

If a heat source supplies 10^6 Btu for every 10 hours of operation, the thermal power rating is

$$10^6 \text{ Btu}/10 \text{ hours} = 10^5 \text{ Btu/hr}$$

(4) Example D

If a turbine can deliver 10^6 ft-lb of mechanical energy in 10 hours, the turbine rating is

$$10^6 \text{ ft-lb}/10 \text{ hours} = 10^5 \text{ ft-lb/hr}$$

Other more common units of power are horsepower (hp), for mechanical power, and kilowatts (kw) and megawatts (Mw) for thermal power. Electric

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power is also quoted in megawatts. Obviously, we need some conversion factors to help us convert from one unit of power to another. Here is a useful set:

$$1 \text{ hp} = 550 \text{ ft-lb/sec} = 0.75 \text{ kw}$$

$$1 \text{ kw} = 3413 \text{ Btu/hr} = 1.34 \text{ hp}$$

$$1 \text{ Mw} = 10^3 \text{ kw}$$

(5) Example E

We found in Example B of this Section that it required $85.58 \times 10^6 \frac{\text{ft-lb}}{^\circ\text{F}}$ to raise the temperature of the reactor vessel considered there. We want to raise the temperature $\frac{100^\circ\text{F}}{\text{hr}}$. What power does this require? Express the answer in hp, kw, and Mw.

c Answer:

$$P = 85.58 \times 10^6 \ \frac{\text{ft-lb}}{^\circ\text{F}} \times \frac{100^\circ\text{F}}{\text{hr}} = 85.58 \times 10^8 \ \frac{\text{ft-lb}}{\text{hr}}$$

$$P = 85.58 \times 10^8 \ \frac{\text{ft-lb}}{\text{hr}} \times \frac{1 \text{ hr}}{3600 \text{ sec}} = 2.38 \times 10^6 \ \frac{\text{ft-lb}}{\text{sec}}$$

$$P = 2.38 \times 10^6 \ \frac{\text{ft-lb}}{\text{sec}} \times \frac{1 \text{ hp}}{5500 \ \frac{\text{ft-lb}}{\text{sec}}} = 4.32 \times 10^3 \text{ hp}$$

$$P = 4.32 \times 10^3 \text{ hp} \times \frac{.75 \text{ kw}}{\text{hp}} = 3.24 \times 10^3 \text{ kw}$$

$$P = 3.24 \times 10^3 \text{ kw} \times \frac{1 \text{ Mw}}{10^3 \text{ kw}} = 3.24 \text{ Mw}$$

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D. Quality, Carryover, and Calorimeter

(1) Quality is the weight fraction of steam in a mixture of steam and water. If a mixture of Mm pounds exists such that Mg pounds of steam is involved, quality can be defined as follows:

$$\text{quality } (x) = \frac{\text{Steam mass}}{\text{Mixture mass}} = \frac{Mg}{Mm}$$

Since the steam mass equals the mixture mass minus the water mass (Mf), we can rewrite our expression as follows:

$$X = \frac{Mg}{Mm} = \frac{Mm - Mf}{Mm}$$

$$X = 1 - \frac{Mf}{Mm}$$

Quality is most often expressed in percent by multiplying the fraction by 100.

(2) Carryover is the weight fraction of water in a steam-water mixture. Using our same symbols we can write the following expression:

$$\text{carryover} = \frac{Mf}{Mm}$$

From a previous expression for quality, we can derive the following relationship between quality and carryover:

$$\text{quality} = 1 - \text{carryover}$$

or

$$\text{carryover} = 1 - \text{quality}$$

We use the term carryover primarily when we talk about the amount of moisture in the steam to the turbine. A typical PWR steam generator specification has a performance requirement that the moisture content (or carryover) of the steam leaving the unit shall not exceed 0.25%. That means no more than 1/4 pound of water for every 100 pounds of mixture.

a Example

If our steam generator is rated at 4×10^6 LB/HR at 0.25% moisture, how much water is leaving the unit at rated conditions?

$$Mf = \text{carryover} \times \text{mixture flow}$$
$$Mf = 0.25\% \times 4 \times 10^6$$
$$Mf = .0025 \times 4 \times 10^6$$
$$Mf = 10,000 \text{ LB/HR}$$

Suppose that we want to find out what the enthalpy is of a steam-water mixture leaving a steam generator. We know a couple of expressions that we can start with. For instance, we know that the total mixture is equal to the sum of its parts:

$$Mm = Mf + Mg$$
$$\text{Mixture (lb)} = \text{Water (lb)} + \text{Steam (lb)}$$

We also worked out a problem earlier where we found that :

$$\text{Heat} = \text{Mass} \times \text{Enthalpy}$$
$$\text{Heat} = M \times h$$

We can combine these expressions and write one that states that the heat of the mixture is equal to the sum of the heats of its parts:

$$Mm\,hm = Mg\,hg + Mf\,hf$$

Since we are looking for the enthalpy of the mixture, hm, it is necessary to divide this expression by Mm to eliminate this term on the left side of the equation.

$$hm = \frac{Mghg}{Mm} + \frac{Mfhf}{Mm}$$

We recognize the Mg/Mm is equal to the quality (x) and the Mf/Mm is equal to the carryover. Carryover can be written as 1 - quality. If we perform the appropriate substitutions, our equation becomes:

$$hm = xhg + (1-x)\ hf$$

- Example

Saturated steam at 1000 psia has a moisture content of 1/2%. What is the enthalpy of the mixture.

$$\text{Moisture content} = \text{carryover} = 1 - x$$
$$1/2\% = 1 - x$$
$$X = 1 - .005$$
$$X = .995$$

At 1000 psia hf = 542.6 BTU/LB
and hg = 1192.9 BTU/LB

$$hm = (.995 \times 1192.9) + (1 - .995)\ (542.6)$$
$$hm = 1186.94 + 2.71 = 1189.65$$

We can take our equation for hm and solve it for X, quality, so that should we know the enthalpy of the mixture, we can calculate the quality.

$$hm = xhg + (1 - x)\ hf$$
$$hm = xhg + (hf - xhf)$$
$$hm = hf + xhg - xhf = hf + x(hg - hf)$$
$$hm - hf = x(hg - hf)$$
$$X = \frac{hm - hf}{hg - hf}$$

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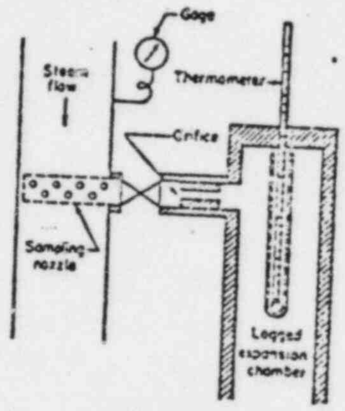
Since hg - hf = hfg we can write·

$$X = \frac{hm - hf}{hfg}$$

(3) Calorimeter. A device that allows us to measure the enthalpy of a mixture of steam and water is a calorimeter. Literally, calorimeter means "heat indicator". Its principle of operation can best be understood by examining a schematic of it.



A sample of the mixture is expanded in steady flow through a valve or other throttling device into a large volume. If the flow is smooth and not varying, this process is one in which the enthalpy (h) is constant. Thus, the expanded mixture becomes superheated at the lower pressure. (Note that we did not say the temperature increased.) Thus, we have

$$h_{exhaust} = h_{pipe} = h_{mixture}$$

As before, the quality inside the pipe can be expressed by

$$X = \frac{h_m - h_f}{h_{fg}}$$

and, therefore,

$$X = \frac{h_{exhaust} - h_f}{h_{fg}}$$

Assume the following conditions exist

pipe pressure = 950 psia

exhaust pressure = 14.7 psia

exhaust temperature = 250°F

From Steam Tables, h_f = 534.7 and h_{fg} = 660.0 BTU/LB

From Steam Tables, $h_{exhaust}$ = 1169.2 BTU/LB*

Plugging these values into the expression for X:

$$X = (1169.2 - 534.7)/660.0$$

$$= 634.5/660.0$$

$$= 0.9613$$

$$= 96.13\%$$

$$\text{Carryover} = 1 - x = 100 - 96.13 = 3.87\%$$

Some cautions should be observed about calorimeters. They are only useful if the moisture content is a few percent, because the mixture has to superheat when expanded. Also, the best accuracy that can be reasonably expected is ±0.5 percent quality.

*Remember, the exhaust steam (expanded mixture) is now superheated at the lower atmospheric pressure, so use the superheated steam tables (pg. 14)

E. Heat Balance

A heat balance could more correctly be called an energy balance, since it involves work as well as just heat. However, the term "heat balance" has been used for many years, so we will also use it in this book.

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A heat balance is an equation which says:

$$\left\{\begin{array}{l}\text{Energy we get}\\\text{out of a piece}\\\text{of hardware}\end{array}\right\} = \left\{\begin{array}{l}\text{Energy we}\\\text{put in}\end{array}\right\} + \left\{\begin{array}{l}\text{Energy we}\\\text{produce}\\\text{inside}\end{array}\right\} \quad (1)$$

The "piece of hardware" mentioned here can be either <u>one</u> piece (such as the reactor vessel) or it can be several joined together by pipes, canals, or electrical wires (such as the complete power station).

The heat balance expressed by equation (1) is useful because if we know any two of the items in it, we can use this equation to find the value of the third item. If we know the <u>energy we put in</u> and the <u>energy we produce inside</u>, we can use equation (1) to calculate the <u>energy we get out</u>. If we know the <u>energy we get out</u> and the energy we produce inside, we can find the <u>energy we put in</u> by re-arranging equation (1) to read:

$$\left\{\begin{array}{l}\text{Energy we get}\\\text{out of a piece}\\\text{of hardware}\end{array}\right\} - \left\{\begin{array}{l}\text{Energy we}\\\text{produce}\\\text{inside}\end{array}\right\} = \left\{\begin{array}{l}\text{Energy we}\\\text{put in}\end{array}\right\} \quad (2)$$

If we know the <u>energy we put in</u> and the <u>energy we get out</u>, we can determine the <u>energy we produce inside</u> by re-arranging equation (1) to read:

$$\left\{\begin{array}{l}\text{Energy we get}\\\text{out of a piece}\\\text{of hardware}\end{array}\right\} - \left\{\begin{array}{l}\text{Energy we}\\\text{put in}\end{array}\right\} = \left\{\begin{array}{l}\text{Energy we}\\\text{produce}\\\text{inside}\end{array}\right\} \quad (3)$$

We have used energy in each term of these equations, but it is just as correct to use power in each term. Then the three equations would read (if we leave out some of the words):

Power out	=	Power in	+	Power produced inside	(4)
Power out	-	Power produced inside	=	Power in	(5)
Power out	-	Power in	=	Power produced inside	(6)

Equation (6) is very important because it provides the most accurate way of finding out how much power our reactor core is producing. We have to know this power because the AEC license sets a maximum allowable core power. In order to determine the core power from equation (6), we would measure the power we get out of the reactor vessel and the power we put into the reactor vessel. The difference between these two would be the power produced inside -- that is, the core power. We will discuss this in more detail in the next section.

Power can be put into a piece of hardware as thermal energy carried in by flowing water or steam:

$$\text{Power in} = w_{in} h_{in} \qquad (7)$$

where

w_{in} is the mass flow rate of the water or steam going in (lbs/hr);

and

h_{in} is the enthalpy of the water or steam going in (Btu/lb).

When there are several pipes bringing in water or steam, we add up the products $w_{in} h_{in}$ for all the pipes. For example, if there are three pipes, the power input would be

1103 1 4 3 8

$$\text{Power in} = w_{in1}h_{in1} + w_{in2}h_{in2} + w_{in3}h_{in3}$$

We can get power out of a piece of hardware as thermal energy carried out by flowing water or steam:

$$\text{Power out} = w_{out}h_{out} \tag{8}$$

where

w_{out} is the mass flow rate of the water or steam going out (lb/hr);

and

h_{out} is the enthalpy of the water or steam going out (Btu/lb).

When there are several pipes carrying out water or steam, we add up the products $w_{out}h_{out}$ for all the pipes. For example, if there are three pipes, the power out would be:

$$\text{Power out} = w_{out1}h_{out1} + w_{out2}h_{out2} + w_{out3}h_{out3}$$

Power can also be gotten out of a piece of hardware in the form of mechanical power along a rotating shaft (such as a turbine). We will not need explicit mathematical expressions for this kind of power, so we will just use the symbol P_w to indicate mechanical power:

P_w = Power output as work

We will use the symbol P_q to indicate the thermal energy produced in a reactor core:

P_q = thermal power produced inside

Our heat balance equations (4-6) can be written in terms of these symbols. For example, equation (4) would be:

$$P_w + w_{out1}h_{out1} + w_{out2}h_{out2} + \ldots = w_{in1}h_{in1} + w_{in2}h_{in2} \ldots + P_q \qquad (9)$$

The black dots here mean you should keep adding up wh's for as many pipes as there are going in and going out. When there is only one pipe going in and one going out, the flow rate must be the same in both. In this special case we do not need the numbers on the w's and h's in equation (9), and this equation becomes:

$$P_w + wh_{out} = wh_{in} + P_q. \qquad (10)$$

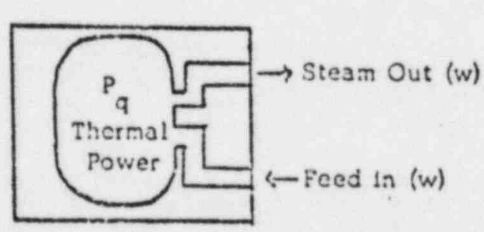
In the further special case when no mechanical power is involved, $P_w = 0$, and we can rewrite equation (10) as:

$$P_q = w(h_{out} - h_{in}). \qquad (11)$$

EXAMPLES

1. Simple Heat Source Heat Balance

In a simplified form, a BWR can be thought of as a heat source in a container with feedwater flowing in and an equal amount of steam flowing out.



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Given: feed flow = steam flow = 3×10^6 lb/hr
feed temperature = $232^\circ F$
steam properties = 1000 psia, 0% moisture saturated

What is the power of the heat source in MW?

Solution:

$$P_q - P_w = w(h_{out} - h_{in})$$

No work is being done, therefore $P_w = 0$.

$$P_q = w(h_{out} - h_{in})$$

From steam tables, feedwater enthalpy is 200.35* and steam enthalpy is 1192.9.

$$\begin{aligned} P_q &= 3 \times 10^6 \text{ lb/hr } (1192.9 - 200.35) \text{ Btu/lb} \\ &= 3 \times 10^6 \ (992.55) \\ &= 2.98 \times 10^9 \text{ Btu/hr} \end{aligned}$$

$$\text{in MW, } P_q = (2.98 \times 10^9 \text{ Btu/hr}) \left(\frac{1 \text{ kW}}{3413 \text{ Btu/hr}}\right) \left(\frac{1 \text{ MW}}{1000 \text{ kW}}\right)$$

$$= 875 \text{ MW}$$

2. Simple Turbine Heat Balance

In a simplified form, a turbine can be thought of as a work sink** with steam flowing in and an equal amount of steam flowing out.

---

*Not exactly correct since the feedwater is compressed liquid, but it is close enough for our purposes here.

**Sink is the opposite of source.

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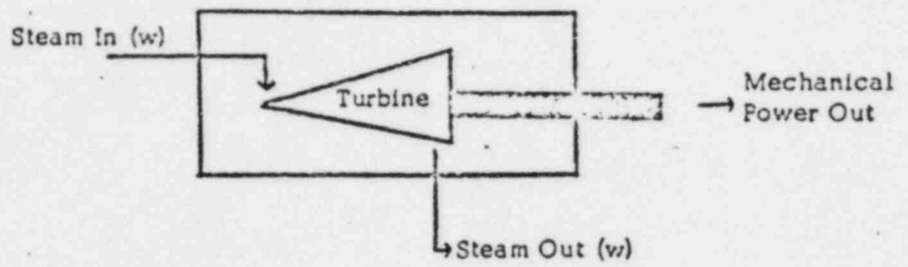

Given: steam flow $= 3 \times 10^6$ lb/hr
steam-in enthalpy $=$ 1190 Btu/lb
steam out enthanpy $=$ 1045 Btu/lb

What is the power of the turbine in hp?

Solution: $P_q - P_w = w(h_{out} - h_{in})$

No heat is being added, so $P_q = 0$

$$
\begin{aligned}
-P_w &= w(h_{out} - h_{in}) \\
&= (3 \times 10^6 \text{ lb/hr})(1045 - 1190 \text{ Btu/lb}) \\
&= -3 \times 10^6 (145) \\
&= -4.35 \times 10^8 \text{ Btu/hr} \\
P_w &= 4.35 \times 10^8 \text{ Btu/hr}
\end{aligned}
$$

in hp

$$
\begin{aligned}
P_w &= (4.35 \times 10^8 \text{ Btu/hr})(3.93 \times 10^{-4} \frac{\text{hp}}{\text{Btu/hr}}) \\
&= 171{,}000 \text{ hp}
\end{aligned}
$$

3. Fuel Assembly Heat Balance

The heat balance can also be applied to a reactor fuel assembly. No work is done, so the governing equation is

$$P_q = w(h_{out} - h_{in})$$

Given: A fuel assembly produces 3 MWt (t stands for thermal)

The flow through the assembly is 120,000 lb/hr.

The saturation pressure is 1000 psia and the inlet water to the fuel assembly has an enthalpy of 520 Btu/lb.

What are the exit conditions of the coolant?

Solution: In this case P_q is known and a solution for h_{out} is required.

$$\begin{aligned} h_{out} &= P_q/w + h_{in} \\ &= \left(\frac{3\text{MWt}}{1.2 \times 10^5 \text{ lb/hr}}\right)\left(3.413 \times 10^6 \frac{\text{Btu/hr}}{\text{MWt}}\right) + 520 \text{ Btu/lb} \\ &= 85.4 + 520 \text{ Btu/lb} \\ &= 605.4 \text{ Btu/lb} \end{aligned}$$

Is the existing coolant steam? Going to Table 2 of the steam tables, we see that the enthalpy of steam at 1000 psia is 1193 Btu/lb and of water is 543 Btu/lb. The above answer is between these two values and thus the existing coolant is a mixture of water and steam. We will now discuss this type of condition.

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F. Reactor Heat Balance

We will illustrate the uses of a heat balance by checking the heat balance presented for a typical PWR with a rating of 1658 MWT. Figure 2A shows the elements of this heat balance. Using equation (11), we can express the power produced in the core as:

$$P_q = w(h_{out} - h_{in})$$

From Figure 2A, we see that

$$w_{in} = w_{out} = 67.4 \times 10^6 \text{ lb/hr}$$

$$h_{in} = 541 \text{ Btu/lb}$$

$$h_{out} = 625 \text{ Btu/lb}$$

Then

$$P_q = 67.4 \times 10^6 \frac{\text{lb}}{\text{hr}} (625 - 541) \frac{\text{Btu}}{\text{lb}}$$

$$P_q = 5.66 \times 10^9 \text{ Btu/hr}$$

We can use the conversion factor from Section 2C to convert this to kw:

$$P_q = 5.66 \times 10^9 \frac{\text{Btu}}{\text{hr}} \times \frac{1 \text{ kw}}{3413 \frac{\text{Btu}}{\text{hr}}} = 1.658 \times 10^6 \text{kw}$$

or

$$P_q = 1.658 \times 10^6 \text{kw} \times \frac{1 \text{MW}}{10^3 \text{kw}} = 1658 \text{MW}$$

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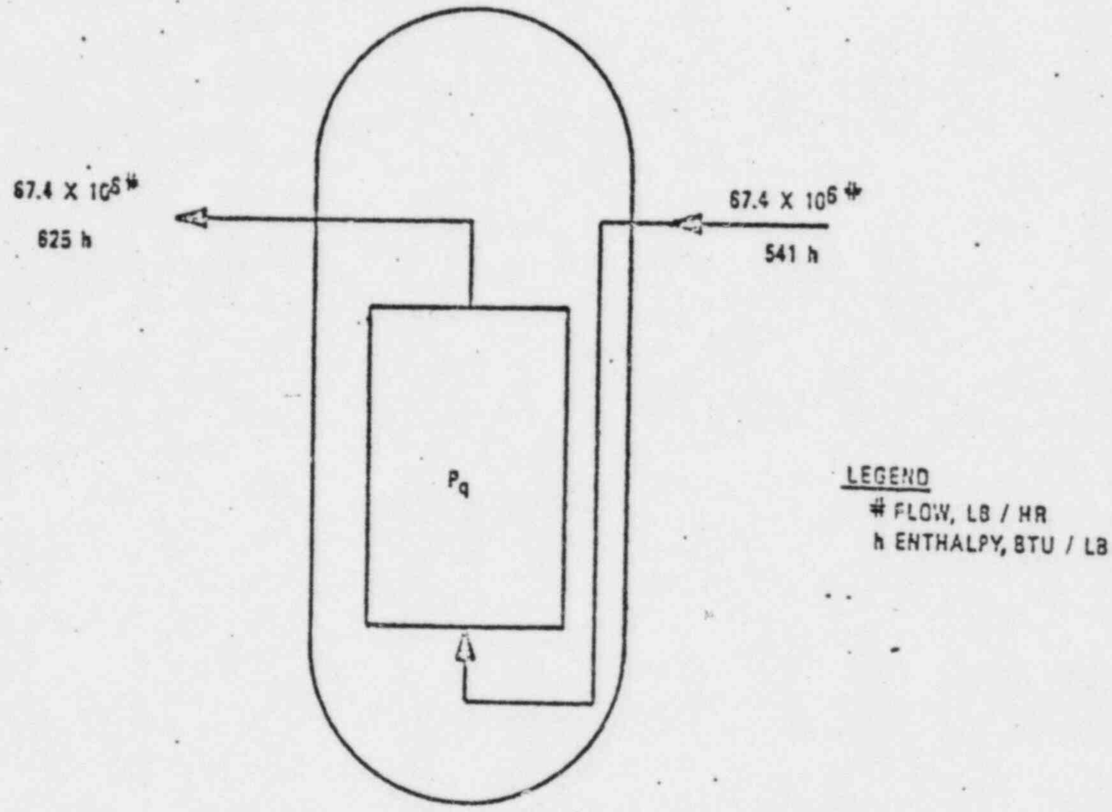

FIGURE 2A
PWR HEAT BALANCE

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G. Efficiency

(1) The efficiency of a piece or of pieces of hardware is defined as:

$$\text{Efficiency} = \frac{\text{Heat Used}}{\text{Net Heat In}} \quad (12)$$

(2) By "heat used" we mean the amount of heat energy which is actually converted into a form we are interested in. In a turbine, for example, the heat used is that which is converted into the energy of rotation of the shaft and blades. If we consider the turbine and condenser together, the heat used is only that which is converted in the energy of rotation of the shaft and blades; the heat rejected in the condenser is not really used.

(3) By "net heat in" we mean the amount of heat energy which enters the hardware through the normal path, less the amount which leaves through the normal path. In a turbine, for example, the net heat in is that which enters in the high pressure steam less that which leaves in the low pressure steam. If we consider turbine and condenser together, the net heat in is that which enters the turbine in the high pressure steam less that which leaves the condenser in the feedwater. In this case, considerable heat is lost by rejection to the condenser.

(4) Since

$$\text{Net Heat In} = \text{Heat Used} + \text{Heat Lost} \quad (13)$$

we can express the efficiency in terms of:

- $\left\{\begin{array}{l}\text{Net Heat In}\\ \text{Heat Used}\end{array}\right.$

or

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or

- $\begin{cases} \text{Net Heat In} \\ \text{Heat Lost} \end{cases}$

or

- $\begin{cases} \text{Heat Used} \\ \text{Heat Lost} \end{cases}$

To do this, solve Equation (13) for heat used:

Heat Used = Net Heat In - Heat Lost

and substitute this into the definition of efficiency:

$$\text{Efficiency} = \frac{\text{Heat Used}}{\text{Net Heat In}} = \frac{\text{Net Heat In} - \text{Heat Lost}}{\text{Net Heat In}}$$

$$\text{Efficiency} = 1 - \frac{\text{Heat Lost}}{\text{Net Heat In}} \qquad (14)$$

We can also substitute Equation (13) for net heat in directly into our definition to get

$$\text{Efficiency} = \frac{\text{Heat Used}}{\text{Net Heat In}} = \frac{\text{Heat Used}}{\text{Heat Used} + \text{Heat Lost}} \qquad (15)$$

(5) As expressed by Equations (12), (14) and (15), the efficiency is a fraction lying between zero and one. It is common to multiply this fraction by 100, so that efficiency is expressed in percent.

(6) Example

The difference between the enthalpy of steam flowing into a turbine and the enthalpy of the water flowing out of its condenser is 1132.9 Btu/lb. Of this enthalpy difference, 833.3 Btu/lb is lost as heat rejected by the condenser. (The flow rates through turbine and condenser are the same.) What is the efficiency?

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Heat Lost = 833.3 Btu/lb

Net Heat In = 1132.9 Btu/lb

$$\text{Efficiency} = 1 - \frac{\text{Heat Lost}}{\text{Net Heat In}}$$

$$\text{Efficiency} = 1 - \frac{833.3 \text{ Btu/lb}}{1132.9 \text{ Btu/lb}}$$

$$\text{Efficiency} = 0.265$$

or, expressed as percent,

$$\text{Efficiency} = 100 (0.265) = 26.5\%$$

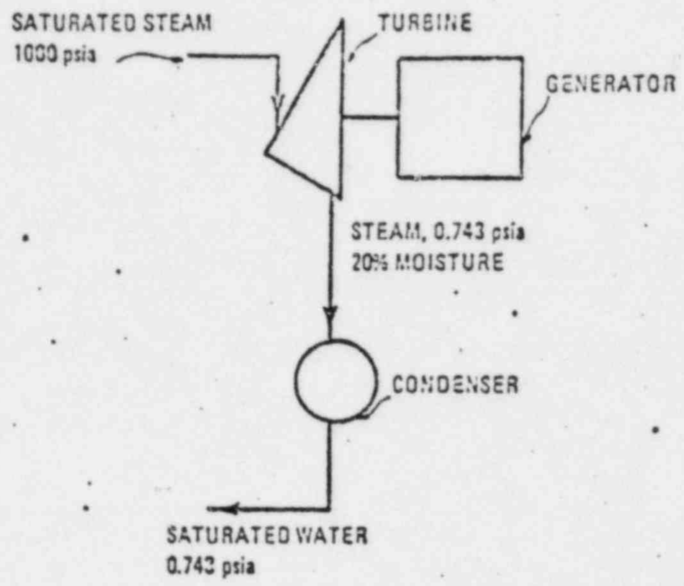
(7) In order to evaluate the

- Net Heat In
- Heat Used
- Heat Lost

we need to know wh, the product of the flow rate and the enthalpy of the water, steam, or mixture of the two at various points in the system. As explained in Section 2E, there will be more than one flow in and one out in a practical example, and in this case, we add up the values of wh for all flows.

(8) Example

The diagram on page 2-35 shows a turbine-condenser system. Saturated steam enters the turbine at 1000 psia and leaves the turbine at 0.743 psia with 20% moisture. The feedwater leaves the condenser as saturated water at 0.743 psia. The flow rate through the whole system is w lb/hr. What is the efficiency of this system?



The heat used is that which leaves the steam through the turbine. The heat lost is that which leaves through the condenser. The net heat in is that which enters in the saturated steam less that which leaves in the saturated water. In order to evaluate these amounts of heat, we first have to determine the enthalpy at the turbine inlet, at the turbine exhaust (which is the same as the condenser inlet), and at the condenser outlet.

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- Determine turbine exhaust enthalpy from steam tables:

  h_f moisture enthalpy = 60.0 Btu/lb

  h_g vapor enthalpy = 1101.6 Btu/lb

  $$h_{exhaust} = (\text{moisture fraction})(h_f) + (\text{vapor fraction})(h_g)$$
  $$= (0.2)(60.0) + (0.8)(1101.6)$$
  $$= 893.3$$

- Determine turbine inlet enthalpy from steam tables:

  $$h_{inlet} = h_g \text{ at } 1000 \text{ psia} = 1192.9$$

- Determine efficiency:

  heat in $= w\ (1192.9 - 60.0) = 1132.9\ w$

  heat lost $= w\ (893.3 - 60.0) = 833.3\ w$

  $$\text{eff} = 1 - \frac{833.3\ w}{1132.9\ w}$$

Flow rate cancels; therefore,

$$\text{eff} = 1 - 0.735$$
$$\text{eff} = 0.265 = 26.5\%$$

H. Effects on Efficiency

You will not be able to operate your plant such that all the parameters that affect efficiency are right on the textbook values. The steam will have some moisture in it as it enters the turbine, and the condensate leaving the condenser will be slightly sub-cooled. In this section we'll examine how varying some of these parameters affects the plant efficiency.

(1) Effect of Carryover

Assume that our plant supplies steam to the turbine that has 5% moisture. How does moisture content affect efficiency?

- Find turbine inlet enthalpy

$$h_m = Xhg + (1 - X)hf = h_{in}$$

$$h_{in} = (0.95)(1192.9) + (.05)(542.6)$$

$$h_{in} = 1133.5 + 27.1 = 1160.6 \text{ BTU/LB}$$

- Find new efficiency by same formula

$$\text{Efficiency} = 1 - \frac{\text{Heat Loss}}{\text{Heat In}}$$

$$\text{Heat Loss} = w(893.3 - 60.0) = 833.3\,w$$

$$\text{Heat In} = w(1160.6 - 60.0) = 1100.6\,w$$

$$\text{Eff} = 1 - \frac{833.3\,w}{1100.6\,w}$$

$$= 1 - .756 = .244 \text{ or } 24.4\%$$

Efficiency is lower when you have carryover.

(2) Effect of Superheat

Unless you have the one plant that is designed to produce superheated steam, you will not have any control or capability to produce superheated steam. However, we will still examine its effect on efficiency. Assume the steam to the turbine has 205°F of superheat.

- Find new turbine inlet steam enthalpy

  h_{in} = 1358.7 BTU/LB

- Find efficiency by same formula

  $$\text{Efficiency} = 1 - \frac{\text{Heat Loss}}{\text{Heat In}}$$

  Heat Loss = w(893.3 - 60.0) = 833.3w
  Heat In = w(1358.7 - 60.0) = 1298.7w

  $$\text{Eff} = 1 - \frac{833.3w}{1298.7w}$$

  $$= 1 - .642 = .358 \text{ or } 35.8\%$$

Superheat increases efficiency.

(3) Effect of Subcooled Condensate

It is difficult to maintain the condensate at saturation temperature, and may even be undesirable because of condensate pump suction head limits (to be discussed later). Assume that the condensate leaving the condenser is cooled below the saturation temperature of 92°F to 80°F.

- New condensate enthalpy hf = 48.0 BTU/LB
- Calculate efficiency as before

$$\text{Efficiency} = 1 - \frac{\text{Heat Loss}}{\text{Heat In}}$$

$$\text{Heat Loss} = w(893.3 - 48.0) = 845.3w$$

$$\text{Heat In} = w(1192.9 - 48.0) = 1144.9w$$

$$\text{Eff.} = 1 - \frac{845.3w}{1144.9w}$$

$$= 1 - .738 = .262 \text{ or } 26.2\%$$

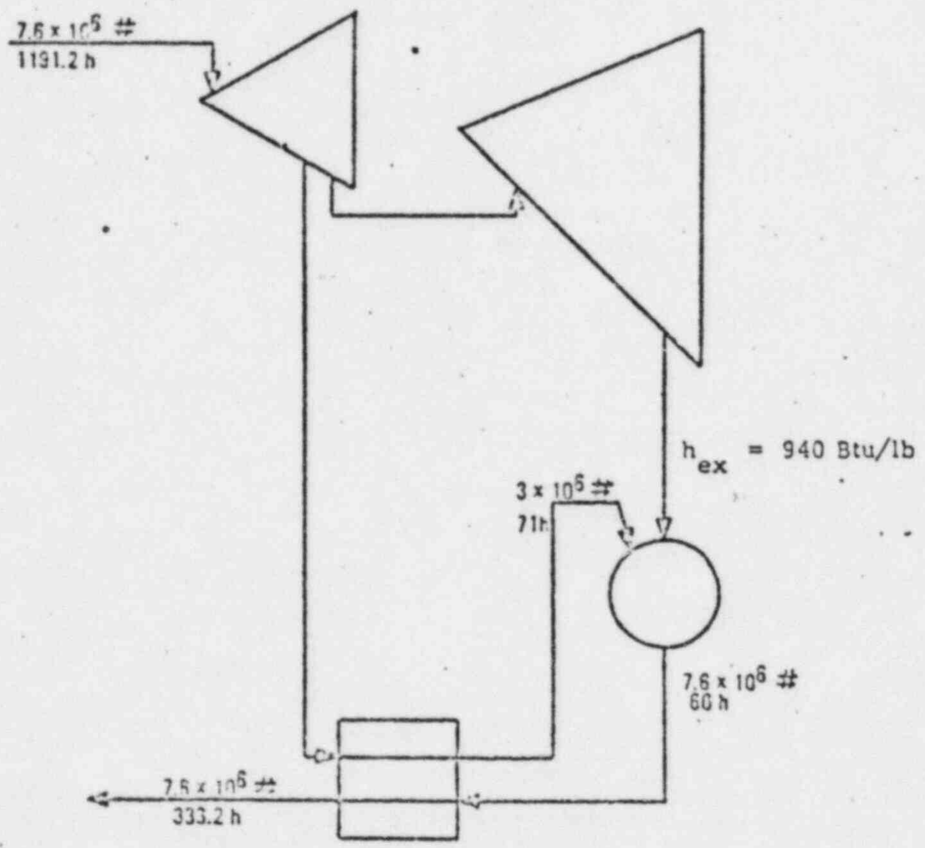
Subcooling the condensate decreases the efficiency but only by a very small amount.

(4) Effect of Feedwater Heating

Feedwater heating is the process of utilizing extraction steam and drains from various stages of the turbine to heat the feedwater prior to its return to the reactor vessel or steam generator. Figure 2B shows a heat balance for a secondary system utilizing feed heating. If we use the numbers shown, we can calculate the effect on efficiency of heating our feedwater. Note that we have new values for h_{in} and h_{ex} as follows:

$$h_{inlet} = 1191.2 \text{ BTU/LB}$$

$$h_{exhaust} = 940 \text{ BTU/LB}$$



CYCLE WITH FEEDWATER HEATING

FIGURE 2B

Also in our calculation of efficiency, we'll have to account for some flows, since their enthalpies are different. These flows are those that enter the condenser.

$$\text{Heat Loss} = 4.6 \times 10^6 \,(940 - 60) = 4.05 \times 10^9 \text{ BTU/HR}$$

$$\text{Heat Loss} = 3 \times 10^6 \,(71 - 60) = 0.03 \times 10^9 \text{ BTU/HR}$$

The first heat loss was that associated with condensing the steam from the L.P. turbines and the second was cooling the drains from the feed heaters.

$$\begin{aligned}\text{Heat In} &= 7.6 \times 10^6 \,(1191.2 - 333.2)\\ &= 6.5 \times 10^9 \text{ BTU/HR}\end{aligned}$$

$$\begin{aligned}\text{Efficiency} &= 1 - \frac{\text{Heat Loss}}{\text{Heat In}}\\ &= 1 - \frac{(4.05 + .03) \times 10^9}{6.5 \times 10^9}\\ &= 1 - .63 = .37 \text{ or } 37\%\end{aligned}$$

Feed heating greatly increases efficiency.

From these examples you can see that in order to keep your plant at maximum efficiency, you want to:

- Minimize carryover
- Maximize superheat (if you have it)
- Decrease condensate subcooling
- Maintain feedwater heating at its maximum

CORE PERFORMANCE

2. Thermodynamics and Heat Balances (cont.)

Problem Set - Chapter 2

1. What is the change in enthalpy of 1 pound of water when it is heated from 100°F to 180°F? 80°F

$\Delta h = -[1\ Btu/lb/°F \times (100-32)°F] + [1\ Btu/lb/°F\ (180-32)]$

$\Delta h = -68 + 148 = 80\ Btu/lbm$

2. If h_{fg} = 1000 Btu/lb at atmospheric pressure, how much heat is required to turn 200 pounds of water at 0 psig and 100°F into steam at atmospheric pressure?

112 × 200 = 22400 + 200000 = 222,400 BTU's

3. If the specific heat of steam is 0.5 Btu/lb/°F, how much more heat is required to make the steam in problem 2 superheated steam at 262°F?

262 − 212 = 50

50 × .5 = 25 × 200 = 5000 BTU's

4. A pressurizer has a volume of 100 ft³ and is filled with steam and water at 2200 psia and 650°F. If there are 2,000 pounds of water in it, how many pounds of steam are there? 287 lbs

5. Convert the following gauge pressures to absolute pressures:

(a) 500 psig
(b) 673 psig
(c) 1000 psig
(d) 0 psig
(e) 2235 psig

add 15 psi to ea.

6. Convert the following absolute pressures to gauge pressures:

(a) 515 psia
(b) 90 psia
(c) 688 psia
(d) 900 psia
(e) 1015 psia
(f) 2000 psia
(g) 15 psia
(h) 1725 psia
(i) 2250 psia

Subtract 15 psi from ea.

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7. Convert the following absolute pressures to inches of Hg, vacuum:

(a) 15 psia 0
(b) 7.5 psia 15
(c) 0 psia 30 in Hg
(d) 12 psia 6 in Hg
(e) - 3 psia
(f) 10 psia 10 in Hg abs

Plot these results on graph paper, using the vertical scale for psia and the horizontal scale for inches Hg, vacuum. Join the points with a straight line. You can use this graph to convert psia to gauge, or vice-versa.

8. What must be the temperature of saturated water at a pressure of:

(a) 1 psia
(b) 14.7 psia
(c) 100 psia
(d) 200 psia
(e) 500 psia
(f) 1000 psia
(g) 2000 psia
(h) 2200 psia

Look in Table STM for ~~saturated~~ Press.

9. What must be the temperature of saturated steam at a pressure of:

(a) 1 psia
(b) 14.7 psia
(c) 100 psia
(d) 200 psia
(e) 500 psia
(f) 1000 psia
(g) 2000 psia
(h) 2200 psia

Look in STM Table for Press.

10. What must be the pressure of saturated water at a temperature of:

(a) 32°F
(b) 70°F
(c) 100°F
(d) 200°F
(e) 300°F
(f) 400°F
(g) 500°F
(h) 600°F
(i) 640°F

Look in STM Table for Temp

11. What must be the pressure of saturated steam at a temperature of:

(a) 32°F
(b) 70°F
(c) 100°F
(d) 200°F
(e) 300°F
(f) 400°F
(g) 500°F
(h) 600°F
(i) 640°F

Look in STM Table for Temp

12. What are the specific volume and enthalpy of water at the following pressures:

(a) 1 psia
(b) 14.7 psia
(c) 100 psia
(d) 200 psia
(e) 500 psia
(f) 1000 psia
(g) 2000 psia
(h) 2200 psia

Look at STM Table for Press.

13. What are the specific volume and enthalpy of steam at the following temperatures:

(a) 32°F
(b) 70°F
(c) 100°F
(d) 200°F
(e) 300°F
(f) 400°F
(g) 500°F
(h) 600°F
(i) 640°F

Look in STM Table for Temp.

14. (a) What is the enthalpy of a pound of steam at 1000 psia and 800°F?

1389.6 BTU/lb

(b) What is the enthalpy of a pound of steam at 1000 psia in a saturated condition?

1192.9 BTU/lb

1192.9

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15. How much work is equivalent to the enthalpy difference between the superheated and saturated steam in problem 14?

$778 \frac{ft\text{-}lb}{BTU} \times 196.7 \frac{BTU}{lb} =$ $\Delta h = 196.7$ ST-16

16. Make yourself a power conversion table. In each square, place the number by which you have to multiply the unit at the side to obtain the unit at the top.

To obtain → / Multiply by ↓	MW	KW	hp
MW	1	10^3	1340
KW	10^{-3}	1	1.34
hp	.00075	.75	1

17. What is the error if the temperature instrument reads 5° higher than the actual temperature of 250°F?

950#   9%   Refer to Lecture for other data

18. What is the error if the pressure instrument reads 915 psig and the actual pressure is 950 psia?

19. What values of quality correspond to the following values of carryover:

(a) 0.25%
(b) 0.10%
(c) 0.75%
(d) 1.20%

1 − X = Carryover
X = 1 − Carryover

20. What values of carryover correspond to the following values of quality:

(a) 99.0%
(b) 99.5%
(c) 99.85%
(d) 99.75%

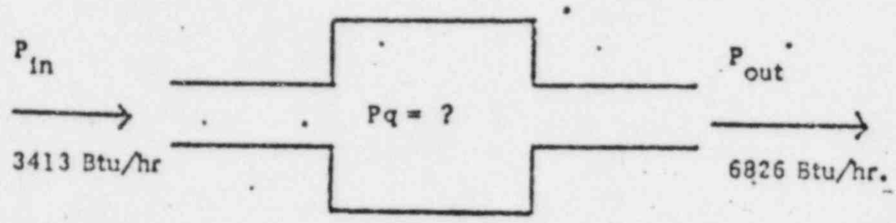
1 − X = Carryover

Sat. temp.

21. Find the enthalpy and specific volume of steam at 1000 psia and having the following moisture content:

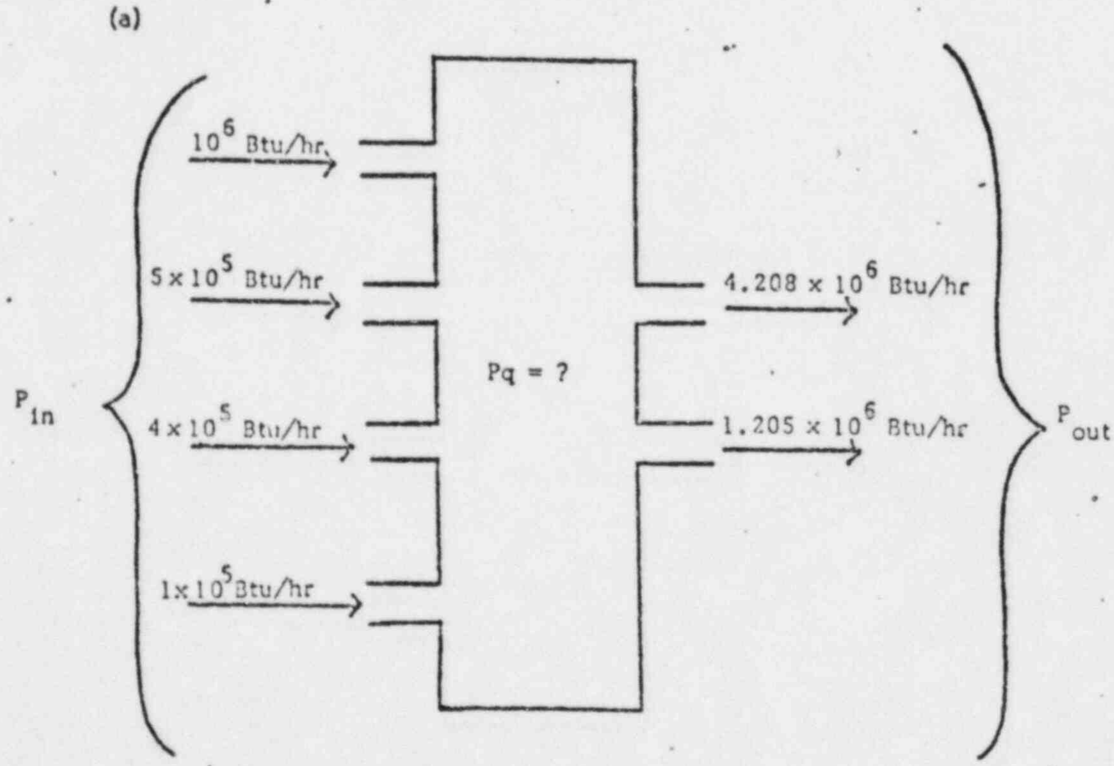
(a) 0.1%
(b) 0.4%
(c) 1.0%

22. What power is produced in the box shown in the following diagram?



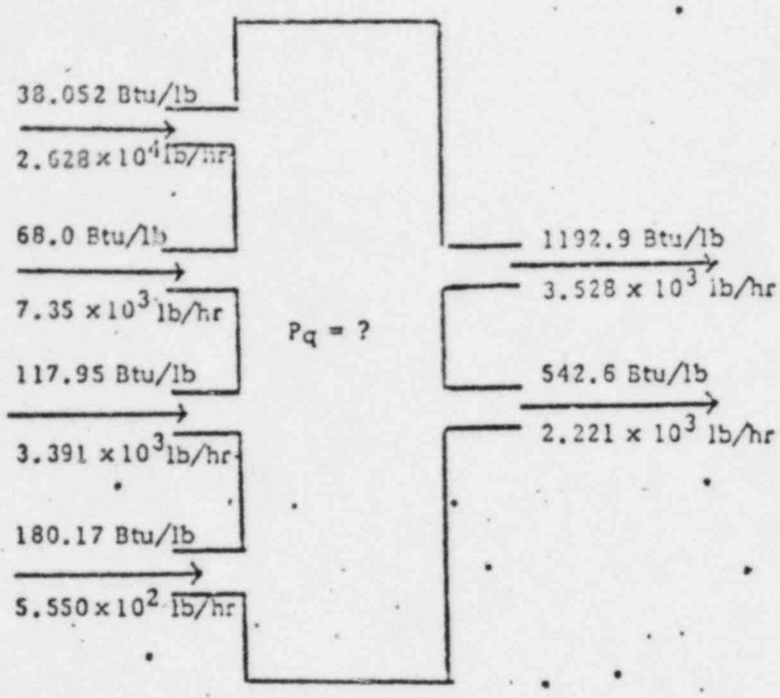
23. What power is produced in the box shown in the following diagrams:

(a)

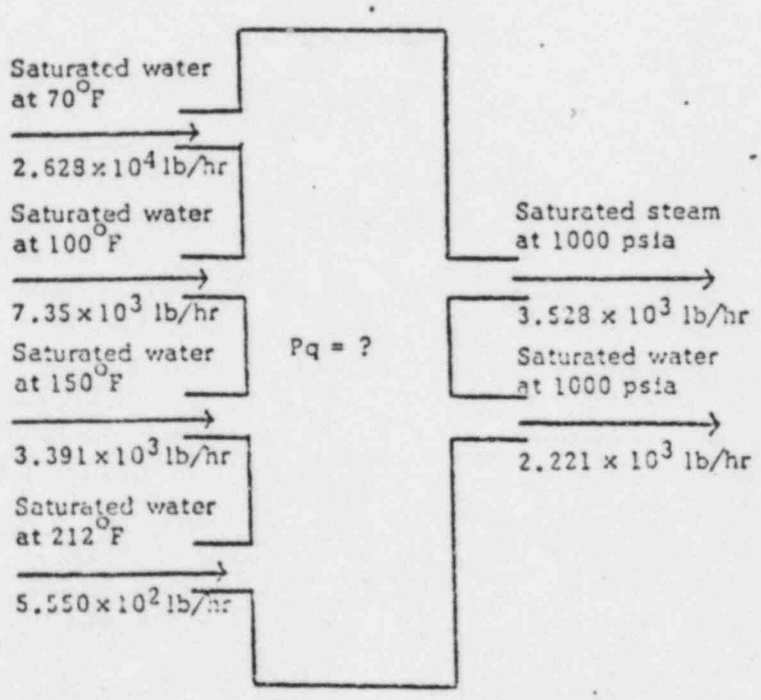


1103 1 4 6 0

(b)


38.052 Btu/lb
2.628×10^4 lb/hr
68.0 Btu/lb
7.35×10^3 lb/hr
117.95 Btu/lb
3.391×10^3 lb/hr
180.17 Btu/lb
5.550×10^2 lb/hr
Pq = ?
1192.9 Btu/lb
3.528×10^3 lb/hr
542.6 Btu/lb
2.221×10^3 lb/hr

(c)


Saturated water at 70°F
2.628×10^4 lb/hr
Saturated water at 100°F
7.35×10^3 lb/hr
Saturated water at 150°F
3.391×10^3 lb/hr
Saturated water at 212°F
5.550×10^2 lb/hr
Pq = ?
Saturated steam at 1000 psia
3.528×10^3 lb/hr
Saturated water at 1000 psia
2.221×10^3 lb/hr

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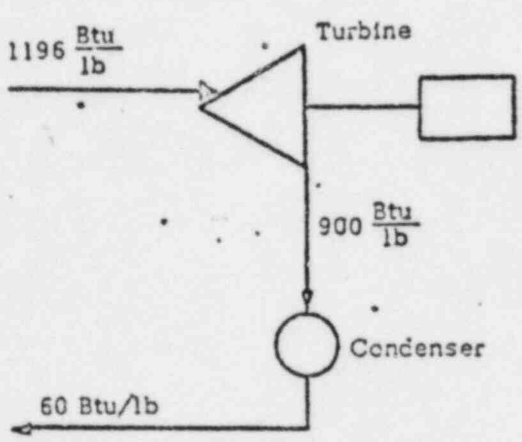
24. The flow rate through a particular PWR fuel assembly is 200,000 lb/hr. The enthalpy is 520 Btu/lb at its inlet and 624 Btu/lb at its outlet. What power is the assembly producing? - ~~2.1 x 10^2 BTU/HR~~ 6.1 MW

25. What are the units of "efficiency"?

26. Consider the following turbine and condenser:



(a) What is the amount of net heat in? 1196-60 = 1136 BTU/lb

(b) What is the amount of heat used? 296 BTU/lb

(c) What is the amount of heat lost? 840 BTU/lb

(d) Calculate the efficiency as:

$$\text{Efficiency} = \frac{\text{Heat Used}}{\text{Net Heat In}}$$

$\frac{296}{1136}$ 100 =

(e) Calculate the efficiency as:

$$\text{Efficiency} = 1 - \frac{\text{Heat Lost}}{\text{Net Heat In}}$$

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26. (f) Calculate the efficiency as:

$$\text{Efficiency} = \frac{\text{Heat Used}}{\text{Heat Used + Heat Lost}}$$

27. Problem 26 was worked in terms of enthalpy (Btu/lb). Suppose the flow rate in that problem is 5×10^6 lb/hr. Repeat problem 26, but now express the amounts of heat in terms of Btu/hr instead of Btu/lb. Remember:

$$\frac{\text{Btu}}{\text{hr}} = \left(\frac{\text{Btu}}{\text{lb}}\right) \times \left(\frac{\text{lb}}{\text{hr}}\right)$$

multiply

3. FLUID FLOW

There are many problems in reactor systems that relate to fluid flow: removal and transport of heat from the core, the pumping power required to move the coolant, the pressure drop through various parts of process systems, and flow distribution within the core.

A. Pressure Drop

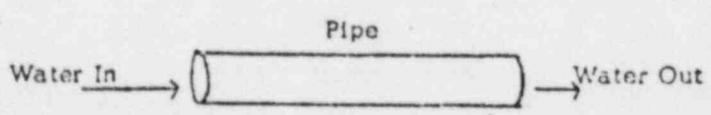
(1) To understand flow, we need to answer the question: What determines the amount and distribution of the flow? The answer is:

Pressure drop determines the amount and distribution of fluid flow.

(2) What is "pressure drop"? Consider two points between which fluid is flowing (in, say, a pipe or fuel assembly). The pressure drop between those points is defined as:

Pressure drop = (Pressure at the point reached by the water earlier) − (Pressure at the point reached by the water later).

(3) Example:

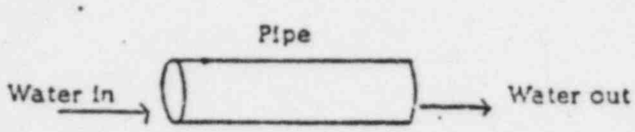


P_{in} = 1000 psia P_{out} = 990 psia

The pressure drop across this pipe is:

$$P_{in} - P_{out} = 1000 \text{ psia} - 990 \text{ psia} = 10 \text{ psi}$$

(4) Example



P_{in} = 985 psig P_{out} = 975 psig

The pressure drop across this pipe is:

$$P_{in} - P_{out} = 985 \text{ psig} - 975 \text{ psig} = 10 \text{ psi}$$

(5) Did you notice that the two examples above are really identical? The first is stated in terms of absolute pressure and the second in terms of gage pressure. But, 1000 psia = 985 psig, and 990 psia = 975 psig, so the physical conditions are really the same in the two problems! Regardless of whether we worked in absolute or gage pressures, we got the same pressure drop.

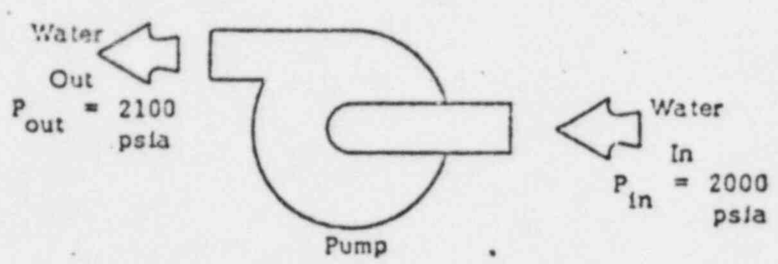
Pressure drop is neither gage nor absolute. Its units are simply psi.

However, we can calculate the pressure drop by working in either gage or psia, as long as we use the same units for both pressures.

(6) Because pressure drop is a difference in pressures, the symbol ΔP is sometimes used for it.

1103 1465

(7) Example



$$\Delta P = P_{in} - P_{out}$$

$$\Delta P = 2000 \text{ psia} - 2100 \text{ psia}$$

$$\Delta P = -100 \text{ psi}$$

(8) Pressure drop can be negative, as in the above example. Under normal conditions the pressure drop across a pump is always negative.

(9) Another way of expressing pressure drop is as "head loss." The units of head or head loss are feet.

$$\underset{(\text{ft})}{\text{Head loss}} = \underset{\frac{\text{lb}}{\text{in}^2}}{\text{Pressure drop}} \times \underset{\frac{\text{in}^2}{\text{ft}^2}}{144} \times \underset{\frac{\text{ft}^3}{\text{lb}}}{\text{Specific volume}}$$

The 144 is just a conversion factor to get everything into the units of feet. The specific volume to use here can be obtained from the steam tables.

(10) Example

The examples in (3) and (4) above yielded a ΔP of 10 psi. The corresponding head loss is:

/1103 1 4 6 6

$$\text{Head loss} = \frac{10\ \text{lb}}{\text{in}^2} \times \frac{144\ \text{in}^2}{\text{ft}^2} \times \frac{.0216\ \text{ft}^3}{\text{lb}} = 31.1\ \text{ft}$$

B. Relation of Pressure Drop to Flow

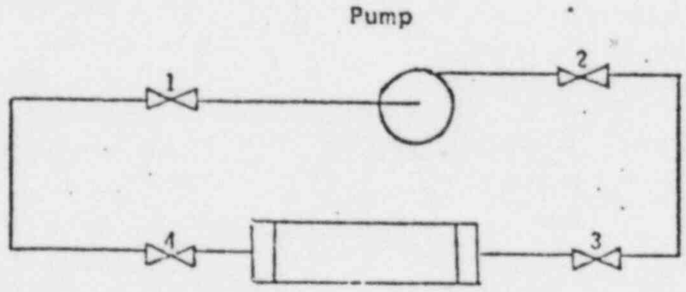
(1) The basic rule relating pressure drop to flow is:

> The flow adjusts itself so that when you go around a flow loop and add up all the pressure drops, you get zero.

(2) A <u>flow loop</u> is a path for the water to flow in, which leads you back to the spot at which you started.

(3) Example:

The figure below shows a flow loop containing a pump, 4 valves, a heat exchanger, and piping. The pressure drops across these components are shown in Figure 3A.



(4) The pressure drop in each component is the sum of the pressure drops due to friction, elevation changes, entrance and exit losses (sometimes called shock losses) and acceleration losses.

$$\Delta P = \Delta P_{\text{friction}} + \Delta P_{\text{elevation}} + \Delta P_{\text{acceleration}} + \Delta P_{\text{shock}}$$

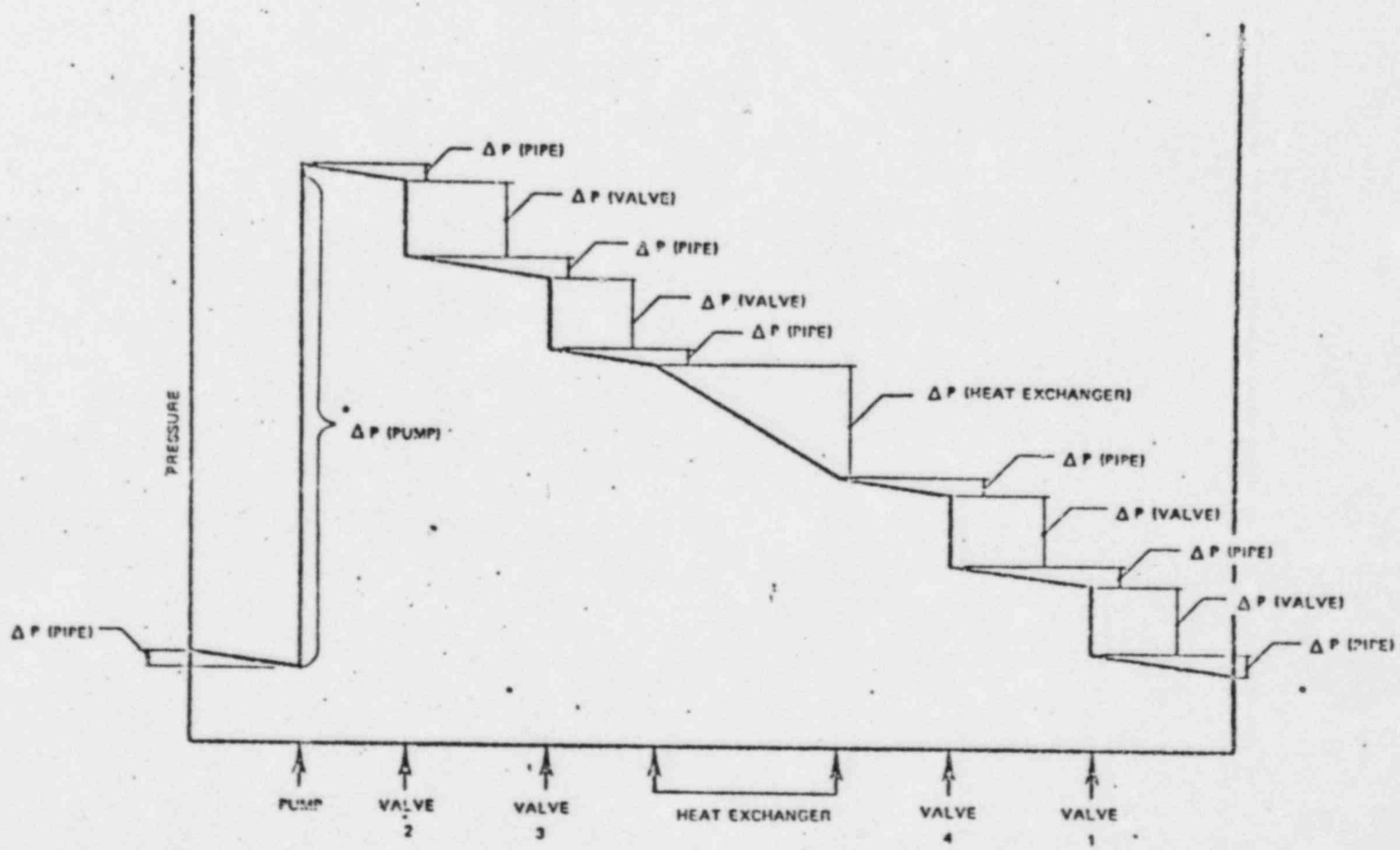


FIGURE 3A
PRESSURE AND PRESSURE DROPS AROUND
A FLOW LOOP

(5) Friction losses are those due to the internal friction of fluid and are directly related to viscosity. Friction losses result in a slight temperature increase of the fluid.

(6) Elevation losses occur if the fluid is pumped uphill. The potential energy of the fluid is increased at the expense of pressure. This does not depend on viscosity.

(7) The entrance and exit losses, or shock losses, are the type of loss across an orifice in a line which is filled on both sides. This results in a slight temperature increase of the fluid. It has a very weak dependence on viscosity.

(8) Acceleration losses are those that occur when the fluid velocity is increased. For example, a converging nozzle, which converts pressure energy to kinetic energy, incurs acceleration losses.

C. What Determines Pressure Drop?

(1) The size of friction losses, shock losses, and acceleration losses depends upon the type of flow. There are two readily discernible types of flow: when the velocity is sufficiently low, flow is laminar. Otherwise, flow is turbulent. The difference is best described by demonstration. The smoke from a burning cigarette or candle in a very still room exhibits laminar flow in the first few inches where the smoke rises in a smooth streamlined manner. Where the flow becomes wavy, the transition to turbulent flow has begun. In fully developed turbulence, the flow is "confused." Considerable sideways motion exists for any particle of smoke.

(2) Reynolds Number - The nature (that is, whether it is laminar or turbulent) of flow in a pipe depends on pipe diameter, density

and viscosity of the flowing fluid, and velocity of flow. The numerical value of a certain dimensionless combination of these four variables is known as the Reynolds number, R_e. It is written as:

$$R_e = \frac{DV\rho}{\mu}$$

where

D = Pipe diameter (ft)
V = Flow velocity (ft/sec)
μ = Viscosity (lb/ft-sec)
ρ = Density (lb/ft^3)

(3) Viscosity - Viscosity is a fluid property that expresses the readiness with which a fluid will flow when acted upon by an external force. It may be thought of as the internal friction of a fluid. Molasses has a high viscosity. Water is much less viscous, and the viscosity of gas is much less than that of water.

Viscosity is an important influence on whether flow is laminar or turbulent. Viscosity affects pressure drop, lubrication characteristics, and heat transfer properties of a flowing fluid.

The viscosity of liquid water depends only on the temperature, and it decreases with temperature. The viscosity of steam increases with temperature, but it also depends upon the pressure.

(4) You will not need exact expressions for the various kinds of pressure drops we have mentioned, but you should realize that they depend upon the Reynolds number. Whenever we change anything in the Reynolds number, the pressure drop and, therefore, the flow, may change.

R_e < 2000 gives laminar flow.

R_e > 4000 gives turbulent flow.

When R_e lies between 2000 and 4000, the flow may be either laminar or turbulent.

Detailed numerical analyses of flow and pressure drop in real hardware are given in "Flow of Fluids Through Valves, Fittings, and Pipes," by the Engineering Division of the Crane Company, Chicago (Technical Paper No. 410).

Bernoulli's eq.

Ideal situation
Non viscous
Non compressible

$$P + \frac{1}{2}\rho v^2 + (\rho g y) = K$$

dynamic effect

Static don't change

every state will have a constant related according to this equation

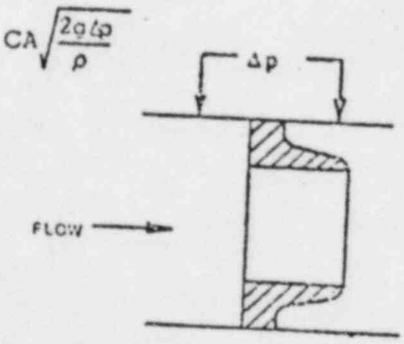
D. FLOW MEASUREMENT

Flow measurement devices occur in many places throughout the plant. It is important to understand the principles of these devices.

There are two common methods for measuring flow. One is a displacement device that moves in direct proportion to the flow rate, as, for example, an impeller whose rotational speed depends on flow rate. Another way is to measure the pressure drop across an orifice or component such as a flow nozzle or pipe elbow.

To use the pressure drop method of flow measurement, we must first determine the flow versus pressure drop characteristic of the component. The volumetric flow rate of any fluid through an orifice or nozzle may be expressed as

$$\text{flow} = CA\sqrt{\frac{2g\Delta p}{\rho}}$$



where

C	=	nozzle flow coefficient
A	=	area of nozzle opening (ft^2)
g	=	32.2 ft/sec^2 = gravitational acceleration
Δp	=	pressure drop across the nozzle (lb/ft^2)
ρ	=	density of fluid
flow	=	volumetric flow (ft^3/sec)

The nozzle or orifice flow coefficient C is determined experimentally for each setup.

With this relationship, if a calibrated orifice or nozzle is used in combination with a differential pressure detector, the flow rate in a system can be measured. To obtain the weight flow rate w the formula becomes

$$w = \rho(\text{flow}) = CA\sqrt{2g\rho\,\Delta p}$$

which will yield the results in lb/sec.

An important point to note is that the density ρ changes with temperature, and the flow meter calibration is good for a given temperature. Corrections for density change must be applied if the temperature changes. Thus, if calibration curves are supplied to an operator, he must insure that the conditions under which the calibration occurred correspond to the actual condition of the desired measurement.

As it turns out, the value of C depends on the Reynolds number if the flow is only slightly turbulent, and thus, for this case, also changes somewhat with temperature. In most applications, however, the Reynolds number is sufficiently high that the value of C is unaffected by temperature. Thus, calibration curves are available for a different temperature than is being measured. A thumb-rule flow estimate can be obtained by

$$w_{actual} = w_{curve}\sqrt{\frac{\rho_{actual}}{\rho_{curve}}}$$

where w_{curve} is the flow rate from the calibration curve based on the measured Δp, ρ_{actual} is the fluid density based on the existing temperature, and ρ_{curve} is the fluid density at the calibration temperature.

E. PUMPS

Before starting our discussions of pumps, it will be necessary to lay some ground work for the material we'll be covering. Let's review the concept of energy first. Remember, we said that the total energy was equal to the kinetic energy + potential energy. Also you'll remember that the kinetic energy was that which a body possessed because of its motion and the potential energy was that which a body possessed because of its position. What does all this have to do with pumps? The total energy of the fluid discharge from any pump is usually referred to as "head" and is normally expressed in feet of water. Essentially this total head is the mathematical sum of the velocity head and the pressure head which the fluid possesses. The velocity head represents the kinetic energy of the moving fluid, whereas the pressure head represents the potential energy of the fluid, measurable by a pressure gauge. Hence, pressure head is referred to as static pressure.

We can write a mathematical expression to illustrate this principle. This is part of Bernoulli's equation and it states:

$$\text{Total Head } H = \frac{V^2}{2g} + \frac{P}{\rho}$$

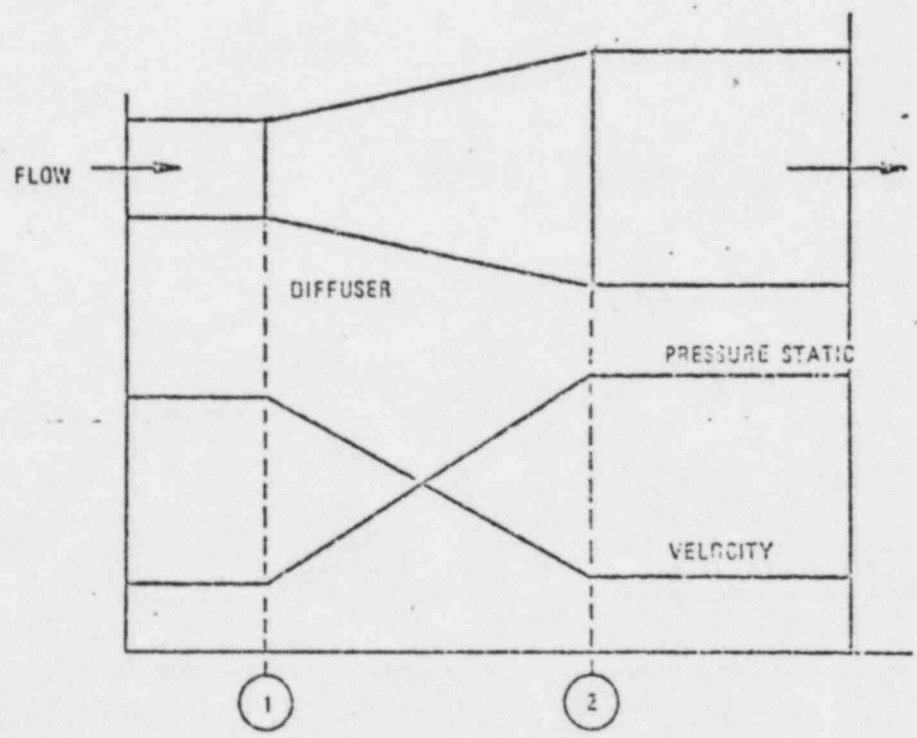
Represents velocity head — Represents pressure head — in ST

Checking the units:

$$\frac{FT\text{-}LB}{LB} = \left(\frac{FT^2}{SEC^2} \times \frac{1}{\frac{FT}{SEC^2}}\right) + \left(\frac{LB}{FT^2} \times \frac{1}{\frac{LB}{FT^3}}\right)$$

$$FT = FT + FT$$

- Let's apply Bernoulli's equation to a diffuser example and see how it works.



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A diffuser converts some of the kinetic energy of the high velocity mass of water into potential energy or pressure in permitting the moving mass to slow down as it passes through the widening channel. As the passage widens toward the outlet, less linear velocity is required to pass the same mass of water in a given length of time. Now, if, at the same time, the discharge is restricted so that the diffuser remains filled with fluid, it can be seen that the impact of the entering fluid upon the retarding mass of water tends to develop a static pressure.

Let's check that out using our equation.

$$H = \frac{V^2}{2g} + \frac{P}{\rho}$$

At Point 1 we can define the total head as:

$$H_1 = \frac{V_1^{\,2}}{2g} + \frac{P_1}{\rho}$$

Similarly, at Point 2 the head is:

$$H_2 = \frac{V_2^{\,2}}{2g} + \frac{P_2}{\rho}$$

Since there is no addition or subtraction of energy between points 1 and 2 (except for some very small friction loss), we can say that:

$$H_1 = H_2$$

$$\frac{V_1^{\,2}}{2g} + \frac{P_1}{\rho} = \frac{V_2^{\,2}}{2g} + \frac{P_2}{\rho}$$

assume $\rho_1 = \rho_2$

assume non viscous solution

$$P_1 + \tfrac{1}{2}\rho V_1^2 + \rho g y_1 = P_2 + \tfrac{1}{2}\rho V_2^2 + \rho g y_2$$

laminar flow

$$\Delta P_{skin} = 64 \frac{\mu}{D V \rho} \cdot \frac{L}{D} \cdot \frac{\rho V^2}{2g_c}$$

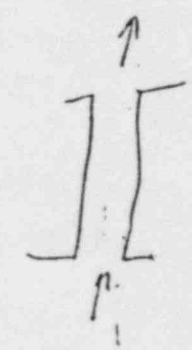
$\frac{64}{Re}$

turbulent flow

$$\Delta P_{skin} = 4f \cdot \frac{L}{D} \cdot \frac{\rho V^2}{2g_c}$$

$\frac{64}{Re} = 4f$

$f = \frac{16}{Re}$



$$D = D_e = 4 \times \frac{\text{Cross-section of stream}}{\text{wetted perimeter}}$$

$$\Delta P = \frac{\rho}{2g}\left(V_1^2 - V_2^2\right) K_c$$

use by Mikes units

$$H_{Total} = \frac{V^2}{2g} + \frac{P}{\rho}$$

$$f = .079\, Re^{-.25}$$

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Rearranging terms gives us:

$$\frac{V_1^2}{2g} - \frac{V_2^2}{2g} = \frac{P_2}{\rho} - \frac{P_1}{\rho}$$

$$\frac{1}{2g}\left(V_1^2 - V_2^2\right) = \frac{1}{\rho}(P_2 - P_1)$$

$$\Delta P \doteq \frac{\rho}{2g}(V_1^2 - V_2^2)$$

We can do an example to illustrate the static pressure gain in a diffuser.

Given: $V_1 = 5$ ft/sec

$V_2 = 3$ ft/sec

$\rho = 62.4$ lb/ft^3

$$\Delta P = \frac{\rho}{2g}(V_1^2 - V_2^2)$$

$$\Delta P = \frac{62.4}{64.4}(25 - 9)$$

$$\Delta P \approx 16 \text{ lb/ft}^2$$

As a result of slowly decreasing the velocity of the fluid from 5 to 3 ft/sec, we realized a gain in static pressure of 16 lb/ft^2. We'll apply this principle later when we examine how a pump works.

Let's continue our ground work and define some terms. The first is something called NET POSITIVE SUCTION HEAD, and the second is CAVITATION.

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(H) Velocity head = $\frac{V^2}{2g_c}$

larger to smaller pipe

(K_c) contraction constant

$$\boxed{\Delta P = \frac{\Delta P_H}{\cancel{\rho}} = K_c \rho \cdot \frac{V^2}{2g_c}}$$

D_2/D_1	.8	.6	.4	.2	0	
K_c	.13		.35	.45	.50	

Small to larger pipe

$$K_e\left(1-\frac{D_1^2}{D_2^2}\right)^2$$

Mike wants

$$\Delta P = \frac{\rho}{2g}\left(V_1^2 - V_2^2\right)K_c$$

D_1 exit pipe

D_2 entrance

$$H = \frac{V^2}{2g} + \frac{P}{\rho}$$

$$f = \frac{16}{Re}$$

$4f \neq \frac{64}{Re}$ but use as an assumed equality

$$H = \frac{\rho V^2}{2g_c}\left(K_c + K_e\right)$$

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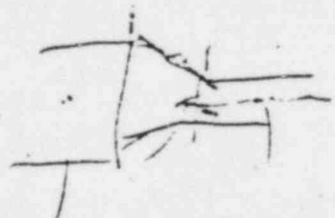
1480



First, we'll define available NPSH. It is the difference between the total pressure at the suction of the pump and the vapor or saturation pressure at the temperature of the pumped fluid. It can be calculated using the formula:

$$NPSH = (P_s \pm Z - h_{fs}) - P_{vp}$$

where

Z	=	head corresponding to height above (+) or below (-) impeller centerline (ft).
P_s	=	absolute pressure (ft) at the level of the pump supply
P_{vp}	=	vapor pressure of fluid at pumping temperature (ft)
h_{fs}	=	friction losses in suction line (ft)

The terms in parentheses define the total head at the suction of the pump. Note that pressure head and friction loss are all in feet of water. Pressure is converted to feet by dividing it by density or by multiplying it by specific volume. Since specific volume can be read directly from the steam tables, our determination of head will be based on:

$$\text{Head (ft)} = P \times v$$

We have not shown how to calculate friction loss, so we will just assign a value for it in the following example.

Pumping Pwr = ΔP X Volume flow rate

$PP = \Delta P_{pump} \times A_{pump} V$

$\Delta P = Hf \frac{L}{D} \frac{\rho V^2}{2g_c}$

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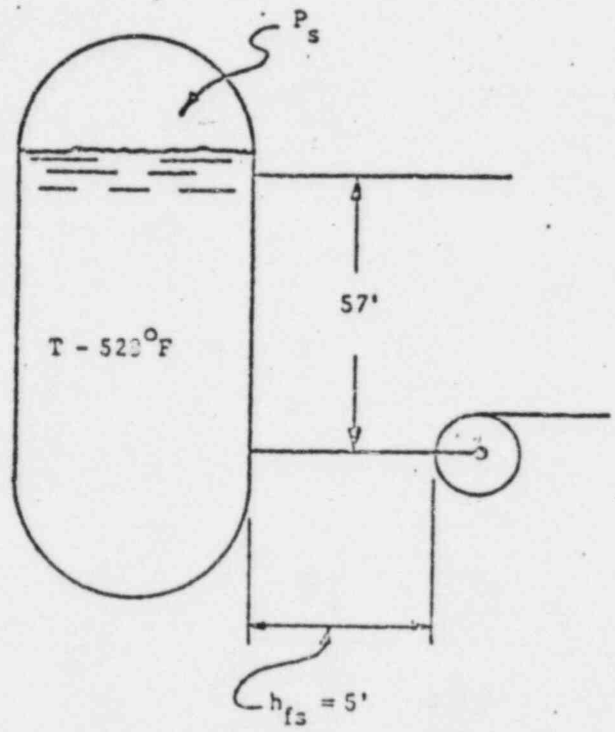
What is the available NPSH for the pump in the following figure?



We must first go to the Steam Tables to obtain the specific volume and pressure at 529°F.

$$\text{At } 529^{\circ}\text{F}, \; v_f = .02115 \; ft^3/lb$$

$$P = 878 \text{ psia}$$

Since our vessel contents are in a saturated state,

$$Ps = Pvp$$

And:

$$NPSH = P_s + Z - hfs - P_{vp}$$

$$= Z - hfs$$

$$= 57 - 5 = 52 \text{ ft.}$$

Let's modify our example and say that there is a gas present in the vessel above the level so that:

$$Ps = 1020 \text{ psia}$$

We can calculate a new NPSH as follows:

$$NPSH = Ps + Z - hfs - Pvp$$

$$= (1020 \times 144 \times .02115) + 57 - 5 - (878 \times 144 \times .02115)$$

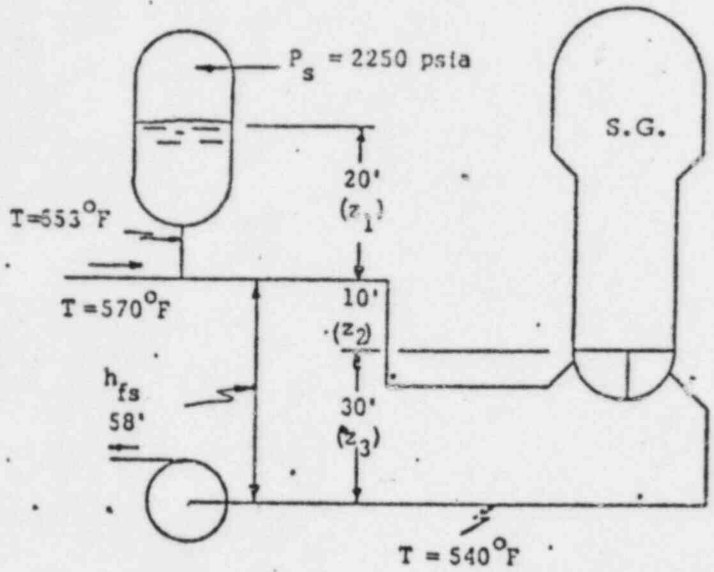
$$= 3108 + 57 - 5 - 2676$$

$$= 3165 - 2681 = 484 \text{ ft.}$$

The number 144 was used to convert psi to psf.

Note that we used a specific volume value for conditions at the pump suction (T=529°F). We would have been more accurate to use specific volume for the compressed liquid, but our error is less than ½%.

Let's look at a more difficult example:



In this problem we must account for the fact that the specific volume of the fluid in the pressurizer and hot leg is different from that at the pump suction. Since everything must be related to the pump suction conditions, Z_1 and Z_2 will have to be restated. For our compressed liquid (and to be most accurate) the following values of specific volume are to be used:

@ P = 2250 psia saturated: vf = .0271 ft^3/lb

@ P = 2250 psia and 570°F: vf = .0219 ft^3/lb

@ P = 2250 psia and 540°F: vf = .0211 ft^3/lb

@ 540°F saturated: vf = .0215 ft^3/lb and P = 963 psia

$$NPSH = Ps + Z_1 + Z_2 + Z_3 - hfs - Pvp$$

Where $Ps = (2250 \times 144 \times .0211) = 6840$ ft.

$$Z_1 = \frac{(20 \times .0211)}{.0271} = 15.6 \text{ ft.}$$

$$Z_2 = \frac{(10 \times .0211)}{.0219} = 9.6 \text{ ft.}$$

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$Z_3 = 30'$

hfs = 58'

Pvp = (963 x 144 x .0215) = 2985 ft.

NPSH = 6840 + 15.6 + 9.6 + 30 - 58 - 2985

= 6895.2 - 3043

= 3852.2 ft.

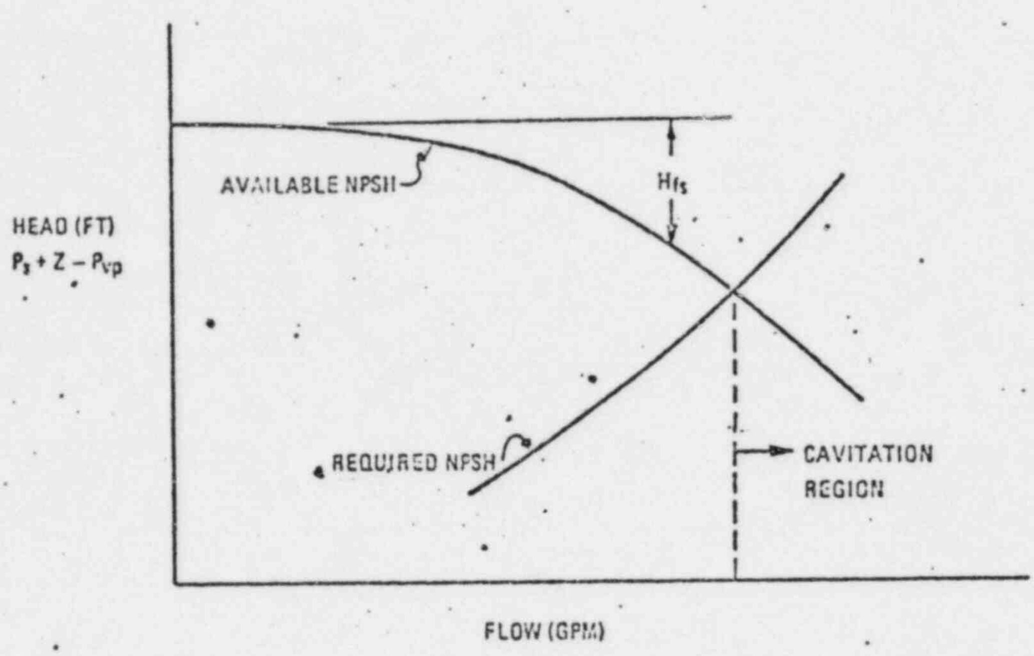
Note that Ps was converted to feet of head using the value of specific volume at the pump suction. Additionally, Z_1 and Z_2 were re-evaluated to give equivalent height at pump suction conditions.

Required NPSH is peculiar to individual pumps and depends on the pump design. The NPSH for the pump is usually determined experimentally and is supplied by the manufacturer. The available NPSH must always be greater than the required NPSH. If it isn't, the pump starts to cavitate. What is cavitation? When the available NPSH is less than the required NPSH, the vapor pressure is greater than the total suction pressure and part of the liquid flashes into vapor; in our case, the vapor is steam.

This vapor displaces an equal amount of fluid, reducing the capacity of the impeller. As the bubbles of vapor reach areas of the impeller where the pressure is increasing, they collapse violently, causing noise, vibration, and sometimes metal damage.

The friction terms in the available NPSH depend on the flow rate. As the flow increases, the friction head gets more negative. This is demonstrated in Figure 3B. When the system and pump flow is such that the required available NPSH curves cross, cavitation occurs.

FIGURE 3B. CAVITATION



Knowing these terms and understanding the principle of a diffuser, we can proceed with our discussion of pumps.

(1) Centrifugal Pumps

Centrifugal pumps, as their name implies, depend for their operation on centrifugal force. This force is generated by the high speed rotation of an impeller, the only moving part of the pump. Essentially, the impeller consists of a shaft upon which are mounted radially a series of curved vanes with suitable means providing for the entry of the liquid at the "eye" or

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center of the impeller. When positive supply of the liquid being pumped is initially introduced at the "eye," the high speed rotation of the vanes literally throws the liquid outward, imparting to it a high velocity head. The pressure head of the liquid at this stage is relatively low as compared to that at discharge. As a result, almost all of the energy which the moving liquid possesses at this point is kinetic.

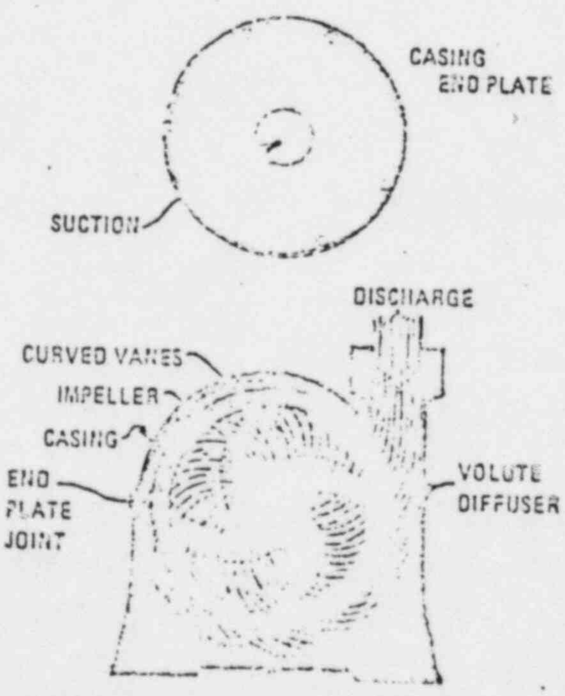

FIGURE 3C. CENTRIFUGAL PUMP

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Upon leaving the curved vanes of the impeller, the high velocity liquid enters the stationary pump casing which encloses the impeller. This section of the casing is termed the "volute," since its shape is derived from the involute of a circle. Referring to Figure 3C, the clearance between the vanes and casing increases as the liquid approaches the discharge connection to the casing. The purpose of this volute is to collect the fluid being discharged from the impeller at a high velocity and gradually bring it to a relatively low velocity, thus converting a large part of the velocity head to static pressure.

The volute is a variation of the straight diffuser illustrated previously in which the straight nozzle has been changed to a nozzle whose axis is bent in a 360° circle. Knowing how the diffuser functions will help to understand the following characteristics of the centrifugal pump: (1) opening the discharge valve wider with no change in speed of rotation will result in a corresponding lowering of the discharge pressure; similarly, closing the discharge will cause an increase in pressure; (2) if the discharge valve of the pump is closed completely, the pressure will attain a certain value, called the shut-off head, after which the impeller will churn and heat the fluid without a further increase in pressure; and (3) increasing the speed of rotation, with a constant discharge opening, will result in an increase in pressure, and decreasing the speed will bring about a lowering of the pressure.

Another characteristic of the centrifugal pump which must always be kept in mind is that the pump is not self-priming. In other words, the pump casing must be flooded before it will function.

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To attain this, most centrifugal pumps are located below the level of the source from which they are to take suction. The same effect can be gained by supplying water to the pump suction under pressure supplied by another pump placed in the suction line.

There is a practical limit in discharge head beyond which it is not economical to use a centrifugal pump having a single impeller. For the highest discharge heads, either a very high rotative speed or a large diameter of impeller must be used. Both of these lead to high mechanical stresses and to lowered efficiency, due primarily to friction and leakage losses. Impeller friction increases very rapidly with increased speed or increased diameter. Going to a higher speed and smaller impeller would give less friction for the same head. However, leakage loss, due to liquid passing back from discharge to suction through clearance spaces, increases with the smaller diameter. At high heads, with the greater pressure difference between discharge and suction, the leakage loss is an important item. As a result of these effects, the efficiency of a high-head pump of the single impeller type is likely to be comparatively low. While single pumps have been built which have delivered a head of 1200 feet or more, practical designs limit the head in many cases to 650 feet per impeller. For the higher heads, two or more impellers are connected in series, the discharge from one impeller being connected to the suction of the next. The total head is the head of one impeller multiplied by the number of impellers. For the sake of economy such an arrangement is built in one casing and is known as a multi-stage pump, each impeller being a stage.

The table below lists some characteristics of the main centrifugal pumps in use in your plant.

Duty	Stages	Head	RPM
Recirculation Pump	1	370 ft.	1800
Feed Pump	3	2620 ft.	3600
Condensate Pump	9	1030 ft.	1200

The condensate pump has a higher than normal number of stages because of its slow speed. Its speed must be kept slow because of the severe available NPSH conditions at the main condenser. Internal pump velocities are kept extremely low to prevent local static pressures from falling below the vapor pressure and causing cavitation.

(2) Jet Pumps

For those of you with steam plant experience, you have already been exposed to jet pumps. The most common use of such a pump is to maintain the vacuum in a condenser by removing non-condensible gases. This is the steam jet air ejector. In this case, steam is the driving force but of course other fluids can be used.

How does a jet pump device work. Let's make use of some of that groundwork that we laid. We said that the total head could be written as the sum of the static pressure head and the velocity head. If we apply that principle to a nozzle, we can begin to explain how a jet pump works.

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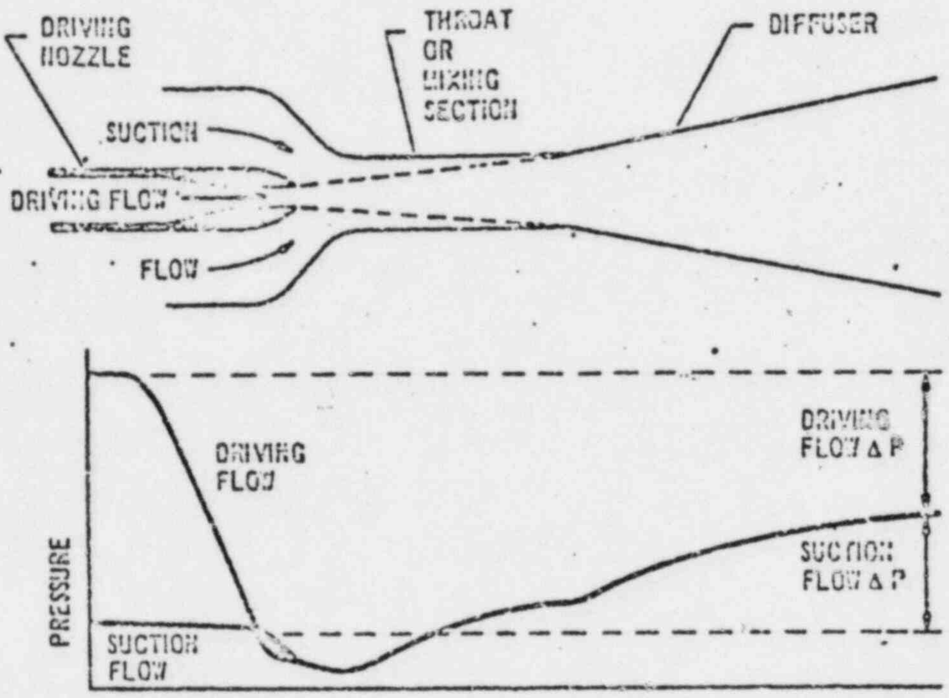

FIGURE 3D. JET PUMP SCHEMATIC

Here is a cross section view of a jet pump. It is made up of three major sections--the nozzle, the throat, and the diffuser. The driving fluid enters the nozzle at a high pressure, is accelerated through the nozzle, and what happens? The velocity head increases and the static pressure drops in order to satisfy our equation. If the increase in velocity is influenced, the static pressure at the outlet of the nozzle can be made less than the pressure of the fluid we want to pump. As soon as the pressure

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in the throat falls below suction pressure, we get flow. As the driving fluid and the pumped fluid pass through the throat and are mixed, we get some increase in static pressure as the velocity of the two fluids stabilizes. Note also that as the suction flow increases and passes through the suction nozzle, its static pressure is reduced to match that existing at the throat entrance. Downstream of the throat, the mixture enters the diffuser and, as we know from the first segment, is further slowed with a resultant increase in static pressure.

The suction flow through the jet pump is dependent upon the driving flow pressure. Normally, the suction flow is at least equal to the driving flow and can be as much as twice the driving flow depending upon the pump design.

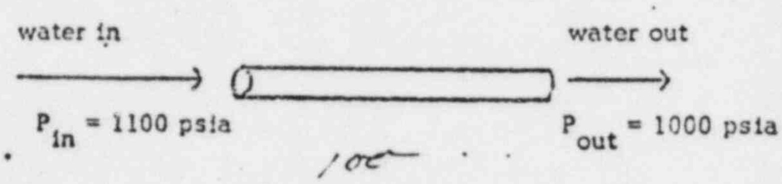
The efficiency of a jet pump is less than that of a centrifugal pump, but its operational simplicity and the lack of moving parts outweigh this deficiency.

As with other components that have fluid flowing through them, jet pumps can be calibrated so that flow can be directly measured.

Problem Set - Chapter 3

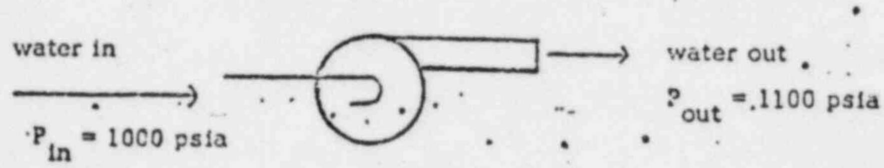
1. Find the pressure drop:

(a)



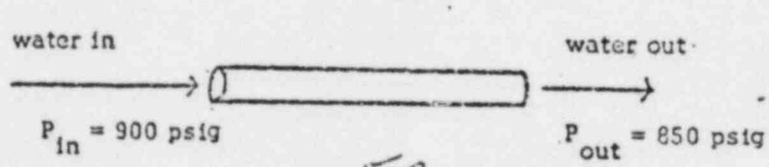
100

(b)



— 100

(c)



50

2. The water referred to in problem 1 is saturated water. Convert the pressure drops to head losses.

Head loss for a. is $100 \frac{lb}{in^2} \times 144 \frac{in^2}{ft^2} \times .022 \frac{ft^3}{lb} = 316.8$ ft
b & c [illegible] similar manner

3. Define laminar flow.

a smooth steady stream line flow no irregularities.

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4. Define turbulent flow. Turbulent is irregular, non uniform mixed flow unsteady.

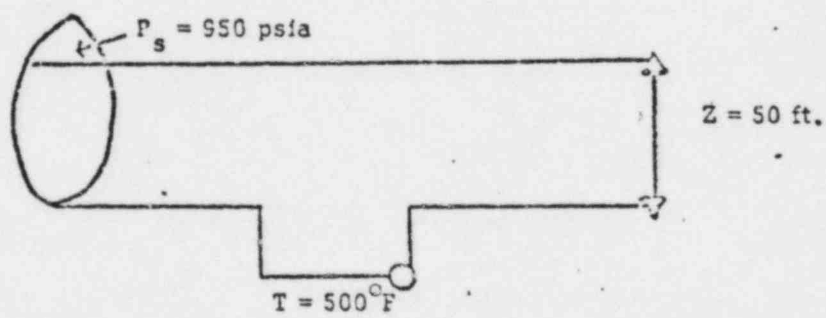
5. At approximately what Reynolds Number does laminar flow turn into turbulent flow? R_e = 2000 to 4000

6. If the flow of saturated water is measured at 500°F using curves appropriate to 68°F, what will be the percentage error in the flow rate if no temperature corrections are made?

$$V_s = \frac{1}{\rho} \qquad \frac{W_{actual}}{W_{curve}} = \sqrt{\frac{\rho_{act\ 500°F}}{\rho_{curve\ 68°F}}}$$

7. What do the letters NPSH stand for?

8. What is the available NPSH?



(no friction losses)

$$NPSH = (P_s + Z - h_{fs}) - P_{vp} = 997 \ldots$$

$$P_s = 950\ psia\ \frac{144\ in^2}{ft^2} \times .0214\ \frac{ft^3}{lb}$$

$$P_s = 2928\ FT$$

$$Z = 50\ FT$$

$$P_{vp} = 681\ \frac{lb}{in^2} \times \frac{144\ in^2}{ft} \quad .0204\ \frac{ft^3}{lb}$$

$$= 2001\ FT$$

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CONTENTS

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Table 1. Saturated Steam: Temperature Table

Temp Fahr t	Abs Press. Lb per Sq In. p	Specific Volume Sat. Liquid v_f	Specific Volume Evap v_{fg}	Specific Volume Sat. Vapor v_g	Enthalpy Sat. Liquid h_f	Enthalpy Evap h_{fg}	Enthalpy Sat. Vapor h_g	Entropy Sat. Liquid s_f	Entropy Evap s_{fg}	Entropy Sat. Vapor s_g	Temp Fahr t
32.0	0.08859	0.016022	3304.7	3304.7	-0.0179	1075.5	1075.5	0.0000	2.1873	2.1873	32.0
34.0	0.09600	0.016021	3061.9	3061.9	1.996	1074.4	1076.4	0.0041	2.1762	2.1802	34.0
36.0	0.10395	0.016020	2839.0	2839.0	4.008	1073.2	1077.2	0.0081	2.1651	2.1732	36.0
38.0	0.11249	0.016019	2634.1	2634.2	6.018	1072.1	1078.1	0.0122	2.1541	2.1663	38.0
40.0	0.12163	0.016019	2445.8	2445.8	8.027	1071.0	1079.0	0.0162	2.1432	2.1594	40.0
42.0	0.13143	0.016019	2272.4	2272.4	10.035	1069.8	1079.9	0.0202	2.1325	2.1527	42.0
44.0	0.14192	0.016019	2112.8	2112.8	12.041	1068.7	1080.7	0.0242	2.1217	2.1459	44.0
46.0	0.15314	0.016020	1965.7	1965.7	14.047	1067.6	1081.6	0.0282	2.1111	2.1393	46.0
48.0	0.16514	0.016021	1830.0	1830.0	16.051	1066.4	1082.5	0.0321	2.1006	2.1327	48.0
50.0	0.17796	0.016023	1704.8	1704.8	18.054	1065.3	1083.4	0.0361	2.0901	2.1262	50.0
52.0	0.19165	0.016024	1589.2	1589.2	20.057	1064.2	1084.2	0.0400	2.0798	2.1197	52.0
54.0	0.20625	0.016026	1482.4	1482.4	22.059	1063.1	1085.1	0.0439	2.0695	2.1134	54.0
56.0	0.22183	0.016028	1383.6	1383.6	24.059	1061.9	1086.0	0.0478	2.0593	2.1070	56.0
58.0	0.23843	0.016031	1292.2	1292.2	26.060	1060.8	1086.9	0.0516	2.0491	2.1008	58.0
60.0	0.25611	0.016033	1207.6	1207.6	28.060	1059.7	1087.7	0.0555	2.0391	2.0946	60.0
62.0	0.27494	0.016036	1129.2	1129.2	30.059	1058.5	1088.6	0.0593	2.0291	2.0885	62.0
64.0	0.29497	0.016039	1055.5	1056.5	32.058	1057.4	1089.5	0.0632	2.0192	2.0824	64.0
66.0	0.31626	0.016043	989.0	989.1	34.056	1056.3	1090.4	0.0670	2.0094	2.0764	66.0
68.0	0.33889	0.016046	926.5	926.5	36.054	1055.2	1091.2	0.0708	1.9996	2.0704	68.0
70.0	0.36292	0.016050	868.3	868.4	38.052	1054.0	1092.1	0.0745	1.9900	2.0645	70.0
72.0	0.38844	0.016054	814.3	814.3	40.049	1052.9	1093.0	0.0783	1.9804	2.0587	72.0
74.0	0.41550	0.016058	764.1	764.1	42.046	1051.8	1093.8	0.0821	1.9708	2.0529	74.0
76.0	0.44420	0.016063	717.4	717.4	44.043	1050.7	1094.7	0.0858	1.9614	2.0472	76.0
78.0	0.47461	0.016067	673.8	673.9	46.040	1049.5	1095.6	0.0895	1.9520	2.0415	78.0
80.0	0.50683	0.016072	633.2	633.3	48.037	1048.4	1096.4	0.0932	1.9426	2.0359	80.0
82.0	0.54093	0.016077	595.5	595.5	50.033	1047.3	1097.3	0.0969	1.9334	2.0303	82.0
84.0	0.57702	0.016082	560.3	560.3	52.029	1046.1	1098.2	0.1006	1.9242	2.0248	84.0
86.0	0.61518	0.016087	527.5	527.5	54.026	1045.0	1099.0	0.1043	1.9151	2.0193	86.0
88.0	0.65551	0.016093	496.8	496.8	56.022	1043.9	1099.9	0.1079	1.9060	2.0139	88.0
90.0	0.69813	0.016099	468.1	468.1	58.018	1042.7	1100.8	0.1115	1.8970	2.0086	90.0
92.0	0.74313	0.016105	441.3	441.3	60.014	1041.6	1101.6	0.1152	1.8881	2.0033	92.0
94.0	0.79062	0.016111	416.3	416.3	62.010	1040.5	1102.5	0.1188	1.8792	1.9980	94.0
96.0	0.84072	0.016117	392.8	392.9	64.006	1039.3	1103.3	0.1224	1.8704	1.9928	96.0
98.0	0.89356	0.016123	370.9	370.9	66.003	1038.2	1104.2	0.1260	1.8617	1.9876	98.0
100.0	0.94924	0.016130	350.4	350.4	67.999	1037.1	1105.1	0.1295	1.8530	1.9825	100.0
102.0	1.00789	0.016137	331.1	331.1	69.995	1035.9	1105.9	0.1331	1.8444	1.9775	102.0
104.0	1.06965	0.016144	313.1	313.1	71.992	1034.8	1106.8	0.1366	1.8358	1.9725	104.0
106.0	1.1347	0.016151	296.16	296.18	73.99	1033.6	1107.6	0.1402	1.8273	1.9675	106.0
108.0	1.2030	0.016158	280.28	280.30	75.98	1032.5	1108.5	0.1437	1.8188	1.9626	108.0
110.0	1.2750	0.016165	265.37	265.39	77.98	1031.4	1109.3	0.1472	1.8105	1.9577	110.0
112.0	1.3505	0.016173	251.37	251.38	79.98	1030.2	1110.2	0.1507	1.8021	1.9528	112.0
114.0	1.4299	0.016180	238.21	238.22	81.97	1029.1	1111.0	0.1542	1.7938	1.9480	114.0
116.0	1.5133	0.016188	225.84	225.85	83.97	1027.9	1111.9	0.1577	1.7856	1.9433	116.0
118.0	1.6009	0.016196	214.20	214.21	85.97	1026.8	1112.7	0.1611	1.7774	1.9386	118.0
120.0	1.6927	0.016204	203.25	203.26	87.97	1025.6	1113.6	0.1646	1.7693	1.9339	120.0
122.0	1.7891	0.016213	192.94	192.95	89.96	1024.5	1114.4	0.1680	1.7613	1.9293	122.0
124.0	1.8901	0.016221	183.23	183.24	91.96	1023.3	1115.3	0.1715	1.7533	1.9247	124.0
126.0	1.9959	0.016229	174.08	174.09	93.96	1022.2	1116.1	0.1749	1.7453	1.9202	126.0
128.0	2.1068	0.016238	165.45	165.47	95.96	1021.0	1117.0	0.1783	1.7374	1.9157	128.0
130.0	2.2230	0.016247	157.32	157.33	97.96	1019.8	1117.8	0.1817	1.7295	1.9112	130.0
132.0	2.3445	0.016256	149.64	149.66	99.95	1018.7	1118.6	0.1851	1.7217	1.9068	132.0
134.0	2.4717	0.016265	142.40	142.41	101.95	1017.5	1119.5	0.1884	1.7140	1.9024	134.0
136.0	2.6047	0.016274	135.55	135.57	103.95	1016.4	1120.3	0.1918	1.7063	1.8980	136.0
138.0	2.7438	0.016284	129.09	129.11	105.95	1015.2	1121.1	0.1951	1.6985	1.8937	138.0
140.0	2.8892	0.016293	122.98	123.00	107.95	1014.0	1122.0	0.1985	1.6910	1.8895	140.0
142.0	3.0411	0.016303	117.21	117.22	109.95	1012.9	1122.8	0.2018	1.6834	1.8852	142.0
144.0	3.1997	0.016312	111.74	111.76	111.95	1011.7	1123.6	0.2051	1.6759	1.8810	144.0
146.0	3.3653	0.016322	106.58	106.59	113.95	1010.5	1124.5	0.2084	1.6684	1.8769	146.0
148.0	3.5381	0.016332	101.68	101.70	115.95	1009.3	1125.3	0.2117	1.6610	1.8727	148.0
150.0	3.7184	0.016343	97.05	97.07	117.95	1008.2	1126.1	0.2150	1.6536	1.8686	150.0
152.0	3.9065	0.016353	92.66	92.68	119.95	1007.0	1126.9	0.2183	1.6463	1.8646	152.0
154.0	4.1025	0.016363	88.50	88.52	121.95	1005.8	1127.7	0.2216	1.6390	1.8606	154.0
156.0	4.3068	0.016374	84.56	84.57	123.95	1004.6	1128.6	0.2248	1.6318	1.8566	156.0
158.0	4.5197	0.016384	80.82	80.83	125.96	1003.4	1129.4	0.2281	1.6245	1.8526	158.0
160.0	4.7414	0.016395	77.27	77.29	127.96	1002.2	1130.2	0.2313	1.6174	1.8487	160.0
162.0	4.9722	0.016406	73.90	73.92	129.96	1001.0	1131.0	0.2345	1.6103	1.8448	162.0
164.0	5.2124	0.016417	70.70	70.72	131.96	999.8	1131.8	0.2377	1.6032	1.8409	164.0
166.0	5.4623	0.016428	67.67	67.68	133.97	998.6	1132.6	0.2409	1.5961	1.8371	166.0
168.0	5.7223	0.016440	64.78	64.80	135.97	997.4	1133.4	0.2441	1.5892	1.8333	168.0
170.0	5.9926	0.016451	62.04	62.06	137.97	996.2	1134.2	0.2473	1.5822	1.8295	170.0
172.0	6.2736	0.016463	59.43	59.45	139.98	995.0	1135.0	0.2505	1.5753	1.8258	172.0
174.0	6.5656	0.016474	56.95	56.97	141.98	993.8	1135.8	0.2537	1.5684	1.8221	174.0
176.0	6.8690	0.016486	54.59	54.61	143.99	992.6	1136.5	0.2568	1.5616	1.8184	176.0
178.0	7.1840	0.016498	52.35	52.36	145.99	991.4	1137.3	0.2600	1.5548	1.8147	178.0

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Table 1. Saturated Steam: Temperature Table—*Continued*

Temp Fahr t	Abs Press Lb per Sq In p	Specific Volume Sat Liquid v_f	Specific Volume Evap v_{fg}	Specific Volume Sat Vapor v_g	Enthalpy Sat Liquid h_f	Enthalpy Evap h_{fg}	Enthalpy Sat Vapor h_g	Entropy Sat Liquid s_f	Entropy Evap s_{fg}	Entropy Sat Vapor s_g	Temp Fahr t
180.0	7.5110	0.016510	50.21	50.22	148.00	990.2	1138.2	0.2631	1.5480	1.8111	180.0
182.0	7.850	0.016522	48.172	48.189	150.01	989.0	1139.0	0.2662	1.5413	1.8075	182.0
184.0	8.203	0.016534	46.232	46.249	152.01	987.8	1139.8	0.2694	1.5346	1.8040	184.0
186.0	8.568	0.016547	44.383	44.400	154.02	986.5	1140.5	0.2725	1.5279	1.8004	186.0
188.0	8.947	0.016559	42.621	42.638	156.03	985.3	1141.3	0.2756	1.5213	1.7969	188.0
190.0	9.340	0.016572	40.941	40.957	158.04	984.1	1142.1	0.2787	1.5148	1.7934	190.0
192.0	9.747	0.016585	39.337	39.354	160.05	982.8	1142.9	0.2818	1.5082	1.7900	192.0
194.0	10.168	0.016598	37.808	37.824	162.05	981.6	1143.7	0.2848	1.5017	1.7865	194.0
196.0	10.605	0.016611	36.348	36.364	164.06	980.4	1144.4	0.2879	1.4952	1.7831	196.0
198.0	11.058	0.016624	34.954	34.970	166.08	979.1	1145.2	0.2910	1.4888	1.7798	198.0
200.0	11.526	0.016637	33.622	33.639	168.09	977.9	1146.0	0.2940	1.4824	1.7764	200.0
204.0	12.512	0.016664	31.135	31.151	172.11	975.4	1147.5	0.3001	1.4697	1.7698	204.0
208.0	13.568	0.016691	28.862	28.878	176.14	972.8	1149.0	0.3061	1.4571	1.7632	208.0
212.0	14.696	0.016719	26.782	26.799	180.17	970.3	1150.5	0.3121	1.4447	1.7568	212.0
216.0	15.901	0.016747	24.878	24.894	184.20	967.8	1152.0	0.3181	1.4323	1.7505	216.0
220.0	17.186	0.016775	23.131	23.148	188.23	965.2	1153.4	0.3241	1.4201	1.7442	220.0
224.0	18.556	0.016805	21.529	21.545	192.27	962.6	1154.9	0.3300	1.4081	1.7380	224.0
228.0	20.015	0.016834	20.056	20.073	196.31	960.0	1156.3	0.3359	1.3961	1.7320	228.0
232.0	21.567	0.016864	18.701	18.718	200.35	957.4	1157.8	0.3417	1.3842	1.7260	232.0
236.0	23.216	0.016895	17.454	17.471	204.40	954.8	1159.2	0.3476	1.3725	1.7201	236.0
240.0	24.968	0.016926	16.304	16.321	208.45	952.1	1160.6	0.3533	1.3609	1.7142	240.0
244.0	26.826	0.016958	15.243	15.260	212.50	949.5	1162.0	0.3591	1.3494	1.7085	244.0
248.0	28.796	0.016990	14.264	14.281	216.56	946.8	1163.4	0.3649	1.3379	1.7028	248.0
252.0	30.883	0.017022	13.358	13.375	220.62	944.1	1164.7	0.3706	1.3266	1.6972	252.0
256.0	33.091	0.017055	12.520	12.538	224.69	941.4	1166.1	0.3763	1.3154	1.6917	256.0
260.0	35.427	0.017089	11.745	11.762	228.76	938.6	1167.4	0.3819	1.3043	1.6862	260.0
264.0	37.894	0.017123	11.025	11.042	232.83	935.9	1168.7	0.3876	1.2933	1.6808	264.0
268.0	40.500	0.017157	10.358	10.375	236.91	933.1	1170.0	0.3932	1.2823	1.6755	268.0
272.0	43.249	0.017193	9.738	9.755	240.99	930.3	1171.3	0.3987	1.2715	1.6702	272.0
276.0	46.147	0.017228	9.162	9.180	245.08	927.5	1172.5	0.4043	1.2607	1.6650	276.0
280.0	49.200	0.017264	8.627	8.644	249.17	924.6	1173.8	0.4098	1.2501	1.6599	280.0
284.0	52.414	0.01730	8.1280	8.1453	253.3	921.7	1175.0	0.4154	1.2395	1.6549	284.0
288.0	55.795	0.01734	7.6634	7.6807	257.4	918.8	1176.2	0.4208	1.2290	1.6498	288.0
292.0	59.350	0.01738	7.2301	7.2475	261.5	915.9	1177.4	0.4263	1.2186	1.6449	292.0
296.0	63.084	0.01741	6.8259	6.8433	265.6	913.0	1178.6	0.4317	1.2082	1.6400	296.0
300.0	67.005	0.01745	6.4483	6.4658	269.7	910.0	1179.7	0.4372	1.1979	1.6351	300.0
304.0	71.119	0.01749	6.0955	6.1130	273.8	907.0	1180.9	0.4426	1.1877	1.6303	304.0
308.0	75.433	0.01753	5.7655	5.7830	278.0	904.0	1182.0	0.4479	1.1776	1.6256	308.0
312.0	79.953	0.01757	5.4566	5.4742	282.1	901.0	1183.1	0.4533	1.1676	1.6209	312.0
316.0	84.668	0.01761	5.1673	5.1849	286.3	897.9	1184.1	0.4586	1.1576	1.6162	316.0
320.0	89.643	0.01766	4.8961	4.9138	290.4	894.8	1185.2	0.4640	1.1477	1.6116	320.0
324.0	94.826	0.01770	4.6418	4.6595	294.6	891.6	1186.2	0.4692	1.1378	1.6071	324.0
328.0	100.245	0.01774	4.4030	4.4208	298.7	888.5	1187.2	0.4745	1.1280	1.6025	328.0
332.0	105.907	0.01779	4.1788	4.1966	302.9	885.3	1188.2	0.4798	1.1183	1.5981	332.0
336.0	111.820	0.01783	3.9681	3.9859	307.1	882.1	1189.1	0.4850	1.1086	1.5936	336.0
340.0	117.992	0.01787	3.7699	3.7878	311.3	878.8	1190.1	0.4902	1.0990	1.5892	340.0
344.0	124.430	0.01792	3.5834	3.6013	315.5	875.5	1191.0	0.4954	1.0894	1.5849	344.0
348.0	131.142	0.01797	3.4078	3.4258	319.7	872.2	1191.9	0.5006	1.0799	1.5806	348.0
352.0	138.138	0.01801	3.2423	3.2603	323.9	868.9	1192.7	0.5058	1.0705	1.5763	352.0
356.0	145.424	0.01806	3.0863	3.1044	328.1	865.5	1193.6	0.5110	1.0611	1.5721	356.0
360.0	153.010	0.01811	2.9392	2.9573	332.3	862.1	1194.4	0.5161	1.0517	1.5678	360.0
364.0	160.903	0.01816	2.8002	2.8184	336.5	858.6	1195.2	0.5212	1.0424	1.5637	364.0
368.0	169.113	0.01821	2.6691	2.6873	340.8	855.1	1195.9	0.5263	1.0332	1.5595	368.0
372.0	177.648	0.01826	2.5451	2.5633	345.0	851.6	1196.7	0.5314	1.0240	1.5554	372.0
376.0	186.517	0.01831	2.4279	2.4462	349.3	848.1	1197.4	0.5365	1.0148	1.5513	376.0
380.0	195.729	0.01836	2.3170	2.3353	353.6	844.5	1198.0	0.5416	1.0057	1.5473	380.0
384.0	205.294	0.01842	2.2120	2.2304	357.9	840.8	1198.7	0.5466	0.9966	1.5432	384.0
388.0	215.220	0.01847	2.1126	2.1311	362.2	837.2	1199.3	0.5516	0.9876	1.5392	388.0
392.0	225.516	0.01853	2.0184	2.0369	366.5	833.4	1199.9	0.5567	0.9786	1.5352	392.0
396.0	236.193	0.01858	1.9291	1.9477	370.8	829.7	1200.4	0.5617	0.9696	1.5313	396.0
400.0	247.259	0.01864	1.8444	1.8630	375.1	825.9	1201.0	0.5667	0.9607	1.5274	400.0
404.0	258.725	0.01870	1.7640	1.7827	379.4	822.0	1201.5	0.5717	0.9518	1.5234	404.0
408.0	270.600	0.01875	1.6877	1.7064	383.8	818.2	1201.9	0.5766	0.9429	1.5195	408.0
412.0	282.894	0.01881	1.6152	1.6340	388.1	814.2	1202.4	0.5816	0.9341	1.5157	412.0
416.0	295.617	0.01887	1.5463	1.5651	392.5	810.2	1202.8	0.5866	0.9253	1.5118	416.0
420.0	308.780	0.01894	1.4808	1.4997	396.9	806.2	1203.1	0.5915	0.9165	1.5080	420.0
424.0	322.391	0.01900	1.4184	1.4374	401.3	802.2	1203.5	0.5964	0.9077	1.5042	424.0
428.0	336.463	0.01906	1.3591	1.3782	405.7	798.0	1203.7	0.6014	0.8990	1.5004	428.0
432.0	351.00	0.01913	1.30266	1.32179	410.1	793.9	1204.0	0.6063	0.8903	1.4966	432.0
436.0	366.03	0.01919	1.24887	1.26806	414.6	789.7	1204.2	0.6112	0.8816	1.4928	436.0
440.0	381.54	0.01926	1.19761	1.21687	419.0	785.4	1204.4	0.6161	0.8729	1.4890	440.0
444.0	397.56	0.01933	1.14874	1.16806	423.5	781.1	1204.6	0.6210	0.8643	1.4853	444.0
448.0	414.09	0.01940	1.10212	1.12152	428.0	776.7	1204.7	0.6259	0.8557	1.4815	448.0
452.0	431.14	0.01947	1.05764	1.07711	432.5	772.3	1204.8	0.6308	0.8471	1.4778	452.0
456.0	448.73	0.01954	1.01518	1.03472	437.0	767.8	1204.8	0.6356	0.8385	1.4741	456.0

Table 1. Saturated Steam: Temperature Table—*Continued*

Temp Fahr t	Abs Press Lb per Sq In. p	Specific Volume Sat. Liquid v_f	Evap v_{fg}	Sat. Vapor v_g	Enthalpy Sat. Liquid h_f	Evap h_{fg}	Sat. Vapor h_g	Entropy Sat. Liquid s_f	Evap s_{fg}	Sat. Vapor s_g	Temp Fahr t
460.0	466.87	0.01961	0.97463	0.99424	441.5	763.2	1204.8	0.6405	0.8299	1.4704	460.0
464.0	485.56	0.01969	0.93588	0.95557	446.1	758.6	1204.7	0.6454	0.8213	1.4667	464.0
468.0	504.83	0.01976	0.89885	0.91862	450.7	754.0	1204.6	0.6502	0.8127	1.4629	468.0
472.0	524.67	0.01984	0.86345	0.88329	455.2	749.3	1204.5	0.6551	0.8042	1.4592	472.0
476.0	545.11	0.01992	0.82958	0.84950	459.9	744.5	1204.3	0.6599	0.7956	1.4555	476.0
480.0	566.15	0.02000	0.79716	0.81717	464.5	739.6	1204.1	0.6648	0.7871	1.4518	480.0
484.0	587.81	0.02009	0.76613	0.78622	469.1	734.7	1203.8	0.6696	0.7785	1.4481	484.0
488.0	610.10	0.02017	0.73641	0.75658	473.8	729.7	1203.5	0.6745	0.7700	1.4444	488.0
492.0	633.03	0.02026	0.70794	0.72820	478.5	724.6	1203.1	0.6793	0.7614	1.4407	492.0
496.0	656.61	0.02034	0.68065	0.70100	483.2	719.5	1202.7	0.6842	0.7528	1.4370	496.0
500.0	680.86	0.02043	0.65448	0.67492	487.9	714.3	1202.2	0.6890	0.7443	1.4333	500.0
504.0	705.78	0.02053	0.62938	0.64991	492.7	709.0	1201.7	0.6939	0.7357	1.4296	504.0
508.0	731.40	0.02062	0.60530	0.62592	497.5	703.7	1201.1	0.6987	0.7271	1.4258	508.0
512.0	757.72	0.02072	0.58218	0.60289	502.3	698.2	1200.5	0.7036	0.7185	1.4221	512.0
516.0	784.76	0.02081	0.55997	0.58079	507.1	692.7	1199.8	0.7085	0.7099	1.4183	516.0
520.0	812.53	0.02091	0.53864	0.55956	512.0	687.0	1199.0	0.7133	0.7013	1.4146	520.0
524.0	841.04	0.02102	0.51814	0.53916	516.9	681.3	1198.2	0.7182	0.6926	1.4108	524.0
528.0	870.31	0.02112	0.49843	0.51955	521.8	675.5	1197.3	0.7231	0.6839	1.4070	528.0
532.0	900.34	0.02123	0.47947	0.50070	526.8	669.6	1196.4	0.7280	0.6752	1.4032	532.0
536.0	931.17	0.02134	0.46123	0.48257	531.7	663.6	1195.4	0.7329	0.6665	1.3993	536.0
540.0	962.79	0.02146	0.44367	0.46513	536.8	657.5	1194.3	0.7378	0.6577	1.3954	540.0
544.0	995.22	0.02157	0.42677	0.44834	541.8	651.3	1193.1	0.7427	0.6489	1.3915	544.0
548.0	1028.49	0.02169	0.41048	0.43217	546.9	645.0	1191.9	0.7476	0.6400	1.3876	548.0
552.0	1062.59	0.02182	0.39479	0.41660	552.0	638.5	1190.6	0.7525	0.6311	1.3837	552.0
556.0	1097.55	0.02194	0.37966	0.40160	557.2	632.0	1189.2	0.7575	0.6222	1.3797	556.0
560.0	1133.38	0.02207	0.36507	0.38714	562.4	625.3	1187.7	0.7625	0.6132	1.3757	560.0
564.0	1170.10	0.02221	0.35099	0.37320	567.6	618.5	1186.1	0.7674	0.6041	1.3716	564.0
568.0	1207.72	0.02235	0.33741	0.35975	572.9	611.5	1184.5	0.7725	0.5950	1.3675	568.0
572.0	1246.26	0.02249	0.32429	0.34678	578.3	604.5	1182.7	0.7775	0.5859	1.3634	572.0
576.0	1285.74	0.02264	0.31162	0.33426	583.7	597.2	1180.9	0.7825	0.5766	1.3592	576.0
580.0	1326.17	0.02279	0.29937	0.32216	589.1	589.9	1179.0	0.7876	0.5673	1.3550	580.0
584.0	1367.7	0.02295	0.28753	0.31048	594.6	582.4	1176.9	0.7927	0.5580	1.3507	584.0
588.0	1410.0	0.02311	0.27608	0.29919	600.1	574.7	1174.8	0.7978	0.5485	1.3464	588.0
592.0	1453.3	0.02328	0.26499	0.28827	605.7	566.8	1172.6	0.8030	0.5390	1.3420	592.0
596.0	1497.8	0.02345	0.25425	0.27770	611.4	558.8	1170.2	0.8082	0.5293	1.3375	596.0
600.0	1543.2	0.02364	0.24384	0.26747	617.1	550.6	1167.7	0.8134	0.5196	1.3330	600.0
604.0	1589.7	0.02382	0.23374	0.25757	622.9	542.2	1165.1	0.8187	0.5097	1.3284	604.0
608.0	1637.3	0.02402	0.22394	0.24796	628.8	533.6	1162.4	0.8240	0.4997	1.3238	608.0
612.0	1686.1	0.02422	0.21442	0.23865	634.8	524.7	1159.5	0.8294	0.4896	1.3190	612.0
616.0	1735.9	0.02444	0.20516	0.22960	640.8	515.6	1156.4	0.8348	0.4794	1.3141	616.0
620.0	1786.9	0.02466	0.19615	0.22081	646.9	506.3	1153.2	0.8403	0.4689	1.3092	620.0
624.0	1839.0	0.02489	0.18737	0.21226	653.1	496.6	1149.8	0.8458	0.4583	1.3041	624.0
628.0	1892.4	0.02514	0.17880	0.20394	659.5	486.7	1146.1	0.8514	0.4474	1.2988	628.0
632.0	1947.0	0.02539	0.17044	0.19583	665.9	476.4	1142.2	0.8571	0.4364	1.2934	632.0
636.0	2002.8	0.02566	0.16226	0.18792	672.4	465.7	1138.1	0.8628	0.4251	1.2879	636.0
640.0	2059.9	0.02595	0.15427	0.18021	679.1	454.6	1133.7	0.8686	0.4134	1.2821	640.0
644.0	2118.3	0.02625	0.14644	0.17269	685.9	443.1	1129.0	0.8746	0.4015	1.2761	644.0
648.0	2178.1	0.02657	0.13876	0.16534	692.9	431.1	1124.0	0.8806	0.3893	1.2699	648.0
652.0	2239.2	0.02691	0.13124	0.15816	700.0	418.7	1118.7	0.8868	0.3767	1.2634	652.0
656.0	2301.7	0.02728	0.12387	0.15115	707.4	405.7	1113.1	0.8931	0.3637	1.2567	656.0
660.0	2365.7	0.02768	0.11663	0.14431	714.9	392.1	1107.0	0.8995	0.3502	1.2498	660.0
664.0	2431.1	0.02811	0.10947	0.13757	722.9	377.7	1100.6	0.9064	0.3361	1.2425	664.0
668.0	2498.1	0.02858	0.10229	0.13087	731.5	362.1	1093.5	0.9137	0.3210	1.2347	668.0
672.0	2566.6	0.02911	0.09514	0.12424	740.2	345.7	1085.9	0.9212	0.3054	1.2266	672.0
676.0	2636.8	0.02970	0.08799	0.11769	749.2	328.5	1077.6	0.9287	0.2892	1.2179	676.0
680.0	2708.6	0.03037	0.08080	0.11117	758.5	310.1	1068.5	0.9365	0.2720	1.2086	680.0
684.0	2782.1	0.03114	0.07349	0.10463	768.2	290.2	1058.4	0.9447	0.2537	1.1984	684.0
688.0	2857.4	0.03204	0.06595	0.09799	778.8	268.2	1047.0	0.9535	0.2337	1.1872	688.0
692.0	2934.5	0.03313	0.05797	0.09110	790.5	243.1	1033.6	0.9634	0.2110	1.1744	692.0
696.0	3013.4	0.03455	0.04916	0.08371	804.4	212.8	1017.2	0.9749	0.1841	1.1591	696.0
700.0	3094.3	0.03662	0.03857	0.07519	822.4	172.7	995.2	0.9901	0.1490	1.1390	700.0
702.0	3135.5	0.03824	0.03173	0.06997	835.0	144.7	979.7	1.0006	0.1246	1.1252	702.0
704.0	3177.2	0.04108	0.02192	0.06300	854.2	102.0	956.2	1.0169	0.0876	1.1046	704.0
705.0	3198.3	0.04427	0.01304	0.05730	873.0	61.4	934.4	1.0329	0.0527	1.0856	705.0
705.47*	3208.2	0.05078	0.00000	0.05078	906.0	0.0	906.0	1.0612	0.0000	1.0612	705.47*

*Critical temperature

Table 2: Saturated Steam: Pressure Table

Abs Press. Lb/Sq In. p	Temp Fahr t	Specific Volume: Sat. Liquid v_f	Specific Volume: Evap v_{fg}	Specific Volume: Sat. Vapor v_g	Enthalpy: Sat. Liquid h_f	Enthalpy: Evap h_{fg}	Enthalpy: Sat. Vapor h_g	Entropy: Sat. Liquid s_f	Entropy: Evap s_{fg}	Entropy: Sat. Vapor s_g	Abs Press. Lb/Sq In. p
0.08865	32.018	0.016022	3302.4	3302.4	0.0003	1075.5	1075.5	0.0000	2.1872	2.1872	0.08865
0.25	59.323	0.016032	1235.5	1235.5	27.382	1060.1	1087.4	0.0542	2.0425	2.0967	0.25
0.50	79.586	0.016071	641.5	641.5	47.623	1048.6	1096.3	0.0925	1.9446	2.0370	0.50
1.0	101.74	0.016136	333.59	333.60	69.73	1036.1	1105.8	0.1326	1.8455	1.9781	1.0
5.0	162.24	0.016407	73.515	73.532	130.20	1000.9	1131.1	0.2349	1.6094	1.8443	5.0
10.0	193.21	0.016592	38.404	38.420	161.26	982.1	1143.3	0.2836	1.5043	1.7879	10.0
14.696	212.00	0.016719	26.782	26.799	180.17	970.3	1150.5	0.3121	1.4447	1.7568	14.696
15.0	213.03	0.016726	26.274	26.290	181.21	969.7	1150.9	0.3137	1.4415	1.7552	15.0
20.0	227.96	0.016834	20.070	20.087	196.27	960.1	1156.3	0.3358	1.3962	1.7320	20.0
30.0	250.34	0.017009	13.7266	13.7436	218.9	945.2	1164.1	0.3682	1.3313	1.6995	30.0
40.0	267.25	0.017151	10.4794	10.4965	236.1	933.6	1169.8	0.3921	1.2844	1.6765	40.0
50.0	281.02	0.017274	8.4967	8.5140	250.2	923.9	1174.1	0.4112	1.2474	1.6586	50.0
60.0	292.71	0.017383	7.1562	7.1736	262.2	915.4	1177.6	0.4273	1.2167	1.6440	60.0
70.0	302.93	0.017482	6.1875	6.2050	272.7	907.8	1180.6	0.4411	1.1905	1.6316	70.0
80.0	312.04	0.017573	5.4536	5.4711	282.1	900.9	1183.1	0.4534	1.1675	1.6208	80.0
90.0	320.28	0.017659	4.8779	4.8953	290.7	894.6	1185.3	0.4643	1.1470	1.6113	90.0
100.0	327.82	0.017740	4.4133	4.4310	298.5	888.6	1187.2	0.4743	1.1284	1.6027	100.0
110.0	334.79	0.01782	4.0306	4.0484	305.8	883.1	1188.9	0.4834	1.1115	1.5950	110.0
120.0	341.27	0.01789	3.7097	3.7275	312.6	877.8	1190.4	0.4919	1.0960	1.5879	120.0
130.0	347.33	0.01795	3.4364	3.4544	319.0	872.8	1191.7	0.4998	1.0815	1.5813	130.0
140.0	353.04	0.01803	3.2010	3.2190	325.0	868.0	1193.0	0.5071	1.0681	1.5752	140.0
150.0	358.43	0.01809	2.9958	3.0139	330.6	863.4	1194.1	0.5141	1.0554	1.5695	150.0
160.0	363.55	0.01815	2.8155	2.8336	336.1	859.0	1195.1	0.5206	1.0435	1.5641	160.0
170.0	368.42	0.01821	2.6556	2.6738	341.2	854.8	1196.0	0.5269	1.0322	1.5591	170.0
180.0	373.08	0.01827	2.5129	2.5312	346.2	850.7	1196.9	0.5328	1.0215	1.5543	180.0
190.0	377.53	0.01833	2.3847	2.4030	350.9	846.7	1197.6	0.5384	1.0113	1.5498	190.0
200.0	381.80	0.01839	2.2689	2.2873	355.5	842.8	1198.3	0.5438	1.0016	1.5454	200.0
210.0	385.91	0.01844	2.16373	2.18217	359.9	839.1	1199.0	0.5490	0.9923	1.5413	210.0
220.0	389.88	0.01850	2.06779	2.08629	364.2	835.4	1199.6	0.5540	0.9834	1.5374	220.0
230.0	393.70	0.01855	1.97991	1.99846	368.3	831.8	1200.1	0.5588	0.9748	1.5336	230.0
240.0	397.39	0.01860	1.89909	1.91769	372.3	828.4	1200.6	0.5634	0.9665	1.5299	240.0
250.0	400.97	0.01865	1.82452	1.84317	376.1	825.0	1201.1	0.5679	0.9585	1.5264	250.0
260.0	404.44	0.01870	1.75548	1.77418	379.9	821.6	1201.5	0.5722	0.9508	1.5230	260.0
270.0	407.80	0.01875	1.69137	1.71013	383.6	818.3	1201.9	0.5764	0.9433	1.5197	270.0
280.0	411.07	0.01880	1.63169	1.65049	387.1	815.1	1202.3	0.5805	0.9361	1.5166	280.0
290.0	414.25	0.01885	1.57597	1.59482	390.6	812.0	1202.6	0.5844	0.9291	1.5135	290.0
300.0	417.35	0.01889	1.52384	1.54274	394.0	808.9	1202.9	0.5882	0.9223	1.5105	300.0
350.0	431.73	0.01912	1.30642	1.32554	409.8	794.2	1204.0	0.6059	0.8909	1.4968	350.0
400.0	444.60	0.01934	1.14162	1.16055	424.2	780.4	1204.6	0.6217	0.8630	1.4847	400.0
450.0	456.28	0.01954	1.01224	1.03179	437.3	767.5	1204.8	0.6360	0.8378	1.4738	450.0
500.0	467.01	0.01975	0.90787	0.92762	449.5	755.1	1204.7	0.6490	0.8148	1.4639	500.0
550.0	476.94	0.01994	0.82183	0.84177	460.9	743.3	1204.3	0.6611	0.7936	1.4547	550.0
600.0	486.20	0.02013	0.74962	0.76975	471.7	732.0	1203.7	0.6723	0.7738	1.4461	600.0
650.0	494.89	0.02032	0.68811	0.70843	481.9	720.9	1202.8	0.6828	0.7552	1.4381	650.0
700.0	503.08	0.02050	0.63505	0.65556	491.6	710.2	1201.8	0.6928	0.7377	1.4304	700.0
750.0	510.84	0.02069	0.58880	0.60949	500.9	699.8	1200.7	0.7022	0.7210	1.4232	750.0
800.0	518.21	0.02087	0.54809	0.56896	509.8	689.6	1199.4	0.7111	0.7051	1.4163	800.0
850.0	525.24	0.02105	0.51197	0.53302	518.4	679.5	1198.0	0.7197	0.6899	1.4096	850.0
900.0	531.95	0.02123	0.47968	0.50091	526.7	669.7	1196.4	0.7279	0.6753	1.4032	900.0
950.0	538.39	0.02141	0.45064	0.47205	534.7	660.0	1194.7	0.7358	0.6612	1.3970	950.0
1000.0	544.58	0.02159	0.42436	0.44596	542.6	650.4	1192.9	0.7434	0.6476	1.3910	1000.0
1050.0	550.53	0.02177	0.40047	0.42224	550.1	640.9	1191.0	0.7507	0.6344	1.3851	1050.0
1100.0	556.28	0.02195	0.37863	0.40058	557.5	631.5	1189.1	0.7578	0.6216	1.3794	1100.0
1150.0	561.82	0.02214	0.35859	0.38073	564.8	622.2	1187.0	0.7647	0.6091	1.3738	1150.0
1200.0	567.19	0.02232	0.34013	0.36245	571.9	613.0	1184.8	0.7714	0.5969	1.3683	1200.0
1250.0	572.38	0.02250	0.32306	0.34556	578.8	603.8	1182.6	0.7780	0.5850	1.3630	1250.0
1300.0	577.42	0.02269	0.30722	0.32991	585.6	594.6	1180.2	0.7843	0.5733	1.3577	1300.0
1350.0	582.32	0.02288	0.29250	0.31537	592.3	585.4	1177.8	0.7906	0.5620	1.3525	1350.0
1400.0	587.07	0.02307	0.27871	0.30178	598.8	576.5	1175.3	0.7966	0.5507	1.3474	1400.0
1450.0	591.70	0.02327	0.26584	0.28911	605.3	567.4	1172.8	0.8026	0.5397	1.3423	1450.0
1500.0	596.20	0.02346	0.25372	0.27719	611.7	558.4	1170.1	0.8085	0.5288	1.3373	1500.0
1550.0	600.59	0.02366	0.24235	0.26601	618.0	549.4	1167.4	0.8142	0.5182	1.3324	1550.0
1600.0	604.87	0.02387	0.23159	0.25545	624.2	540.3	1164.5	0.8199	0.5076	1.3274	1600.0
1650.0	609.05	0.02407	0.22143	0.24551	630.4	531.3	1161.6	0.8254	0.4971	1.3225	1650.0
1700.0	613.13	0.02428	0.21178	0.23607	636.5	522.2	1158.6	0.8309	0.4867	1.3176	1700.0
1750.0	617.12	0.02450	0.20263	0.22713	642.5	513.1	1155.6	0.8363	0.4765	1.3128	1750.0
1800.0	621.02	0.02472	0.19390	0.21861	648.5	503.8	1152.3	0.8417	0.4662	1.3079	1800.0
1850.0	624.83	0.02495	0.18558	0.21052	654.5	494.6	1149.0	0.8470	0.4561	1.3030	1850.0
1900.0	628.56	0.02517	0.17761	0.20278	660.4	485.2	1145.6	0.8522	0.4459	1.2981	1900.0
1950.0	632.22	0.02541	0.16999	0.19540	666.3	475.8	1142.0	0.8574	0.4358	1.2931	1950.0
2000.0	635.80	0.02565	0.16266	0.18831	672.1	466.2	1138.3	0.8625	0.4256	1.2881	2000.0
2100.0	642.76	0.02615	0.14885	0.17501	683.8	446.7	1130.5	0.8727	0.4053	1.2780	2100.0
2200.0	649.45	0.02669	0.13603	0.16272	695.5	426.7	1122.2	0.8828	0.3848	1.2676	2200.0
2300.0	655.89	0.02727	0.12406	0.15133	707.2	406.0	1113.2	0.8929	0.3640	1.2569	2300.0
2400.0	662.11	0.02790	0.11287	0.14076	719.0	384.8	1103.7	0.9031	0.3430	1.2460	2400.0
2500.0	668.11	0.02859	0.10209	0.13068	731.7	361.6	1093.3	0.9139	0.3206	1.2345	2500.0
2600.0	673.91	0.02938	0.09172	0.12110	744.5	337.6	1082.0	0.9247	0.2977	1.2225	2600.0
2700.0	679.53	0.03029	0.08165	0.11194	757.3	312.3	1069.7	0.9356	0.2741	1.2097	2700.0
2800.0	684.96	0.03134	0.07171	0.10305	770.7	285.1	1055.8	0.9468	0.2491	1.1958	2800.0
2900.0	690.22	0.03262	0.06158	0.09420	785.1	254.7	1039.8	0.9588	0.2215	1.1803	2900.0
3000.0	695.33	0.03428	0.05073	0.08500	801.8	218.4	1020.3	0.9728	0.1891	1.1619	3000.0
3100.0	700.28	0.03681	0.03771	0.07452	824.0	169.3	993.3	0.9914	0.1460	1.1373	3100.0
3200.0	705.08	0.04472	0.01191	0.05663	875.5	56.1	931.6	1.0351	0.0482	1.0832	3200.0
3208.2*	705.47	0.05078	0.00000	0.05078	906.0	0.0	906.0	1.0612	0.0000	1.0612	3208.2*

*Critical pressure

1103 1500

Table 3. Superheated Steam

Abs Press Lb Sq In (Sat. Temp)		Sat. Water	Sat. Steam	200	250	300	350	400	450	500	600	700	800*	900	1000	1100	1200
1 (101.74)	Sh			98.26	148.26	198.26	248.26	298.26	348.26	398.26	498.26	598.26	698.26	798.26	898.26	998.26	1098.26
	v	0.01614	333.6	392.5	422.4	452.3	482.1	[illegible]	541.7	[illegible]	631.1	690.7	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	h	69.73	1105.8	1150.2	1172.9	1195.7	1218.7	[illegible]	1265.1	[illegible]	1336.1	1384.5	[illegible]	[illegible]	1531.4	1583.0	1635.4
	s	0.1326	1.9781	2.0509	2.0841	2.1152	2.1445	2.1722	2.1985	2.2237	2.2708	2.3144	[illegible]	[illegible]	2.4251	2.4592	2.4918
5 (162.24)	Sh			37.76	87.76	137.76	187.76	237.76	287.76	337.76	437.76	537.76	637.76	737.76	837.76	937.76	1037.76
	v	0.01641	73.53	78.14	84.21	90.24	96.25	102.24	108.23	[illegible]	126.15	138.03	150.01	161.94	173.85	185.78	197.70
	h	130.20	1131.1	1148.6	1171.7	1194.8	1218.0	1241.3	1264.7	[illegible]	1335.9	1384.3	1433.6	1483.7	1534.7	1586.7	1639.5
	s	0.2349	1.8443	1.8716	1.9054	1.9369	1.9664	1.9943	2.0208	2.0460	2.0932	2.1369	2.1776	2.2159	2.2521	2.2866	2.3194
10 (193.21)	Sh			6.79	56.79	106.79	156.79	206.79	256.79	306.79	406.79	506.79	606.79	706.79	806.79	906.79	1006.79
	v	0.01659	38.42	38.84	41.93	44.98	48.02	51.03	54.04	57.04	63.03	69.00	74.98	80.94	86.91	92.87	98.84
	h	161.26	1143.3	1146.6	1170.2	1193.7	1217.1	1240.6	1264.1	1287.8	1335.5	1384.0	1433.4	1483.5	1534.5	1586.6	1639.5
	s	0.2836	1.7875	1.7928	1.8273	1.8593	1.8892	1.9173	1.9439	1.9692	2.0166	2.0603	2.1011	2.1394	2.1757	2.2101	2.2430
14.696* (212.00)	Sh				38.00	88.00	138.00	188.00	238.00	288.00	388.00	488.00	588.00	688.00	788.00	888.00	988.00
	v	0.0167	26.828		28.44	30.52	32.61	34.65	36.73	38.75	42.83	46.91	50.97	55.03	59.09	63.19	67.25
	h	180.07	1150.4		1169.2	1192.0	1215.4	1239.5	1262.1	1286.4	1333.0	1381.4	1430.5	1480.4	1531.1	1582.7	1635.1
	s	0.3120	1.7566		1.7838	1.8148	1.8446	1.8727	1.8989	1.9238	1.9709	2.0145	2.0551	2.0932	2.1292	2.1634	2.1960
15 (213.03)	Sh				36.97	86.97	136.97	186.97	236.97	286.97	386.97	486.97	586.97	686.97	786.97	886.97	986.97
	v	0.01673	26.290		27.837	29.899	31.939	33.963	35.977	37.985	41.986	45.978	49.964	53.946	57.926	61.905	65.882
	h	181.21	1150.9		1168.7	1192.5	1216.2	1239.9	1263.6	1287.3	1335.2	1383.8	1433.2	1483.4	1534.5	1586.5	1639.4
	s	0.3137	1.7552		1.7809	1.8134	1.8437	1.8720	1.8988	1.9242	1.9717	2.0155	2.0563	2.0946	2.1309	2.1653	2.1982
20 (227.96)	Sh				22.04	72.04	122.04	172.04	222.04	272.04	372.04	472.04	572.04	672.04	772.04	872.04	972.04
	v	0.01683	20.087		20.788	22.356	23.900	25.428	26.946	28.457	31.466	34.465	37.458	40.447	43.435	46.420	49.405
	h	196.27	1156.3		1167.1	1191.4	1215.4	1239.2	1263.0	1286.9	1334.9	1383.5	1432.9	1483.2	1534.3	1586.3	1639.3
	s	0.3358	1.7320		1.7475	1.7805	1.8111	1.8397	1.8666	1.8921	1.9397	1.9836	2.0244	2.0628	2.0991	2.1336	2.1665
25 (240.07)	Sh				9.93	59.93	109.93	159.93	209.93	259.93	359.93	459.93	559.93	659.93	759.93	859.93	959.93
	v	0.01693	16.301		16.558	17.829	19.076	20.307	21.527	22.740	25.153	27.557	29.954	32.348	34.740	37.130	39.518
	h	208.52	1160.6		1165.6	1190.2	1214.5	1238.5	1262.5	1286.4	1334.6	1383.3	1432.7	1483.0	1534.2	1586.2	1639.2
	s	0.3535	1.7141		1.7212	1.7547	1.7856	1.8145	1.8415	1.8672	1.9149	1.9588	1.9997	2.0381	2.0744	2.1089	2.1418
30 (250.34)	Sh					49.66	99.66	149.66	199.66	249.66	349.66	449.66	549.66	649.66	749.66	849.66	949.66
	v	0.01701	13.744			14.810	15.859	16.892	17.914	18.929	20.945	22.951	24.952	26.949	28.943	30.936	32.927
	h	218.93	1164.1			1189.0	1213.6	1237.8	1261.9	1286.0	1334.2	1383.0	1432.5	1482.8	1534.0	1586.1	1639.0
	s	0.3682	1.6995			1.7334	1.7647	1.7937	1.8210	1.8467	1.8946	1.9386	1.9795	2.0179	2.0543	2.0888	2.1217
35 (259.29)	Sh					40.71	90.71	140.71	190.71	240.71	340.71	440.71	540.71	640.71	740.71	840.71	940.71
	v	0.01708	11.896			12.654	13.562	14.453	15.334	16.207	17.939	19.662	21.379	23.092	24.803	26.512	28.220
	h	228.03	1167.1			1187.8	1212.7	1237.1	1261.3	1285.5	1333.9	1382.8	1432.3	1482.7	1533.9	1586.0	1638.9
	s	0.3809	1.6872			1.7152	1.7468	1.7761	1.8035	1.8294	1.8774	1.9214	1.9624	2.0009	2.0372	2.0717	2.1046
40 (267.25)	Sh					32.75	82.75	132.75	182.75	232.75	332.75	432.75	532.75	632.75	732.75	832.75	932.75
	v	0.01715	10.497			11.035	11.838	12.624	13.398	14.165	15.685	17.195	18.699	20.199	21.697	23.194	24.689
	h	236.14	1169.8			1186.6	1211.7	1236.4	1260.8	1285.0	1333.6	1382.5	1432.1	1482.5	1533.7	1585.8	1638.8
	s	0.3921	1.6765			1.6992	1.7312	1.7608	1.7883	1.8143	1.8624	1.9065	1.9476	1.9860	2.0224	2.0569	2.0899
45** (274.43)	Sh					25.57	75.57	125.57	175.57	225.57	325.57	425.57	525.57	625.57	725.57	825.57	925.57
	v	0.01722	9.403			9.762	10.503	11.226	11.897	12.584	13.939	15.264	16.573	17.903	19.292	20.623	21.954
	h	243.47	1172.1			1185.4	1210.9	1235.7	1260.2	1284.6	1333.3	1382.3	1432.0	1482.4	1533.6	1585.7	1638.8
	s	0.4021	1.6671			1.6849	1.7174	1.7472	1.7749	1.8010	1.8492	1.8934	1.9345	1.9730	2.0054	2.0439	2.0769
50 (281.02)	Sh					18.98	68.98	118.98	168.98	218.98	318.98	418.98	518.98	618.98	718.98	818.98	918.98
	v	0.01727	8.514			8.769	9.424	10.062	10.688	11.306	12.529	13.741	14.947	16.150	17.350	18.549	19.746
	h	250.21	1174.1			1184.1	1209.9	1234.9	1259.6	1284.1	1332.9	1382.0	1431.7	1482.2	1533.4	1585.6	1638.6
	s	0.4112	1.6585			1.6720	1.7048	1.7349	1.7628	1.7890	1.8374	1.8816	1.9227	1.9613	1.9977	2.0322	2.0652
55** (287.07)	Sh					12.93	62.93	112.93	162.93	212.93	312.93	412.93	512.93	612.93	712.93	812.93	912.93
	v	0.01733	7.787			7.947	8.550	9.134	9.706	[illegible]	11.385	12.489	13.587	14.682	15.775	16.865	17.954
	h	255.42	1175.9			1182.9	1208.9	1234.3	1259.1	[illegible]	1332.6	1381.5	1431.5	1482.0	1533.3	1585.5	1638.5
	s	0.4196	1.6510			1.6602	1.6934	1.7238	1.7518	1.7781	1.8267	1.8710	1.9123	1.9507	1.9871	2.0217	2.0546
60 (292.71)	Sh					7.29	57.29	107.29	157.29	207.29	307.29	407.29	507.29	607.29	707.29	807.29	907.29
	v	0.01738	7.174			7.257	7.815	8.354	8.881	9.400	10.425	11.438	12.446	13.450	14.452	15.452	16.450
	h	262.21	1177.6			1181.6	1208.0	1233.5	1258.5	1283.2	1332.3	1381.5	1431.3	1481.8	1533.2	1585.3	1638.4
	s	0.4273	1.6440			1.6492	1.6834	1.7134	1.7417	1.7681	1.8168	1.8612	1.9024	1.9410	1.9774	2.0120	2.0450
65 (297.98)	Sh					2.02	52.02	102.02	152.02	202.02	302.02	402.02	502.02	602.02	702.02	802.02	902.02
	v	0.01743	6.653			6.675	7.195	7.697	8.186	8.667	9.615	10.552	11.484	12.412	13.337	14.261	15.183
	h	267.63	1179.1			1180.3	1207.0	1232.7	1257.9	1282.7	1331.9	1381.3	1431.1	1481.6	1533.0	1585.2	1638.3
	s	0.4344	1.6375			1.6390	1.6731	1.7040	1.7324	1.7590	1.8077	1.8522	1.8935	1.9321	1.9685	2.0031	2.0361
70 (302.93)	Sh						47.07	97.07	147.07	197.07	297.07	397.07	497.07	597.07	697.07	797.07	897.07
	v	0.01742	6.205				6.694	7.133	7.590	8.039	8.922	9.793	10.659	11.522	12.382	13.240	14.097
	h	272.74	1180.6				1206.0	1232.0	1257.3	1282.2	1331.6	1381.0	1430.9	1481.5	1532.9	1585.1	1638.2
	s	0.4411	1.6316				1.6640	1.6951	1.7237	1.7504	1.7993	1.8439	1.8852	1.9238	1.9603	1.9949	2.0279
75 (307.61)	Sh						42.39	92.39	142.39	192.39	292.39	392.39	492.39	592.39	692.39	792.39	892.39
	v	0.01753	5.814				6.204	6.645	7.074	7.494	8.320	9.135	9.945	10.750	11.553	12.355	13.155
	h	277.56	1181.9				1205.0	1231.2	1256.7	1281.7	1331.3	1380.7	1430.7	1481.3	1532.7	1585.0	1638.1
	s	0.4474	1.6260				1.6554	1.6868	1.7156	1.7424	1.7915	1.8361	1.8774	1.9161	1.9526	1.9872	2.0202

Sh = superheat, F
v = specific volume, cu ft per lb
h = enthalpy, Btu per lb
s = entropy, Btu per F per lb

*Values from STEAM TABLES, Properties of Saturated and Superheated Steam Published by COMBUSTION ENGINEERING, INC. Copyright 1940
**Values interpolated from ASME STEAM TABLES

Table 3. Superheated Steam – Continued

Abs Press Lb/Sq In (Sat Temp)		Sat Water	Sat Steam	350	400	450	500	550	600	700	800	900	1000	1100	1200	1300	1400
80 (312.04)	Sh			37.96	87.96	137.96	187.96	237.96	287.96	387.96	487.96	587.96	687.96	787.96	887.96	987.96	1087.96
	v	0.01757	5.471	5.801	6.218	6.622	7.018	7.408	7.794	8.560	9.319	10.075	10.829	11.581	12.331	13.081	13.829
	h	282.15	1183.1	1204.0	1230.5	1256.1	1281.3	1306.2	1330.9	1380.5	1430.5	1481.1	1532.6	1584.9	1638.0	1692.0	1746.8
	s	0.4534	1.6208	1.6473	1.6790	1.7080	1.7349	1.7602	1.7842	1.8289	1.8702	1.9089	1.9454	1.9800	2.0131	2.0446	2.0750
85 (316.26)	Sh			33.74	83.74	133.74	183.74	233.74	283.74	383.74	483.74	583.74	683.74	783.74	883.74	983.74	1083.74
	v	0.01762	5.167	5.445	5.840	6.223	6.597	6.966	7.330	8.052	8.768	9.480	10.190	10.898	11.604	12.310	13.014
	h	286.52	1184.2	1202.7	1229.7	1255.5	1280.8	1305.8	1330.6	1380.3	1430.3	1481.0	1532.4	1584.7	1637.9	1691.8	1746.7
	s	0.4590	1.6159	1.6394	1.6716	1.7008	1.7279	1.7532	1.7772	1.8220	1.8634	1.9021	1.9386	1.9733	2.0063	2.0379	2.0682
90 (320.28)	Sh			29.72	79.72	129.72	179.72	229.72	279.72	379.72	479.72	579.72	679.72	779.72	879.72	979.72	1079.72
	v	0.01766	4.895	5.128	5.505	5.869	6.223	6.572	6.917	7.600	8.277	8.950	9.621	10.290	10.958	11.625	12.290
	h	290.69	1185.3	1202.5	1228.9	1254.9	1280.3	1305.4	1330.2	1380.0	1430.1	1480.8	1532.3	1584.6	1637.8	1691.8	1746.7
	s	0.4643	1.6113	1.6323	1.6645	1.6940	1.7212	1.7467	1.7707	1.8156	1.8570	1.8957	1.9323	1.9669	2.0000	2.0316	2.0619
95 (324.13)	Sh			25.87	75.87	125.87	175.87	225.87	275.87	375.87	475.87	575.87	675.87	775.87	875.87	975.87	1075.87
	v	0.01770	4.651	4.845	5.205	5.551	5.889	6.221	6.548	7.196	7.838	8.477	9.113	9.747	10.380	11.012	11.643
	h	294.70	1186.2	1200.9	1228.1	1254.4	1279.8	1305.0	1329.9	1379.7	1429.9	1480.6	1532.1	1584.5	1637.7	1691.7	1746.6
	s	0.4694	1.6069	1.6253	1.6580	1.6878	1.7152	1.7408	1.7649	1.8098	1.8513	1.8901	1.9267	1.9613	1.9944	2.0260	2.0559
100 (327.82)	Sh			22.18	72.18	122.18	172.18	222.18	272.18	372.18	472.18	572.18	672.18	772.18	872.18	972.18	1072.18
	v	0.01774	4.431	4.590	4.935	5.266	5.588	5.904	6.216	6.833	7.443	8.050	8.655	9.258	9.860	10.460	11.060
	h	298.54	1187.2	1199.3	1227.4	1253.7	1279.3	1304.6	1329.6	1379.5	1429.7	1480.4	1532.0	1584.4	1637.6	1691.6	1746.5
	s	0.4743	1.6027	1.6187	1.6516	1.6814	1.7088	1.7344	1.7586	1.8036	1.8451	1.8839	1.9205	1.9552	1.9883	2.0199	2.0502
105 (331.37)	Sh			18.63	68.63	118.63	168.63	218.63	268.63	368.63	468.63	568.63	668.63	768.63	868.63	968.63	1068.63
	v	0.01778	4.231	4.359	4.690	5.007	5.315	5.617	5.915	6.504	7.086	7.665	8.241	8.816	9.389	9.961	10.532
	h	302.24	1188.0	1197.8	1226.6	1253.1	1278.8	1304.2	1329.2	1379.2	1429.4	1480.3	1531.8	1584.2	1637.5	1691.5	1746.4
	s	0.4790	1.5988	1.6122	1.6455	1.6755	1.7031	1.7288	1.7530	1.7981	1.8396	1.8785	1.9151	1.9498	1.9828	2.0145	2.0446
110 (334.79)	Sh			15.21	65.21	115.21	165.21	215.21	265.21	365.21	465.21	565.21	665.21	765.21	865.21	965.21	1065.21
	v	0.01782	4.048	4.149	4.468	4.772	5.068	5.357	5.642	6.205	6.761	7.314	7.865	8.413	8.961	9.507	10.053
	h	305.80	1188.9	1197.7	1225.8	1252.5	1278.3	1303.8	1328.9	1379.0	1429.2	1480.1	1531.7	1584.1	1637.4	1691.4	1746.4
	s	0.4834	1.5950	1.6061	1.6396	1.6698	1.6975	1.7233	1.7476	1.7928	1.8344	1.8732	1.9099	1.9446	1.9777	2.0093	2.0397
115 (338.08)	Sh			11.92	61.92	111.92	161.92	211.92	261.92	361.92	461.92	561.92	661.92	761.92	861.92	961.92	1061.92
	v	0.01785	3.881	3.957	4.265	4.558	4.841	5.119	5.392	5.932	6.465	6.994	7.521	8.046	8.570	9.093	9.615
	h	309.25	1189.6	1196.7	1225.0	1251.8	1277.9	1303.4	1328.6	1378.7	1429.0	1479.9	1531.5	1584.0	1637.2	1691.4	1746.3
	s	0.4877	1.5913	1.6001	1.6340	1.6644	1.6922	1.7181	1.7425	1.7877	1.8294	1.8682	1.9049	1.9396	1.9727	2.0044	2.0347
120 (341.27)	Sh			8.73	58.73	108.73	158.73	208.73	258.73	358.73	458.73	558.73	658.73	758.73	858.73	958.73	1058.73
	v	0.01789	3.7275	3.7815	4.0786	4.3610	4.6341	4.9009	5.1637	5.6813	6.1928	6.7006	7.2060	7.7096	8.2119	8.7130	9.2134
	h	312.58	1190.4	1195.6	1224.1	1251.2	1277.4	1302.9	1328.2	1378.4	1428.8	1479.8	1531.4	1582.9	1637.1	1691.3	1746.2
	s	0.4919	1.5879	1.5943	1.6286	1.6592	1.6872	1.7132	1.7376	1.7829	1.8246	1.8635	1.9001	1.9349	1.9680	1.9996	2.0300
130 (347.33)	Sh			2.67	52.67	102.67	152.67	202.67	252.67	352.67	452.67	552.67	652.67	752.67	852.67	952.67	1052.67
	v	0.01796	3.4544	3.4699	3.7489	4.0129	4.2672	4.5151	4.7589	5.2384	5.7118	6.1814	6.6486	7.1140	7.5781	8.0411	8.5033
	h	318.95	1191.7	1193.4	1222.5	1249.9	1276.4	1302.1	1327.5	1377.9	1428.4	1479.4	1531.1	1583.6	1636.9	1691.1	1746.1
	s	0.4998	1.5813	1.5833	1.6182	1.6493	1.6775	1.7037	1.7283	1.7737	1.8155	1.8545	1.8911	1.9259	1.9591	1.9907	2.0211
140 (353.04)	Sh				46.96	96.96	146.96	196.96	246.96	346.96	446.96	546.96	646.96	746.96	846.96	946.96	1046.96
	v	0.01803	3.2190		3.4661	3.7143	3.9526	4.1844	4.4119	4.8588	5.2995	5.7364	6.1709	6.6036	7.0349	7.4652	7.8946
	h	324.96	1193.0		1220.8	1248.7	1275.3	1301.3	1326.8	1377.4	1428.0	1479.1	1530.8	1583.4	1636.7	1690.9	1745.9
	s	0.5071	1.5752		1.6085	1.6400	1.6686	1.6949	1.7196	1.7652	1.8071	1.8461	1.8828	1.9176	1.9508	1.9825	2.0129
150 (358.43)	Sh				41.57	91.57	141.57	191.57	241.57	341.57	441.57	541.57	641.57	741.57	841.57	941.57	1041.57
	v	0.01809	3.0139		3.2208	3.4555	3.6799	3.8978	4.1112	4.5298	4.9421	5.3507	5.7568	6.1612	6.5642	6.9661	7.3671
	h	330.65	1194.1		1219.1	1247.4	1274.3	1300.5	1326.1	1376.9	1427.6	1478.7	1530.5	1583.1	1636.5	1690.7	1745.7
	s	0.5141	1.5695		1.5993	1.6313	1.6602	1.6867	1.7115	1.7573	1.7992	1.8383	1.8751	1.9099	1.9431	1.9748	2.0052
160 (363.55)	Sh				36.45	86.45	136.45	186.45	236.45	336.45	436.45	536.45	636.45	736.45	836.45	936.45	1036.45
	v	0.01815	2.8336		3.0060	3.2288	3.4413	3.6469	3.8480	4.2420	4.6295	5.0132	5.3945	5.7741	6.1522	6.5293	6.9055
	h	336.07	1195.1		1217.4	1246.0	1273.3	1299.6	1325.4	1376.4	1427.2	1478.4	1530.3	1582.9	1636.3	1690.5	1745.5
	s	0.5206	1.5641		1.5906	1.6231	1.6522	1.6790	1.7039	1.7499	1.7919	1.8310	1.8678	1.9027	1.9359	1.9676	1.9980
170 (368.42)	Sh				31.58	81.58	131.58	181.58	231.58	331.58	431.58	531.58	631.58	731.58	831.58	931.58	1031.58
	v	0.01821	2.6738		2.8162	3.0288	3.2306	3.4255	3.6158	3.9879	4.3536	4.7155	5.0749	5.4325	5.7888	6.1440	6.4983
	h	341.24	1196.0		1215.6	1244.7	1272.2	1298.8	1324.7	1375.8	1426.8	1478.0	1530.0	1582.6	1636.1	1690.4	1745.4
	s	0.5269	1.5591		1.5823	1.6152	1.6447	1.6717	1.6968	1.7428	1.7850	1.8241	1.8610	1.8959	1.9291	1.9608	1.9913
180 (373.08)	Sh				26.92	76.92	126.92	176.92	226.92	326.92	426.92	526.92	626.92	726.92	826.92	926.92	1026.92
	v	0.01827	2.5312		2.6474	2.8508	3.0433	3.2286	3.4093	3.7621	4.1084	4.4508	4.7907	5.1289	5.4657	5.8014	6.1363
	h	346.19	1196.9		1213.8	1243.4	1271.2	1297.9	1324.0	1375.3	1426.3	1477.7	1529.7	1582.4	1635.9	1690.2	1745.3
	s	0.5328	1.5543		1.5743	1.6078	1.6376	1.6647	1.6900	1.7362	1.7784	1.8176	1.8545	1.8894	1.9227	1.9545	1.9849
190 (377.53)	Sh				22.47	72.47	122.47	172.47	222.47	322.47	422.47	522.47	622.47	722.47	822.47	922.47	1022.47
	v	0.01833	2.4030		2.4961	2.6915	2.8756	3.0525	3.2246	3.5601	3.8889	4.2140	4.5365	4.8572	5.1766	5.4949	5.8124
	h	350.94	1197.8		1212.0	1242.0	1270.1	1297.1	1323.3	1374.8	1425.9	1477.4	1529.4	1582.2	1635.7	1690.0	1745.1
	s	0.5384	1.5498		1.5667	1.6006	1.6307	1.6581	1.6835	1.7299	1.7722	1.8115	1.8484	1.8834	1.9166	1.9484	1.9789
200 (381.80)	Sh				18.20	68.20	118.20	168.20	218.20	318.20	418.20	518.20	618.20	718.20	818.20	918.20	1018.20
	v	0.01839	2.2873		2.3598	2.5480	2.7247	2.8939	3.0583	3.3783	3.6915	4.0008	4.3077	4.6128	4.9165	5.2191	5.5209
	h	355.51	1198.4		1210.1	1240.6	1269.0	1296.2	1322.6	1374.3	1425.5	1477.0	1529.1	1581.9	1635.4	1689.8	1745.0
	s	0.5438	1.5454		1.5593	1.5938	1.6242	1.6518	1.6773	1.7239	1.7663	1.8057	1.8426	1.8776	1.9109	1.9427	1.9732

Sh = superheat, F
v = specific volume, cu ft per lb
h = enthalpy, Btu per lb
s = entropy, Btu per F per lb

1103 1502

Table 3. Superheated Steam – Continued

Abs Press Lb/Sq In (Sat. Temp)		Sat. Water	Sat. Steam	400	450	500	550	600	700	800	900	1000	1100	1200	1300	1400	1500
				Temperature – Degrees Fahrenheit													
210 (385.91)	Sh			14.09	64.09	114.09	164.09	214.09	314.09	414.09	514.09	614.09	714.09	814.09	914.09	1014.09	1114.09
	v	0.01844	2.1822	2.2364	2.4181	2.5880	2.7504	2.9078	3.2137	3.5128	3.8080	4.1007	4.3915	4.6811	4.9695	5.2571	5.5440
	h	359.91	1199.0	1204.02	1232.2	1260.0	1285.3	1321.9	1373.7	1425.1	1476.7	1528.3	1581.6	1635.2	1689.6	1744.6	1800.9
	s	0.5402	1.5413	1.5522	1.5877	1.6180	1.6455	1.6715	1.7182	1.7607	1.8001	1.8371	1.8721	1.9054	1.9372	1.9677	1.9970
220 (389.88)	Sh			10.12	60.12	110.12	160.12	210.12	310.12	410.12	510.12	610.12	710.12	810.12	910.12	1010.12	1110.12
	v	0.01850	2.0863	2.1240	2.2993	2.4638	2.6199	2.7710	3.0642	3.3504	3.6327	3.9125	4.1905	4.4671	4.7426	5.0173	5.2913
	h	364.17	1199.6	1206.3	1237.8	1265.9	1291.5	1321.2	1373.2	1424.7	1476.3	1528.5	1581.4	1635.0	1689.4	1744.7	1800.6
	s	0.5440	1.5374	1.5453	1.5808	1.6120	1.6400	1.6658	1.7128	1.7553	1.7943	1.8318	1.8668	1.9002	1.9320	1.9625	1.9919
230 (393.70)	Sh			6.30	56.30	106.30	156.30	206.30	306.30	406.30	506.30	606.30	706.30	806.30	906.30	1006.30	1106.30
	v	0.01855	1.9985	2.0212	2.1919	2.3503	2.5008	2.6461	2.9276	3.2020	3.4726	3.7406	4.0068	4.2717	4.5355	4.7984	5.0606
	h	368.28	1200.1	1204.4	1235.3	1263.7	1289.6	1320.4	1372.7	1424.2	1476.0	1528.2	1581.1	1634.8	1689.3	1744.5	1800.5
	s	0.5568	1.5336	1.5385	1.5747	1.6062	1.6344	1.6604	1.7075	1.7502	1.7897	1.8268	1.8618	1.8952	1.9270	1.9576	1.9869
240 (397.39)	Sh			2.61	52.61	102.61	152.61	202.61	302.61	402.61	502.61	602.61	702.61	802.61	902.61	1002.61	1102.61
	v	0.01860	1.9177	1.9268	2.0928	2.2462	2.3915	2.5316	2.8024	3.0661	3.3259	3.5831	3.8385	4.0926	4.3456	4.5977	4.8492
	h	372.27	1200.6	1202.4	1232.9	1261.5	1287.7	1319.7	1372.1	1423.8	1475.6	1527.9	1580.9	1634.6	1689.1	1744.3	1800.4
	s	0.5634	1.5299	1.5320	1.5687	1.6006	1.6291	1.6552	1.7025	1.7452	1.7848	1.8219	1.8570	1.8904	1.9223	1.9528	1.9822
250 (400.97)	Sh				49.03	99.03	149.03	199.03	299.03	399.03	499.03	599.03	699.03	799.03	899.03	999.03	1099.03
	v	0.01865	1.8432		2.0016	2.1504	2.2909	2.4262	2.6872	2.9410	3.1909	3.4382	3.6837	3.9278	4.1709	4.4131	4.6546
	h	376.14	1201.1		1233.4	1263.5	1291.5	1319.0	1371.6	1423.4	1475.3	1527.6	1580.6	1634.4	1688.9	1744.2	1800.2
	s	0.5679	1.5264		1.5629	1.5951	1.6239	1.6502	1.6976	1.7405	1.7801	1.8173	1.8524	1.8858	1.9177	1.9482	1.9776
260 (404.44)	Sh				45.56	95.56	145.56	195.56	295.56	395.56	495.56	595.56	695.56	795.56	895.56	995.56	1095.56
	v	0.01870	1.7742		1.9173	2.0619	2.1981	2.3273	2.5823	2.8265	3.0663	3.3044	3.5408	3.7758	4.0097	4.2427	4.4750
	h	379.90	1201.5		1231.9	1262.4	1290.9	1318.2	1371.1	1423.0	1474.9	1527.3	1580.4	1634.2	1688.7	1744.0	1800.1
	s	0.5722	1.5230		1.5573	1.5899	1.6189	1.6453	1.6930	1.7359	1.7756	1.8128	1.8480	1.8814	1.9133	1.9439	1.9732
270 (407.80)	Sh				42.20	92.20	142.20	192.20	292.20	392.20	492.20	592.20	692.20	792.20	892.20	992.20	1092.20
	v	0.01875	1.7101		1.8391	1.9799	2.1121	2.2388	2.4824	2.7186	2.9509	3.1806	3.4084	3.6349	3.8603	4.0849	4.3087
	h	383.56	1201.9		1230.4	1261.2	1290.0	1317.5	1370.5	1422.6	1474.6	1527.1	1580.1	1634.0	1688.5	1742.9	1800.0
	s	0.5764	1.5197		1.5518	1.5848	1.6140	1.6406	1.6885	1.7315	1.7713	1.8085	1.8437	1.8771	1.9090	1.9396	1.9690
280 (411.07)	Sh				38.93	88.93	138.93	188.93	288.93	388.93	488.93	588.93	688.93	788.93	888.93	988.93	1088.93
	v	0.01880	1.6505		1.7665	1.9037	2.0322	2.1551	2.3909	2.6194	2.8437	3.0655	3.2855	3.5042	3.7217	3.9384	4.1543
	h	387.12	1202.3		1228.8	1260.0	1289.1	1316.8	1370.0	1422.1	1474.2	1526.8	1579.9	1633.8	1688.4	1743.7	1799.8
	s	0.5805	1.5166		1.5464	1.5798	1.6093	1.6361	1.6841	1.7273	1.7671	1.8043	1.8395	1.8730	1.9050	1.9356	1.9649
290 (414.25)	Sh				35.75	85.75	135.75	185.75	285.75	385.75	485.75	585.75	685.75	785.75	885.75	985.75	1085.75
	v	0.01885	1.5948		1.6988	1.8327	1.9578	2.0772	2.3058	2.5269	2.7440	2.9585	3.1711	3.3824	3.5926	3.8019	4.0105
	h	390.60	1202.6		1227.3	1258.9	1288.1	1316.0	1369.5	1421.7	1473.9	1526.5	1579.6	1633.5	1688.2	1743.6	1799.7
	s	0.5844	1.5135		1.5412	1.5750	1.6046	1.6317	1.6793	1.7232	1.7630	1.8003	1.8356	1.8690	1.9010	1.9316	1.9610
300 (417.35)	Sh				32.65	82.65	132.65	182.65	282.65	382.65	482.65	582.65	682.65	782.65	882.65	982.65	1082.65
	v	0.01889	1.5427		1.6356	1.7665	1.8883	2.0044	2.2263	2.4407	2.6509	2.8585	3.0643	3.2688	3.4721	3.6746	3.8764
	h	393.99	1202.9		1225.7	1257.7	1287.2	1315.2	1368.9	1421.3	1473.6	1526.2	1579.4	1633.3	1688.0	1743.4	1799.6
	s	0.5882	1.5105		1.5361	1.5703	1.6003	1.6274	1.6758	1.7192	1.7591	1.7964	1.8317	1.8652	1.8972	1.9278	1.9572
310 (420.36)	Sh				29.64	79.64	129.64	179.64	279.64	379.64	479.64	579.64	679.64	779.64	879.64	979.64	1079.64
	v	0.01894	1.4939		1.5763	1.7044	1.8233	1.9363	2.1520	2.3600	2.5638	2.7650	2.9644	3.1625	3.3594	3.5555	3.7509
	h	397.30	1203.2		1224.1	1256.5	1286.3	1314.5	1368.4	1420.9	1473.2	1525.9	1579.2	1633.1	1687.8	1743.3	1799.4
	s	0.5920	1.5076		1.5311	1.5657	1.5960	1.6233	1.6719	1.7153	1.7553	1.7927	1.8280	1.8615	1.8935	1.9241	1.9536
320 (423.31)	Sh				26.69	76.69	126.69	176.69	276.69	376.69	476.69	576.69	676.69	776.69	876.69	976.69	1076.69
	v	0.01899	1.4480		1.5207	1.6462	1.7623	1.8725	2.0823	2.2843	2.4821	2.6774	2.8708	3.0628	3.2538	3.4438	3.6332
	h	400.53	1203.4		1222.5	1255.2	1285.3	1313.7	1367.8	1420.5	1472.9	1525.6	1578.9	1632.9	1687.6	1743.1	1799.3
	s	0.5956	1.5048		1.5261	1.5612	1.5918	1.6192	1.6680	1.7116	1.7516	1.7890	1.8243	1.8579	1.8899	1.9206	1.9500
330 (426.18)	Sh				23.82	73.82	123.82	173.82	273.82	373.82	473.82	573.82	673.82	773.82	873.82	973.82	1073.82
	v	0.01903	1.4048		1.4684	1.5915	1.7050	1.8125	2.0168	2.2132	2.4054	2.5950	2.7828	2.9692	3.1545	3.3389	3.5227
	h	403.70	1203.6		1220.9	1254.0	1284.4	1313.0	1367.3	1420.0	1472.5	1525.3	1578.7	1632.7	1687.5	1742.9	1799.2
	s	0.5991	1.5021		1.5213	1.5568	1.5876	1.6153	1.6643	1.7079	1.7420	1.7855	1.8208	1.8544	1.8864	1.9171	1.9466
340 (428.99)	Sh				21.01	71.01	121.01	171.01	271.01	371.01	471.01	571.01	671.01	771.01	871.01	971.01	1071.01
	v	0.01908	1.3640		1.4191	1.5399	1.6511	1.7561	1.9552	2.1463	2.3333	2.5175	2.7000	2.8811	3.0611	3.2402	3.4186
	h	406.80	1203.8		1219.2	1252.8	1283.4	1312.2	1366.7	1419.6	1472.2	1525.0	1578.4	1632.5	1687.3	1742.8	1799.0
	s	0.6026	1.4994		1.5165	1.5525	1.5836	1.6114	1.6606	1.7044	1.7445	1.7820	1.8174	1.8510	1.8831	1.9138	1.9432
350 (431.72)	Sh				18.27	68.27	118.27	168.27	268.27	368.27	468.27	568.27	668.27	768.27	868.27	968.27	1068.27
	v	0.01912	1.3255		1.3725	1.4913	1.6002	1.7028	1.8970	2.0832	2.2652	2.4445	2.6219	2.7980	2.9730	3.1471	3.3205
	h	409.83	1204.0		1217.5	1251.5	1282.4	1311.4	1366.2	1419.2	1471.8	1524.7	1578.2	1632.3	1687.1	1742.6	1798.9
	s	0.6059	1.4968		1.5119	1.5483	1.5797	1.6077	1.6571	1.7009	1.7411	1.7787	1.8141	1.8477	1.8798	1.9105	1.9400
360 (434.41)	Sh				15.59	65.59	115.59	165.59	265.59	365.59	465.59	565.59	665.59	765.59	865.59	965.59	1065.59
	v	0.01917	1.2891		1.3285	1.4454	1.5521	1.6525	1.8421	2.0237	2.2009	2.3755	2.5482	2.7196	2.8898	3.0592	3.2279
	h	412.81	1204.1		1215.8	1250.3	1281.5	1310.6	1365.6	1418.7	1471.5	1524.4	1577.9	1632.1	1686.9	1742.5	1798.8
	s	0.6092	1.4943		1.5073	1.5441	1.5758	1.6040	1.6536	1.6976	1.7379	1.7754	1.8109	1.8445	1.8766	1.9073	1.9368
[illegible] (435.[illegible])	Sh				10.39	60.39	110.39	[illegible]	260.39	360.39	460.39	560.39	660.39	760.39	[illegible]	960.39	1060.39
	v	0.01925	1.2218		1.2472	1.3605	1.4635	[illegible]	1.7410	1.9139	2.0825	2.2451	2.4124	2.5750	2.7366	2.8973	3.0572
	h	416.53	1204.4		1212.4	1247.7	1279.5	[illegible]	1364.5	1417.9	1470.5	1523.8	1577.4	1631.6	[illegible]	1742.2	1798.5
	s	0.6156	1.4894		1.4982	1.5360	1.5683	[illegible]	1.6470	1.6911	1.7315	1.7692	1.8047	1.8384	1.8705	1.9012	1.9307

Sh = superheat, F
v = specific volume, cu ft per lb
h = enthalpy, Btu per lb
s = entropy, Btu per F per lb

Table 3. Superheated Steam – Continued

Abs Press Lb/Sq In (Sat. Temp)		Sat Water	Sat Steam	450	500	550	600	650	700	800	900	1000	1100	1200	1300	1400	1500
400	Sh			5.40	55.40	105.40	155.40	205.40	255.40	355.40	455.40	555.40	655.40	755.40	855.40	955.40	1055.40
(444.60)	v	0.01934	1.1610	1.1738	1.2841	1.3836	1.4763	1.5646	1.6499	1.8151	1.9759	2.1339	2.2901	2.4450	2.5987	2.7515	2.9037
	h	424.17	1204.5	1208.8	1245.1	1277.5	1306.9	1335.9	1362.7	1417.0	1470.1	1523.3	1576.9	1631.2	1686.2	1741.9	1798.2
	s	0.6217	1.4847	1.4894	1.5282	1.5611	1.5901	1.6163	1.6406	1.6850	1.7255	1.7632	1.7988	1.8325	1.8647	1.8955	1.9250
420	Sh			0.60	50.60	100.60	150.60	200.60	250.60	350.60	450.60	550.60	650.60	750.60	850.60	950.60	1050.60
(449.40)	v	0.01942	1.1057	1.1071	1.2148	1.3113	1.4007	1.4856	1.5676	1.7258	1.8795	2.0304	2.1795	2.3273	2.4739	2.6195	2.7647
	h	429.56	1204.7	1205.2	1242.4	1275.4	1305.3	1334.5	1362.3	1416.7	1469.4	1522.7	1576.4	1630.8	1685.8	1741.6	1798.0
	s	0.6276	1.4802	1.4808	1.5206	1.5542	1.5835	1.6100	1.6345	1.6791	1.7197	1.7575	1.7932	1.8269	1.8591	1.8899	1.9195
440	Sh				45.97	95.97	145.97	195.97	245.97	345.97	445.97	545.97	645.97	745.97	845.97	945.97	1045.97
(454.03)	v	0.01950	1.0554		1.1517	1.2454	1.3319	1.4138	1.4926	1.6445	1.7918	1.9363	2.0790	2.2203	2.3605	2.4998	2.6384
	h	434.77	1204.8		1239.7	1273.4	1304.2	1333.2	1361.1	1415.3	1468.7	1522.1	1575.9	1630.4	1685.5	1741.2	1797.7
	s	0.6332	1.4759		1.5132	1.5474	1.5772	1.6043	1.6286	1.6734	1.7142	1.7521	1.7878	1.8216	1.8538	1.8847	1.9143
460	Sh				41.50	91.50	141.50	191.50	241.50	341.50	441.50	541.50	641.50	741.50	841.50	941.50	1041.50
(458.50)	v	0.01959	1.0092		1.0939	1.1852	1.2691	1.3482	1.4242	1.5703	1.7117	1.8504	1.9872	2.1226	2.2569	2.3903	2.5230
	h	439.83	1204.8		1236.9	1271.3	1302.5	1331.8	1360.0	1414.4	1468.0	1521.5	1575.4	1629.9	1685.1	1740.9	1797.4
	s	0.6387	1.4718		1.5060	1.5409	1.5711	1.5982	1.6230	1.6680	1.7089	1.7469	1.7826	1.8165	1.8488	1.8797	1.9093
480	Sh				37.18	87.18	137.18	187.18	237.18	337.18	437.18	537.18	637.18	737.18	837.18	937.18	1037.18
(462.82)	v	0.01967	0.9668		1.0409	1.1300	1.2115	1.2881	1.3615	1.5023	1.6384	1.7716	1.9030	2.0330	2.1619	2.2900	2.4173
	h	444.75	1204.8		1234.1	1269.1	1300.8	1330.5	1358.8	1413.6	1467.3	1520.9	1574.9	1629.5	1684.7	1740.6	1797.2
	s	0.6439	1.4677		1.4990	1.5346	1.5652	1.5925	1.6176	1.6628	1.7038	1.7419	1.7777	1.8116	1.8439	1.8748	1.9045
500	Sh				32.99	82.99	132.99	182.99	232.99	332.99	432.99	532.99	632.99	732.99	832.99	932.99	1032.99
(467.01)	v	0.01975	0.9276		0.9919	1.0791	1.1584	1.2327	1.3037	1.4397	1.5708	1.6992	1.8256	1.9507	2.0746	2.1977	2.3200
	h	449.52	1204.7		1231.2	1267.0	1299.1	1329.1	1357.7	1412.7	1466.6	1520.3	1574.4	1629.1	1684.4	1740.3	1796.9
	s	0.6490	1.4639		1.4921	1.5284	1.5595	1.5871	1.6123	1.6578	1.6990	1.7371	1.7730	1.8069	1.8393	1.8702	1.8998
520	Sh				28.93	78.93	128.93	178.93	228.93	328.93	428.93	528.93	628.93	728.93	828.93	928.93	1028.93
(471.07)	v	0.01982	0.8914		0.9466	1.0321	1.1094	1.1816	1.2504	1.3819	1.5085	1.6323	1.7542	1.8746	1.9940	2.1125	2.2302
	h	454.18	1204.5		1228.3	1264.8	1297.4	1327.7	1356.5	1411.8	1465.9	1519.7	1573.9	1628.7	1684.0	1740.0	1796.7
	s	0.6540	1.4601		1.4853	1.5223	1.5539	1.5818	1.6072	1.6530	1.6943	1.7325	1.7684	1.8024	1.8348	1.8657	1.8954
540	Sh				24.99	74.99	124.99	174.99	224.99	324.99	424.99	524.99	624.99	724.99	824.99	924.99	1024.99
(475.01)	v	0.01990	0.8577		0.9045	0.9884	1.0640	1.1342	1.2010	1.3284	1.4508	1.5704	1.6880	1.8042	1.9193	2.0336	2.1471
	h	458.71	1204.4		1225.3	1262.5	1295.7	1326.3	1355.3	1410.9	1465.1	1519.1	1573.4	1628.2	1683.6	1739.7	1796.4
	s	0.6587	1.4565		1.4786	1.5164	1.5485	1.5767	1.6023	1.6483	1.6897	1.7280	1.7640	1.7981	1.8305	1.8615	1.8911
560	Sh				21.16	71.16	121.16	171.16	221.16	321.16	421.16	521.16	621.16	721.16	821.16	921.16	1021.16
(478.84)	v	0.01998	0.8264		0.8653	0.9479	1.0217	1.0902	1.1552	1.2787	1.3972	1.5129	1.6266	1.7388	1.8500	1.9603	2.0699
	h	463.14	1204.2		1222.2	1260.3	1293.9	1324.9	1354.2	1410.0	1464.4	1518.6	1572.9	1627.8	1683.3	1739.4	1796.1
	s	0.6634	1.4529		1.4720	1.5106	1.5431	1.5717	1.5975	1.6438	1.6853	1.7237	1.7598	1.7939	1.8263	1.8573	1.8870
580	Sh				17.43	67.43	117.43	167.43	217.43	317.43	417.43	517.43	617.43	717.43	817.43	917.43	1017.43
(482.57)	v	0.02006	0.7971		0.8287	0.9100	0.9824	1.0492	1.1125	1.2324	1.3473	1.4593	1.5693	1.6780	1.7855	1.8921	1.9980
	h	467.47	1203.9		1219.1	1258.0	1292.1	1323.4	1353.0	1409.2	1463.7	1518.0	1572.4	1627.4	1682.9	1739.1	1795.9
	s	0.6679	1.4495		1.4654	1.5049	1.5380	1.5668	1.5929	1.6394	1.6811	1.7196	1.7556	1.7898	1.8223	1.8533	1.8831
600	Sh				13.80	63.80	113.80	163.80	213.80	313.80	413.80	513.80	613.80	713.80	813.80	913.80	1013.80
(486.20)	v	0.02013	0.7697		0.7944	0.8746	0.9456	1.0109	1.0726	1.1892	1.3008	1.4093	1.5160	1.6211	1.7252	1.8284	1.9309
	h	471.70	1203.7		1215.9	1255.6	1290.3	1322.0	1351.8	1408.3	1463.0	1517.4	1571.9	1627.0	1682.6	1738.8	1795.6
	s	0.6723	1.4461		1.4590	1.4993	1.5329	1.5621	1.5884	1.6351	1.6769	1.7155	1.7517	1.7859	1.8184	1.8494	1.8792
650	Sh				5.11	55.11	105.11	155.11	205.11	305.11	405.11	505.11	605.11	705.11	805.11	905.11	1005.11
(494.89)	v	0.02032	0.7084		0.7173	0.7954	0.8634	0.9254	0.9835	1.0929	1.1969	1.2979	1.3969	1.4944	1.5909	1.6864	1.7813
	h	481.89	1202.8		1207.6	1249.6	1285.7	1318.3	1348.7	1406.0	1461.2	1515.9	1570.7	1625.9	1681.6	1738.0	1794.9
	s	0.6828	1.4381		1.4430	1.4858	1.5207	1.5507	1.5775	1.6249	1.6671	1.7059	1.7422	1.7765	1.8092	1.8403	1.8701
700	Sh					46.92	96.92	146.92	196.92	296.92	396.92	496.92	596.92	696.92	796.92	896.92	996.92
(503.08)	v	0.02050	0.6556			0.7271	0.7928	0.8520	0.9072	1.0102	1.1078	1.2023	1.2948	1.3858	1.4757	1.5647	1.6530
	h	491.60	1201.8			1243.4	1281.0	1314.6	1345.6	1403.7	1459.4	1514.4	1569.4	1624.8	1680.7	1737.2	1794.3
	s	0.6926	1.4304			1.4726	1.5090	1.5399	1.5673	1.6154	1.6580	1.6970	1.7335	1.7679	1.8006	1.8318	1.8617
750	Sh					39.16	89.16	139.16	189.16	289.16	389.16	489.16	589.16	689.16	789.16	889.16	989.16
(510.84)	v	0.02069	0.6095			0.6676	0.7313	0.7882	0.8409	0.9386	1.0306	1.1195	1.2063	1.2916	1.3759	1.4592	1.5419
	h	500.89	1200.7			1236.9	1276.1	1310.7	1342.5	1401.5	1457.6	1512.9	1568.2	1623.8	1679.9	1736.4	1793.6
	s	0.7022	1.4232			1.4598	1.4977	1.5296	1.5577	1.6065	1.6494	1.6886	1.7252	1.7598	1.7926	1.8239	1.8538
800	Sh					31.79	81.79	131.79	181.79	281.79	381.79	481.79	581.79	681.79	781.79	881.79	981.79
(518.21)	v	0.02087	0.5690			0.6151	0.6774	0.7323	0.7828	0.8759	0.9631	1.0470	1.1289	1.2093	1.2885	1.3669	1.4446
	h	509.81	1199.4			1230.1	1271.1	1306.8	1339.3	1399.1	1455.8	1511.4	1566.9	1622.7	1678.9	1735.7	1792.9
	s	0.7111	1.4163			1.4472	1.4869	1.5198	1.5484	1.5980	1.6413	1.6807	1.7175	1.7522	1.7851	1.8164	1.8464
850	Sh					24.76	74.76	124.76	174.76	274.76	374.76	474.76	574.76	674.76	774.76	874.76	974.76
(525.24)	v	0.02105	0.5330			0.5683	0.6296	0.6829	0.7315	0.8205	0.9034	0.9830	1.0606	1.1366	1.2115	1.2855	1.3588
	h	518.40	1198.0			1223.0	1265.9	1302.8	1336.0	1396.8	1454.0	1510.0	1565.7	1621.6	1678.0	1734.9	1792.3
	s	0.7197	1.4096			1.4347	1.4763	1.5102	1.5396	1.5900	1.6336	1.6733	1.7102	1.7450	1.7780	1.8094	1.8395
900	Sh					18.05	68.05	118.05	168.05	268.05	368.05	468.05	568.05	668.05	768.05	868.05	968.05
(531.95)	v	0.02123	0.5009			0.5263	0.5869	0.6388	0.6858	0.7713	0.8504	0.9262	0.9998	1.0720	1.1430	1.2131	1.2825
	h	526.70	1196.4			1215.5	1260.6	1298.6	1332.7	1394.4	1452.2	1508.5	1564.4	1620.6	1677.1	1734.1	1791.6
	s	0.7279	1.4032			1.4223	1.4659	1.5010	1.5311	1.5822	1.6263	1.6662	1.7033	1.7382	1.7713	1.8028	1.8329

Sh – superheat, F
v – specific volume, cu ft per lb
h – enthalpy, Btu per lb
s – entropy, Btu per F per lb

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Table 3. Superheated Steam—Continued

Abs Press Lb Sq In (Sat Temp)		Sat Water	Sat Steam	550	600	650	700	750	800	850	900	1000	1100	1200	1300	1400	1500
950 (538.39)	Sh			11.61	61.61	111.61	[illegible]	211.61	261.61	311.61	361.61	461.61	561.61	661.61	761.61	861.61	961.61
	v	0.02141	0.4721	[illegible]	0.5435	0.5993	[illegible]	[illegible]	[illegible]	0.7656	[illegible]	[illegible]	0.9455	1.0142	1.0817	1.1484	1.2143
	h	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1676.7	[illegible]	[illegible]
	s	0.7389	[illegible]	[illegible]	1.4557	1.4921	[illegible]	[illegible]	[illegible]	1.5977	[illegible]	[illegible]	[illegible]	[illegible]	1.7643	[illegible]	[illegible]
1000 (544.58)	Sh			5.42	55.42	105.42	155.42	205.42	255.42	305.42	355.42	455.42	555.42	655.42	755.42	855.42	955.42
	v	0.02159	0.4460	[illegible]	0.5137	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.0266	[illegible]	[illegible]
	h	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1675.3	1732.5	[illegible]
	s	[illegible]	[illegible]	[illegible]	1.4457	1.4833	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.6905	1.7255	1.7559	[illegible]	[illegible]
1050 (550.53)	Sh				49.47	99.47	149.47	199.47	249.47	299.47	349.47	449.47	549.47	649.47	749.47	849.47	949.47
	v	0.02177	0.4222		0.4821	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.9767	[illegible]	[illegible]
	h	[illegible]	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1674.4	[illegible]	[illegible]
	s	[illegible]	1.3851		1.4358	1.4746	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.7531	[illegible]	[illegible]
1100 (556.28)	Sh				43.72	93.72	[illegible]	[illegible]	[illegible]	[illegible]	343.72	443.72	543.72	643.72	743.72	843.72	943.72
	v	0.02195	0.4006		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	h	[illegible]	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	s	[illegible]	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
1150 (561.82)	Sh				38.18	88.18	[illegible]	[illegible]	[illegible]	[illegible]	338.18	[illegible]	538.18	638.18	738.18	838.18	938.18
	v	0.02214	0.3807		0.4263	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	h	[illegible]	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	s	[illegible]	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
1200 (567.19)	Sh				32.81	82.81	[illegible]	[illegible]	[illegible]	[illegible]	332.81	432.81	532.81	632.81	732.81	832.81	932.81
	v	0.02232	0.3624		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	h	[illegible]	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	s	[illegible]	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
1300 (577.42)	Sh				22.58	72.58	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	422.58	522.58	622.58	722.58	822.58	922.58
	v	0.02269	0.3299		0.3570	0.4052	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	h	[illegible]	[illegible]		1259.9	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	s	[illegible]	1.3577		1.3650	1.4540	[illegible]	[illegible]	[illegible]	1.5544	[illegible]	[illegible]	[illegible]	[illegible]	1.7275	[illegible]	[illegible]
1400 (587.07)	Sh				12.93	62.93	112.93	162.93	[illegible]	262.93	312.93	412.93	512.93	612.93	712.93	812.93	912.93
	v	0.02307	0.3018		0.3176	0.3667	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.7272	[illegible]	[illegible]
	h	[illegible]	1175.3		1194.1	1251.4	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1668.0	[illegible]	[illegible]
	s	0.7956	1.3474		1.3652	1.4161	1.4575	[illegible]	[illegible]	1.5436	1.5670	[illegible]	[illegible]	[illegible]	1.7185	[illegible]	1.7415
1500 (596.20)	Sh				3.80	53.80	103.80	153.80	203.80	253.80	303.80	403.80	503.80	603.80	703.80	803.80	903.80
	v	0.02346	0.2772		0.2820	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	h	[illegible]	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	s	[illegible]	1.3373		1.3431	1.4022	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.7101	1.7425	1.7724
1600 (604.87)	Sh					45.13	95.13	145.13	195.13	245.13	295.13	395.13	495.13	595.13	695.13	795.13	895.13
	v	0.02387	0.2555			[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	h	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	s	[illegible]	1.3274			1.3861	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
1700 (613.13)	Sh					36.87	86.87	136.87	186.87	236.87	286.87	386.87	486.87	586.87	686.87	786.87	886.87
	v	0.02428	0.2361			0.2754	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	h	[illegible]	[illegible]			1215.3	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	s	[illegible]	1.3176			1.3697	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.6947	[illegible]	1.7555
1800 (621.02)	Sh					28.98	78.98	128.98	178.98	228.98	278.98	378.98	478.98	578.98	678.98	778.98	878.98
	v	0.02472	0.2186			[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	h	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	s	[illegible]	1.3079			1.3526	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.7516
1900 (628.56)	Sh					21.44	71.44	121.44	171.44	221.44	271.44	371.44	471.44	571.44	671.44	771.44	871.44
	v	0.02517	0.2028			0.2274	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.5303	[illegible]	[illegible]
	h	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	s	[illegible]	[illegible]			1.3346	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
2000 (635.80)	Sh					14.20	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	664.20	[illegible]	[illegible]
	v	0.02565	0.1883			[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	h	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	s	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
2100 (642.76)	Sh					7.24	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	657.24	[illegible]	[illegible]
	v	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	h	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	s	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
2200 (649.45)	Sh					0.55	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	650.55	[illegible]	[illegible]
	v	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	h	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	s	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
2300 (655.89)	Sh						[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	v	[illegible]	[illegible]				[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	h	[illegible]	[illegible]				[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	s	[illegible]	[illegible]				[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

Sh = superheat, F
v = specific volume, cu ft per lb
h = enthalpy, Btu per lb
s = entropy, Btu per F per lb

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Table 3. Superheated Steam—Continued

Abs Press Lb Sq In (Sat Temp)		Sat Water	Sat Steam	Temperature—Degrees Fahrenheit 700	750	800	850	900	950	1000	1050	1100	1150	1200	1300	1400	1500
2400 (662.11)	Sh			37.89	87.89	137.89	187.89	237.89	287.89	337.89	387.89	437.89	487.89	537.89	637.89	737.89	837.89
	v	0.02790	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.4155	0.4442	[illegible]
	h	718.95	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1645.6	[illegible]	[illegible]
	s	0.9022	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.6509	1.6847	1.7167
2500 (668.11)	Sh			31.89	81.89	131.89	181.89	231.89	281.89	331.89	381.89	431.89	481.89	531.89	631.89	731.89	831.89
	v	0.02859	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.4253	0.4505
	h	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1647.6	1709.2	1772.8
	s	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.5029	[illegible]	1.5492	1.5702	[illegible]	[illegible]	1.6456	1.6796	1.7116
2600 (673.91)	Sh			26.09	76.09	126.09	176.09	226.09	276.09	326.09	376.09	426.09	476.09	526.09	626.09	726.09	826.09
	v	0.02938	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.2755	[illegible]	[illegible]	[illegible]	[illegible]	0.3540	0.3819	0.4036	[illegible]
	h	744.47	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1419.2	[illegible]	[illegible]	[illegible]	[illegible]	1583.7	1646.0	1707.7	[illegible]
	s	0.9247	1.2125	[illegible]	[illegible]	1.4042	[illegible]	[illegible]	1.4964	1.5218	1.5434	1.5646	[illegible]	[illegible]	1.6405	1.6746	[illegible]
2700 (679.53)	Sh			20.47	70.47	120.47	170.47	220.47	270.47	320.47	370.47	420.47	470.47	520.47	620.47	720.47	820.47
	v	0.03019	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.3570	[illegible]	[illegible]
	h	757.34	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	s	[illegible]	[illegible]	[illegible]	1.3481	[illegible]	[illegible]	[illegible]	1.4900	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.6355	1.6697	[illegible]
2800 (684.96)	Sh			15.04	65.04	115.04	165.04	215.04	265.04	315.04	365.04	415.04	465.04	515.04	615.04	715.04	815.04
	v	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.2531	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.3765	[illegible]
	h	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1642.2	1704.5	1766.5
	s	0.9468	[illegible]	1.2527	[illegible]	[illegible]	1.4245	[illegible]	1.4835	1.5089	1.5327	[illegible]	1.5742	1.5938	1.6306	1.6651	1.6975
2900 (690.22)	Sh			9.78	59.78	109.78	159.78	209.78	259.78	309.78	359.78	409.78	459.78	509.78	609.78	709.78	809.78
	v	0.03262	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.2407	[illegible]	0.2734	[illegible]	[illegible]	[illegible]	0.3403	[illegible]	[illegible]
	h	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1577.0	1640.4	1703.0	[illegible]
	s	[illegible]	[illegible]	1.2283	[illegible]	[illegible]	[illegible]	[illegible]	1.4777	1.5032	[illegible]	[illegible]	[illegible]	[illegible]	1.6259	1.6605	[illegible]
3000 (695.33)	Sh			4.67	54.67	104.67	154.67	204.67	254.67	304.67	354.67	404.67	454.67	504.67	604.67	704.67	804.67
	v	0.03428	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.3753
	h	801.84	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1701.4	[illegible]
	s	0.9728	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.4717	1.4976	1.5213	1.5434	[illegible]	1.5841	1.6214	1.6561	[illegible]
3100 (700.28)	Sh				49.72	99.72	149.72	199.72	249.72	299.72	349.72	399.72	449.72	499.72	599.72	699.72	799.72
	v	0.03681	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	0.2237	[illegible]	0.2533	[illegible]	[illegible]	0.2927	0.3170	0.3403	0.3628
	h	823.97	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1572.5	1636.7	[illegible]	1762.5
	s	0.9914	1.1373		1.3007	[illegible]	[illegible]	[illegible]	1.4658	[illegible]	1.5161	[illegible]	[illegible]	1.5794	1.6169	[illegible]	1.6847
3200 (705.08)	Sh				44.92	94.92	144.92	194.92	244.92	294.92	344.92	394.92	444.92	494.92	594.92	694.92	794.92
	v	0.04472	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	0.2151	0.2301	0.2442	0.2576	[illegible]	0.2827	0.3065	0.3291	0.3510
	h	875.54	[illegible]		1172.3	[illegible]	[illegible]	[illegible]	1394.9	[illegible]	1469.2	[illegible]	[illegible]	1570.3	1634.8	1698.3	1761.2
	s	1.0351	1.0832		1.2877	[illegible]	[illegible]	[illegible]	1.4600	1.4866	1.5110	1.5335	1.5547	1.5749	1.6126	1.6477	1.6806
3300	Sh																
	v				0.1213	0.1510	0.1727	[illegible]	0.2070	0.2218	0.2357	0.2489	0.2613	0.2734	0.2966	0.3187	0.3400
	h				1155.2	1242.5	[illegible]	[illegible]	1390.7	1429.5	1466.1	[illegible]	[illegible]	[illegible]	1632.9	1696.7	1759.9
	s				1.2742	1.3425	1.3879	[illegible]	1.4542	1.4813	1.5059	1.5287	1.5501	1.5704	1.6084	1.6435	1.6767
3400	Sh																
	v				0.1129	0.1435	0.1653	[illegible]	0.1994	0.2140	0.2276	0.2405	0.2528	0.2646	0.2872	0.3088	0.3296
	h				[illegible]	[illegible]	[illegible]	[illegible]	1386.4	1425.9	1462.9	1498.3	[illegible]	[illegible]	1631.1	1695.1	[illegible]
	s				1.2600	1.3334	1.3807	[illegible]	1.4486	1.4761	1.5010	1.5240	1.5456	1.5660	1.6042	1.6396	[illegible]
3500	Sh																
	v				0.1046	0.1364	0.1583	0.1764	0.1922	0.2066	0.2200	0.2326	0.2447	0.2563	0.2784	0.2995	0.3198
	h				1137.1	1224.6	1287.8	[illegible]	1382.2	[illegible]	1459.7	1495.5	[illegible]	1563.6	1629.2	1693.6	1757.2
	s				1.2450	1.3242	1.3734	1.4112	1.4433	1.4709	1.4962	1.5194	1.5412	1.5616	1.6002	1.6358	1.6691
3600	Sh																
	v				0.0966	0.1296	0.1517	[illegible]	0.1854	[illegible]	0.2128	0.2252	[illegible]	0.2485	0.2702	0.2908	0.3106
	h				[illegible]	1215.3	[illegible]	[illegible]	1377.9	1418.6	1456.5	1492.6	[illegible]	[illegible]	[illegible]	[illegible]	1755.9
	s				1.2281	1.3148	1.3662	[illegible]	1.4374	1.4653	1.4914	1.5149	[illegible]	1.5576	1.5962	1.6320	1.6654
3800	Sh																
	v				[illegible]	[illegible]	[illegible]	[illegible]	0.1729	[illegible]	[illegible]	0.2116	[illegible]	0.2340	0.2549	0.2746	0.2936
	h				1064.2	1195.5	[illegible]	[illegible]	[illegible]	1411.2	1450.1	[illegible]	[illegible]	[illegible]	1623.6	1688.9	1753.2
	s				[illegible]	1.2955	[illegible]	[illegible]	1.4265	1.4558	1.4821	[illegible]	1.5284	1.5495	[illegible]	1.6247	[illegible]
4000	Sh																
	v				0.0631	[illegible]	[illegible]	[illegible]	0.1615	0.1752	0.1877	0.1994	[illegible]	[illegible]	0.2411	[illegible]	[illegible]
	h				1007.4	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1685.7	[illegible]
	s				1.1396	1.2754	1.3371	[illegible]	1.4159	1.4461	1.4730	1.4976	1.5203	1.5417	1.5812	1.6177	1.6516
4200	Sh																
	v				[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.2287	0.2470	0.2645
	h				[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1682.6	1748.0
	s				[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.5341	1.5742	[illegible]	1.6452
4400	Sh																
	v				[illegible]	[illegible]	[illegible]	[illegible]	0.1420	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	h				[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	s				[illegible]	[illegible]	[illegible]	[illegible]	1.3949	[illegible]	1.4556	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

Sh = superheat, F
v = specific volume, cu ft per lb
h = enthalpy, Btu per lb
s = entropy, Btu per F per lb

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Table 3. Superheated Steam – Continued

Abs Press Lb/Sq In (Sat Temp)		Sat Water	Sat Steam	750	800	850	900	950	1000	1050	1100	1150	1200	1250	1300	1400	1500
4600	Sh																
	v			0.0360	0.0751	0.1025	0.1185	0.1329	0.1465	0.1587	0.1697	0.1792	0.1893	0.1982	0.2071	0.2242	0.2404
	h			795.5	[illegible]	1423.3	1277.2	[illegible]	1362.5	1403.7	1439.9	1500.9	1533.4	1570.3	1603.5	1676.3	1742.7
	s			1.0321	1.2134	1.2912	1.3446	1.3847	1.4181	1.4472	1.4734	1.4974	1.5197	1.5407	1.5607	1.5552	1.6330
4800	Sh																
	v			0.0355	0.0665	0.0927	0.1129	0.1257	0.1385	0.1500	0.1605	0.1704	0.1800	0.1890	0.1977	0.2142	0.2299
	h			815.9	[illegible]	[illegible]	1265.7	[illegible]	[illegible]	1417.0	1458.0	1494.7	[illegible]	1565.7	1604.7	1673.1	1740.0
	s			10.85	1.1835	1.2748	1.3327	1.3745	1.4090	1.4393	1.4657	1.4901	1.5123	1.5341	1.5543	1.5921	1.6272
5000	Sh																
	v			0.0338	0.0591	0.0875	0.1025	0.1161	0.1312	0.1425	0.1529	0.1626	0.1718	0.1806	0.1890	0.2050	0.2203
	h			854.9	1042.4	1221.6	1252.9	[illegible]	1363.9	1410.2	1467.1	[illegible]	1523.1	1565.9	1600.9	1670.9	1737.4
	s			1.0070	1.1833	1.2602	1.3207	1.3645	1.4001	1.4309	1.4582	1.4831	1.5061	1.5277	1.5481	1.5863	1.6216
5200	Sh																
	v			0.0326	0.0531	0.0733	0.0973	[illegible]	0.1244	0.1355	0.1459	0.1557	0.1642	0.1728	0.1810	0.1965	0.2114
	h			845.9	1016.4	1159.0	1240.4	[illegible]	1354.5	1403.4	1447.2	1496.3	1524.5	1561.3	1597.2	1666.8	1734.7
	s			0.9985	1.1373	1.2455	1.3024	1.3565	1.3914	1.4225	1.4505	1.4762	1.4995	1.5214	1.5420	1.5806	1.6161
5400	Sh																
	v			0.0317	0.0483	0.0773	0.0912	0.1058	0.1182	0.1292	0.1392	0.1485	0.1572	0.1656	0.1736	0.1888	0.2031
	h			835.5	994.3	1133.1	1227.7	1292.7	1346.4	1396.5	1440.3	1480.1	1517.6	1557.1	1593.4	1663.7	1732.1
	s			0.9915	1.1175	1.2296	1.2989	1.3444	1.3827	1.4151	1.4433	1.4696	1.4931	1.5153	1.5362	1.5750	1.6109
5600	Sh																
	v			0.0309	0.0447	0.0672	0.0856	0.1001	0.1124	0.1232	0.1330	0.1422	0.1508	0.1589	0.1667	0.1815	0.1954
	h			822.4	975.3	1113.9	1214.8	1283.7	1342.2	1389.5	1434.3	1475.9	1513.7	1552.9	1589.6	1660.5	1729.5
	s			0.9855	1.1003	1.2137	1.2850	1.3348	1.3742	1.4075	1.4366	1.4625	1.4869	1.5093	1.5304	1.5697	1.6058
5800	Sh																
	v			0.0303	0.0415	0.0622	0.0805	0.0949	0.1070	0.1177	0.1274	0.1363	0.1447	0.1527	0.1603	0.1747	0.1883
	h			827.3	958.9	1101.3	1201.8	1273.6	1332.0	1382.6	1428.3	1472.4	1510.5	1548.7	1585.8	1657.4	1726.8
	s			0.9803	1.0887	1.1981	1.2732	1.3253	1.3658	1.3999	1.4297	1.4564	1.4808	1.5035	1.5248	1.5644	1.6004
6000	Sh																
	v			0.0293	0.0397	0.0579	0.0757	0.0900	0.1020	0.1126	0.1221	0.1309	0.1391	0.1469	0.1544	0.1684	0.1817
	h			822.9	945.1	1084.6	1196.8	1261.4	1323.6	1375.7	1422.3	1465.4	1505.9	1544.6	1582.0	1654.2	1724.2
	s			0.9758	1.0745	1.1833	1.2615	1.3154	1.3574	1.3925	1.4229	1.4500	1.4748	1.4978	1.5194	1.5593	1.5960
6500	Sh																
	v			0.0287	0.0358	0.0495	0.0655	0.0793	0.0909	0.1012	0.1104	0.1188	0.1266	0.1340	0.1411	0.1544	0.1669
	h			813.9	919.5	1045.7	1156.3	1237.8	1302.7	1358.1	1407.3	1452.2	1494.2	1534.1	1572.5	1646.4	1717.6
	s			0.9661	1.0515	1.1506	1.2328	1.2917	1.3370	1.3743	1.4064	1.4347	1.4604	1.4841	1.5062	1.5471	1.5844
7000	Sh																
	v			0.0279	0.0334	0.0438	0.0573	0.0704	0.0816	0.0915	0.1004	0.1085	0.1160	0.1231	0.1298	0.1424	0.1542
	h			825.9	[illegible]	1016.5	1124.9	1212.6	1281.7	1340.5	1392.2	1439.1	1482.6	1523.7	1563.1	1638.6	1711.1
	s			0.9582	1.0350	1.1243	1.2055	1.2689	1.3171	1.3567	1.3904	1.4200	1.4466	1.4710	1.4938	1.5355	1.5735
7500	Sh																
	v			0.0272	0.0318	0.0399	0.0512	0.0631	0.0737	0.0833	0.0918	0.0996	0.1068	0.1136	0.1200	0.1321	0.1433
	h			831.3	886.0	992.9	1097.7	1188.2	1261.0	1322.9	1377.2	1425.6	1470.3	1513.3	1553.7	1630.8	1704.6
	s			0.9514	1.0224	1.1035	1.1818	1.2473	1.2980	1.3397	1.3751	1.4053	1.4335	1.4586	1.4819	1.5245	1.5632
8000	Sh																
	v			0.0267	0.0306	0.0371	0.0465	0.0571	0.0671	0.0762	0.0845	0.0920	0.0989	0.1054	0.1115	0.1230	0.1338
	h			796.5	875.1	971.4	1074.3	1165.4	1241.0	1305.5	1362.2	1412.0	1459.5	1502.1	1544.5	1623.1	1698.1
	s			0.9455	1.0122	1.0864	1.1613	1.2271	1.2798	1.3233	1.3603	1.3924	1.4203	1.4457	1.4705	1.5140	1.5533
8500	Sh																
	v			0.0262	0.0296	0.0350	0.0429	0.0522	0.0615	0.0701	0.0780	0.0852	0.0919	0.0982	0.1041	0.1151	0.1254
	h			792.7	867.2	955.8	1054.5	1144.0	1221.5	1288.5	1347.5	1400.2	1448.2	1492.9	1535.3	1615.4	1691.7
	s			0.9402	1.0037	1.0727	1.1437	1.2084	1.2627	1.3076	1.3460	1.3793	1.4087	1.4352	1.4597	1.5040	1.5439
9000	Sh																
	v			0.0258	0.0288	0.0335	0.0402	0.0483	0.0565	0.0649	0.0724	0.0794	0.0858	0.0918	0.0975	0.1081	0.1179
	h			789.3	864.7	944.5	1037.6	1125.4	1204.1	1272.1	1333.0	1387.5	1437.1	1482.9	1526.3	1607.9	1685.3
	s			0.9354	0.9964	1.0613	1.1285	1.1918	1.2468	1.2926	1.3323	1.3667	1.3970	1.4243	1.4492	1.4944	1.5349
9500	Sh																
	v			0.0254	0.0282	0.0322	0.0380	0.0451	0.0528	0.0603	0.0675	0.0742	0.0804	0.0862	0.0917	0.1019	0.1113
	h			786.4	859.2	939.3	1022.4	1105.9	1186.7	1256.5	1318.9	1375.1	1426.1	1473.1	1517.3	1600.4	1679.0
	s			0.9310	0.9905	1.0506	1.1153	1.1771	1.2320	1.2785	1.3191	1.3546	1.3853	1.4137	1.4392	1.4851	1.5263
10000	Sh																
	v			0.0251	0.0276	0.0312	0.0362	0.0425	0.0495	0.0565	0.0633	0.0697	0.0757	0.0812	0.0865	0.0963	0.1054
	h			783.8	849.5	930.2	1011.3	1094.2	1170.6	1242.0	1305.3	1362.9	1415.3	1463.4	1508.6	1592.1	1672.8
	s			0.9270	0.9842	1.0432	1.1049	1.1636	1.2185	1.2652	1.3065	1.3429	1.3749	1.4035	1.4295	1.4763	1.5180
10500	Sh																
	v			0.0248	0.0271	0.0303	0.0347	0.0404	0.0467	0.0532	0.0595	0.0655	0.0714	0.0769	0.0818	0.0913	0.1001
	h			781.5	845.5	923.4	1001.0	[illegible]	1155.5	1228.4	1292.4	[illegible]	1404.7	1453.3	1500.0	1555.8	1665.7
	s			0.9232	0.9790	1.0354	1.0939	1.1519	1.2060	1.2529	1.2945	1.3311	1.3644	1.3937	1.4202	1.4677	1.5100

Sh = superheat, F
v = specific volume, cu ft per lb

h = enthalpy, Btu per lb
s = entropy, Btu per F per lb

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Table 3. Superheated Steam – *Continued*

Abs Press. Lb Sq In (Sat. Temp)		Sat Water	Sat Steam	Temperature – Degrees Fahrenheit 750	800	850	900	950	1000	1050	1100	1150	1200	1250	1300	1400	1500
11000	v			0.0245	0.0267	0.0296	0.0335	0.0385	0.0442	0.0503	0.0550	0.0620	0.0676	0.0777	0.0776	0.0868	0.0952
	h			779.5	846.3	917.5	993.1	1033.9	1146.3	1216.9	1290.2	1339.7	1354.4	1444.6	1491.5	1578.7	1660.5
	s			0.9195	0.9742	1.0202	1.0851	1.1412	1.1945	1.2414	1.2833	1.3200	1.3544	1.3842	1.4112	1.4595	1.5023
11500	v			0.0243	0.0263	0.0290	0.0325	0.0370	0.0423	0.0478	0.0531	0.0583	0.0641	0.0691	0.0739	0.0827	0.0909
	h			777.7	838.8	912.4	964.5	1019.8	1124.9	1204.3	1259.7	1328.8	1384.4	1435.5	1483.2	1571.8	1655.7
	s			0.9163	0.9605	1.0232	1.0772	1.1316	1.1840	1.2356	1.2727	1.3107	1.3446	1.3750	1.4025	1.4515	1.4949
12000	v			0.0241	0.0259	0.0288	0.0317	0.0357	0.0406	0.0455	0.0508	0.0560	0.0610	0.0659	0.0704	0.0790	0.0869
	h			776.1	[illegible]	907.9	977.8	1005.9	[illegible]	1196.7	1259.0	1312.5	1374.7	1426.6	1475.1	1564.9	1643.8
	s			0.9131	0.9657	1.0177	1.0701	1.1229	1.1742	1.2209	1.2627	1.3010	1.3353	1.3662	1.3941	1.4438	1.4877
12500	v			0.0238	0.0256	0.0279	0.0307	0.0345	0.0390	0.0437	0.0486	0.0535	0.0583	0.0629	0.0673	0.0755	0.0832
	h			774.7	835.6	903.9	971.5	1043.1	1115.2	1184.1	1247.9	1305.8	1365.4	1416.0	1467.2	1555.2	1641.1
	s			0.9103	0.9618	1.0127	1.0637	1.1151	1.1653	1.2117	1.2534	1.2916	1.3264	1.3576	1.3820	1.4363	1.4808
13000	v			0.0236	0.0253	0.0275	0.0302	0.0336	0.0376	0.0420	0.0466	0.0517	0.0558	0.0602	0.0645	0.0725	0.0799
	h			773.5	835.3	900.4	956.8	1018.2	1105.7	1174.8	1238.5	1299.6	1356.5	1409.6	1459.4	1551.6	1627.4
	s			0.9073	0.9582	1.0033	1.0578	1.1079	1.1571	1.2030	1.2445	1.2831	1.3179	1.3494	1.3761	1.4291	1.4742
13500	v			0.0235	0.0251	0.0271	0.0297	0.0328	0.0364	0.0405	0.0448	0.0492	0.0535	0.0577	0.0619	0.0695	0.0768
	h			772.3	824.4	897.2	952.2	1030.0	1099.1	1165.3	1225.7	1291.0	1348.1	1401.5	1451.8	1545.2	[illegible]
	s			0.9045	0.9548	1.0037	1.0524	1.1014	1.1495	1.1948	1.2361	1.2749	1.3098	1.3415	1.3705	1.4221	1.4675
14000	v			0.0233	0.0248	0.0267	0.0291	0.0320	0.0354	0.0392	0.0432	0.0474	0.0515	0.0555	0.0595	0.0670	0.0740
	h			771.3	832.6	894.3	955.0	1024.5	1032.3	1156.5	1221.4	1253.0	1340.2	1393.8	1444.4	1539.8	1626.5
	s			0.9019	0.9515	0.9996	1.0473	1.0953	1.1426	1.1872	1.2282	1.2671	1.3021	1.3339	1.3631	1.4153	1.4512
14500	v			0.0231	0.0246	0.0264	0.0287	0.0314	0.0345	0.0380	0.0418	0.0458	0.0495	0.0534	0.0573	0.0646	0.0714
	h			770.4	831.0	891.7	954.3	1019.6	1086.2	1151.4	1213.5	1275.4	1332.9	1385.4	1437.3	1532.6	1621.1
	s			0.8994	0.9484	0.9957	1.0426	1.0897	1.1362	1.1801	1.2203	1.2597	1.2943	1.3266	1.3560	1.4097	1.4551
15000	v			0.0230	0.0244	0.0261	0.0282	0.0309	0.0337	0.0369	0.0405	0.0443	0.0479	0.0516	0.0552	0.0624	0.0690
	h			769.6	829.5	889.1	950.9	1015.1	1033.6	1144.9	1205.8	1267.1	1326.0	1379.4	1430.3	1525.4	1615.9
	s			0.8970	0.9455	0.9920	1.0382	1.0846	1.1302	1.1735	1.2139	1.2525	1.2850	1.3197	1.3491	1.4022	1.4491
15500	v			0.0228	0.0242	0.0258	0.0278	0.0302	0.0329	0.0360	0.0393	0.0429	0.0464	0.0499	0.0531	0.0603	0.0668
	h			768.9	828.2	887.2	947.8	1011.1	1075.7	1132.9	1205.3	1261.1	1319.5	1372.8	1423.6	1520.4	1610.8
	s			0.8946	0.9427	0.9886	1.0340	1.0797	1.1247	1.1674	1.2073	1.2457	1.2815	1.3131	1.3424	1.3959	1.4433

Sh = superheat, F
v = specific volume, cu ft per lb
h = enthalpy, Btu per lb
s = entropy, Btu per F per lb

CONVERSION FACTORS

TO OBTAIN	MULTIPLY	BY
Acres	Sq miles	640.0
Atmospheres	Cm of Hg @ 0 deg C	0.013158
Atmospheres	Ft of H_2O @ 39.2 F	0.029499
Atmospheres	Grams/sq cm	0.00096784
Atmospheres	In. Hg @ 32 F	0.033421
Atmospheres	In. H_2O @ 39.2 F	0.0024583
Atmospheres	Pounds/sq ft	0.00047254
Atmospheres	Pounds/sq in.	0.068046
Btu	Ft-lb	0.0012854
Btu	Hp-hr	2545.1
Btu	Kg-cal.	3.9685
Btu	Kw-hr	3413
Btu	Watt-hr	3.4130
Btu/(cu ft) (hr)	Kw/liter	96,650.6
Btu/hr	Mech. hp	2545.1
Btu/hr	Kw	3413
Btu/hr	Tons of refrigeration	12,000
Btu/hr	Watts	3.4127
Btu/kw hr	Kg cal/kw hr	3.9685
Btu/(hr) (ft) (deg F)	Cal/(sec) (cm) (deg C)	241.90
Btu/(hr) (ft) (deg F)	Joules/(sec) (cm) (deg C)	57.803
Btu/(hr) (ft) (deg F)	Watts/(cm) (deg C)	57.803
Btu/(hr) (sq ft)	Cal/(sec) (sq cm)	13,273.0
Btu/min	Ft-lb/min	0.0012854
Btu/min	Mech. hp	42.418
Btu/min	Kw	56.896
Btu/lb	Cal/gram	1.8
Btu/lb	Kg cal/kg	1.8
Btu/(lb) (deg F)	Cal/(gram) (deg C)	1.0
Btu/(lb) (deg F)	Joules/(gram) (deg C)	0.23889
Btu/sec	Mech. hp	0.70696
Btu/sec	Mech. hp (metric)	0.6971
Btu/sec	Kg-cal/hr	0.0011024
Btu/sec	Kw	0.94827
Btu/sq ft	Kg-cal/sq meter	0.36867
Calories	Ft-lb	0.32389
Calories	Joules	0.23889
Calories	Watt-hr	860.01
Cal/(cu cm) (sec)	Kw/liter	0.23868
Cal/gram	Btu/lb	0.55556
Cal/(gram) (deg C)	Btu/(lb) (deg F)	1.0
Cal/(sec) (cm) (deg C)	Btu/(hr) (ft) (deg F)	0.0041336
Cal/(sec) (sq cm)	Btu/(hr) (sq ft)	0.000075341
Cal/(sec) (sq cm) (deg C)	Btu/(hr) (sq ft) (deg F)	0.0001355
Centimeters	Inches	2.540
Centimeters	Microns	0.0001
Centimeters	Mils	0.002540
Cm of Hg @ 0 deg C	Atmospheres	76.0
Cm of Hg @ 0 deg C	Ft of H_2O @ 39.2 F	2.242
Cm of Hg @ 0 deg C	Grams/sq cm	0.07356
Cm of Hg @ 0 deg C	In. of H_2O @ 4 C	0.1868

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CONVERSION FACTORS—(Continued)

TO OBTAIN	MULTIPLY	BY
Cm of Hg @ 0 deg C	Lb/sq in.	5.1715
Cm of Hg @ 0 deg C	Lb/sq ft	0.035913
Cm/deg C	In./deg F	4.5720
Cm/sec	Ft/min	0.508
Cm/sec	Ft/sec	30.48
Cm/(sec) (sec)	Gravity	980.665
Cm of H_2O @ 39.2 F	Atmospheres	1033.24
Cm of H_2O @ 39.2 F	Lb/sq in.	70.31
Centipoises	Centistokes	Density
Centistokes	Centipoises	1/density
Cu cm	Cu ft	28,317
Cu cm	Cu in.	16.387
Cu cm	Gal. (USA, liq.)	3785.43
Cu cm	Liters	1000.03
Cu cm	Ounces (USA, liq.)	29.573730
Cu cm	Quarts (USA, liq.)	946.358
Cu cm/sec	Cu ft/min	472.0
Cu ft	Cords (wood)	128.0
Cu ft	Cu meters	35.314
Cu ft	Cu yards	27.0
Cu ft	Gal. (USA, liq.)	0.13368
Cu ft	Liters	0.03532
Cu ft/min	Cu meters/sec	2118.9
Cu ft/min	Gal. (USA, liq./sec)	8.0192
Cu ft/lb	Cu meters/kg	16.02
Cu ft/lb	Liters/kg	0.01602
Cu ft/sec	Cu meters/min	0.5886
Cu ft/sec	Gal. (USA, liq.)/min	0.0022280
Cu ft/sec	Liters/min	0.0005886
Cu in.	Cu centimeters	0.061023
Cu in.	Gal. (USA, liq.)	231.0
Cu in.	Liters	61.03
Cu in.	Ounces (USA, liq.)	1.805
Cu meters	Cu ft	0.028317
Cu meters	Cu yards	0.7646
Cu meters	Gal. (USA, liq.)	0.0037854
Cu meters	Liters	0.001000028
Cu meters/hr	Gal./min	0.22712
Cu meters/kg	Cu ft/lb	0.062428
Cu meters/min	Cu ft/min	0.02832
Cu meters/min	Gal./sec	0.22712
Cu meters/sec	Gal./min	0.000063088
Cu yards	Cu meters	1.3079
Dynes	Grams	980.66
Dynes	Pounds (avoir.)	444820.0
Dyne-centimeters	Ft-lb	13,558,000
Dynes/sq cm	Lb/sq in.	68947
Ergs	Joules	10,000,000
Feet	Meters	3.281
Ft of H_2O @ 39.2 F	Atmospheres	33.899
Ft of H_2O @ 39.2 F	Cm of Hg @ 0 deg C	0.44604

CONVERSION FACTORS—(Continued)

TO OBTAIN	MULTIPLY	BY
Ft of H_2O @ 39.2 F	In. of Hg @ 32 deg F	1.1330
Ft of H_2O @ 39.2 F	Lb/sq ft	0.016018
Ft of H_2O @ 39.2 F	Lb/sq in.	2.3066
Ft/min	Cm/sec	1.9685
Ft/min	Miles (USA, statute)/hr	88.0
Ft/sec	Knots	1.6889
Ft/sec	Meters/sec	3.2808
Ft/sec	Miles (USA, statute)/hr	1.4667
Ft/(sec) (sec)	Gravity (sea level)	32.174
Ft/(sec) (sec)	Meters/(sec) (sec)	3.2808
Ft-lb	Btu	778.0
Ft-lb	Joules	0.73756
Ft-lb	Kg-calories	3087.4
Ft-lb	Kw-hr	2,655,200
Ft-lb	Mech. hp-hr	1,980,000
Ft-lb/min	Btu/min	778.0
Ft-lb/min	Kg cal/min	3087.4
Ft-lb/min	Kw	44,254.0
Ft-lb/min	Mech. hp	33,000
Ft-lb/sec	Btu/min	12.96
Ft-lb/sec	Kw	737.56
Ft-lb/sec	Mech. hp	550.0
Gal. (Imperial, liq.)	Gal. (USA, liq.)	0.83268
Gal. (USA, liq.)	Barrels (petroleum, USA)	42
Gal. (USA, liq.)	Cu ft	7.4805
Gal. (USA, liq.)	Cu meters	264.173
Gal. (USA, liq.)	Cu yards	202.2
Gal. (USA, liq.)	Gal. (Imperial, liq.)	1.2010
Gal. (USA, liq.)	Liters	0.2642
Gal. (USA, liq.)/min	Cu ft/sec	448.83
Gal. (USA, liq.)/min	Cu meters/hr	4.4029
Gal. (USA, liq.)/sec	Cu ft/min	0.12468
Gal. (USA, liq.)/sec	Liters/min	0.0044028
Grains	Grams	15.432
Grains	Ounces (avoir.)	437.5
Grains	Pounds (avoir.)	7000
Grains/gal. (USA, liq.)	Parts/million	0.0584
Grams	Grains	0.0648
Grams	Ounces (avoir.)	28.350
Grams	Pounds (avoir.)	453.5924
Grams/cm	Pounds/in.	178.579
Grams/(cm) (sec)	Centipoises	0.01
Grams/cu cm	Lb/cu ft	0.016018
Grams/cu cm	Lb/cu in.	27.680
Grams/cu cm	Lb/gal.	0.119826
Gravity (at sea level)	Ft/(sec) (sec)	0.03108
Inches	Centimeters	0.3937
Inches	Microns	0.00003937
Inches of Hg @ 32 F	Atmospheres	29.921
Inches of Hg @ 32 F	Ft of H_2O @ 39.2 F	0.88265
Inches of Hg @ 32 F	Lb/sq in.	2.0360

CONVERSION FACTORS – (Continued)

TO OBTAIN	MULTIPLY	BY
Inches of Hg @ 32 F	In. of H_2O @ 4 C	0.07355
Inches of H_2O @ 4 C	In. of Hg @ 32 F	13.60
Inches of H_2O @ 39.2 F	Lb/sq in.	27.673
Inches/deg F	Cm/deg C	0.21872
Joules	Btu	1054.8
Joules	Calories	4.186
Joules	Ft-lb	1.35582
Joules	Kg-meters	9.807
Joules	Kw-hr	3,600,000
Joules	Mech. hp-hr	2,684,500
Kg	Pounds (avoir.)	0.45359
Kg-cal	Btu	0.2520
Kg-cal	Ft-lb	0.00032389
Kg-cal	Joules	0.0002389
Kg-cal	Kw-hr	860.01
Kg-cal	Mech. hp-hr	641.3
Kg-cal/kg	Btu/lb	0.5556
Kg-cal/kw hr	Btu/kw hr	0.2520
Kg-cal/min	Ft-lb/min	0.0003239
Kg-cal/min	Kw	14.33
Kg-cal/min	Mech. hp	10.70
Kg-cal/sq meter	Btu/sq ft.	2.712
Kg/cu meter	Lb/cu ft	16.018
Kg/(hr) (meter)	Centipoises	3.60
Kg/liter	Lb/gal. (USA, liq.)	0.11983
Kg/meter	Lb/ft	1.488
Kg/sq cm	Atmospheres	1.0332
Kg/sq cm	Lb/sq in.	0.0703
Kg/sq meter	Lb/sq ft	4.8824
Kg/sq meter	Lb/sq in.	703.07
Km	Miles (USA, statute)	1.6093
Kw	Btu/min	0.01758
Kw	Ft-lb/min	0.00002259
Kw	Ft-lb/sec	0.00135582
Kw	Kg-cal/hr	0.0011628
Kw	Kg-cal/min	0.069767
Kw	Mech. hp	0.7457
Kw-hr	Btu	0.000293
Kw-hr	Ft-lb	0.0000003766
Kw-hr	Kg-cal	0.0011628
Kw-hr	Mech. hp-hr	0.7457
Knots	Ft/sec	0.5921
Knots	Miles/hr	0.8684
Liters	Cu ft	28.316
Liters	Cu in.	0.01639
Liters	Cu meters	999.973
Liters	Gal. (Imperial, liq.)	4.546
Liters	Gal. (USA, liq.)	3.78533
Liters/kg	Cu ft/lb	62.42621
Liters/min	Cu ft/sec	1699.3
Liters/min	Gal. (USA, liq.)/min	3.785

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CONVERSION FACTORS—(Continued)

TO OBTAIN	MULTIPLY	BY
Liters/sec	Cu ft/min	0.47193
Liters/sec	Gal./min	0.063083
Mech. hp	Btu/hr	0.0003929
Mech. hp	Btu/min	0.023575
Mech. hp	Ft-lb/sec	0.0018182
Mech. hp	Kg-cal/min	0.093557
Mech. hp	Kw	1.3410
Mech. hp-hr	Btu	0.00039292
Mech. hp-hr	Ft-lb	0.00000050505
Mech. hp-hr	Kg-calories	0.0015593
Mech. hp-hr	Kw-hr	1.3410
Meters	Feet	0.3048
Meters	Inches	0.0254
Meters	Miles (Int., nautical)	1852.0
Meters	Miles (USA, statute)	1609.344
Meters/min	Ft/min	0.3048
Meters/min	Miles (USA, statute)/hr	26.82
Meters/sec	Ft/sec	0.3048
Meters/sec	Km/hr	0.2778
Meters/sec	Knots	0.5148
Meters/sec	Miles (USA, statute)/hr	0.44704
Meters/(sec) (sec)	Ft/(sec) (sec)	0.3048
Microns	Inches	25,400
Microns	Mils	25.4
Miles (Int., nautical)	Km	0.54
Miles (Int., nautical)	Miles (USA, statute)	0.8690
Miles (Int., nautical)/hr	Knots	1.0
Miles (USA, statute)	Km	0.6214
Miles (USA, statute)	Meters	0.0006214
Miles (USA, statute)	Miles (Int., nautical)	1.151
Miles (USA, statute)/hr	Knots	1.151
Miles (USA, statute)/hr	Ft/min	0.011364
Miles (USA, statute)/hr	Ft/sec	0.68182
Miles (USA, statute)/hr	Meters/min	0.03728
Miles (USA, statute)/hr	Meters/sec	2.2369
Milliliters/gram	Cu ft/lb	62.42621
Millimeters	Microns	0.001
Mils	Centimeters	393.7
Mils	Inches	1000
Mils	Microns	0.03937
Minutes	Radians	3437.75
Ounces (avoir.)	Grains (avoir.)	0.0022857
Ounces (avoir.)	Grams	0.035274
Ounces (USA, liq.)	Gal. (USA, liq.)	128.0
Parts/million	Gr/gal. (USA, liq.)	17.118
Percent grade	Ft/100 ft	1.0
Pounds (avoir.)	Grains	0.0001429
Pounds (avoir.)	Grams	0.0022046
Pounds (avoir.)	Kg	2.2046
Pounds (avoir.)	Tons, long	2240
Pounds (avoir.)	Tons, metric	2204.6

CONVERSION FACTORS—(Continued)

TO OBTAIN	MULTIPLY	BY
Pounds (avoir.)	Tons, short	2000
Pounds/cu ft	Grams/cu cm	62.428
Pounds/cu ft	Kg/cu meter	0.062428
Pounds/cu ft	Pounds/gal.	7.48
Pounds/cu in.	Grams/cu cm	0.036127
Pounds/ft	Kg/meter	0.67197
Pounds/hr	Kg/min	132.28
Pounds/(hr) (ft)	Centipoises	2.42
Pounds/inch	Grams/cm	0.0056
Pounds/(sec) (ft)	Centipoises	0.000672
Pounds/sq inch	Atmospheres	14.696
Pounds/sq inch	Cm of Hg @ 0 deg C	0.19337
Pounds/sq inch	Ft of H_2O @ 39.2 F	0.43352
Pounds/sq inch	In. Hg @ 32 F	0.491
Pounds/sq inch	In. H_2O @ 39.2 F	0.0361
Pounds/sq inch	Kg/sq cm	14.223
Pounds/sq inch	Kg/sq meter	0.0014223
Pounds/gal. (USA, liq.)	Kg/liter	8.3452
Pounds/gal. (USA, liq.)	Pounds/cu ft	0.1337
Pounds/gal. (USA, liq.)	Pounds/cu inch	231
Quarts (USA, liq.)	Cu cm	0.0010567
Quarts (USA, liq.)	Cu in.	0.01732
Quarts (USA, liq.)	Liters	1.057
Sq centimeters	Sq ft	929.0
Sq centimeters	Sq inches	6.4516
Sq ft	Acres	43,560
Sq ft	Sq meters	10.764
Sq inches	Sq centimeters	0.155
Sq meters	Acres	4046.9
Sq meters	Sq ft	0.0929
Sq miles (USA, statute)	Acres	0.001562
Sq mils	Sq cm	155,000
Sq mils	Sq inches	1,000,000
Tons (metric)	Tons (short)	0.9072
Tons (short)	Tons (metric)	1.1023
Watts	Btu/sec	1054.8
Yards	Meters	1.0936

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Table 3. Superheated Steam

Abs Press. Lb Sq In (Sat. Temp)		Sat. Water	Sat. Steam	Temperature — Degrees Fahrenheit 200	250	300	350	400	450	500	600	700	800	900	1000	1100	1200
1	Sh			98.26	148.26	198.26	248.26	298.26	348.26	398.26	498.26	598.26	698.26	798.26	898.26	998.26	1098.26
(101.74)	v	0.01614	333.6	392.5	422.4	452.3	482.1	511.9	541.7	571.5	631.1	690.7	750.3	809.9	869.4	929.1	988.7
	h	69.70	1105.8	1150.2	1172.9	1195.7	1218.7	1241.8	1265.1	1288.6	1336.1	1384.5	1431.0	1484.9	1534.4	1588.0	1635.4
	s	0.1326	1.9781	2.0509	2.0841	2.1152	2.1445	2.1722	2.1985	2.2237	2.2708	2.3144	2.3512	2.3882	2.4231	2.4592	2.4918
5	Sh			37.76	87.76	137.76	187.76	237.76	287.76	337.76	437.76	537.76	637.76	737.76	837.76	937.76	1037.76
(162.24)	v	0.01641	73.53	78.14	84.21	90.24	96.25	102.24	108.22	114.21	126.15	138.08	150.01	161.94	173.86	185.76	197.70
	h	130.20	1131.1	1148.6	1171.7	1194.8	1218.0	1241.3	1264.7	1288.2	1335.9	1384.3	1433.5	1483.7	1534.7	1586.7	1639.6
	s	0.2349	1.8443	1.8716	1.9054	1.9369	1.9664	1.9943	2.0208	2.0460	2.0932	2.1369	2.1776	2.2159	2.2521	2.2866	2.3194
10	Sh			6.79	56.79	106.79	156.79	206.79	256.79	306.79	406.79	506.79	606.79	706.79	806.79	906.79	1006.79
(193.21)	v	0.01659	38.42	38.84	41.93	44.98	48.02	51.03	54.04	57.04	63.03	69.00	74.98	80.94	86.91	92.87	98.84
	h	161.26	1143.3	1146.6	1170.2	1193.7	1217.1	1240.6	1264.1	1287.4	1335.5	1384.0	1433.4	1483.5	1534.5	1586.5	1639.5
	s	0.2836	1.7876	1.7928	1.8273	1.8593	1.8892	1.9173	1.9439	1.9692	2.0166	2.0603	2.1011	2.1394	2.1757	2.2102	2.2430
14.696*	Sh				38.00	88.00	138.00	188.00	238.00	288.00	388.00	488.00	588.00	688.00	788.00	888.00	988.00
(212.00)	v	0.01672	26.828		28.44	30.52	32.61	34.65	36.73	38.75	42.83	46.91	50.97	55.03	59.09	63.19	67.25
	h	180.07	1150.4		1168.2	1192.0	1215.4	1239.9	1263.1	1288.4	1333.0	1381.2	1432.5	1481.4	1533.1	1582.7	1635.1
	s	0.3120	1.7566		1.7838	1.8148	1.8446	1.8727	1.8989	1.9238	1.9709	2.0145	2.0551	2.0932	2.1292	2.1634	2.1960
15	Sh				36.97	86.97	136.97	186.97	236.97	286.97	386.97	486.97	586.97	686.97	786.97	886.97	986.97
(213.03)	v	0.01673	26.290		27.837	29.899	31.939	33.963	35.977	37.985	41.986	45.978	49.964	53.946	57.926	61.905	65.882
	h	181.21	1150.9		1168.7	1192.5	1216.2	1239.9	1263.6	1287.3	1335.2	1383.8	1433.2	1483.4	1534.5	1586.5	1639.4
	s	0.3137	1.7552		1.7809	1.8134	1.8437	1.8720	1.8988	1.9242	1.9717	2.0155	2.0563	2.0946	2.1309	2.1653	2.1982
20	Sh				22.04	72.04	122.04	172.04	222.04	272.04	372.04	472.04	572.04	672.04	772.04	872.04	972.04
(227.96)	v	0.01683	20.087		20.788	22.356	23.900	25.428	26.946	28.457	31.466	34.465	37.458	40.447	43.435	46.420	49.405
	h	196.27	1156.3		1167.1	1191.4	1215.4	1239.2	1263.0	1286.9	1334.9	1383.5	1432.9	1483.2	1534.3	1586.3	1639.3
	s	0.3358	1.7320		1.7475	1.7805	1.8111	1.8397	1.8666	1.8921	1.9397	1.9836	2.0244	2.0628	2.0991	2.1336	2.1665
25	Sh				9.93	59.93	109.93	159.93	209.93	259.93	359.93	459.93	559.93	659.93	759.93	859.93	959.93
(240.07)	v	0.01693	16.301		16.558	17.829	19.076	20.307	21.527	22.740	25.153	27.557	29.954	32.348	34.740	37.130	39.518
	h	208.52	1160.6		1165.6	1190.2	1214.5	1238.5	1262.5	1286.4	1334.6	1383.3	1432.7	1483.0	1534.2	1586.2	1639.2
	s	0.3535	1.7141		1.7212	1.7547	1.7856	1.8145	1.8415	1.8672	1.9149	1.9588	1.9997	2.0381	2.0744	2.1089	2.1418
30	Sh					49.66	99.66	149.66	199.66	249.66	349.66	449.66	549.66	649.66	749.66	849.66	949.66
(250.34)	v	0.01701	13.744			14.810	15.859	16.892	17.914	18.929	20.945	22.951	24.952	26.949	28.943	30.936	32.927
	h	218.93	1164.1			1189.0	1213.6	1237.8	1261.9	1286.0	1334.2	1383.0	1432.5	1482.8	1534.0	1586.1	1639.0
	s	0.3682	1.6995			1.7334	1.7647	1.7937	1.8210	1.8467	1.8946	1.9386	1.9795	2.0179	2.0543	2.0888	2.1217
35	Sh					40.71	90.71	140.71	190.71	240.71	340.71	440.71	540.71	640.71	740.71	840.71	940.71
(259.29)	v	0.01708	11.896			12.654	13.562	14.453	15.334	16.207	17.939	19.662	21.379	23.092	24.803	26.512	28.220
	h	228.03	1167.1			1187.8	1212.7	1237.1	1261.3	1285.5	1333.9	1382.8	1432.3	1482.7	1533.9	1586.0	1638.9
	s	0.3809	1.6872			1.7152	1.7468	1.7761	1.8035	1.8294	1.8774	1.9214	1.9624	2.0009	2.0372	2.0717	2.1046
40	Sh					32.75	82.75	132.75	182.75	232.75	332.75	432.75	532.75	632.75	732.75	832.75	932.75
(267.25)	v	0.01715	10.497			11.036	11.838	12.624	13.398	14.165	15.685	17.195	18.699	20.199	21.697	23.194	24.689
	h	236.14	1169.8			1186.6	1211.7	1236.4	1260.8	1285.0	1333.6	1382.5	1432.1	1482.5	1533.7	1585.8	1638.8
	s	0.3921	1.6765			1.6992	1.7312	1.7608	1.7883	1.8143	1.8624	1.9065	1.9476	1.9860	2.0224	2.0569	2.0899
45**	Sh					25.57	75.57	125.57	175.57	225.57	325.57	425.57	525.57	625.57	725.57	825.57	925.57
(274.43)	v	0.01722	9.403			9.782	10.503	11.206	11.897	12.584	13.939	15.284	16.623	17.959	19.292	20.623	21.954
	h	243.47	1172.1			1185.4	1210.8	1235.7	1260.2	1284.6	1333.3	1382.2	1432.0	1482.4	1533.6	1585.7	1638.8
	s	0.4021	1.6671			1.6849	1.7174	1.7472	1.7749	1.8010	1.8492	1.8934	1.9345	1.9730	2.0094	2.0439	2.0769
50	Sh					18.98	68.98	118.98	168.98	218.98	318.98	418.98	518.98	618.98	718.98	818.98	918.98
(281.02)	v	0.01727	8.514			8.769	9.424	10.062	10.688	11.306	12.529	13.741	14.947	16.150	17.350	18.549	19.746
	h	250.21	1174.1			1184.1	1209.9	1234.9	1259.6	1284.1	1332.9	1382.0	1431.7	1482.2	1533.4	1585.6	1638.6
	s	0.4112	1.6586			1.6720	1.7048	1.7349	1.7628	1.7890	1.8374	1.8816	1.9227	1.9613	1.9977	2.0322	2.0652
55**	Sh					12.93	62.93	112.93	162.93	212.93	312.93	412.93	512.93	612.93	712.93	812.93	912.93
(287.07)	v	0.01733	7.787			7.947	8.550	9.134	9.706	10.270	11.385	12.489	13.587	14.682	15.775	16.865	17.954
	h	256.42	1175.9			1182.9	1208.9	1234.3	1259.1	1283.6	1332.6	1381.8	1431.6	1482.0	1533.3	1585.5	1638.5
	s	0.4196	1.6510			1.6602	1.6934	1.7238	1.7518	1.7781	1.8267	1.8710	1.9123	1.9507	1.9871	2.0217	2.0546
60	Sh					7.29	57.29	107.29	157.29	207.29	307.29	407.29	507.29	607.29	707.29	807.29	907.29
(292.71)	v	0.01738	7.174			7.257	7.815	8.354	8.881	9.400	10.425	11.438	12.446	13.450	14.452	15.452	16.450
	h	262.21	1177.6			1181.6	1208.0	1233.5	1258.5	1283.2	1332.3	1381.5	1431.3	1481.8	1533.2	1585.3	1638.4
	s	0.4273	1.6440			1.6492	1.6824	1.7134	1.7417	1.7681	1.8168	1.8612	1.9024	1.9410	1.9774	2.0120	2.0450
65	Sh					2.02	52.02	102.02	152.02	202.02	302.02	402.02	502.02	602.02	702.02	802.02	902.02
(297.98)	v	0.01743	6.653			6.675	7.195	7.697	8.186	8.667	9.615	10.552	11.484	12.412	13.337	14.261	15.183
	h	267.63	1179.1			1180.3	1207.0	1232.7	1257.9	1282.7	1331.9	1381.3	1431.1	1481.6	1533.0	1585.2	1638.3
	s	0.4344	1.6375			1.6390	1.6731	1.7040	1.7324	1.7590	1.8077	1.8522	1.8935	1.9321	1.9685	2.0031	2.0361
70	Sh						47.07	97.07	147.07	197.07	297.07	397.07	497.07	597.07	697.07	797.07	897.07
(302.93)	v	0.01748	6.205				6.664	7.133	7.590	8.039	8.922	9.793	10.659	11.522	12.382	13.240	14.097
	h	272.74	1180.6				1206.0	1232.0	1257.3	1282.2	1331.6	1381.0	1430.9	1481.5	1532.9	1585.1	1638.2
	s	0.4411	1.6316				1.6640	1.6951	1.7237	1.7504	1.7993	1.8439	1.8852	1.9238	1.9603	1.9949	2.0279
75	Sh						42.39	92.39	142.39	192.39	292.39	392.39	492.39	592.39	692.39	792.39	892.39
(307.61)	v	0.01753	5.814				6.204	6.645	7.074	7.494	8.320	9.135	9.945	10.750	11.553	12.355	13.155
	h	277.56	1181.9				1205.0	1231.2	1256.7	1281.7	1331.3	1380.7	1430.7	1481.3	1532.7	1585.0	1638.1
	s	0.4474	1.6260				1.6554	1.6868	1.7156	1.7424	1.7915	1.8361	1.8774	1.9161	1.9526	1.9872	2.0202

Sh = superheat, F
v = specific volume, cu ft per lb
h = enthalpy, Btu per lb
s = entropy, Btu per F per lb

*Values from STEAM TABLES, Properties of Saturated and Superheated Steam, Published by COMBUSTION ENGINEERING, INC., Copyright 1940
**Values interpolated from ASME STEAM TABLES

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Table 3. Superheated Steam – *Continued*

Abs Press. Lb/Sq In (Sat. Temp)		Sat. Water	Sat. Steam	Temperature – Degrees Fahrenheit 350	400	450	500	550	600	700	800	900	1000	1100	1200	1300	1400
80 (312.04)	Sh			37.95	87.95	137.95	187.95	237.95	287.95	387.95	487.95	587.95	687.95	787.95	887.95	987.95	1087.95
	v	0.01757	5.471	5.801	6.218	6.622	7.018	7.408	7.794	8.560	9.319	10.075	10.829	11.581	12.331	13.081	13.829
	h	282.15	1183.1	1204.0	1230.5	1256.1	1281.3	1306.2	1330.9	1380.5	1430.5	1481.1	1532.6	1584.9	1638.0	1692.0	1746.3
	s	0.4534	1.6208	1.6473	1.6790	1.7080	1.7349	1.7602	1.7842	1.8289	1.8702	1.9089	1.9454	1.9800	2.0131	2.0446	2.0750
85 (316.25)	Sh			33.74	83.74	133.74	183.74	233.74	283.74	383.74	483.74	583.74	683.74	783.74	883.74	983.74	1083.74
	v	0.01762	5.167	5.445	5.840	6.223	6.597	6.966	7.330	8.052	8.768	9.480	10.193	10.898	11.604	12.309	13.014
	h	286.52	1184.2	1203.0	1229.7	1255.5	1280.8	1305.8	1330.6	1380.2	1430.3	1481.0	1532.4	1584.7	1637.9	1691.9	1746.8
	s	0.4590	1.6159	1.6396	1.6715	1.7009	1.7279	1.7532	1.7772	1.8220	1.8634	1.9021	1.9386	1.9733	2.0063	2.0379	2.0682
90 (320.28)	Sh			29.72	79.72	129.72	179.72	229.72	279.72	379.72	479.72	579.72	679.72	779.72	879.72	979.72	1079.72
	v	0.01766	4.895	5.123	5.505	5.850	6.223	6.572	6.917	7.600	8.277	8.950	9.621	10.290	10.958	11.625	12.290
	h	290.69	1185.3	1202.0	1228.9	1254.9	1280.3	1305.4	1330.2	1380.0	1430.1	1480.8	1532.3	1584.6	1637.8	1691.8	1746.7
	s	0.4643	1.6113	1.6323	1.6646	1.6940	1.7212	1.7467	1.7707	1.8156	1.8570	1.8957	1.9323	1.9669	2.0000	2.0316	2.0619
95 (324.13)	Sh			25.87	75.87	125.87	175.87	225.87	275.87	375.87	475.87	575.87	675.87	775.87	875.87	975.87	1075.87
	v	0.01770	4.651	4.845	5.205	5.551	5.889	6.221	6.548	7.196	7.838	8.477	9.113	9.747	10.380	11.012	11.643
	h	294.70	1186.2	1200.9	1228.1	1254.3	1279.8	1305.0	1329.9	1379.7	1429.9	1480.6	1532.1	1584.5	1637.7	1691.7	1746.5
	s	0.4694	1.6069	1.6253	1.6580	1.6876	1.7149	1.7404	1.7645	1.8094	1.8509	1.8897	1.9262	1.9609	1.9940	2.0256	2.0559
100 (327.82)	Sh			22.18	72.18	122.18	172.18	222.18	272.18	372.18	472.18	572.18	672.18	772.18	872.18	972.18	1072.18
	v	0.01774	4.431	4.590	4.935	5.266	5.588	5.904	6.216	6.833	7.443	8.050	8.655	9.258	9.860	10.460	11.060
	h	298.54	1187.2	1199.9	1227.4	1253.7	1279.3	1304.6	1329.6	1379.5	1429.7	1480.4	1532.0	1584.4	1637.6	1691.6	1746.5
	s	0.4743	1.6027	1.6187	1.6516	1.6814	1.7088	1.7344	1.7586	1.8036	1.8451	1.8839	1.9205	1.9552	1.9883	2.0199	2.0502
105 (331.37)	Sh			18.63	68.63	118.63	168.63	218.63	268.63	368.63	468.63	568.63	668.63	768.63	868.63	968.63	1068.63
	v	0.01778	4.231	4.359	4.690	5.007	5.315	5.617	5.915	6.504	7.085	7.665	8.241	8.816	9.389	9.961	10.532
	h	302.24	1188.0	1198.8	1226.6	1253.1	1278.8	1304.2	1329.2	1379.2	1429.4	1480.3	1531.8	1584.2	1637.5	1691.5	1746.4
	s	0.4790	1.5988	1.6122	1.6455	1.6755	1.7031	1.7288	1.7530	1.7981	1.8396	1.8785	1.9151	1.9498	1.9828	2.0145	2.0448
110 (334.79)	Sh			15.21	65.21	115.21	165.21	215.21	265.21	365.21	465.21	565.21	665.21	765.21	865.21	965.21	1065.21
	v	0.01782	4.048	4.149	4.468	4.772	5.068	5.357	5.642	6.205	6.761	7.314	7.865	8.413	8.961	9.507	10.053
	h	305.80	1188.9	1197.7	1225.8	1252.5	1278.3	1303.8	1328.9	1379.0	1429.2	1480.1	1531.7	1584.1	1637.4	1691.4	1746.4
	s	0.4834	1.5950	1.6061	1.6396	1.6698	1.6975	1.7233	1.7476	1.7928	1.8344	1.8732	1.9099	1.9446	1.9777	2.0093	2.0397
115 (338.08)	Sh			11.92	61.92	111.92	161.92	211.92	261.92	361.92	461.92	561.92	661.92	761.92	861.92	961.92	1061.92
	v	0.01785	3.881	3.957	4.265	4.558	4.841	5.119	5.392	5.932	6.465	6.994	7.521	8.046	8.570	9.093	9.615
	h	309.25	1189.6	1196.7	1225.0	1251.8	1277.9	1303.3	1328.6	1378.7	1429.0	1479.9	1531.6	1584.0	1637.2	1691.4	1746.3
	s	0.4877	1.5913	1.6001	1.6340	1.6644	1.6922	1.7181	1.7425	1.7877	1.8294	1.8682	1.9049	1.9396	1.9727	2.0044	2.0347
120 (341.27)	Sh			8.73	58.73	108.73	158.73	208.73	258.73	358.73	458.73	558.73	658.73	758.73	858.73	958.73	1058.73
	v	0.01789	3.7275	3.7815	4.0786	4.3610	4.6341	4.9009	5.1637	5.6813	6.1928	6.7006	7.2060	7.7096	8.2119	8.7130	9.2134
	h	312.58	1190.4	1195.6	1224.1	1251.2	1277.4	1302.9	1328.2	1378.4	1428.8	1479.8	1531.4	1583.9	1637.1	1691.3	1746.2
	s	0.4919	1.5879	1.5943	1.6286	1.6592	1.6872	1.7132	1.7376	1.7829	1.8246	1.8635	1.9001	1.9349	1.9680	1.9996	2.0300
130 (347.33)	Sh			2.67	52.67	102.67	152.67	202.67	252.67	352.67	452.67	552.67	652.67	752.67	852.67	952.67	1052.67
	v	0.01796	3.4544	3.4699	3.7489	4.0129	4.2672	4.5151	4.7589	5.2384	5.7118	6.1814	6.6486	7.1140	7.5781	8.0411	8.5033
	h	318.95	1191.7	1193.4	1222.5	1249.9	1276.4	1302.1	1327.5	1377.9	1428.4	1479.5	1531.1	1583.6	1636.9	1691.1	1746.1
	s	0.4998	1.5813	1.5833	1.6182	1.6493	1.6775	1.7037	1.7283	1.7737	1.8155	1.8545	1.8911	1.9259	1.9591	1.9907	2.0211
140 (353.04)	Sh				46.96	96.96	146.96	196.96	246.96	346.96	446.96	546.96	646.96	746.96	846.96	946.96	1046.96
	v	0.01803	3.2190		3.4661	3.7143	3.9526	4.1844	4.4119	4.8559	5.2995	5.7364	6.1709	6.6036	7.0349	7.4652	7.8946
	h	324.82	1193.0		1220.8	1248.7	1275.3	1301.3	1326.8	1377.4	1428.0	1479.1	1530.8	1583.4	1636.7	1690.9	1745.9
	s	0.5071	1.5752		1.6085	1.6400	1.6686	1.6949	1.7196	1.7652	1.8071	1.8461	1.8828	1.9176	1.9508	1.9825	2.0129
150 (358.43)	Sh				41.57	91.57	141.57	191.57	241.57	341.57	441.57	541.57	641.57	741.57	841.57	941.57	1041.57
	v	0.01809	3.0139		3.2208	3.4555	3.6799	3.8978	4.1112	4.5298	4.9421	5.3507	5.7568	6.1612	6.5642	6.9661	7.3671
	h	330.65	1194.1		1219.1	1247.4	1274.3	1300.5	1326.1	1376.9	1427.6	1478.7	1530.5	1583.1	1636.5	1690.7	1745.7
	s	0.5141	1.5695		1.5993	1.6313	1.6602	1.6867	1.7115	1.7573	1.7992	1.8383	1.8751	1.9099	1.9431	1.9748	2.0052
160 (363.55)	Sh				36.45	86.45	136.45	186.45	236.45	336.45	436.45	536.45	636.45	736.45	836.45	936.45	1036.45
	v	0.01815	2.8336		3.0060	3.2288	3.4413	3.6469	3.8480	4.2420	4.6295	5.0132	5.3945	5.7741	6.1522	6.5293	6.9055
	h	336.07	1195.1		1217.4	1246.0	1273.3	1299.6	1325.4	1376.4	1427.2	1478.4	1530.3	1582.9	1636.3	1690.5	1745.6
	s	0.5206	1.5641		1.5906	1.6231	1.6522	1.6790	1.7039	1.7499	1.7919	1.8310	1.8678	1.9027	1.9359	1.9676	1.9980
170 (368.42)	Sh				31.58	81.58	131.58	181.58	231.58	331.58	431.58	531.58	631.58	731.58	831.58	931.58	1031.58
	v	0.01821	2.6738		2.8162	3.0288	3.2306	3.4255	3.6158	3.9879	4.3536	4.7155	5.0749	5.4325	5.7888	6.1440	6.4983
	h	341.24	1196.0		1215.6	1244.7	1272.2	1298.8	1324.7	1375.8	1426.8	1478.0	1530.0	1582.6	1636.1	1690.4	1745.4
	s	0.5269	1.5591		1.5823	1.6152	1.6447	1.6717	1.6968	1.7428	1.7850	1.8241	1.8610	1.8959	1.9291	1.9608	1.9913
180 (373.08)	Sh				26.92	76.92	126.92	176.92	226.92	326.92	426.92	526.92	626.92	726.92	826.92	926.92	1026.92
	v	0.01827	2.5312		2.6474	2.8508	3.0433	3.2286	3.4093	3.7621	4.1084	4.4508	4.7907	5.1289	5.4657	5.8014	6.1363
	h	346.19	1196.9		1213.8	1243.4	1271.2	1297.9	1324.0	1375.3	1426.3	1477.7	1529.7	1582.4	1635.9	1690.2	1745.3
	s	0.5328	1.5543		1.5743	1.6078	1.6376	1.6647	1.6900	1.7362	1.7784	1.8176	1.8545	1.8894	1.9227	1.9545	1.9849
190 (377.53)	Sh				22.47	72.47	122.47	172.47	222.47	322.47	422.47	522.47	622.47	722.47	822.47	922.47	1022.47
	v	0.01833	2.4030		2.4961	2.6915	2.8756	3.0525	3.2246	3.5601	3.8889	4.2140	4.5365	4.8572	5.1766	5.4949	5.8124
	h	350.94	1197.6		1212.0	1242.0	1270.1	1297.1	1323.3	1374.8	1425.9	1477.4	1529.4	1582.1	1635.7	1690.0	1745.1
	s	0.5384	1.5498		1.5667	1.6006	1.6307	1.6581	1.6835	1.7299	1.7722	1.8115	1.8484	1.8834	1.9166	1.9484	1.9789
200 (381.80)	Sh				18.20	68.20	118.20	168.20	218.20	318.20	418.20	518.20	618.20	718.20	818.20	918.20	1018.20
	v	0.01839	2.2873		2.3598	2.5480	2.7247	2.8939	3.0583	3.3783	3.6915	4.0008	4.3077	4.6128	4.9165	5.2191	5.5209
	h	355.51	1198.3		1210.1	1240.6	1269.0	1296.2	1322.6	1374.3	1425.5	1477.0	1529.1	1581.9	1635.4	1689.8	1745.0
	s	0.5438	1.5454		1.5593	1.5938	1.6242	1.6518	1.6773	1.7239	1.7663	1.8057	1.8426	1.8776	1.9109	1.9427	1.9732

Sh = superheat, F
v = specific volume, cu ft per lb
h = enthalpy, Btu per lb
s = entropy, Btu per F per lb

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Table 3. Superheated Steam – Continued

Abs Press. Lb Sq In (Sat Temp)		Sat Water	Sat Steam	Temperature – Degrees Fahrenheit 400	450	500	550	600	700	800	900	1000	1100	1200	1300	1400	1500
	Sh			14.09	64.09	114.09	164.09	214.09	314.09	414.09	514.09	614.09	714.09	814.09	914.09	1014.09	1114.09
210	v	0.01844	2.1722	2.2273	2.4101	2.5803	2.7504	2.9078	3.2137	3.5124	3.8080	4.1007	4.3915	4.6811	4.9695	5.2571	5.5440
(385.91)	h	359.91	1199.0	1204.3	1238.2	1268.0	1296.3	1323.9	1373.7	1425.1	1475.7	1525.8	1579.6	1635.2	1689.6	1744.8	1800.8
	s	0.5450	1.5443	1.5522	1.5877	1.6183	1.6455	1.6715	1.7192	1.7637	1.8001	1.8371	1.8721	1.9054	1.9372	1.9677	1.9970
	Sh			10.12	60.12	110.12	160.12	210.12	310.12	410.12	510.12	610.12	710.12	810.12	910.12	1010.12	1110.12
220	v	0.01850	2.0903	2.1240	2.3058	2.4638	2.6119	2.7710	3.0642	3.3504	3.6327	3.9125	4.1905	4.4671	4.7425	5.0173	5.2913
(389.88)	h	364.17	1199.6	1204.3	1237.8	1266.9	1295.3	1322.2	1373.2	1424.7	1475.3	1527.5	1581.4	1635.0	1689.4	1744.7	1800.6
	s	0.5540	1.5374	1.5453	1.5823	1.6120	1.6403	1.6655	1.7128	1.7553	1.7948	1.8319	1.8668	1.9002	1.9320	1.9625	1.9919
	Sh			6.30	56.30	106.30	156.30	206.30	306.30	406.30	506.30	606.30	706.30	806.30	906.30	1006.30	1106.30
230	v	0.01855	1.9945	2.0212	2.1913	2.3593	2.5081	2.6461	2.9276	3.2020	3.4726	3.7406	4.0058	4.2717	4.5355	4.7934	5.0606
(393.70)	h	368.28	1200.1	1204.4	1235.3	1265.7	1293.6	1320.4	1372.7	1424.7	1476.0	1528.2	1581.1	1634.8	1689.3	1744.5	1800.5
	s	0.5588	1.5335	1.5385	1.5747	1.6062	1.6344	1.6604	1.7075	1.7502	1.7897	1.8268	1.8618	1.8952	1.9270	1.9576	1.9869
	Sh			2.61	52.61	102.61	152.61	202.61	302.61	402.61	502.61	602.61	702.61	802.61	902.61	1002.61	1102.61
240	v	0.01860	1.9177	1.9268	2.1162	2.2682	2.4115	2.5316	2.8024	3.0661	3.3259	3.5831	3.8385	4.0926	4.3456	4.5977	4.8492
(397.39)	h	372.27	1200.6	1202.4	1234.9	1264.6	1292.7	1319.7	1372.1	1423.8	1475.6	1527.0	1580.9	1634.6	1689.1	1744.3	1800.4
	s	0.5631	1.5299	1.5320	1.5687	1.6006	1.6291	1.6552	1.7025	1.7452	1.7847	1.8219	1.8570	1.8904	1.9223	1.9528	1.9822
	Sh				49.03	99.03	149.03	199.03	299.03	399.03	499.03	599.03	699.03	799.03	899.03	999.03	1099.03
250	v	0.01865	1.8432		2.0016	2.1504	2.2909	2.4262	2.6872	2.9410	3.1903	3.4362	3.6837	3.9278	4.1700	4.4131	4.6546
(400.97)	h	376.14	1201.1		1233.4	1263.5	1291.9	1319.0	1371.6	1423.4	1475.3	1527.6	1580.6	1634.4	1688.9	1744.2	1800.2
	s	0.5679	1.5264		1.5625	1.5951	1.6239	1.6502	1.6976	1.7405	1.7801	1.8173	1.8524	1.8858	1.9177	1.9482	1.9776
	Sh				45.56	95.56	145.56	195.56	295.56	395.56	495.56	595.56	695.56	795.56	895.56	995.56	1095.56
260	v	0.01870	1.7742		1.9173	2.0619	2.1981	2.3289	2.5808	2.8256	3.0663	3.3044	3.5408	3.7758	4.0097	4.2427	4.4750
(404.44)	h	379.93	1201.5		1231.9	1262.4	1290.9	1318.2	1371.1	1423.0	1474.9	1527.3	1580.4	1634.2	1688.7	1744.0	1800.1
	s	0.5722	1.5230		1.5573	1.5903	1.6193	1.6453	1.6930	1.7359	1.7756	1.8128	1.8480	1.8814	1.9133	1.9439	1.9732
	Sh				42.20	92.20	142.20	192.20	292.20	392.20	492.20	592.20	692.20	792.20	892.20	992.20	1092.20
270	v	0.01875	1.7101		1.8391	1.9799	2.1121	2.2385	2.4824	2.7186	2.9509	3.1806	3.4084	3.6349	3.8603	4.0849	4.3087
(407.80)	h	383.56	1201.9		1230.4	1261.2	1290.0	1317.5	1370.5	1422.6	1474.6	1527.1	1580.1	1634.0	1688.5	1743.9	1800.0
	s	0.5764	1.5197		1.5518	1.5848	1.6143	1.6406	1.6885	1.7315	1.7713	1.8085	1.8437	1.8771	1.9090	1.9396	1.9690
	Sh				38.93	88.93	138.93	188.93	288.93	388.93	488.93	588.93	688.93	788.93	888.93	988.93	1088.93
280	v	0.01880	1.6505		1.7665	1.9037	2.0322	2.1551	2.3909	2.6194	2.8437	3.0655	3.2855	3.5042	3.7217	3.9384	4.1543
(411.07)	h	387.12	1202.3		1228.8	1260.0	1289.1	1316.8	1370.0	1422.1	1474.2	1526.8	1579.9	1633.8	1688.4	1743.7	1799.8
	s	0.5805	1.5166		1.5464	1.5798	1.6093	1.6361	1.6841	1.7273	1.7671	1.8043	1.8395	1.8730	1.9050	1.9356	1.9649
	Sh				35.75	85.75	135.75	185.75	285.75	385.75	485.75	585.75	685.75	785.75	885.75	985.75	1085.75
290	v	0.01885	1.5948		1.6965	1.8327	1.9578	2.0772	2.3058	2.5269	2.7440	2.9585	3.1711	3.3824	3.5926	3.8019	4.0106
(414.25)	h	390.60	1202.6		1227.3	1258.9	1288.1	1316.0	1369.5	1421.7	1473.9	1526.5	1579.6	1633.5	1688.2	1743.6	1799.7
	s	0.5844	1.5135		1.5412	1.5750	1.6048	1.6317	1.6799	1.7232	1.7630	1.8003	1.8356	1.8690	1.9010	1.9316	1.9610
	Sh				32.65	82.65	132.65	182.65	282.65	382.65	482.65	582.65	682.65	782.65	882.65	982.65	1082.65
300	v	0.01889	1.5427		1.6356	1.7665	1.8883	2.0044	2.2263	2.4407	2.6509	2.8585	3.0643	3.2688	3.4721	3.6746	3.8764
(417.35)	h	393.99	1202.9		1225.7	1257.7	1287.2	1315.2	1368.9	1421.3	1473.6	1526.2	1579.4	1633.3	1688.0	1743.4	1799.6
	s	0.5882	1.5105		1.5361	1.5703	1.6003	1.6274	1.6758	1.7192	1.7591	1.7964	1.8317	1.8652	1.8972	1.9278	1.9572
	Sh				29.64	79.64	129.64	179.64	279.64	379.64	479.64	579.64	679.64	779.64	879.64	979.64	1079.64
310	v	0.01894	1.4939		1.5763	1.7044	1.8233	1.9363	2.1520	2.3600	2.5638	2.7650	2.9644	3.1625	3.3594	3.5555	3.7509
(420.36)	h	397.30	1203.2		1224.1	1256.5	1286.3	1314.5	1368.4	1420.9	1473.2	1525.9	1579.2	1633.1	1687.8	1743.3	1799.4
	s	0.5920	1.5076		1.5311	1.5657	1.5960	1.6233	1.6719	1.7153	1.7553	1.7927	1.8280	1.8615	1.8935	1.9241	1.9536
	Sh				26.69	76.69	126.69	176.69	276.69	376.69	476.69	576.69	676.69	776.69	876.69	976.69	1076.69
320	v	0.01899	1.4480		1.5207	1.6462	1.7623	1.8725	2.0823	2.2843	2.4821	2.6774	2.8708	3.0628	3.2538	3.4438	3.6332
(423.31)	h	400.53	1203.4		1222.5	1255.2	1285.3	1313.7	1367.8	1420.5	1472.9	1525.6	1578.9	1632.9	1687.6	1743.1	1799.3
	s	0.5956	1.5048		1.5261	1.5612	1.5918	1.6192	1.6680	1.7116	1.7516	1.7890	1.8243	1.8579	1.8899	1.9205	1.9500
	Sh				23.82	73.82	123.82	173.82	273.82	373.82	473.82	573.82	673.82	773.82	873.82	973.82	1073.82
330	v	0.01903	1.4048		1.4684	1.5915	1.7050	1.8125	2.0168	2.2132	2.4054	2.5950	2.7828	2.9692	3.1545	3.3389	3.5227
(426.18)	h	403.70	1203.6		1220.9	1254.0	1284.4	1313.0	1367.3	1420.0	1472.5	1525.3	1578.7	1632.7	1687.5	1742.9	1799.2
	s	0.5991	1.5021		1.5213	1.5568	1.5876	1.6153	1.6643	1.7079	1.7480	1.7855	1.8208	1.8544	1.8864	1.9171	1.9466
	Sh				21.01	71.01	121.01	171.01	271.01	371.01	471.01	571.01	671.01	771.01	871.01	971.01	1071.01
340	v	0.01908	1.3640		1.4191	1.5399	1.6511	1.7561	1.9552	2.1463	2.3333	2.5175	2.7000	2.8811	3.0611	3.2402	3.4186
(428.99)	h	406.80	1203.8		1219.2	1252.8	1283.4	1312.2	1366.7	1419.6	1472.2	1525.0	1578.4	1632.5	1687.3	1742.8	1799.0
	s	0.6026	1.4994		1.5165	1.5525	1.5836	1.6114	1.6606	1.7044	1.7445	1.7820	1.8174	1.8510	1.8831	1.9138	1.9432
	Sh				18.27	68.27	118.27	168.27	268.27	368.27	468.27	568.27	668.27	768.27	868.27	968.27	1068.27
350	v	0.01912	1.3255		1.3725	1.4913	1.6002	1.7028	1.8970	2.0832	2.2652	2.4445	2.6219	2.7980	2.9730	3.1471	3.3205
(431.73)	h	409.83	1204.0		1217.5	1251.5	1282.4	1311.4	1366.2	1419.2	1471.8	1524.7	1578.2	1632.3	1687.1	1742.6	1798.9
	s	0.6059	1.4968		1.5119	1.5483	1.5797	1.6077	1.6571	1.7009	1.7411	1.7787	1.8141	1.8477	1.8798	1.9105	1.9400
	Sh				15.59	65.59	115.59	165.59	265.59	365.59	465.59	565.59	665.59	765.59	865.59	965.59	1065.59
360	v	0.01917	1.2891		1.3285	1.4454	1.5521	1.6525	1.8421	2.0237	2.2009	2.3755	2.5482	2.7196	2.8898	3.0592	3.2279
(434.41)	h	412.81	1204.1		1215.8	1250.3	1281.5	1310.6	1365.6	1418.7	1471.5	1524.4	1577.9	1632.1	1686.9	1742.5	1798.8
	s	0.6092	1.4943		1.5073	1.5441	1.5758	1.6040	1.6536	1.6976	1.7379	1.7754	1.8109	1.8445	1.8766	1.9073	1.9368
	Sh				10.39	60.39	110.39	160.39	260.39	360.39	460.39	560.39	660.39	760.39	860.39	960.39	1060.39
380	v	0.01925	1.2218		1.2472	1.3606	1.4635	1.5598	1.7410	1.9139	2.0825	2.2484	2.4124	2.5750	2.7366	2.8973	3.0572
(439.61)	h	418.59	1204.4		1212.4	1247.7	1279.5	1309.0	1364.5	1417.8	1470.8	1523.8	1577.4	1631.6	1686.5	1742.2	1798.5
	s	0.6156	1.4894		1.4982	1.5360	1.5683	1.5969	1.6470	1.6911	1.7315	1.7692	1.8047	1.8384	1.8705	1.9012	1.9307

Sh = superheat, F
v = specific volume, cu ft per lb
h = enthalpy, Btu per lb
s = entropy, Btu per F per lb

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Table 3.   Superheated Steam—Continued

Abs Press Lb Sq In (Sat. Temp)		Sat Water	Sat Steam	450	500	550	600	650	700	800	900	1000	1100	1200	1300	1400	1500
400 (444.60)	Sh			5.40	55.40	105.40	155.40	205.40	255.40	355.40	455.40	555.40	655.40	755.40	855.40	955.40	1055.40
	v	0.01934	[illegible]	[illegible]	1.2841	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	2.1333	2.2901	[illegible]	2.5327	2.7515	2.4037
	h	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1633.1	[illegible]	1631.2	1686.2	1741.9	1798.2
	s	0.6217	1.4847	1.4534	1.5282	1.5611	[illegible]	1.6283	1.6425	[illegible]	1.7255	1.7632	1.7336	1.8325	1.8647	1.8955	1.9250
420 (448.40)	Sh			0.60	50.60	100.60	150.60	200.60	250.60	350.60	450.60	550.60	650.60	750.60	850.60	950.60	1050.60
	v	0.01942	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	2.0024	2.1735	2.3273	2.4739	2.6195	2.7647
	h	423.56	1204.7	[illegible]	1242.4	[illegible]	[illegible]	[illegible]	1387.3	[illegible]	[illegible]	1632.7	[illegible]	[illegible]	1685.8	1741.6	1799.0
	s	0.6276	1.4802	1.4504	1.5206	1.5543	1.5825	1.6103	1.6345	[illegible]	1.7197	1.7575	1.7932	[illegible]	1.8581	1.8889	1.9095
440 (454.03)	Sh				45.97	95.97	145.97	195.97	245.97	345.97	445.97	545.97	645.97	745.97	845.97	945.97	1045.97
	v	0.01960	1.0554		1.1817	1.7434	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.8462	2.0790	[illegible]	2.3875	2.4898	2.6384
	h	[illegible]	[illegible]		1225.7	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1522.1	1570.9	[illegible]	1655.5	1741.2	1797.7
	s	0.6332	1.4759		1.5132	1.4876	[illegible]	1.6040	1.6286	1.6724	1.7142	1.7921	1.7878	[illegible]	1.8539	1.8847	1.9143
460 (458.50)	Sh				41.50	91.50	[illegible]	[illegible]	241.50	341.50	441.50	541.50	641.50	741.50	841.50	941.50	1041.50
	v	0.01959	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.8514	[illegible]	[illegible]	2.2593	[illegible]	2.5236
	h	433.83	[illegible]		1224.9	[illegible]	[illegible]	[illegible]	[illegible]	1414.8	1468.0	[illegible]	1572.4	[illegible]	1685.1	1742.9	1797.4
	s	0.6387	1.4718		1.5043	1.5409	[illegible]	1.5999	1.6220	1.6680	1.7029	1.7459	1.7826	[illegible]	1.8438	1.8797	1.9093
480 (462.82)	Sh				37.18	87.18	137.18	187.18	237.18	337.18	437.18	537.18	637.18	737.18	837.18	937.18	1037.18
	v	0.01967	0.9563		1.0473	1.1393	[illegible]	[illegible]	1.3615	[illegible]	1.6784	1.7716	1.5230	2.0322	2.1619	2.2920	2.4173
	h	441.75	[illegible]		1226.1	[illegible]	[illegible]	[illegible]	[illegible]	1416.5	1467.3	1520.9	1574.9	1623.5	1684.7	1740.4	1797.2
	s	0.6443	1.4677		1.4990	1.5346	1.5662	1.5925	1.6176	1.6626	1.7026	1.7415	1.7777	1.8116	1.8439	1.8749	1.9045
500 (467.01)	Sh				32.99	82.99	132.99	182.99	232.99	332.99	432.99	532.99	632.99	732.99	832.99	932.99	1032.99
	v	0.01975	0.9276		0.9919	1.0791	1.1584	1.2327	1.3037	1.4397	1.5715	1.8992	1.8736	1.9367	2.0746	2.1977	2.3203
	h	449.52	1204.7		1231.2	1267.0	[illegible]	[illegible]	1357.7	1412.7	1475.8	1520.3	1574.2	[illegible]	1684.4	1740.3	1796.9
	s	0.6490	1.4639		1.4921	1.5284	1.5588	1.5871	1.6123	1.6578	1.6990	1.7371	1.7730	1.8059	1.8393	1.8702	1.8998
520 (471.07)	Sh				28.93	78.93	128.93	178.93	228.93	328.93	428.93	528.93	628.93	728.93	828.93	928.93	1028.93
	v	0.01982	0.8914		0.9466	1.0321	1.1094	1.1816	1.2504	1.3819	1.5096	1.6333	1.7542	1.8746	1.9940	2.1125	2.2302
	h	456.28	1204.5		1228.3	1264.3	[illegible]	[illegible]	1356.5	1411.8	1465.9	1519.7	1573.9	1628.7	1684.0	1740.0	1796.7
	s	0.6545	1.4601		1.4853	1.5223	1.5539	1.5828	1.6072	1.6530	1.6943	1.7325	1.7684	1.8024	1.8348	1.8657	1.8954
540 (475.01)	Sh				24.99	74.99	124.99	174.99	224.99	324.99	424.99	524.99	624.99	724.99	824.99	924.99	1024.99
	v	0.01990	0.8577		0.9049	0.9894	[illegible]	1.1382	1.2010	1.3274	1.4508	1.5704	1.6877	1.8043	1.9193	2.0336	2.1471
	h	[illegible]	1204.4		1225.3	1261.5	[illegible]	1324.3	1355.3	[illegible]	1465.1	1519.1	1573.4	1627.2	1683.6	1739.7	1796.4
	s	0.6587	1.4565		1.4786	1.5164	1.5485	1.5767	1.6023	1.6483	1.6897	1.7280	1.7640	1.7981	1.8305	1.8615	1.8911
560 (478.84)	Sh				21.16	71.16	121.16	171.16	221.16	321.16	421.16	521.16	621.16	721.16	821.16	921.16	1021.16
	v	0.01998	0.8264		0.8653	0.9479	[illegible]	[illegible]	1.1552	1.2787	1.3972	1.5129	1.6265	1.7399	1.8500	1.9603	2.0699
	h	463.14	1204.2		1222.2	[illegible]	[illegible]	[illegible]	1354.2	1410.0	[illegible]	[illegible]	1572.3	[illegible]	1683.3	1739.4	1796.1
	s	0.6634	1.4529		1.4722	[illegible]	[illegible]	1.5717	1.5975	1.6435	1.6855	1.7237	1.7598	1.7939	1.8263	1.8573	1.8870
580 (482.57)	Sh				17.43	67.43	117.43	167.43	217.43	317.43	417.43	517.43	617.43	717.43	817.43	917.43	1017.43
	v	0.02006	0.7971		0.8287	[illegible]	[illegible]	[illegible]	1.1136	1.2324	1.3473	1.4593	1.5693	1.6780	1.7855	1.8921	1.9980
	h	497.47	1203.9		1213.1	[illegible]	[illegible]	[illegible]	[illegible]	1409.2	1463.7	1518.6	1572.1	1627.2	1682.9	1739.1	1795.9
	s	0.6679	1.4495		1.4654	1.5049	1.5383	1.5668	1.5929	1.6394	1.6811	1.7196	1.7556	1.7898	1.8223	1.8532	1.8831
600 (486.20)	Sh				13.80	63.80	113.80	163.80	213.80	313.80	413.80	513.80	613.80	713.80	813.80	913.80	1013.80
	v	0.02013	0.7697		0.7944	[illegible]	[illegible]	[illegible]	1.0775	[illegible]	[illegible]	1.4093	1.5160	[illegible]	1.7252	1.8283	1.9309
	h	471.59	1203.7		1215.9	1255.6	[illegible]	[illegible]	1351.8	1408.3	[illegible]	1517.4	1571.9	[illegible]	1682.5	1738.8	1795.6
	s	0.6723	1.4461		1.4590	[illegible]	1.5329	1.5621	1.5884	1.6351	1.6769	1.7155	1.7517	1.7859	1.8184	1.8494	1.8792
650 (494.89)	Sh				5.11	55.11	105.11	155.11	205.11	305.11	405.11	505.11	605.11	705.11	805.11	905.11	1005.11
	v	0.02032	0.7084		0.7173	0.7954	[illegible]	[illegible]	0.9835	1.0929	1.1969	1.2979	1.3969	1.4944	1.5909	1.6864	1.7813
	h	481.37	1203.8		1207.6	1249.6	[illegible]	[illegible]	1348.7	1406.0	1461.2	1515.9	1570.7	1625.9	1681.6	1738.0	1794.9
	s	0.6828	1.4381		1.4430	1.4858	1.5207	1.5507	1.5775	1.6249	1.6671	1.7059	1.7422	1.7765	1.8092	1.8403	1.8701
700 (503.08)	Sh					46.92	96.92	146.92	196.92	296.92	396.92	496.92	596.92	696.92	796.92	896.92	996.92
	v	0.02050	0.6556			0.7271	[illegible]	[illegible]	[illegible]	1.0122	1.1078	1.2023	1.2946	[illegible]	1.4757	[illegible]	1.6530
	h	491.50	1201.8			1243.4	[illegible]	[illegible]	1345.6	1402.7	1459.4	1514.4	1569.4	1624.4	1680.7	1737.2	1794.3
	s	0.6928	1.4304			1.4675	1.5080	[illegible]	1.5673	1.6154	1.6580	1.6970	1.7335	1.7679	1.8006	1.8318	1.8617
750 (510.84)	Sh					39.16	89.16	139.16	189.16	289.16	389.16	489.16	589.16	689.16	789.16	889.16	989.16
	v	0.02069	0.6095			0.6574	[illegible]	0.7882	0.8479	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.3759	[illegible]	1.5419
	h	500.89	[illegible]			1233.9	[illegible]	1311.7	1342.5	1401.5	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1793.6
	s	0.7022	1.4232			1.4558	1.4977	1.5336	1.5577	1.6065	1.6494	1.6886	1.7252	1.7598	1.7926	1.8239	1.8538
800 (518.21)	Sh					31.79	81.79	131.79	181.79	281.79	381.79	481.79	581.79	681.79	781.79	881.79	981.79
	v	0.02087	0.5690			[illegible]	[illegible]	[illegible]	0.7828	0.8759	0.9631	1.0470	1.1289	1.2093	1.2885	1.3669	1.4446
	h	[illegible]	1199.4			[illegible]	[illegible]	[illegible]	1339.3	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1673.9	1735.7	1792.9
	s	0.7111	1.4163			1.4442	1.4869	1.5195	1.5484	1.5980	1.6413	1.6807	1.7175	1.7522	1.7851	1.8164	1.8464
850 (525.24)	Sh					24.76	74.76	124.76	174.76	274.76	374.76	474.76	574.76	674.76	774.76	874.76	974.76
	v	0.02105	0.5330			0.5021	0.6036	[illegible]	0.7315	[illegible]	[illegible]	0.9893	1.0665	1.1366	1.2115	1.2855	1.3588
	h	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]	1336.0	[illegible]	[illegible]	1510.0	[illegible]	[illegible]	1676.0	[illegible]	[illegible]
	s	0.7197	1.4096			1.4327	1.4763	1.5102	1.5396	1.5899	1.6336	1.6733	1.7102	1.7450	1.7780	1.8094	1.8395
900 (531.95)	Sh					18.05	68.05	118.05	168.05	268.05	368.05	468.05	568.05	668.05	768.05	868.05	968.05
	v	0.02123	0.5009			[illegible]	[illegible]	[illegible]	0.6903	[illegible]	[illegible]	[illegible]	0.9974	1.0712	1.1427	[illegible]	1.2825
	h	[illegible]	1198.4			1202.5	[illegible]	[illegible]	1332.7	1394.4	[illegible]	1505.5	1564.4	[illegible]	1677.1	[illegible]	1791.6
	s	0.7279	1.4032			1.4723	1.4659	[illegible]	1.5311	1.5822	1.6263	1.6662	1.7033	1.7382	1.7713	1.8028	1.8329

Sh = superheat, F
v = specific volume, cu ft per lb
h = enthalpy, Btu per lb
s = entropy, Btu per F per lb

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Table 3. Superheated Steam – Continued

Abs Press. Lb Sq In (Sat. Temp)		Sat Water	Sat Steam	550	600	650	700	750	800	850	900	1000	1100	1200	1300	1400	1500
				Temperature – Degrees Fahrenheit													
950 (538.39)	Sh			11.61	61.61	111.61	161.61	211.61	261.61	311.61	361.61	461.61	561.61	661.61	761.61	861.61	961.61
	v	0.02141	0.4721	0.4883	0.5485	0.5993	0.6449	0.6871	0.7272	0.7656	0.8030	0.8753	0.9455	1.0142	1.0817	1.1483	1.2143
	h	534.74	1194.7	1207.6	1255.1	1294.4	1329.3	1361.5	1392.0	1421.5	1450.3	1507.0	1563.2	1619.5	1676.2	1733.3	1791.0
	s	0.7358	1.3970	1.4098	1.4557	1.4921	1.5228	1.5500	1.5748	1.5977	1.6193	1.6595	1.6967	1.7317	1.7649	1.7965	1.8267
1000 (544.58)	Sh			5.42	55.42	105.42	155.42	205.42	255.42	305.42	355.42	455.42	555.42	655.42	755.42	855.42	955.42
	v	0.02159	0.4460	0.4535	0.5137	0.5636	0.6080	0.6489	0.6875	0.7245	0.7603	0.8295	0.8966	0.9622	1.0266	1.0901	1.1529
	h	542.55	1192.9	1199.3	1249.3	1290.1	1325.9	1358.7	1389.6	1419.4	1448.5	1505.4	1561.9	1618.4	1675.3	1732.5	1790.3
	s	0.7434	1.3910	1.3973	1.4457	1.4833	1.5149	1.5426	1.5677	1.5908	1.6126	1.6530	1.6905	1.7255	1.7589	1.7905	1.8207
1050 (550.53)	Sh				49.47	99.47	149.47	199.47	249.47	299.47	349.47	449.47	549.47	649.47	749.47	849.47	949.47
	v	0.02177	0.4222		0.4821	0.5312	0.5745	0.6142	0.6515	0.6872	0.7216	0.7881	0.8524	0.9151	0.9767	1.0373	1.0973
	h	550.15	1191.0		1243.4	1285.7	1322.4	1355.8	1387.2	1417.3	1446.6	1503.9	1560.7	1617.4	1674.4	1731.8	1789.6
	s	0.7507	1.3851		1.4358	1.4748	1.5072	1.5354	1.5608	1.5842	1.6062	1.6469	1.6845	1.7197	1.7531	1.7848	1.8151
1100 (556.28)	Sh				43.72	93.72	143.72	193.72	243.72	293.72	343.72	443.72	543.72	643.72	743.72	843.72	943.72
	v	0.02195	0.4006		0.4531	0.5017	0.5440	0.5826	0.6188	0.6533	0.6865	0.7505	0.8121	0.8723	0.9313	0.9894	1.0468
	h	557.55	1189.1		1237.3	1281.2	1318.8	1352.9	1384.7	1415.2	1444.7	1502.4	1559.4	1616.3	1673.5	1731.0	1789.0
	s	0.7578	1.3794		1.4259	1.4664	1.4996	1.5284	1.5542	1.5779	1.6000	1.6410	1.6787	1.7141	1.7475	1.7793	1.8097
1150 (561.82)	Sh				38.18	88.18	138.18	188.18	238.18	288.18	338.18	438.18	538.18	638.18	738.18	838.18	938.18
	v	0.02214	0.3807		0.4263	0.4746	0.5162	0.5538	0.5889	0.6223	0.6544	0.7161	0.7754	0.8332	0.8899	0.9456	1.0007
	h	564.78	1187.0		1230.9	1276.6	1315.2	1349.9	1382.2	1413.0	1442.8	1500.9	1558.1	1615.2	1672.6	1730.2	1788.3
	s	0.7647	1.3738		1.4160	1.4582	1.4923	1.5216	1.5478	1.5717	1.5941	1.6353	1.6732	1.7087	1.7422	1.7741	1.8045
1200 (567.19)	Sh				32.81	82.81	132.81	182.81	232.81	282.81	332.81	432.81	532.81	632.81	732.81	832.81	932.81
	v	0.02232	0.3624		0.4016	0.4497	0.4905	0.5273	0.5615	0.5939	0.6250	0.6845	0.7418	0.7974	0.8519	0.9055	0.9584
	h	571.85	1184.8		1224.2	1271.8	1311.5	1346.9	1379.7	1410.8	1440.9	1499.4	1556.9	1614.2	1671.6	1729.4	1787.6
	s	0.7714	1.3683		1.4061	1.4501	1.4851	1.5150	1.5415	1.5658	1.5883	1.6298	1.6679	1.7035	1.7371	1.7691	1.7996
1300 (577.42)	Sh				22.58	72.58	122.58	172.58	222.58	272.58	322.58	422.58	522.58	622.58	722.58	822.58	922.58
	v	0.02269	0.3299		0.3570	0.4052	0.4451	0.4804	0.5129	0.5436	0.5729	0.6287	0.6822	0.7341	0.7847	0.8345	0.8836
	h	585.58	1180.2		1209.9	1261.9	1303.9	1340.8	1374.6	1406.4	1437.1	1496.3	1554.3	1612.0	1669.8	1727.9	1786.3
	s	0.7843	1.3577		1.3860	1.4340	1.4711	1.5022	1.5296	1.5544	1.5773	1.6194	1.6578	1.6937	1.7275	1.7596	1.7902
1400 (587.07)	Sh				12.93	62.93	112.93	162.93	212.93	262.93	312.93	412.93	512.93	612.93	712.93	812.93	912.93
	v	0.02307	0.3018		0.3176	0.3667	0.4059	0.4400	0.4712	0.5004	0.5282	0.5809	0.6311	0.6798	0.7272	0.7737	0.8195
	h	598.83	1175.3		1194.1	1251.4	1296.1	1334.5	1369.3	1402.0	1433.2	1493.2	1551.8	1609.9	1668.0	1726.3	1785.0
	s	0.7966	1.3474		1.3652	1.4181	1.4575	1.4900	1.5182	1.5436	1.5670	1.6096	1.6484	1.6845	1.7185	1.7508	1.7815
1500 (596.20)	Sh				3.80	53.80	103.80	153.80	203.80	253.80	303.80	403.80	503.80	603.80	703.80	803.80	903.80
	v	0.02346	0.2772		0.2820	0.3328	0.3717	0.4049	0.4350	0.4629	0.4894	0.5394	0.5869	0.6327	0.6773	0.7210	0.7639
	h	611.68	1170.1		1176.3	1240.2	1287.9	1328.0	1364.0	1397.4	1429.2	1490.1	1549.2	1607.7	1666.2	1724.8	1783.7
	s	0.8085	1.3373		1.3431	1.4022	1.4443	1.4782	1.5073	1.5333	1.5572	1.6004	1.6395	1.6759	1.7101	1.7425	1.7734
1600 (604.87)	Sh					45.13	95.13	145.13	195.13	245.13	295.13	395.13	495.13	595.13	695.13	795.13	895.13
	v	0.02387	0.2555			0.3026	0.3415	0.3741	0.4032	0.4301	0.4555	0.5031	0.5482	0.5915	0.6336	0.6748	0.7153
	h	624.20	1164.5			1228.3	1279.4	1321.4	1358.5	1392.8	1425.2	1486.9	1546.6	1605.6	1664.3	1723.2	1782.3
	s	0.8199	1.3274			1.3861	1.4312	1.4667	1.4968	1.5235	1.5478	1.5916	1.6312	1.6678	1.7022	1.7347	1.7657
1700 (613.13)	Sh					36.87	86.87	136.87	186.87	236.87	286.87	386.87	486.87	586.87	686.87	786.87	886.87
	v	0.02428	0.2361			0.2754	0.3147	0.3468	0.3751	0.4011	0.4255	0.4711	0.5140	0.5552	0.5951	0.6341	0.6724
	h	636.45	1158.6			1215.3	1270.5	1314.5	1352.9	1388.1	1421.2	1483.8	1544.0	1603.4	1662.5	1721.7	1781.0
	s	0.8309	1.3176			1.3697	1.4183	1.4555	1.4867	1.5140	1.5388	1.5833	1.6232	1.6601	1.6947	1.7274	1.7585
1800 (621.02)	Sh					28.98	78.98	128.98	178.98	228.98	278.98	378.98	478.98	578.98	678.98	778.98	878.98
	v	0.02472	0.2186			0.2505	0.2906	0.3223	0.3500	0.3752	0.3988	0.4426	0.4836	0.5229	0.5609	0.5980	0.6343
	h	648.49	1152.3			1201.2	1261.1	1307.4	1347.2	1383.3	1417.1	1480.6	1541.4	1601.2	1660.7	1720.1	1779.7
	s	0.8417	1.3079			1.3526	1.4054	1.4446	1.4768	1.5049	1.5302	1.5753	1.6156	1.6528	1.6876	1.7204	1.7516
1900 (628.56)	Sh					21.44	71.44	121.44	171.44	221.44	271.44	371.44	471.44	571.44	671.44	771.44	871.44
	v	0.02517	0.2028			0.2274	0.2687	0.3004	0.3275	0.3521	0.3749	0.4171	0.4565	0.4940	0.5303	0.5656	0.6002
	h	660.36	1145.6			1185.7	1251.3	1300.2	1341.4	1378.4	1412.9	1477.4	1538.8	1599.1	1658.8	1718.6	1778.4
	s	0.8522	1.2981			1.3346	1.3925	1.4338	1.4672	1.4960	1.5219	1.5677	1.6084	1.6458	1.6808	1.7138	1.7451
2000 (635.80)	Sh					14.20	64.20	114.20	164.20	214.20	264.20	364.20	464.20	564.20	664.20	764.20	864.20
	v	0.02565	0.1883			0.2056	0.2488	0.2805	0.3072	0.3312	0.3534	0.3942	0.4320	0.4680	0.5027	0.5365	0.5695
	h	672.11	1138.3			1168.3	1240.9	1292.6	1335.4	1373.5	1408.7	1474.1	1536.2	1596.9	1657.0	1717.0	1777.1
	s	0.8625	1.2881			1.3154	1.3794	1.4231	1.4578	1.4874	1.5138	1.5603	1.6014	1.6391	1.6743	1.7075	1.7389
2100 (642.76)	Sh					7.24	57.24	107.24	157.24	207.24	257.24	357.24	457.24	557.24	657.24	757.24	857.24
	v	0.02615	0.1750			0.1847	0.2304	0.2624	0.2888	0.3123	0.3339	0.3733	0.4099	0.4445	0.4778	0.5101	0.5418
	h	683.79	1130.5			1148.5	1229.8	1284.9	1329.3	1368.4	1404.4	1470.9	1533.6	1594.7	1655.2	1715.4	1775.7
	s	0.8727	1.2780			1.2942	1.3661	1.4125	1.4486	1.4790	1.5060	1.5532	1.5948	1.6327	1.6681	1.7014	1.7330
2200 (649.45)	Sh					0.55	50.55	100.55	150.55	200.55	250.55	350.55	450.55	550.55	650.55	750.55	850.55
	v	0.02669	0.1627			0.1636	0.2134	0.2458	0.2720	0.2950	0.3161	0.3545	0.3897	0.4231	0.4551	0.4862	0.5165
	h	695.46	1122.2			1123.9	1218.0	1276.8	1323.1	1363.3	1400.0	1467.6	1530.9	1592.5	1653.3	1713.9	1774.4
	s	0.8828	1.2676			1.2691	1.3523	1.4020	1.4395	1.4708	1.4984	1.5463	1.5883	1.6266	1.6622	1.6956	1.7273
2300 (655.89)	Sh						44.11	94.11	144.11	194.11	244.11	344.11	444.11	544.11	644.11	744.11	844.11
	v	0.02727	0.1513				0.1975	0.2305	0.2566	0.2793	0.2999	0.3372	0.3714	0.4035	0.4344	0.4643	0.4935
	h	707.18	1113.2				1205.3	1268.4	1316.7	1358.1	1395.7	1464.2	1528.3	1590.3	1651.5	1712.3	1773.1
	s	0.8929	1.2569				1.3381	1.3914	1.4305	1.4628	1.4910	1.5397	1.5821	1.6207	1.6565	1.6901	1.7219

Sh = superheat, F
v = specific volume, cu ft per lb
h = enthalpy, Btu per lb
s = entropy, Btu per F per lb

1103 1519

Table 3. Superheated Steam – Continued

Abs Press. Lb/Sq In. (Sat. Temp)		Sat Water	Sat Steam	Temperature – Degrees Fahrenheit 700	750	800	850	900	950	1000	1050	1100	1150	1200	1300	1400	1500
	Sh			37.89	87.89	137.89	187.89	237.89	287.89	337.89	387.89	437.89	487.89	537.89	637.89	737.89	837.89
2400 (662.11)	v	0.02790	0.1402	0.1824	0.2164	0.2424	0.2646	0.2850	0.3037	0.3214	0.3382	0.3545	0.3703	0.3856	0.4155	0.4443	0.4724
	h	715.95	1101.7	1191.6	1253.7	[illegible]	1342.3	1391.2	1427.5	1460.9	1492.7	1525.6	1557.0	1588.1	1649.6	1710.8	1771.8
	s	0.9031	1.2460	1.3232	1.3808	1.4217	1.4549	1.4837	1.5095	1.5332	1.5553	1.5761	1.5959	1.6149	1.6509	1.6847	1.7167
	Sh			31.89	81.89	131.89	181.89	231.89	281.89	331.89	381.89	431.89	481.89	531.89	631.89	731.89	831.89
2500 (668.11)	v	0.02859	0.1307	0.1681	0.2032	[illegible]	0.2514	0.2712	0.2896	0.3069	0.3232	0.3390	0.3543	0.3692	0.3980	0.4259	0.4529
	h	731.71	1093.3	1176.7	1240.6	1293.4	1337.4	1384.7	1423.1	1457.5	1490.7	1522.5	1554.5	1585.9	1647.8	1709.2	1770.4
	s	0.9123	1.2345	1.3076	1.3701	1.4129	1.4472	1.4785	1.5009	1.5269	1.5492	1.5703	1.5903	1.6094	1.6456	1.6796	1.7116
	Sh			26.09	76.09	126.09	176.09	226.09	276.09	326.09	376.09	426.09	476.09	526.09	626.09	726.09	826.09
2600 (673.91)	v	0.02938	0.1211	0.1544	0.1909	0.2171	0.2390	0.2585	0.2766	0.2933	0.3093	0.3247	0.3395	0.3540	0.3819	0.4086	0.4350
	h	744.47	1082.0	1160.2	1224.1	[illegible]	[illegible]	1380.1	[illegible]	1454.1	1487.7	1520.2	[illegible]	1583.7	1645.9	1707.7	1769.1
	s	0.9247	1.2225	1.2908	1.3592	1.4042	1.4395	1.4696	1.4964	1.5203	1.5434	1.5646	1.5848	1.6040	1.6405	1.6746	1.7068
	Sh			20.47	70.47	120.47	170.47	220.47	270.47	320.47	370.47	420.47	470.47	520.47	620.47	720.47	820.47
2700 (679.53)	v	0.03073	0.1119	0.1411	0.1794	[illegible]	0.2275	0.2465	0.2644	0.2809	0.2965	0.3114	0.3259	0.3399	0.3670	0.3931	0.4184
	h	757.34	1067.7	1142.0	1221.1	1279.5	1326.3	1377.5	[illegible]	1450.7	1484.6	1517.5	1549.8	1581.5	1644.1	1706.1	1767.8
	s	0.9353	1.2097	1.2727	1.3481	1.3964	1.4319	1.4628	1.4900	1.5149	1.5376	1.5591	1.5794	1.5988	1.6355	1.6697	1.7021
	Sh			15.04	65.04	115.04	165.04	215.04	265.04	315.04	365.04	415.04	465.04	515.04	615.04	715.04	815.04
2800 (684.96)	v	0.03124	0.1010	0.1278	0.1655	0.1957	0.2188	0.2368	0.2531	0.2693	0.2845	0.2991	0.3132	0.3268	0.3532	0.3755	0.4030
	h	770.69	1055.8	1121.2	[illegible]	1272.2	1320.7	1372.8	1411.2	1447.2	1481.6	1514.6	1547.3	1579.3	1642.2	1704.5	1766.5
	s	0.9486	1.1958	1.2537	1.3263	1.3867	1.4245	1.4581	1.4838	1.5039	1.5321	1.5537	1.5742	1.5938	1.6306	1.6651	1.6975
	Sh			9.78	59.78	109.78	159.78	209.78	259.78	309.78	359.78	409.78	459.78	509.78	609.78	709.78	809.78
2900 (690.22)	v	0.03262	0.0942	0.1135	0.1591	0.1853	0.2048	0.2236	0.2427	0.2585	0.2734	0.2877	0.3014	0.3147	0.3403	0.3649	0.3887
	h	785.13	1039.8	1095.1	1183.6	1264.7	1324.9	1368.0	1407.2	1443.7	1478.5	1512.1	1544.9	1577.0	1640.4	1703.0	1765.2
	s	0.9566	1.1803	1.2263	1.3251	1.3782	1.4171	1.4494	1.4777	1.5032	1.5268	1.5485	1.5692	1.5889	1.6259	1.6605	1.6931
	Sh			4.67	54.67	104.67	154.67	204.67	254.67	304.67	354.67	404.67	454.67	504.67	604.67	704.67	804.67
3000 (695.33)	v	0.03428	0.0850	0.0982	0.1483	0.1759	0.1975	0.2161	0.2329	0.2484	0.2632	0.2773	0.2904	0.3033	0.3282	0.3522	0.3753
	h	801.33	1020.3	1060.5	1197.9	1267.0	1318.0	1363.2	1403.1	1440.2	1475.4	1509.4	1542.4	1574.8	1638.5	1701.4	1763.8
	s	0.9728	1.1619	1.1966	1.3131	1.3692	1.4097	1.4429	1.4717	1.4976	1.5213	1.5434	1.5642	1.5841	1.6214	1.6561	1.6888
	Sh				49.72	99.72	149.72	199.72	249.72	299.72	349.72	399.72	449.72	499.72	599.72	699.72	799.72
3100 (700.28)	v	0.03681	0.0745		0.1389	0.1671	0.1887	0.2071	0.2237	0.2390	0.2533	0.2670	0.2800	0.2927	0.3170	0.3403	0.3628
	h	823.57	993.3		1185.4	1259.1	1313.0	1355.4	1399.0	1436.7	1472.3	1506.6	1539.9	1572.6	1636.7	1699.8	1762.5
	s	0.9914	1.1373		1.3007	1.3604	1.4028	1.4366	1.4658	1.4920	1.5161	1.5384	1.5594	1.5794	1.6169	1.6518	1.6847
	Sh				44.92	94.92	144.92	194.92	244.92	294.92	344.92	394.92	444.92	494.92	594.92	694.92	794.92
3200 (705.08)	v	0.04472	0.0566		0.1300	0.1558	0.1804	0.1967	0.2151	0.2301	0.2442	0.2576	0.2704	0.2827	0.3065	0.3291	0.3510
	h	875.54	931.6		1172.3	1252.9	1306.9	1353.4	1394.9	1433.1	1469.2	1503.8	1537.4	1570.3	1634.8	1698.3	1761.2
	s	1.0351	1.0832		1.2877	1.3515	1.3951	1.4300	1.4600	1.4866	1.5110	1.5335	1.5547	1.5749	1.6126	1.6477	1.6806
	Sh																
3300	v				0.1213	0.1510	0.1727	0.1908	0.2070	0.2218	0.2357	0.2489	0.2613	0.2734	0.2966	0.3187	0.3400
	h				1158.2	1242.5	1297.7	1348.4	1390.7	1429.5	1466.1	1501.0	1534.9	1568.1	1632.9	1696.7	1759.9
	s				1.2742	1.3425	1.3879	1.4237	1.4542	1.4813	1.5059	1.5287	1.5501	1.5704	1.6084	1.6436	1.6767
	Sh																
3400	v				0.1129	0.1435	0.1653	0.1834	0.1994	0.2140	0.2276	0.2405	0.2528	0.2646	0.2872	0.3088	0.3295
	h				1143.2	1233.7	1294.3	1343.4	1386.4	1425.9	1462.9	1498.3	1532.4	1565.8	1631.1	1695.1	1758.5
	s				1.2600	1.3334	1.3807	1.4174	1.4486	1.4761	1.5010	1.5240	1.5456	1.5660	1.6042	1.6395	1.6728
	Sh																
3500	v				0.1048	0.1364	0.1583	0.1764	0.1922	0.2066	0.2200	0.2326	0.2447	0.2563	0.2784	0.2995	0.3198
	h				1127.1	1224.6	1287.8	1335.2	1382.2	1422.2	1459.7	1495.5	1529.9	1563.6	1629.2	1693.6	1757.2
	s				1.2450	1.3242	1.3734	1.4112	1.4430	1.4709	1.4962	1.5194	1.5412	1.5618	1.6002	1.6358	1.6691
	Sh																
3600	v				0.0986	0.1296	0.1517	0.1697	0.1854	0.1996	0.2128	0.2252	0.2371	0.2485	0.2702	0.2908	0.3106
	h				1110.6	1215.3	1281.2	1330.0	1377.9	1418.6	1456.5	1492.6	1527.4	1561.3	1627.3	1692.0	1755.9
	s				1.2281	1.3148	1.3662	1.4050	1.4374	1.4658	1.4914	1.5149	1.5369	1.5576	1.5962	1.6320	1.6654
	Sh																
3800	v				0.0790	0.1169	0.1395	0.1574	0.1729	0.1868	0.1995	0.2116	0.2231	0.2340	0.2549	0.2746	0.2936
	h				1064.2	1195.5	1267.6	1322.4	1369.1	1411.2	1450.1	1487.0	1522.4	1556.8	1623.6	1688.9	1753.2
	s				1.1868	1.2955	1.3517	1.3928	1.4265	1.4558	1.4821	1.5061	1.5284	1.5495	1.5885	1.6247	1.6584
	Sh																
4000	v				0.0631	0.1052	0.1284	0.1463	0.1616	0.1752	0.1877	0.1994	0.2105	0.2210	0.2411	0.2601	0.2783
	h				1007.4	1174.3	1253.4	1311.5	1363.2	1403.6	1443.6	1481.3	1517.3	1552.2	1619.8	1685.7	1750.6
	s				1.1396	1.2754	1.3371	1.3807	1.4158	1.4461	1.4730	1.4976	1.5203	1.5417	1.5812	1.6177	1.6516
	Sh																
4200	v				0.0498	0.0945	0.1183	0.1362	0.1513	0.1647	0.1769	0.1883	0.1991	0.2093	0.2287	0.2470	0.2645
	h				950.1	1151.6	1238.6	1300.4	1351.2	1396.0	1437.1	1475.5	1512.2	1547.6	1616.1	1682.6	1748.0
	s				1.0905	1.2544	1.3223	1.3686	1.4053	1.4366	1.4642	1.4893	1.5124	1.5341	1.5742	1.6109	1.6452
	Sh																
4400	v				0.0421	0.0846	0.1090	0.1270	0.1420	0.1552	0.1671	0.1782	0.1887	0.1986	0.2174	0.2351	0.2519
	h				909.5	1127.3	1223.3	1289.0	1342.3	1388.3	1430.8	1469.7	1507.1	1543.3	1612.3	1679.4	1745.3
	s				1.0556	1.2325	1.3073	1.3565	1.3943	1.4272	1.4556	1.4812	1.5048	1.5268	1.5673	1.6044	1.6389

Sh = superheat, F
v = specific volume, cu ft per lb

h = enthalpy, Btu per lb
s = entropy, Btu per F per lb

11/03 1.520

Table 3. Superheated Steam—*Continued*

Abs Press. Lb/Sq In. (Sat. Temp)		Sat Water	Sat Steam	750	800	850	900	950	1000	1050	1100	1150	1200	1250	1300	1400	1500
				Temperature - Degrees Fahrenheit													
4600	Sh																
	v			0.0370	0.0751	0.1025	0.1156	0.1335	0.1465	0.1582	0.1691	0.1792	0.1889	0.1982	0.2071	0.2242	0.2404
	h			823.8	1165.0	1207.3	1272.2	1322.6	1380.5	1422.7	1463.9	1501.9	1535.4	1573.8	1603.5	1676.3	1742.7
	s			1.0331	1.2054	1.2922	1.3446	1.3847	1.4181	1.4472	1.4734	1.4974	1.5197	1.5407	1.5607	1.5952	1.6330
4800	Sh																
	v			0.0355	0.0665	0.0927	0.1109	0.1257	0.1385	0.1500	0.1605	0.1705	0.1800	0.1890	0.1977	0.2142	0.2299
	h			866.9	1071.7	1193.7	1265.2	1323.1	1372.6	[illegible]	1458.0	1496.7	1533.8	1569.7	1604.7	1673.1	1740.0
	s			1.0180	1.1835	1.2758	1.3327	1.3745	[illegible]	1.4390	1.4657	1.4901	1.5128	1.5341	1.5543	1.5921	1.6272
5000	Sh																
	v			0.0338	0.0591	0.0855	0.1038	0.1185	0.1312	0.1425	0.1529	0.1626	0.1718	0.1805	0.1890	0.2050	0.2203
	h			854.9	1042.9	1171.6	1252.9	1313.5	1364.6	1410.2	1452.1	1491.5	1529.1	1565.5	1600.9	1670.0	1737.4
	s			1.0070	1.1593	1.2612	1.3207	1.3645	1.4001	1.4309	1.4582	1.4831	1.5061	1.5277	1.5481	1.5853	1.6216
5200	Sh																
	v			0.0326	0.0531	0.0789	0.0973	0.1129	0.1244	0.1356	0.1453	0.1553	0.1642	0.1728	0.1810	0.1966	0.2114
	h			845.8	1016.9	1156.0	1240.4	1303.7	1356.6	1403.4	1446.2	1486.3	1524.5	1561.3	1597.2	1665.8	[illegible]
	s			0.9985	1.1370	1.2455	1.3063	1.3545	1.3914	1.4223	1.4509	1.4762	1.4995	1.5214	1.5420	1.5806	1.6161
5400	Sh																
	v			0.0317	0.0483	0.0728	0.0912	0.1058	0.1182	0.1292	0.1392	0.1485	0.1572	0.1656	0.1736	0.1888	0.2031
	h			838.5	994.3	1128.1	1227.7	1293.7	1348.4	1396.5	1440.3	1481.1	1519.8	1557.1	1593.4	1663.7	1732.1
	s			0.9915	1.1175	1.2296	1.2969	1.3468	1.3827	1.4151	1.4437	1.4694	1.4931	1.5153	1.5362	1.5750	1.6109
5600	Sh																
	v			0.0309	0.0447	0.0672	0.0856	0.1001	0.1124	0.1232	0.1331	0.1422	0.1508	0.1589	0.1667	0.1815	0.1954
	h			832.4	975.0	1119.9	1214.5	1282.7	1340.2	1389.6	1434.3	1475.9	1515.2	1552.9	1589.6	1660.5	1729.5
	s			0.9855	1.1009	1.2137	1.2850	1.3348	1.3742	1.4075	1.4366	1.4628	1.4869	1.5093	1.5304	1.5697	1.6058
5800	Sh																
	v			0.0303	0.0419	0.0622	0.0805	0.0949	0.1070	0.1177	0.1274	0.1363	0.1447	0.1527	0.1603	0.1747	0.1883
	h			827.3	955.3	1101.8	1201.8	1273.6	1332.0	1382.6	1428.3	1470.6	1510.5	1548.7	1585.8	1657.4	1726.8
	s			0.9803	1.0862	1.1981	1.2732	[illegible]	1.3658	1.3999	1.4297	1.4564	1.4808	1.5035	1.5248	1.5644	1.6008
6000	Sh																
	v			0.0298	0.0397	0.0579	0.0757	0.0900	0.1020	0.1126	0.1221	0.1308	0.1391	0.1469	0.1544	0.1684	0.1817
	h			822.9	915.1	1084.6	1188.8	1263.4	1323.6	1375.7	1422.3	1465.4	1505.9	1544.6	1582.0	1654.2	1724.2
	s			0.9758	1.0746	1.1833	1.2615	1.3154	1.3574	1.3925	1.4229	1.4500	1.4743	1.4978	1.5194	1.5593	1.5960
6500	Sh																
	v			0.0287	0.0358	0.0495	0.0655	0.0793	0.0909	0.1012	0.1104	0.1188	0.1265	0.1340	0.1411	0.1544	0.1669
	h			813.9	919.5	1046.7	1156.3	1237.8	1302.7	1358.1	1407.3	1452.2	1494.2	1534.1	1572.5	1645.4	1717.6
	s			0.9661	1.0515	1.1506	1.2328	1.2917	1.3370	1.3743	1.4064	1.4347	1.4604	1.4841	1.5062	1.5471	1.5844
7000	Sh																
	v			0.0279	0.0334	0.0438	0.0573	0.0704	0.0816	0.0915	0.1004	0.1085	0.1160	0.1231	0.1298	0.1424	0.1542
	h			805.9	901.8	1016.5	1124.9	1212.6	1281.7	1340.5	1392.2	1439.1	1482.6	1523.7	1563.1	1638.6	1711.1
	s			0.9582	1.0350	1.1243	1.2055	1.2689	1.3171	1.3567	1.3904	1.4200	1.4466	1.4710	1.4938	1.5355	1.5735
7500	Sh																
	v			0.0272	0.0318	0.0399	0.0512	0.0631	0.0737	0.0833	0.0918	0.0996	0.1068	0.1136	0.1200	0.1321	0.1433
	h			801.3	889.0	992.9	1097.7	1188.3	1261.0	1322.9	1377.2	1426.0	1471.0	1513.3	1553.7	1630.8	1704.6
	s			0.9514	1.0224	1.1033	1.1818	1.2473	1.2980	1.3397	1.3751	1.4059	1.4335	1.4586	1.4819	1.5245	1.5632
8000	Sh																
	v			0.0267	0.0305	0.0371	0.0465	0.0571	0.0671	0.0762	0.0845	0.0920	0.0989	0.1054	0.1115	0.1230	0.1338
	h			796.6	879.1	974.4	1074.3	1165.4	1241.0	1305.5	1362.2	1413.0	1459.6	1503.1	1544.5	1623.1	1698.1
	s			0.9455	1.0122	1.0864	1.1613	1.2271	1.2798	1.3233	1.3603	1.3924	1.4208	1.4467	1.4705	1.5140	1.5533
8500	Sh																
	v			0.0262	0.0296	0.0350	0.0429	0.0522	0.0615	0.0701	0.0780	0.0853	0.0919	0.0982	0.1041	0.1151	0.1254
	h			792.7	871.2	959.8	1054.5	1144.0	1221.9	1288.5	1347.5	1400.2	1448.7	1493.9	1535.3	1615.4	1691.7
	s			0.9402	1.0037	1.0727	1.1437	1.2084	1.2627	1.3076	1.3460	1.3793	1.4087	1.4352	1.4597	1.5040	1.5439
9000	Sh																
	v			0.0256	0.0288	0.0335	0.0402	0.0483	0.0568	0.0649	0.0724	0.0794	0.0858	0.0918	0.0975	0.1081	0.1179
	h			789.3	864.7	948.0	1037.6	1125.4	1204.1	1272.1	1333.0	1387.5	1437.1	1482.9	1526.3	1607.9	1685.3
	s			0.9354	0.9964	1.0613	1.1285	1.1918	1.2468	1.2925	1.3323	1.3667	1.3970	1.4243	1.4492	1.4944	1.5349
9500	Sh																
	v			0.0254	0.0282	0.0322	0.0380	0.0451	0.0528	0.0603	0.0675	0.0742	0.0804	0.0862	0.0917	0.1019	0.1113
	h			786.4	859.2	938.3	1023.4	1107.9	1187.7	1255.6	1318.9	1375.1	1426.1	1473.1	1517.3	1600.4	1679.0
	s			0.9310	0.9900	1.0516	1.1153	1.1771	1.2320	1.2785	1.3191	1.3546	1.3858	1.4137	1.4392	1.4851	1.5263
10000	Sh																
	v			0.0251	0.0276	0.0312	0.0362	0.0425	0.0495	0.0565	0.0633	0.0697	0.0757	0.0812	0.0865	0.0963	0.1054
	h			783.8	854.5	930.2	1011.3	1094.2	1172.6	1240.0	1305.3	1362.9	1415.3	1463.4	1508.6	1592.1	1672.3
	s			0.9270	0.9842	1.0432	1.1039	1.1638	1.2185	1.2652	1.3065	1.3429	1.3749	1.4035	1.4295	1.4763	1.5180
10500	Sh																
	v			0.0249	0.0271	0.0303	0.0347	0.0404	0.0467	0.0532	0.0595	0.0656	0.0714	0.0768	0.0818	0.0913	0.1001
	h			781.5	850.5	923.4	1001.0	1081.3	1158.9	1226.1	1292.4	1351.1	1404.7	1453.9	1500.0	1585.8	1665.7
	s			0.9232	0.9790	1.0358	1.0939	1.1519	1.2060	1.2529	1.2946	1.3371	1.3644	1.3937	1.4202	1.4677	1.5100

Sh = superheat, F
v = specific volume, cu ft per lb
h = enthalpy, Btu per lb
s = entropy, Btu per F per lb

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Table 3. Superheated Steam – Continued

Abs Press Lb/Sq In. (Sat. Temp)		Sat Water	Sat Steam	750	800	850	900	950	1000	1050	1100	1150	1200	1250	1300	1400	1500
11000	v			0.0245	0.0267	0.0296	0.0335	0.0386	0.0444	0.0503	0.0562	0.0620	0.0679	0.0727	0.0776	0.0863	0.0952
	h			779.5	846.9	917.5	992.1	1069.9	1146.3	1215.9	1272.2	1329.7	1384.4	1438.6	1491.5	1578.7	1660.6
	s			0.9196	0.9742	1.0292	1.0851	1.1412	1.1945	1.2414	1.2833	1.3203	1.3544	1.3842	1.4112	1.4595	1.5023
11500	v			0.0243	0.0263	0.0290	0.0325	0.0370	0.0423	0.0478	0.0534	0.0588	0.0641	0.0691	0.0739	0.0827	0.0909
	h			777.7	843.9	912.4	984.5	1061.8	1136.9	1204.3	1258.7	1308.3	1354.4	1425.5	1483.2	1571.8	1654.7
	s			0.9153	0.9698	1.0232	1.0772	1.1326	1.1840	1.2308	1.2727	1.3107	1.3446	1.3750	1.4025	1.4515	1.4949
12000	v			0.0241	0.0262	0.0284	0.0317	0.0357	0.0405	0.0456	0.0506	0.0560	0.0610	0.0659	0.0704	0.0790	0.0869
	h			776.1	841.0	907.9	977.8	1052.9	1124.5	1195.7	1258.0	1318.5	1374.7	1426.6	1479.1	1564.9	1648.8
	s			0.9131	0.9657	1.0177	1.0701	1.1229	1.1742	1.2209	1.2627	1.3010	1.3353	1.3662	1.3941	1.4438	1.4877
12500	v			0.0238	0.0256	0.0279	0.0309	0.0345	0.0390	0.0437	0.0486	0.0535	0.0583	0.0629	0.0673	0.0755	0.0832
	h			774.7	839.6	903.9	971.9	1043.1	1115.3	1184.1	1247.0	1302.3	1365.4	1420.0	1467.2	1558.2	1643.1
	s			0.9101	0.9618	1.0127	1.0637	1.1151	1.1652	1.2117	1.2538	1.2918	1.3264	1.3576	1.3860	1.4363	1.4808
13000	v			0.0236	0.0253	0.0275	0.0302	0.0335	0.0375	0.0420	0.0466	0.0512	0.0558	0.0602	0.0645	0.0725	0.0799
	h			773.5	836.3	900.4	966.8	1035.2	1106.7	1174.8	1233.5	1293.6	1355.5	1409.6	1459.4	1551.6	1637.4
	s			0.9073	0.9582	1.0083	1.0578	1.1079	1.1571	1.2030	1.2445	1.2831	1.3179	1.3494	1.3781	1.4291	1.4741
13500	v			0.0235	0.0251	0.0271	0.0297	0.0328	0.0364	0.0405	0.0448	0.0492	0.0535	0.0577	0.0619	0.0696	0.0769
	h			772.3	834.4	897.2	962.2	1028.0	1096.1	1165.3	1229.7	1291.0	1346.1	1402.5	1451.3	1545.2	1631.9
	s			0.9045	0.9548	1.0037	1.0524	1.1014	1.1495	1.1948	1.2361	1.2749	1.3098	1.3415	1.3705	1.4221	1.4675
14000	v			0.0233	0.0248	0.0267	0.0291	0.0320	0.0354	0.0392	0.0432	0.0474	0.0515	0.0555	0.0595	0.0670	0.0740
	h			771.3	832.6	894.3	959.0	1024.5	1092.3	1155.5	1221.4	1283.0	1342.2	1393.5	1444.4	1539.8	1626.5
	s			0.9019	0.9515	0.9996	1.0473	1.0953	1.1429	1.1872	1.2282	1.2671	1.3021	1.3339	1.3631	1.4153	1.4612
14500	v			0.0231	0.0246	0.0264	0.0287	0.0314	0.0345	0.0380	0.0418	0.0458	0.0496	0.0534	0.0573	0.0646	0.0714
	h			770.4	831.0	891.7	954.3	1019.6	1085.2	1151.4	1213.8	1275.4	1332.9	1375.4	1437.3	1532.6	1621.1
	s			0.8994	0.9484	0.9957	1.0425	1.0897	1.1352	1.1801	1.2203	1.2597	1.2949	1.3266	1.3560	1.4087	1.4551
15000	v			0.0230	0.0244	0.0261	0.0282	0.0308	0.0337	0.0369	0.0405	0.0443	0.0479	0.0515	0.0552	0.0624	0.0690
	h			769.6	829.5	889.3	952.0	1015.1	1078.6	1144.9	1206.8	1268.1	1322.0	1379.4	1432.3	1526.4	1615.9
	s			0.8970	0.9455	0.9920	1.0382	1.0846	1.1302	1.1735	1.2139	1.2525	1.2880	1.3197	1.3491	1.4022	1.4491
15500	v			0.0228	0.0242	0.0259	0.0278	0.0302	0.0329	0.0360	0.0393	0.0429	0.0464	0.0499	0.0534	0.0603	0.0668
	h			768.8	828.2	887.7	947.8	1011.1	1075.7	1139.0	1200.3	1261.1	1313.6	1372.8	1422.5	1520.4	1610.8
	s			0.8946	0.9427	0.9886	1.0340	1.0797	1.1247	1.1674	1.2073	1.2457	1.2815	1.3131	1.3424	1.3959	1.4433

* Sh = superheat, F
v = specific volume, cu ft per lb
h = enthalpy, Btu per lb
s = entropy, Btu per F per lb

$N = N_0 e^{-\lambda \Delta t} = N_0 e^{nt/T_{1/2}}$

NUCLEAR PHYSICS

I. Natural Radioactivity

A. Alpha Particles (α, ${}_2He^4$) A particle emitted from the nucleus of certain radioactive atoms. Has the same characteristics as that of a helium atom nucleus. Consists of a packet containing 2 protons plus two neutrons with a mass of 6.6442×10^{-24} grams and a charge of +2. Travels through air with an approximate velocity of 1.6×10^9 cm/sec (1/20 speed of light) and has a range (in air) of approximately 2 inches (mev).

B. Beta Particle (${}_{-1}\beta^0$) A particle emitted from the nucleus of certain radioactive atoms. Has the same characteristics as that of an electron. It has approximately no mass (9×10^{-28} grms) and a charge of -1. Travels with extremely high velocity (98% of the speed of light). Its range in air is approximately 100 inches (1mev). 500 X more penetrating than α

C. Gamma Photons (γ) : A packet of energy released from the nucleus of certain radioactive atoms. Consists of a packet of electomagnetic radiation similar to X-rays but of shorter wavelength. It has no mass or charge and is very penetrating in matter.

II. Radioactive Change

A. Alpha Decay :

Ex.

$$ {}_{92}U^{238} \longrightarrow {}_{90}Th^{234} + {}_2He^4 + {}_0\gamma^0 $$

II. Radioactive Change (Con't)

B. Beta Decay :

Ex.

$$_{83}Bi^{214} \longrightarrow {}_{84}Po^{214} + {}_{-1}\beta^{0} + {}_{0}\gamma^{0}$$

$$\overbrace{N \longrightarrow {}_{1}P^{1} + {}_{-1}\beta^{0}}$$ (Internal Conversion)

C. Gamma Decay

Ex.

$$_{55}^{*}Cs^{137} \longrightarrow {}_{55}Cs^{137} + {}_{0}\gamma^{0}$$

For ea. decay it is unlikely of one type of particel per decay.

D. Rate of Radioactive Decay

1. Each radioactive isotope disintegrates at a precise rate.

2. The decay process is a statistical one.

3. Decay is an exponential process.

4. Rate of decay equals:

$$\boxed{\text{Rate of decay} = \lambda N}$$

where : λ = a decay constant ($time^{-1}$)

N = number of atoms present

5. Variation of "N" with time:

$$\boxed{N = No\, e^{-\lambda t}} \quad \lambda t$$

where: No = atoms present at t = o

N = atoms present at some later time "t"

Δt = time of decay $\Delta t = t_1 - t_0$

λ = decay constant ($time^{-1}$)

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Faust

1. Cm^{242}_{96}

2. Pu^{238}_{94}

3. a. $N = 10^{24}$ atoms

$T_{1/2} = 16$ h. $T_{1/2} = .693/\lambda = \frac{.693}{T_{1/2}} = \lambda$

Decay Rate $= N\lambda = N \frac{.693}{T_{1/2}}$

$= 10^{24} \frac{.693}{16 \text{ hrs}}$ $\quad 16 \text{ hrs} \times \frac{3600 \text{ sec}}{hr}$

$= \frac{6.93 \times 10^{23}}{5.75 \times 10^{4}} = \underline{1.205 \times 10^{19}}$ dis/sec $\quad 5.75 \times 10^4$

b. $1c = 3.7 \times 10^{10}$ dis/sec

$?c = \frac{1.205 \times 10^{19}}{3.7 \times 10^{10}} = .3255 \times 10^{9}$ or

3.255×10^{8} Curies

c. $10 \text{ hrs} \frac{3600 \text{ sec}}{hr}$ 3.6×10^{4} sec $= \Delta t$

$\lambda = 1.205 \times 10^{-5}$ dis/sec

$N = N_0 e^{-\lambda \Delta t}$

$= 10^{24} e^{-1.205 \times 10^{-5} \text{dis/sec} \cdot 3.6 \times 10^{4} \text{sec}}$

$N = 10^{24} e^{-.434} = 10^{24} .648 = 6.48 \times 10^{23}$ atoms

Rate decay $= 6.48 \times 10^{23}$ atoms $\times 1.205 \times 10^{-5}$ dis/sec $= \underline{5.37 \times 10^{18}}$ dis/sec

$C = \frac{5.37 \times 10^{18}}{3.7 \times 10^{10}} = \underline{1.45 \times 10^{8}}$ Curies

8003 1045

$^{235}_{92}U$

↓

$^{231}_{90}Th$ α

↓

$^{227}_{88}Ra$ α

↓

$^{227}_{89}Ac$ β^-

↓

$^{227}_{90}Th$ β^-

↓

$^{223}_{88}Ra$ α

↓

$^{219}_{86}Rn$ α

↓

$^{215}_{84}Po$ α

↓

$^{211}_{82}Pb$ α

↓

$^{211}_{83}Bi$ β^-

↓

$^{207}_{81}Tl$ α → $^{207}_{82}Pb$ β^- Stable

1 curie = 3.7×10^{10} dis/sec

6. Half Life : The length of time required for one-half of the atoms in a sample of a radioactive isotope to decay.

7. $T_{1/2}$ vs. λ

$$\boxed{\lambda = \frac{0.693}{T_{1/2}}} \qquad T_{1/2} = \frac{.693}{\lambda}$$

$$\boxed{N = N_0 e^{-\lambda t} = N_0 e^{-\frac{t\,.693}{T_{1/2}}}}$$ Rate of decay by half life.

III. Interaction of Charged Particles with Matter

A. Alpha Particles : lose energy chiefly by the interaction with orbital electrons of the atoms making up the substance through which it travels (attractive forces).

1. Energy required to produce one ion pair in most gases equals 35 ev.

(a) Electron volt (ev) is a unit of energy equivalent to that energy acquired by an electron transversing a potential of 1 volt. One mev = 10^6ev.

2. An alpha particle having an energy of 5 mev would result in the formation of 5×10^5 ev/32 ev = 1.4×10^5 ion pairs as it travels through a gas.

3. Length of track is governed by the rate expressed in ion pairs per cm of travel. The higher the rate, the more rapidly will the alpha be slowed down, the shorter will be the track length. (Track for 5 mev = 40,000 ion pairs/cm)

(a) Rate ∝ electron density of target (the greater the density, the greater the ionization).

A. Alpha Particles: (Con't)

3. (a) Result : The denser the material, the faster it will be stopped, the shorter the track length.

   (b) The greater the charge, the greater the force on the electrons, ∴ the greater the rate of ionization.

   (c) The higher the velocity, the smaller will be the rate of ionization. $E = \frac{1}{2} M V^2 \rightarrow V = \sqrt{\frac{2E}{M}}$

4. Range of Alpha Particle

Range (in)

Energy (mev)	Air	Tissue	Concrete	Lead
1	0.2"	0.003"	0.0002"	0.00005"
2	0.4"	0.0006"	0.0003"	0.00010"
5	1.4"	0.0017"	0.0010"	0.00028"

B. Beta Particles:

1. Interacts with electron structure of material which it penetrates (repulsive forces)

2. Since very small, it travels very fast and is subjected to scattering events.

3. Rate of ionization much less than alpha.

4. A 5 mev Beta will produce 1.4×10^5 ion pairs in air at a rate of 80 ion pairs per cm.

3. Beta Particles: (Con't)

5. Since subjected to scattering events, it is considered not to have a definite range as does an alpha particle.

6. Approximate range of a Beta Particle :

Energy (mev)	Range (in) Air	Tissue	Concrete	Lead
1	115	0.132	0.060	0.014
2	260	0.305	0.180	0.034
5	665	0.830	0.598	0.089

C. Gamma Photons

1. Gamma Photons interact with matter in three ways:

(a) Photoelectric Effect

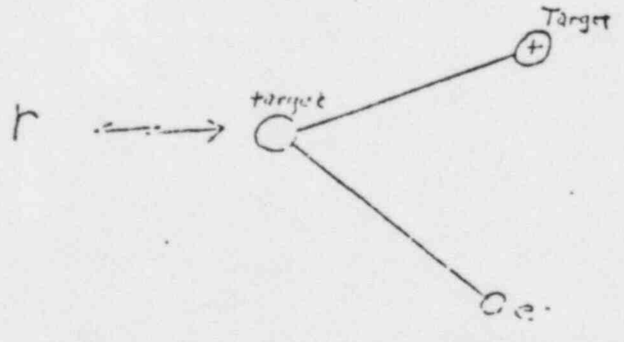
(b) Compton Effect

(c) Pair Production

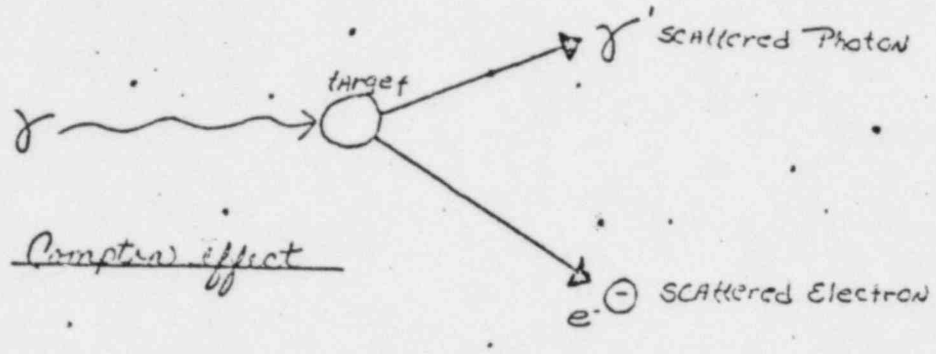
2. Photoelectric Effect: gamma photon is absorbed by one of the orbital electrons in a target atom. Electron thus acquires enough energy to eject from orbit and atom and thru causes additional ionization as it travels through matter.

(a) Photoelectric effect is most likely to occur with gammas of low energy. (< 1mev)

PE effect:



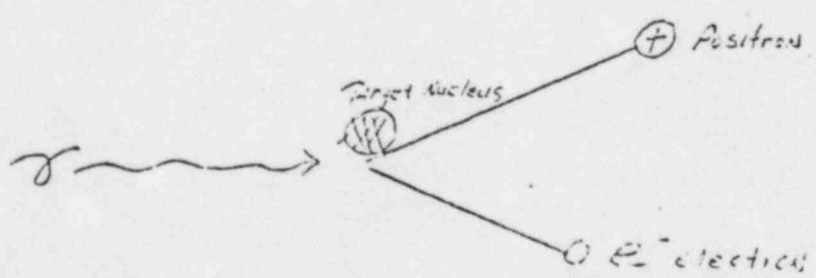
3. Compton Effect : Consists of a billiard ball type of collision. A portion of the incident γ photons is absorbed by an orbital electron. The result of which is an ejection of an orbital electron from the target atom and the scattered γ photon (of lower energy).

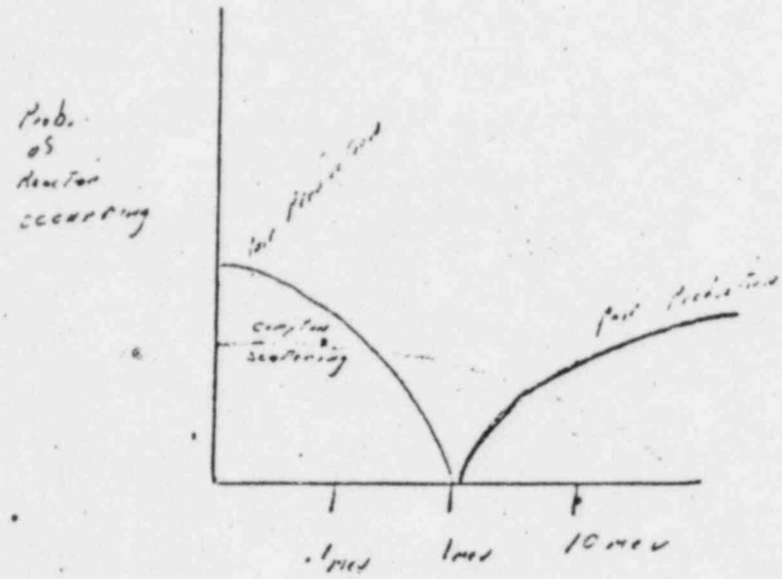


Compton effect

4. Pair Production : At energies ≧ 1.02mev gamma photons penetrate close to the nucleus of target atoms, disapperars and gives rise to the creation of a pair of electrons. One a negative charged electron and the other a positive charged electron (called a positron).

PAIR PRODUCTION





Graph illustrates the area of coverage at the different energies the a probable reaction would occure.

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

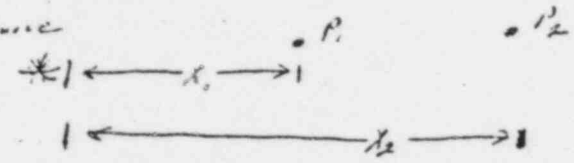
6"

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6″

Gamma Radiation Point Source



(ϕ) <u>Gamma flux</u> - is the number of γ's passing through a unit area per second (γ/cm^2-sec)

flux symbole

$\phi \approx 1/x^2$

(from above illustration above)

To calculate flux from Point Source

$\phi_1 = \frac{\text{Source } (S_0)}{4\pi x_1^2}$ ← abbr.

$\phi_2 = \frac{S_0}{4\pi x_2^2}$

Generally stated

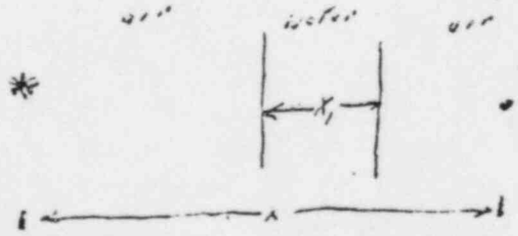
$$\boxed{\phi = \frac{S_0}{4\pi x^2}}$$

<u>flux</u>

Distance to object for point.

Through a vacuum

Taking shielding into account. ~~and~~ those that make it through without having any interactions.



μ { dependent on initial energy of γ for material through material

uncollided flux

$$\boxed{\phi = \frac{S_0 \, e^{-\mu_w x_1}}{4\pi x^2}}$$

S_0 - Source Strength

x - dist. from S_0 to pt. P

μ_w - linear attenuation coefficient of water

x_1 - thickness of water

8603 1046

found on graph

(must be treated as one term by definition) ρ density of shield material

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

To get a proper value for μ the equation this must be solved for

$\mu \equiv$ mass absorption coefficient

35 ev / ion Pair

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Dose Rates.

$1\ R/hr \approx 1\ Rem/hr$ thumb rule

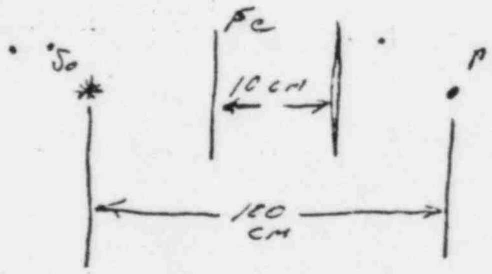
units cancel out so don't worry about them.

means a rate — Empty quantity read off graph (R/hr per photon/cm² sec)

Dose rate uncollided — $\dot{D}_u = \left[k(E)\right] E \left[\phi_u(E)\right]$

uncollided flux (γ/cm²-sec)

Energy of photon or γ (MEV)



$S_o = 1.0 \times 10^{10}\ \gamma/sec$; 2 MEV

$\rho = 7.86\ gm/cm^3$

$\mu = \left(\frac{\mu}{\rho}\right)\rho = (.041)(7.86\ gm/cm^3) = .322\ cm^{-1}$

$\phi_u = \dfrac{S_o\, e^{-\mu_o x_1}}{4\pi r^2} = \dfrac{(10^{10}\ \gamma/sec)\, e^{-.322\ cm^{-1}(10\ cm)}}{(4)(3.14)(100\ cm)^2} = 3.2 \times 10^{3}\ \gamma/cm^2\text{-sec}$

$\dot{D}_u = \left[k(E)\right] E \left[\phi_u(E)\right]$ whole term means flux is dependent on energy, treat as just ϕ_u.

$= (1.6 \times 10^{-6})(2)(3.2 \times 10^{3}) = 10.24 \times 10^{-3}\ R/hr$ = Same in mrem/hr, close to ≈

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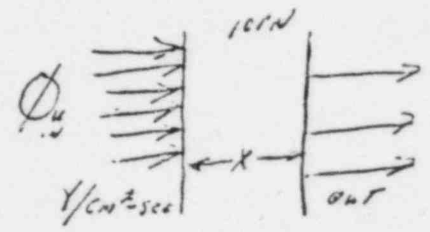
Dose Rates Cont.



Flux in = Flux out $\times e^{-\mu x}$

$$\boxed{\phi_{out} = \phi_{in}\, e^{-\mu_a x}}$$

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

Total Flux & Total Dose Rate.

When considering Photo-electric & Pair Production the gamma is considered to be absorbed or just never reaches the other side of shield. When Compton scattering (mainly) and other's, the gamma makes it through the shield at a lower ~~energy~~ energy but adds to the flux seen on the other side of shield. This increase is called Build up Factor (B).

$$\boxed{\dot{D}_{Total} = \underbrace{K(E)}_{\text{found on graph (function of Energy)}}\, E\, \phi_a\, B}$$

value used to determine line on graph to use for finding B

$\mu x = b$

$e^{-\mu x} = e^{-b}$

= # of MFP

average dist. a γ travels before a collision

MFP → mean free path

$\mu_0 r$ Same as μx

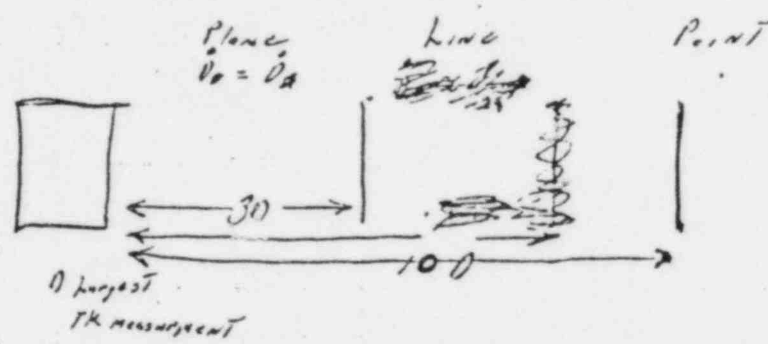
To find Z number on graph for molecule add up protons (atomic no.) total

Cont. Dose Rate

Concrete-Z = 26

$\dot{D}$ = Dose Rate

other than Point



<30 Plane source $\dot{D}_1 = \dot{D}_2$

$>30 - <100$ Line Source $\dot{D}_1 x_1 = \dot{D}_2 x_2$

>100 Point $\dot{D}_1 x_1^2 = \dot{D}_2 x_2^2$

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Frost

3.c. $\mu = \left(\frac{\mu}{\rho}\right)\rho = (.036)(2.3) = .0828\ cm^{-1}$

$$\Phi_u = \frac{S_0\, e^{-\mu x_1}}{4\pi x^2} = \frac{3.9\times10^{12}\ \gamma/sec\ \ e^{-(.0828)(6'')}}{(4)(3.14)(18.'')^2} \quad 15.2\ cm \quad 1.258\ cm$$

$(45.7\ cm)^2$

10.52×10^{4}

$$\doteq \frac{3.9\times10^{12}\ \gamma/sec\ \ \cdot 2845\ cm}{(12.58)(45.7\ cm)^2}$$

$2100\ cm^2$

$2.64\times10^{4}\ cm^2$

$$\Phi_u = 3.99\times10^{7}\ \gamma/cm^2\text{-}sec$$

$\dot{D}_u = K(E)\ E\ \Phi_u$ units

$= 1.43\times10^{-6}\times 3\ mev \times 3.99\times10^{7}\ \gamma/cm^2\text{-}sec$

$171.2\ R/hr$ or $\approx 171.2\ rem/hr.$

4. $\mu = (\mu/\rho)\rho = (.04)(1) = .04\ cm^{-1}$

$$\Phi_w = \frac{10^{6}\ \gamma/sec \times e^{-(.04)(30.48\ cm)}}{4\pi\ (45.7\ cm)^2} = \frac{.2955\times10^{6}}{2.64\times10^{4}}$$

-1.22

$2.64\times10^{4}\ cm^2 \qquad = 1.12\times10^{1}$

$= 11.2\ \gamma/cm^2\text{-}sec$

$\dot{D}_{total} = (KE)\ E\ (\Phi)\ B$

$= (1.43\times10^{-6})\times(3)(11.2)(5.7) = 13\times10^{-5}\ R/hr = 13\times10^{-5}\ rem/hr$

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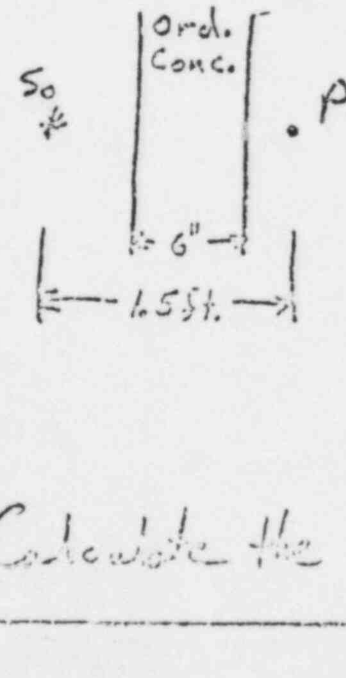
Problems on Lecture #3

1) What is the flux 50 cm. away from a 50 curie point source shielded only by air?

2) If the flux 100 cm. from a point source is 10^6 γ's/cm^2-sec, what is the source strength (γ's/sec) if air is the medium?

3) a) If the density of ordinary concrete is 2.3 gm/cm^3, calculate the value of μ for 5 Mev gammas.

b. Calculate the uncollided flux for the following figure at point P



1 ft. = 30.48 cm

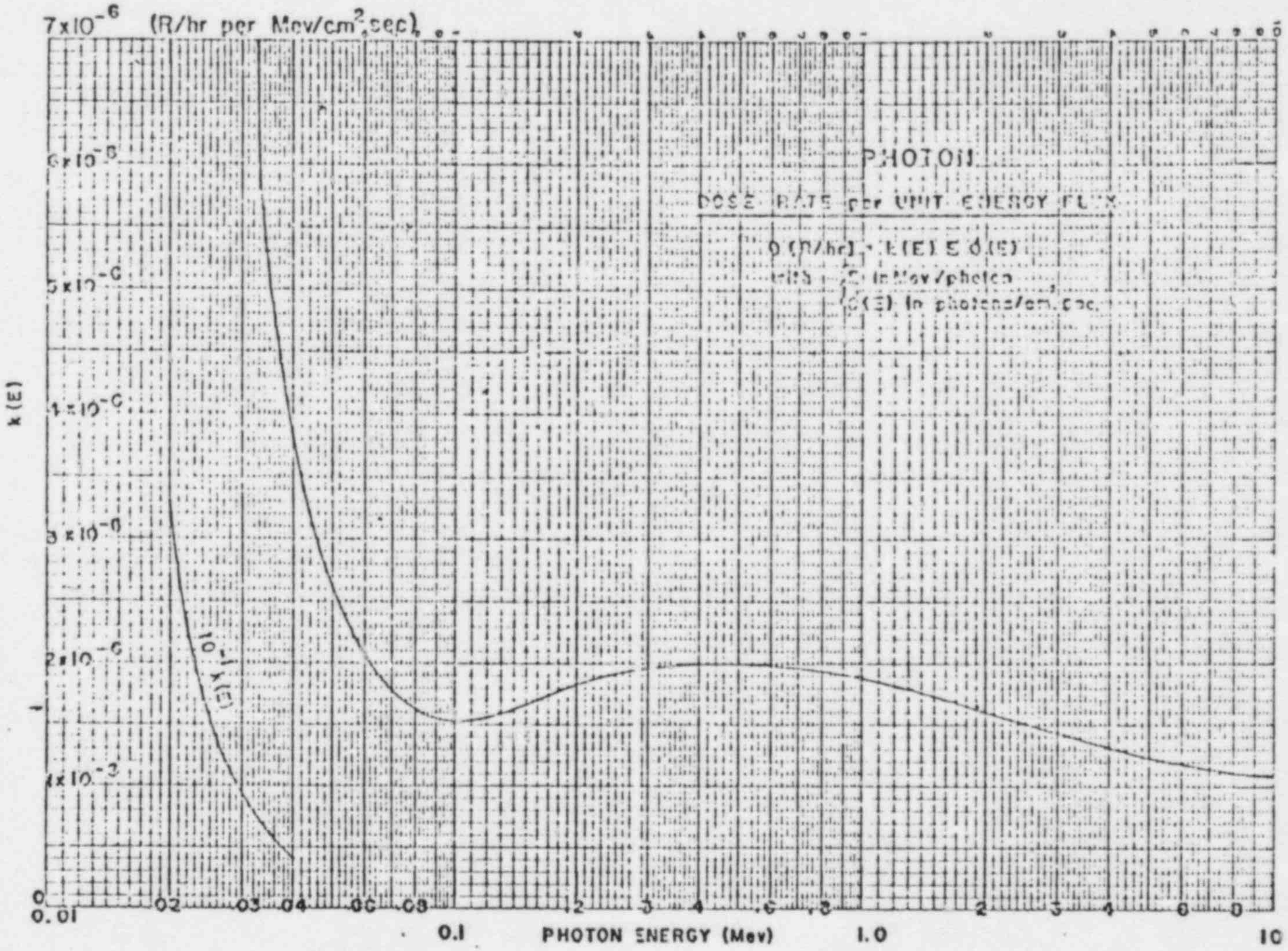
S_0 is a 100 Curie source of 3 Mev gammas. (1 three Mev γ released per disintegration)

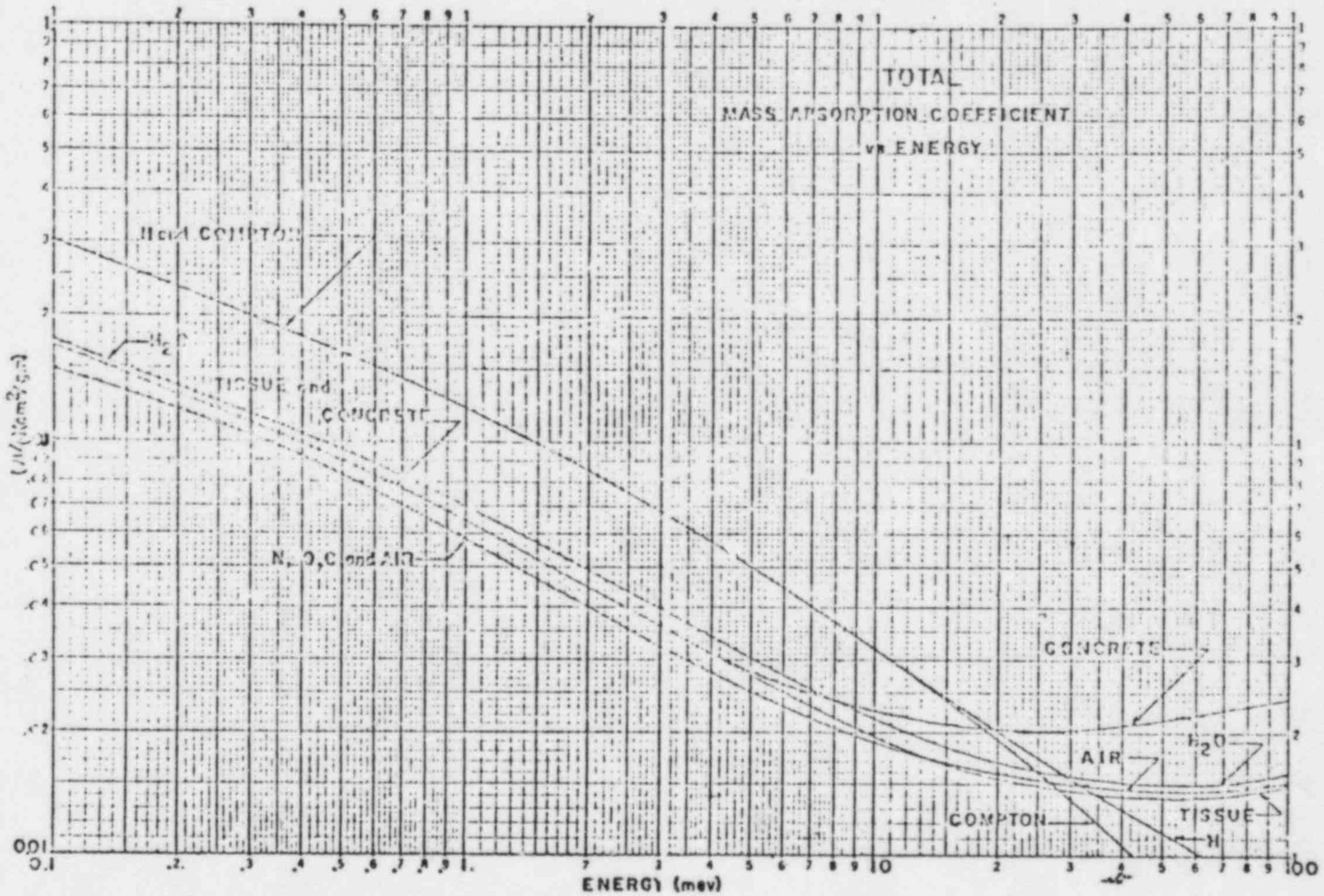
c. Calculate the uncollided dose rate for part b.

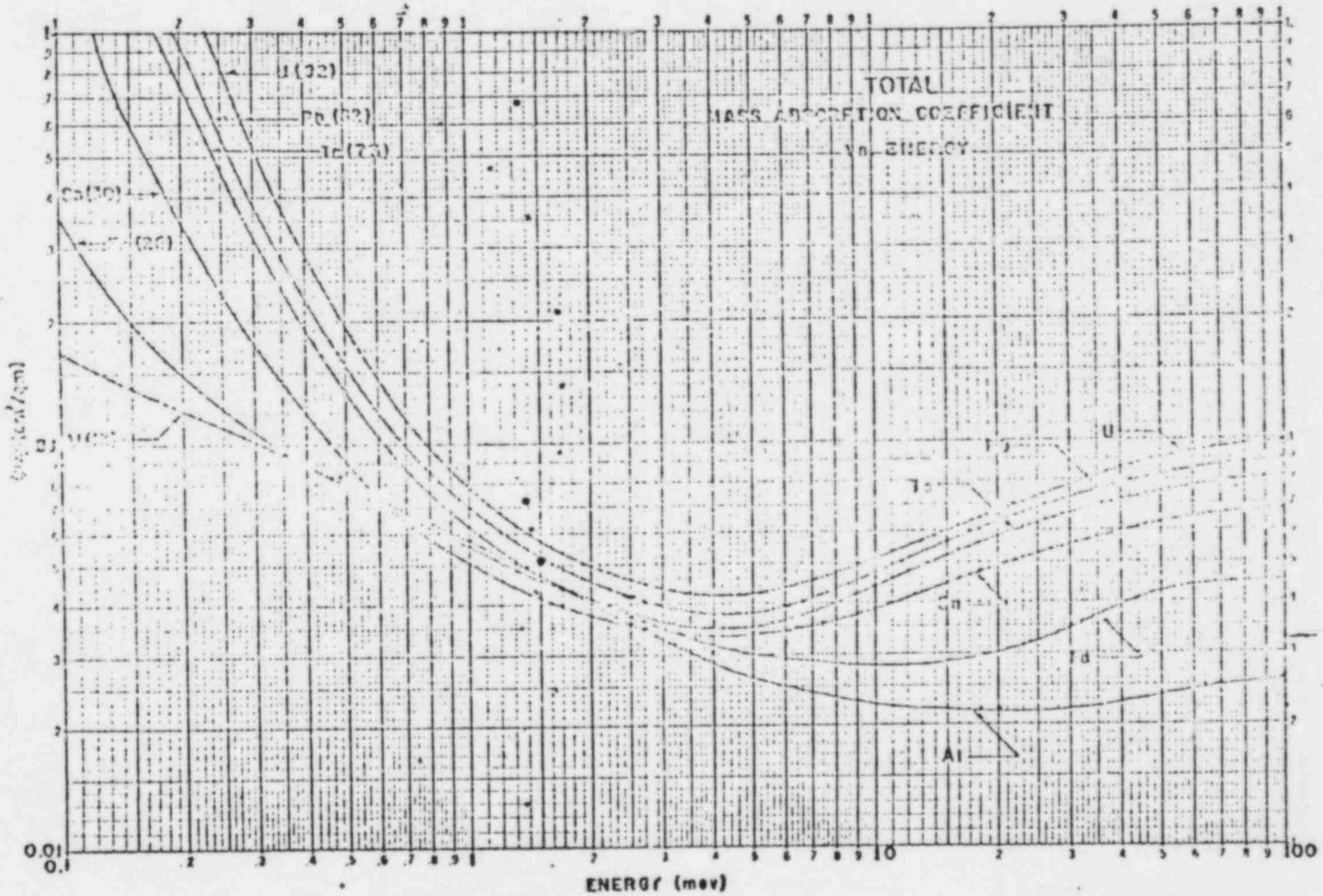
1. If you have a 5 Curie source of Cobalt-60 at some time t_o, how many atoms remain in the source 389 days after t_o?

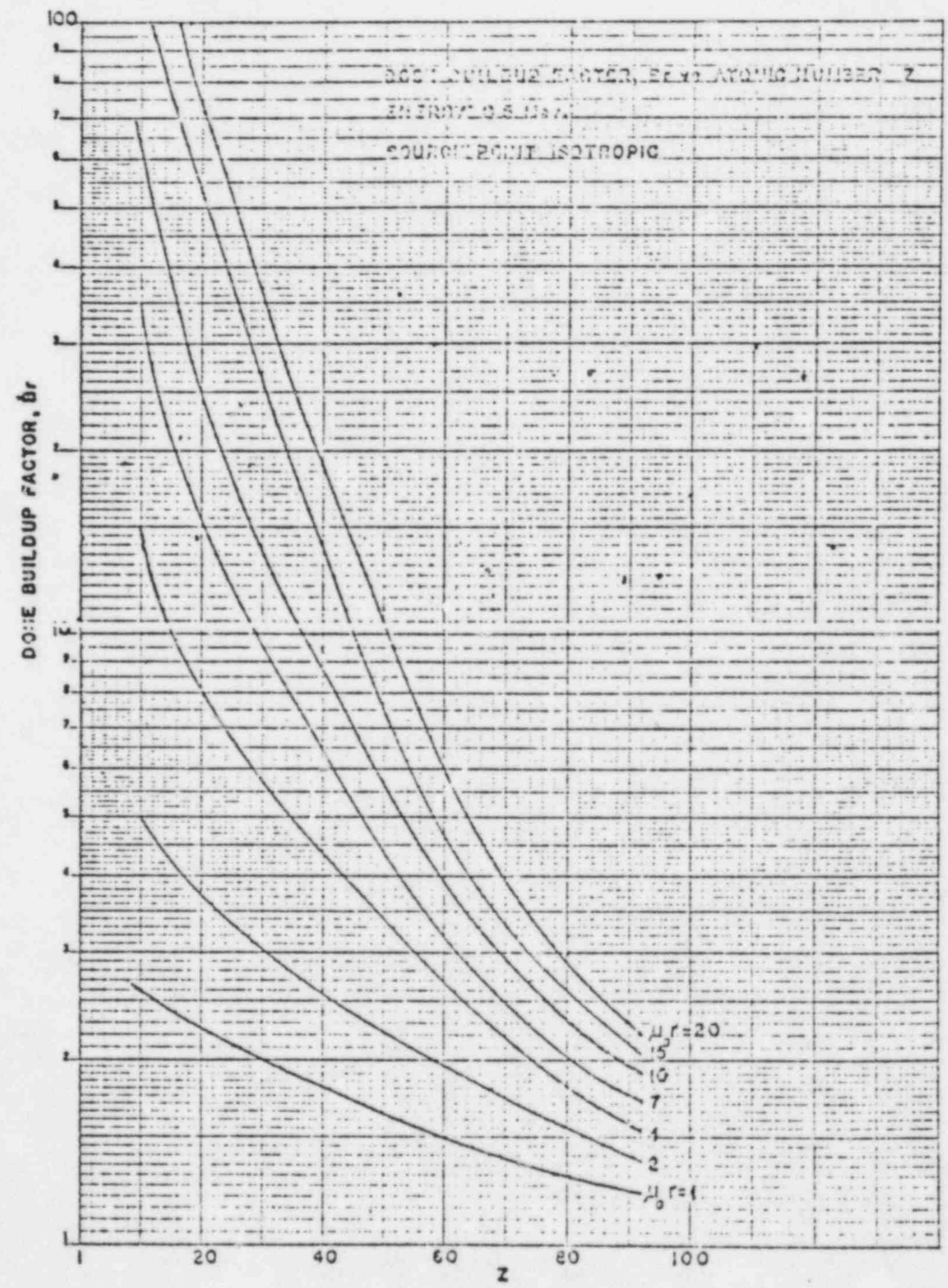
2. For the source in the problem above, what is the total # of gammas emitted from time t_o to 5 min. after t_o? 2 γ's given off per co. disintegration

3. I^{135} decays with a half life of 6.7h to form Xe^{135} which has a half life of 9.16 hrs. At time t_o you have 10 curies of I^{135} and 1 curie of Xe^{135}. How many minutes after time t_o does the number of Xe^{135} atoms equal the number of I^{135} atoms?

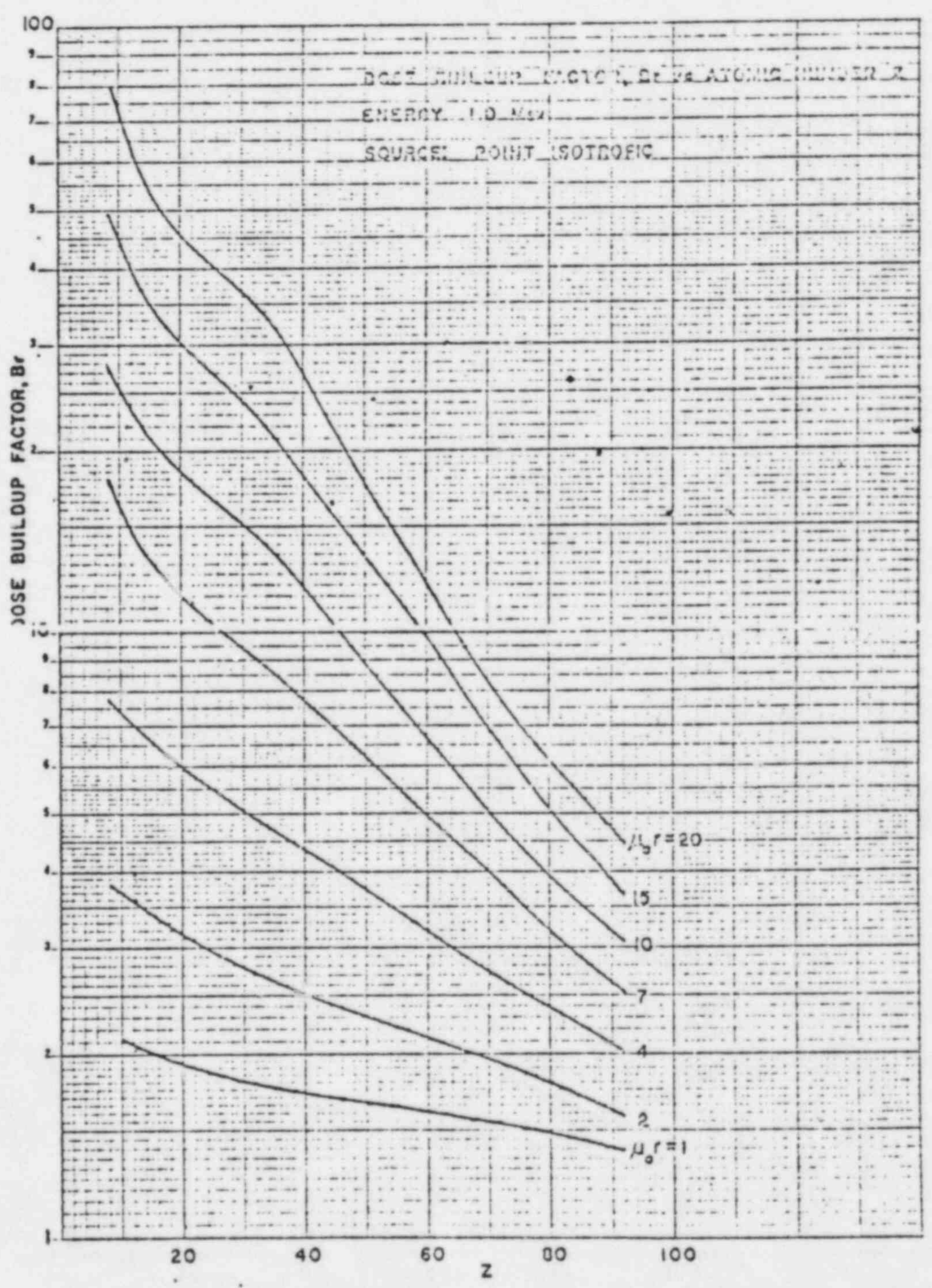

7x10^-6 (R/hr per Mev/cm²,sec)
6x10^-6
5x10^-6
4x10^-6
3x10^-6
2x10^-6
1x10^-3
0
k(E)
PHOTON
DOSE RATE per UNIT ENERGY FLUX
with E in Mev/photon
φ(E) in photons/cm² sec
10^-1 k(E)
0.01
0.1
1.0
10
PHOTON ENERGY (Mev)

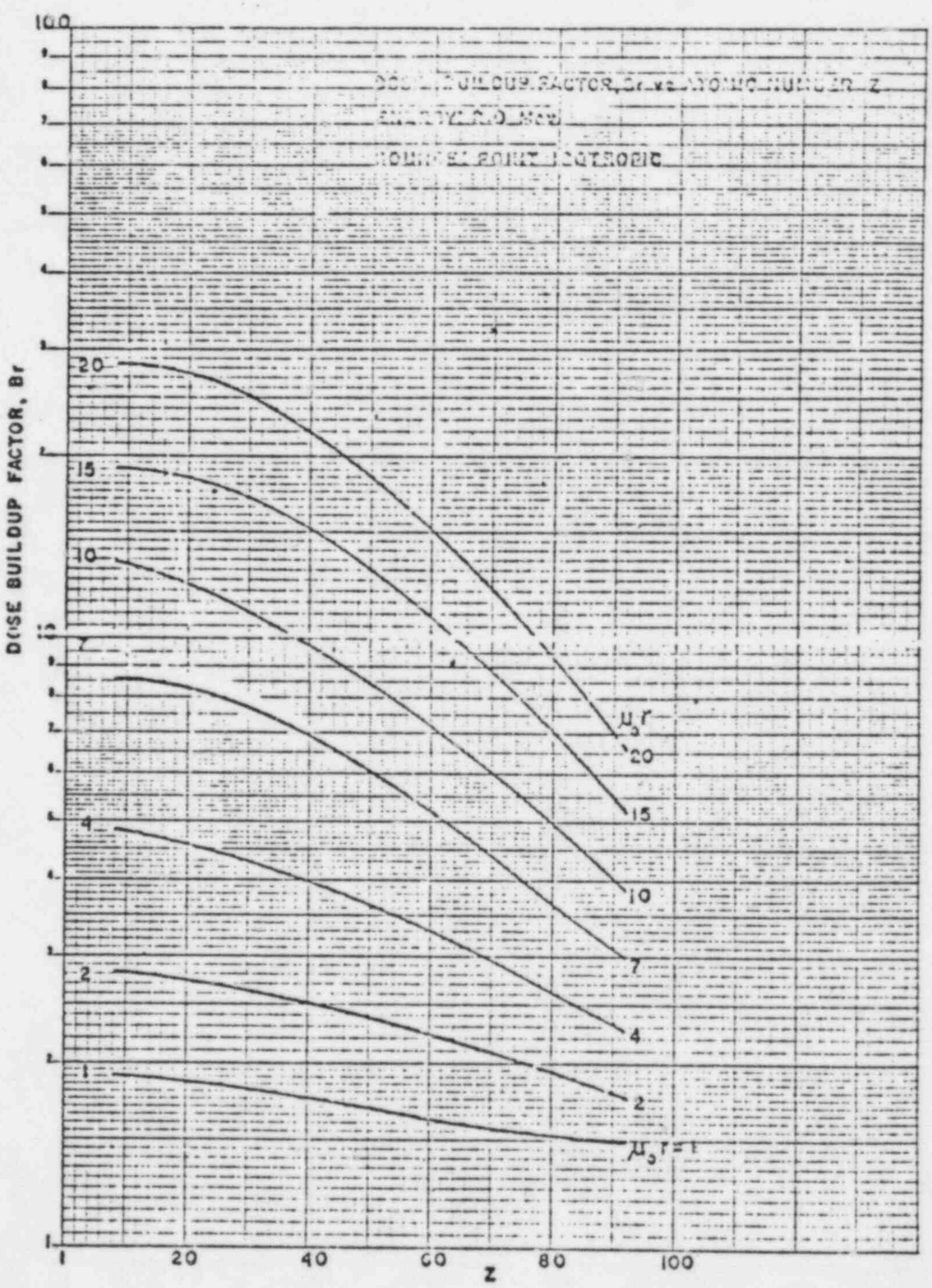

TOTAL
MASS ABSORPTION COEFFICIENT
vs ENERGY
H and COMPTON
H_2O
TISSUE and
CONCRETE
N, O, C and AIR
CONCRETE
AIR
H_2O
COMPTON
TISSUE
H
ENERGY (mev)

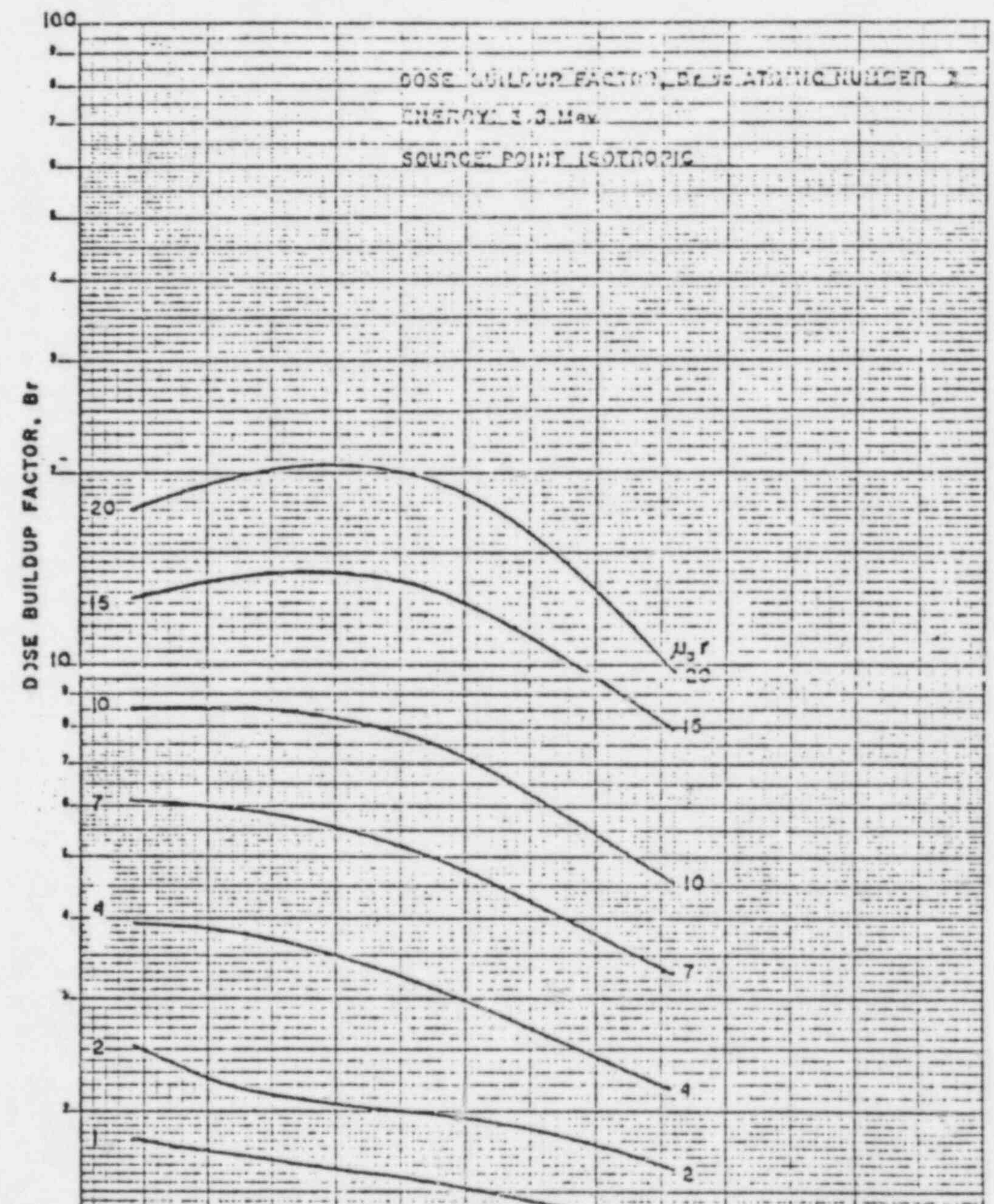

TOTAL
vs ENERGY
U(92)
Pb(82)
U
Al
ENERGY (mev)



ISOTROPIC
DOSE BUILDUP FACTOR, B_r
100
1
20
40
60
80
100
Z
$\mu_o r=20$
15
10
7
4
2
$\mu_o r=1$


DOSE BUILDUP FACTOR, Br vs ATOMIC NUMBER Z
ENERGY 1.0 Mev
SOURCE: POINT ISOTROPIC
DOSE BUILDUP FACTOR, Br
100
10
1
20
40
60
80
100
Z
$\mu_o r = 20$
15
10
7
4
2
$\mu_o r = 1$


DOSE BUILDUP FACTOR, Br
100
10
1
20
15
10
7
4
2
1
$\mu_0 r$
20
15
10
7
4
2
$\mu_0 r = 1$
1
20
40
60
80
100
Z



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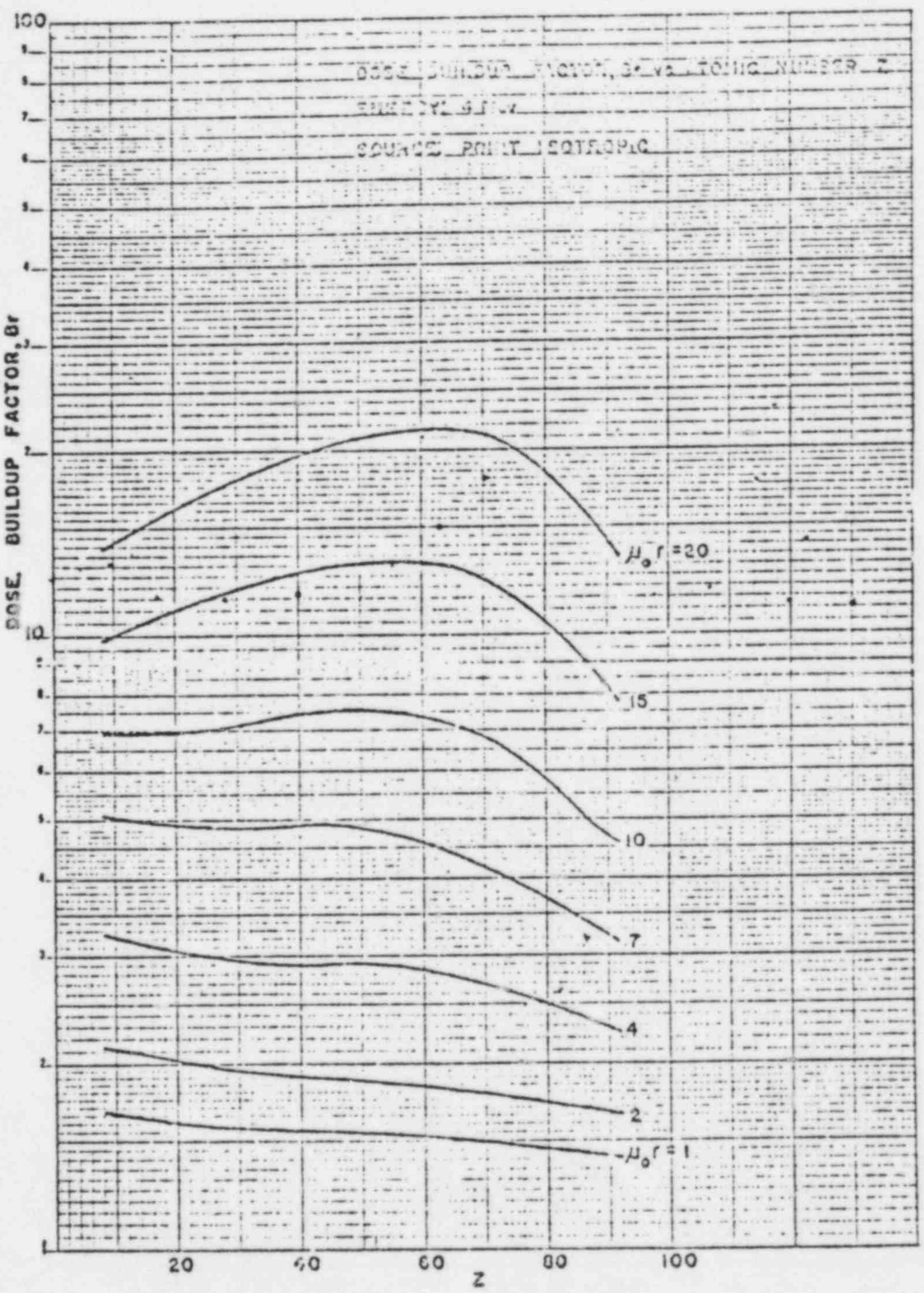
SOURCE: POINT ISOTROPIC
DOSE BUILDUP FACTOR, Br
Z
100
10
1
20
40
60
80
100
$\mu_0 r = 20$
15
10
7
4
2
$\mu_0 r = 1$

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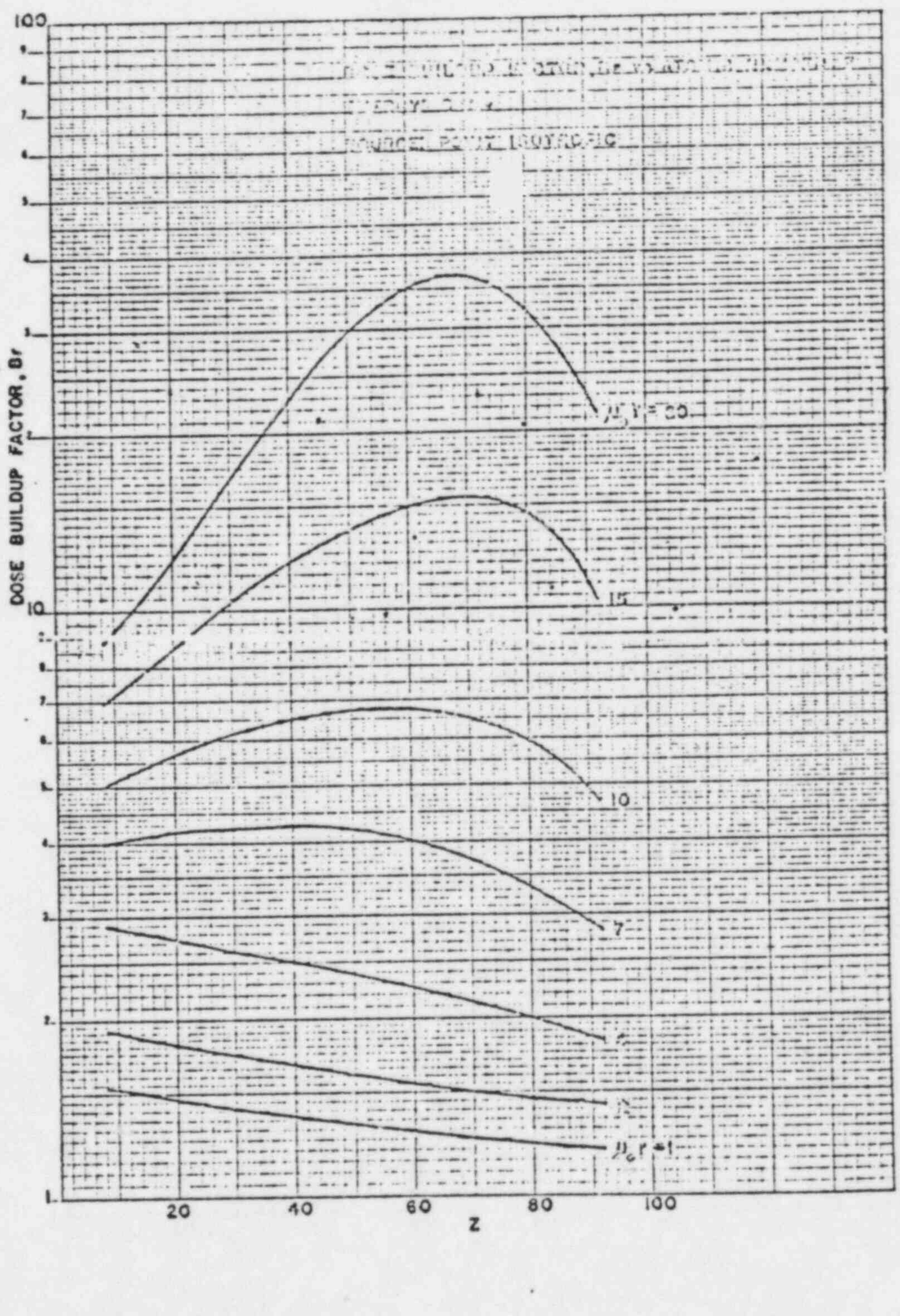
DOSE BUILDUP FACTOR, B_r
Z
$\mu_0 r = 20$
15
10
7
$\mu_0 r = 1$
100
10
1
20
40
60
80
100

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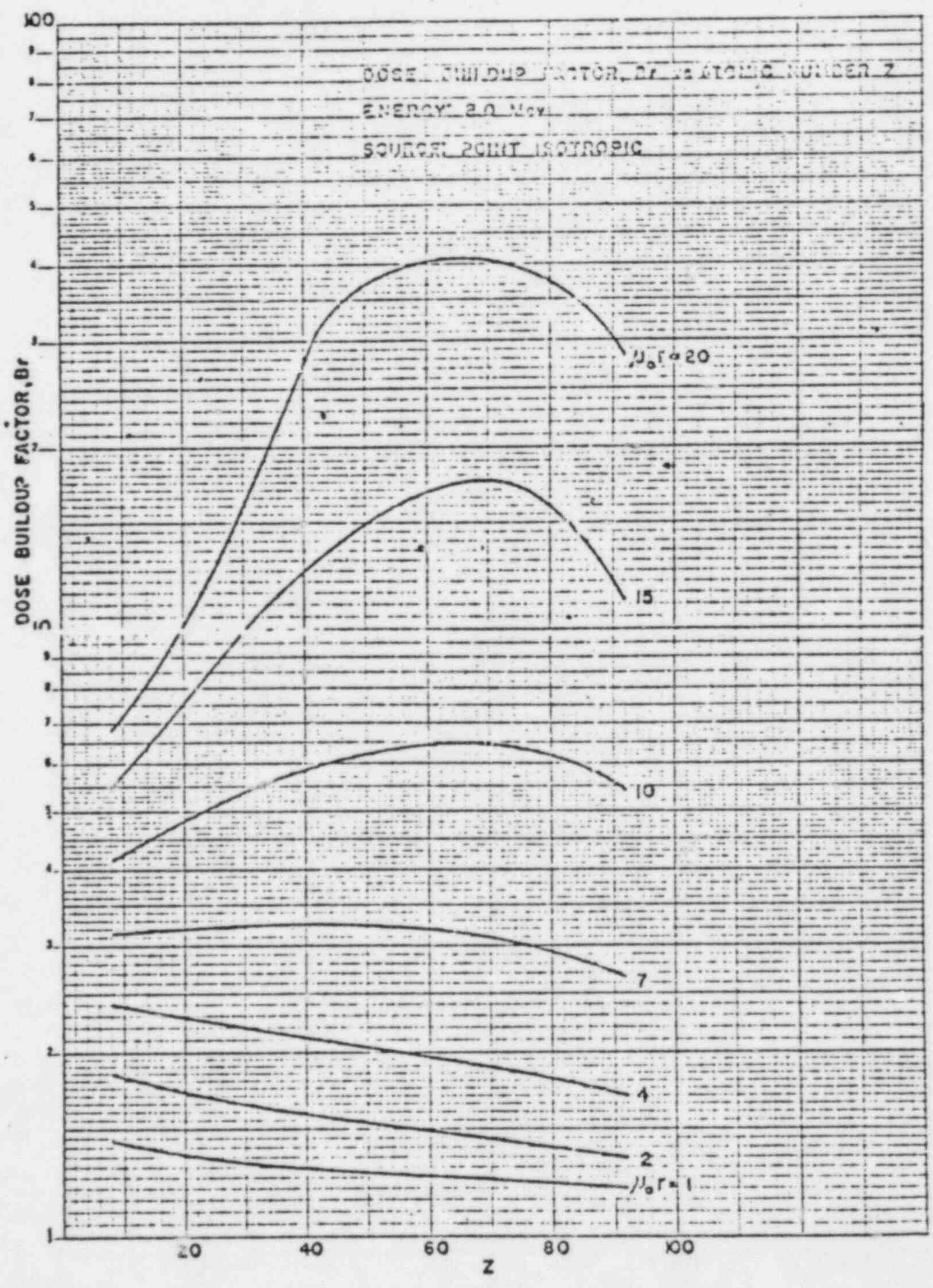
DOSE BUILDUP FACTOR, Br vs ATOMIC NUMBER Z
ENERGY: 2.0 Mev
SOURCE: POINT ISOTROPIC
DOSE BUILDUP FACTOR, Br
$\mu_0 r = 20$
15
10
7
4
2
$\mu_0 r = 1$
20
40
60
80
100
Z

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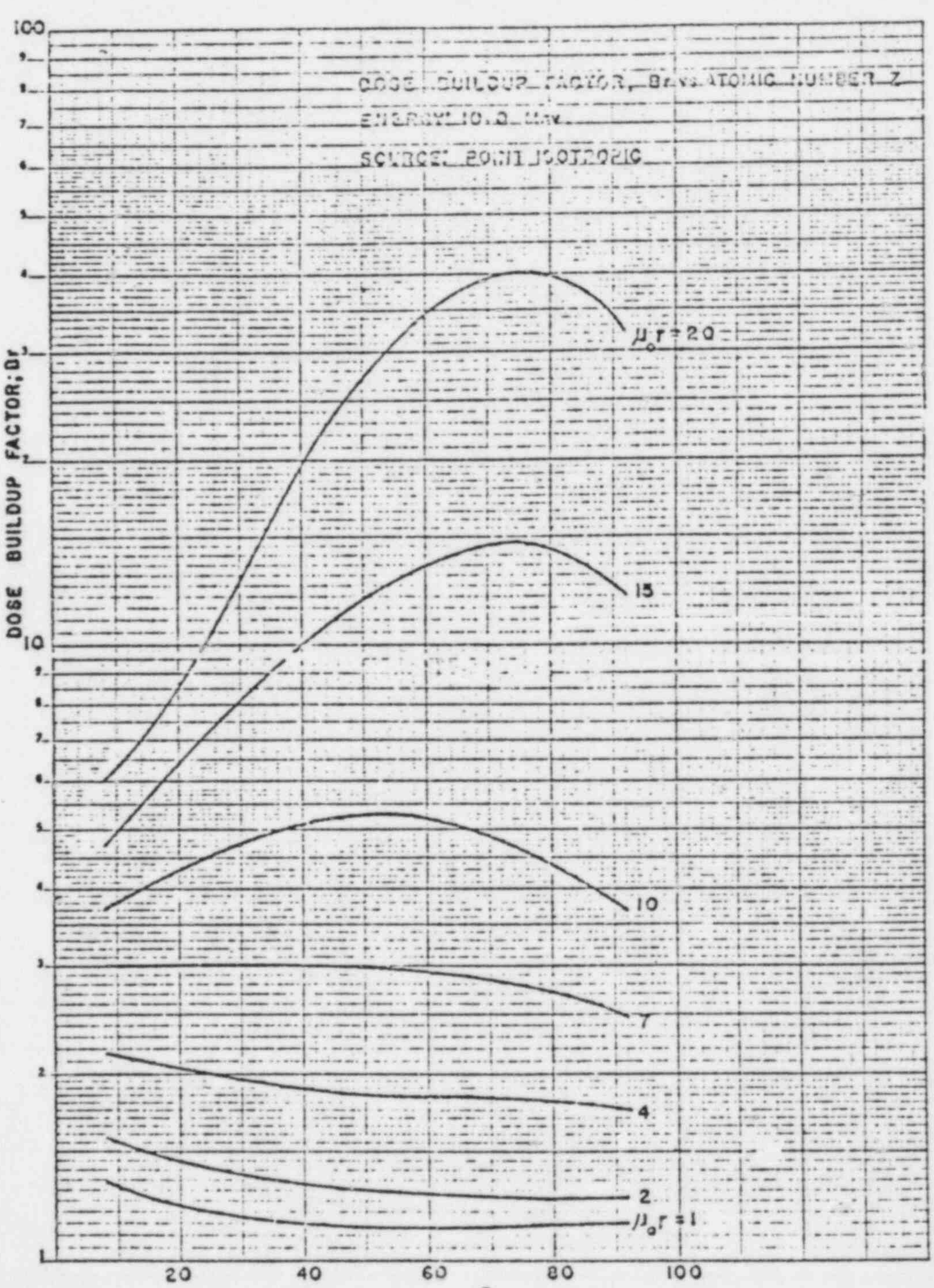
DOSE BUILDUP FACTOR, B vs ATOMIC NUMBER Z
ENERGY: 10.0 Mev
SOURCE: POINT ISOTROPIC
DOSE BUILDUP FACTOR; Dr
Z
100
10
1
20
40
60
80
100
$\mu_0 r = 20$
15
10
7
4
2
$\mu_0 r = 1$

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5. Range Of "γ" Photons

(1) When traveling through matter gamma photons lose their intensity exponentially with distance. Therefore, in theory, the intensity never reaches zero but approaches it asymptotically. Therefore range is meaningless.

(2) In practice one thinks of a finite amount of shielding material which would be required to reduce the intensity to as low as possible.

(3) Listed below is the tenth-value thickness of various materials. This is the amount of material required to reduce the intensity of the gamma by one tenth.

D. *Tenth - Value thickness (inches)

Energy	Air (Miles)	Water	Concrete	Iron	Lead
1	0.18	13.0	6.1	1.92	1.15
2	0.25	18.5	8.7	2.70	1.75
5	0.41	30.0	13.2	3.68	1.80

For rule - of thumb memorization the following should be committed to memory.

" TO be given "

E. Shielding Equations

1. Gamma Ray Flux $\emptyset_0$: The number of gamma rays passing through a unit area per second.

Gamma dose rate at a particular location is dependent upon the

1. Gamma Ray Flux $\emptyset_G$: (Con't)

gamma ray energy and the gamma ray flux.

2. <u>Point Source</u> : is a source which will emit gamma photons uniformly in all directions.

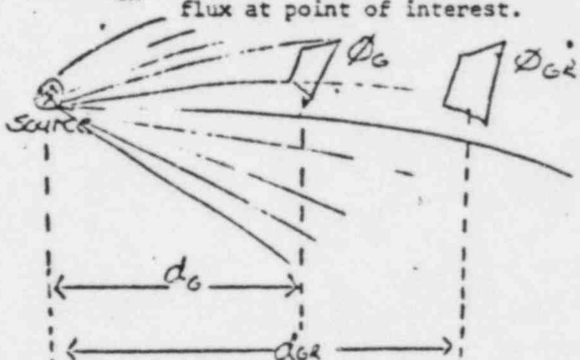
$$\emptyset_{GR} = \emptyset_G \left[\frac{d_G^2}{d_{GR}^2} \right]$$

where : $\emptyset_{GR}$ = gamma flux at reference point

$\emptyset_G$ = gamma flux at point of interest

d_G = distance between source location and reference point.

d_{GR} = distance between source location and gamma flux at point of interest.



3. <u>Parallel or Collimated beam</u> : Indicates very little spreading of the beam.

$$\emptyset_{GR} = \emptyset_G e^{-ux}$$

where : $\emptyset_G$ = gamma flux at reference point.

$\emptyset_{GR}$ = gamma flux at point of interest.

U = Linear attenuation coefficient (thickness $^{-1}$)

X = shield thickness.

bombarding particle a + x target → heavy y + b light product

X(a, b)Y ← short hand form

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NUCLEAR PHYSICS

WEEKLY EXAM 2

4/11/74

1\. A. $_{15}P^{31} + \underline{_{1}d^{2}} \xrightarrow{[_{16}S^{33}]^{*}} {_{15}P^{32}} + {_{1}H^{1}}$ (2 pts)

B. $_{28}Ni^{58} + \underline{_{1}p^{1}} \xrightarrow{[_{29}Cu^{59}]^{*}} {_{29}Cu^{58}} + {_{0}n^{1}}$ (2 pts)

C. $_{11}Na^{23} + \underline{_{2}\alpha^{4}} \xrightarrow{[_{13}Al^{27}]^{*}} {_{13}Al^{26}} + {_{0}n^{1}}$ (2 pts)

D. $_{15}P^{30} + \underline{_{0}\gamma^{0}} \xrightarrow{[_{15}P^{30}]^{*}} {_{15}P^{29}} + {_{0}n^{1}}$ (2 pts)

2\. Why are nuclear reactions harder to produce than atomic reactions? (2 reasons) −2 (6 pts)

a) Due to very strong, short range nuclear attractive forces, in the neucleus of an atom that have to be over come. b) Changes in mass-energy of a nuclei in forming it to fissioning it, that need to be overcome.

3\. Is the following reaction exothermic or endothermic, and by what amount of energy? (10 pts)

Ar^{40} (α, n)Ca^{43} on yellow sheet

4\. What two main elements make up the TMI source? Write down the pertinent equations. (5 pts)

a) americium & beryllium

$_{95}Am \longrightarrow {_{96}Ce} + \beta^{-}$, $_{96}Ce \longrightarrow {_{94}Pu} + \alpha$

$_{4}Be^{9} + \alpha \longrightarrow [_{6}He^{13}]^{*} \longrightarrow {_{3}He^{4}} + {_{0}n^{1}}$

5\. What is the difference between σ and Σ? (5 pts)

σ is the probability of a reaction occuring per atom, in cm^2. Σ is the probability of a reaction per cubic cm of a material, in cm^{-1}.

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6. Find the macroscopic cross section of scattering for Be_2C (10 pts)

Given: $Be_2^9C^{12}$

density = 1.9 $\frac{\text{grams}}{\text{cc}}$

$\sigma_s(Be)$ = 6.06 barns

$\sigma_s(C)$ = 4.78 barns

$\Sigma = N\sigma_s = \frac{\rho N_0}{m}\sigma_s$

(on yellow sheet)

7. The neutron energy range from 1eV to 10 keV is generally referred to as -2 (2 pts)
Resonance energy Range.

8. Why are neutrons the best projectiles for nuclear reactions? Due to No (5 pts)
Charge and fairly large mass (for a particle) the neutron can be slowed down from higher energys without being lost and directed in use for fissioning materials in which not only large amount of energy is Released but so are more neutrons per fissioning of a given material.

9. If a .032" cadmium sheet is placed in a neutron beam containing both thermal and fast neutrons. It is noted that there are essentially no thermal neutrons in the exit beam, while the fast component is unchanged. Why or explain. -2 (7 pts)

Cadium has an extremly high σ_a (thermal neutron absorbtion cross section) for Thermal neutrons (2460 b), and Effectivly none for fast neutrons. So the fast neutrons are not absorbed in the Cadium, while mostly all of the Thermal neutrons are, in passing through.

10. What is the flux for 10 $\frac{\text{neutrons}}{\text{cc}}$ of thermal neutrons? (5 pts)

$$\phi = nv = 10\,\frac{\text{neutrons}}{\text{cc}} \times 2200\,\frac{\text{m}}{\text{sec}} \times 100\,\frac{\text{cm}}{\text{m}}$$

$$= \underline{2.2 \times 10^6\ \frac{\text{neutrons}}{\text{cm}^2\text{-sec}}}$$

4-9

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11. What is the reaction rate for the flux of problem (#10) interacting with the Be_2C of problem (#6)? Assume that Σ_a for Be_2C is negligible. (≈ 0) (10 pts)

$$AR = \Sigma_s \phi$$

$$= 2.2 \times 10^6 \frac{n}{cm^2 sec} \times .664 \frac{}{cm} = 1.46 \times 10^6 \frac{n}{cm^3\text{-}sec}$$

12. TRUE - FALSE

A. Kinetic energy is conserved in elastic collisions? *True* (2 pts)

B. Inelastic collisions occur mainly with heavy nuclei and high energy neutrons. *False* (2 pts)

C. Most charged-particle emissions occur from nuclear reactions with slow neutrons. *True* (2 pts)

D. Because light nuclei cause elastic collisions with neutrons, they would be good moderators. *False* (2 pts)

E. The thermal region obeys the $1/v^3$ law. *False* (2 pts)

13. Given: Power = 25 $\frac{\text{Watts}}{\text{cc}}$

$\Sigma_f = .033\ cm^{-1}$

Conversion = 1 watt = 3.1×10^{10} $\frac{\text{fissions}}{\text{sec.}}$

Find the neutron flux

$$\boxed{5.7 \times 10^{13} \frac{n}{cm^2 sec}} = \left(2.43 \frac{neutrons}{fission}\right)\left(2.34 \times 10^{13} \frac{fissions}{cm^3\text{-}sec}\right)$$

(10 pts)

$$\phi_n = \frac{25 \frac{W}{cm^3} \times 3.1 \times 10^{10} \frac{fissions}{sec\text{-}watt}}{.033\ cm^{-1}} = \frac{7.74 \times 10^{11}}{3.3 \times 10^{-2}} = 2.345 \times 10^{13} \frac{fissions}{cm^2\text{-}sec}$$

14. What are the 3 regions for variations of cross sections with neutron energy? (7 pts)

~~Slow~~ thermal Neutron energy region < 1ev

Resonance Neutron energy region 1ev ↔ 10kev

Fast neutron energy region > 10kev

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3. $Ar^{40} = 39.96238$ amu $\quad$ $Ca^{43} = 42.95878$ amu

$\alpha = 4.00260$ amu $\quad$ $n = 1.00866$ amu

43.96498 amu $\quad$ 43.96744

4. endothermic (Energy added)

Δ mass $= 43.96744$

43.96498

$.00246$ amu

$B.E. = (931 \text{ mev}/\text{amu})(.00246 \text{ amu}) = \underline{2.295 \text{ mev}}$

9.31×10^{2} $\quad$ 2.46×10^{-3}

6. $\Sigma = g\rho \frac{N_0}{m} (2\sigma_{s(O_2)} + \sigma_{s(C)})$ $\quad$ $m = 2(4) + 12$

$= (1)(1.9)\left(\frac{6.02 \times 10^{23}}{30}\right)\left(2[6.06 \times 10^{-24}]\right)$ $\quad$ $= 18 + 12$

$= 30$ g/amu

$1 \text{ barn} = 10^{-24} \text{ cm}$ $\quad$ 4.78×10^{-24}

$= (1.9)(2.066 \times 10^{22})(12.12 \times 10^{-24} + 4.78 \times 10^{-24})$

$= (3.925 \times 10^{22})(1.690 \times 10^{-23})$

$= 6.64 \times 10^{-1} / \text{cm} = \underline{.664 \text{ cm}^{-1}}$

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3. Fission : The process inwhich the nucleus of a fissionable material such as U-235 absorbs a neutron and splits into two large fragments.

C. Cross Section : Is a measure of the probability that a nucleus will undergo a nuclear reaction. The greater the cross section of a nucleus, the greater the probability that a neutron will react with that nucleus.

1. Microscopic Cross Section (σ) : is the probability of an interaction with a single nucleus and has units of cm^2.

2. Macroscopic Cross Section (Σ) : is the probability of an interaction to occur in a cubic centimeter of target material.

$$\Sigma = N\sigma$$

where: Σ = macroscopic cross section (cm^{-1}).

N = target atom density (atom/cm^3)

σ = microscopic cross section (cm^2)

Density

$N = \rho \frac{N_0}{M}$ ← 6.02 × 10²³ atoms/gram ← (grams/gram) (found by experimental ways) Atomic mass No. Ar. 16. A.W.

$\Sigma = \frac{atom}{cm^3} \times cm^2 = cm^{-1}$

Percent abundance of isotope $\Sigma = f \rho \frac{N_0}{m} \sigma$ $N = f \rho \frac{N_0}{m}$

3. Total Macroscopic Cross Section (Σ_t)

$$\Sigma_t = \Sigma_s + \Sigma_{n,\gamma} + \Sigma_{cpc} + \Sigma_f$$

$\Sigma_{tot} = \Sigma_s + \Sigma_a$

$\Sigma_a = \Sigma_{n,\gamma} + \Sigma_{cpc} + \Sigma_f$

$\Sigma_s = \Sigma_e + \Sigma_{ie}$

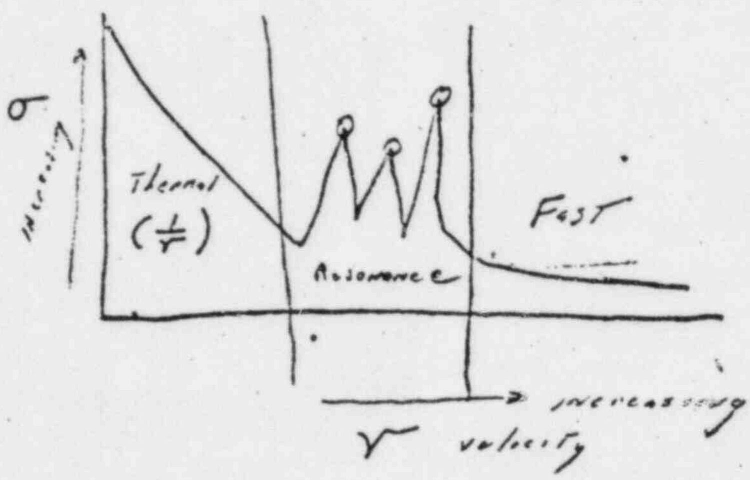
where: Σ_s = Macroscopic Cross Section for scattering

$\Sigma_{n,\gamma}$ = Macroscopic Cross Section for radioactive capture

Σ_{cpc} = Macroscopic Cross Section for charged particle emission

a compound chemically going to produce a 100% abundance (F) and a mixture would have to figure in individual abundances (F). 1 Barn = 10^{-24} cm^2

Crosssection Vrs Energy



$v = D/t$ distance time

4. Buildup Factor : Factor used to account for in - scattering.

$$\phi_{GR} = \phi_0 B e^{-\mu x}$$

where : B = buildup factor (Unitless)

5. Shielded Point Source

$$\phi_{GR} = (\phi_R)\left(\frac{d_R^2}{d_{GR}^2}\right)(B)(e^{-\mu x})$$

IV. Equivalence of Mass and Energy

A. Einstein's equation :

$E = MC^2$

where : M = mass of body

C = speed of light

E = equivalent energy of the body of mass M.

B. AMU vrs Mev

1AMU = 931Mev

C. Nuclear Forces: Short range attractive forces acting between nuclear within an atoms nucleus.

D. Nuclear Stability : is determined by the relative number of neutrons and protons in the nucleus.

1. In lighter elements, stability is achieved when the number of neutrons and protons in the nucleus are equal.
2. In heavier elements, it requires more and more neutrons in the nucleus to obtain stability.
3. In heavier elements :

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3. Heavier elements : (con't)

   (a) If excess neutrons, stability is achieved usually by beta decay.

   (b) If excess protons, stability is achieved usually by positron (B^+) decay or by orbital electron capture.

E. Energy and Mass Considerations

   1. Atomic particles of atoms if assembled into an atom will weigh less than the sum of their individual weights. The difference in weight is called "mass defect".

      (a) Mass defect : Represents the amount of mass which must be lost at the instant a atom is created.

V. Nuclear Reactions

A. Types of reactions

   1. Thermonuclear reactions : occur when a temperature of several million degrees is achieved and nuclei have enough kinetic energy to overcome their mutual electrostatic repulsion.

   2. Acceleration of charged particles : Protons, alpha particles, electrons, can be accelerated in atom smashers until they have enough kinetic energy to cause a nuclear reaction when they hit a target nucleus.

   3. Nuclear reactions : reaction involves a neutron interaction with fissionable material.

   4. Radioactive sources : reactions involve the exposure of a substance to radioactivity from a radioactive source.

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B. Transmutition Process

1. Target nucleus + incident particle $\longrightarrow$ compound nucleus

2. Compound nucleus $\longrightarrow$ Product nucleus + ejected particle

C. Atomic Projectiles

1. Types :

Particle	Symbol
(a) Proton	P or $_1H^1$
(b) Denteron	d or $_1H^2$ or $_1d^2$
(c) Alpha particle	α or $_2He^4$
(d) Neutron	n or $_0n^1$
(e) Gamma photon	γ or $_0\gamma^0$

D. Protor Bombardment $_9F^{19}(P,\alpha)_8O^{16}$

ex.

$$_9F^{19} + {}_1H^1 \longrightarrow [{}_{10}Ne^{20}] \longrightarrow {}_8O^{16} + {}_2He^4$$

E. Alpha Particle Bombardment $_4Be^9(\alpha, n)_6C^{12}$

$$_4Be^9 + {}_2He^4 \longrightarrow {}_6C^{12} + {}_0n^1$$

F. Denteron Bombardment $_6C^{12}(D^2, n)_7N^{13}$

$$_6C^{12} + {}_1H^2 \longrightarrow {}_7N^{13} + {}_0n^1$$

endothermic - energy added to
System initially to cause
~~or start~~ Reaction.

exothermic ~~- energy~~ given off during
initial reaction - occurrence.

energy given off from
system.

Am Be Cm - our source

photo nuclear (γ-ray)
Reaction

$^{9}_{4}Be + {}_{0}\gamma^{0} \longrightarrow 2\,{}_{2}\alpha^{4} + {}_{0}n^{1}$

Gamma of 1.6 mev needed or
>, to cause Reaction.

* Test Q.
We use the source we have in use because
due to its natural decay which produces
${}_{0}n^{1}$'s from no energy input.

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G. Gamma Photon Bombardment $_{12}Mg^{25}(\gamma, P)_{11}Na^{24}$

$$_{12}Mg^{25} + _0\gamma^0 \longrightarrow _{11}Na^{24} + _1H^1$$

H. Energy and Mass Considerations

1. Rule: " The sum total of mass and energy must remain constant throughout the process of a nuclear reaction.

2. Use of this rule can be applied in determining if a reaction will take place.

alpha } Two good
Gamma } sources
for Producing
Reactions to
emit Neutrons.

Know test

VI. Neutron Behavior

A. Neutron Sources

TMI source (ABC)

$$_{95}Am^{242} \xrightarrow{16h} \beta^- + _{96}Cm^{242}$$

$$_{96}Cm^{242} \xrightarrow{162.5d} _2\alpha^4 + _{94}Pu^{238}$$

$$_2\alpha^4 + _4Be^9 \longrightarrow _6C^{12} + n^1$$

Production of $_0n^1$ From our Source

B. Neutron Reactions

1. Radiative Capture (n, γ)

Are reaction inwhich the compound nucleus emits its excess energy as gamma radiation.

ex.

$$_{49}In^{115} + _0n^1 \longrightarrow [_{49}In^{116}] \longrightarrow _{49}In^{116} + _0\gamma^0$$

absorbtion Reactions

This Reaction used for Detection Mainly

2. Charged Particle emission [(n , p) , (n , α) , etc.] n, γ, n, f

Charged particle radiative capture fission

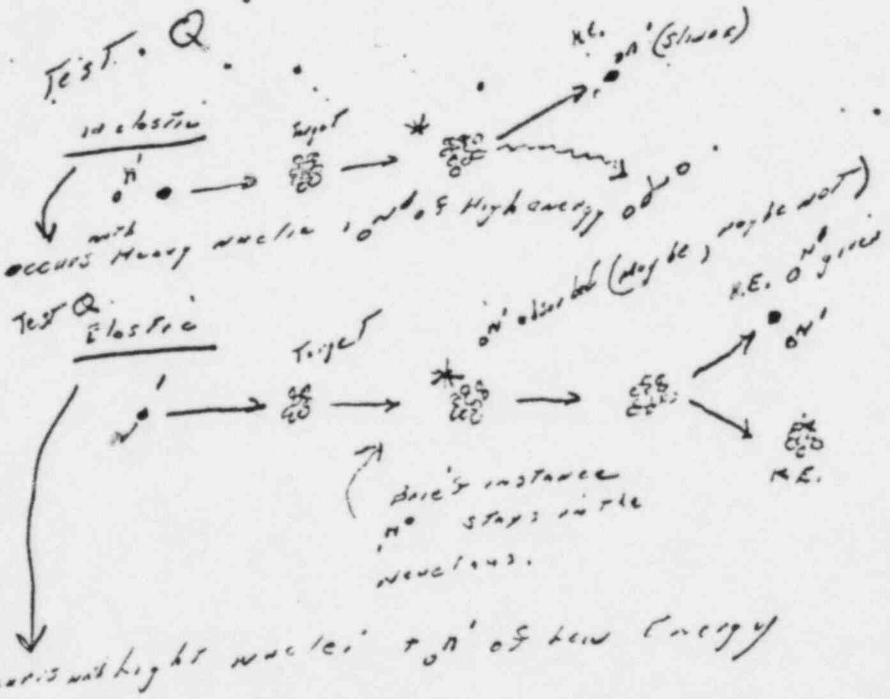
Are reactions inwhich the compound nucleus decays by the emission of a charged particle.

ex. $_5B^{10} + _0n^1 \longrightarrow _3Li^7 + _2He^4$

Scattering Reactions:

Inelastic	Elastic
momentum is conserved	momentum is conserved
K.E. is <u>not</u> conserved	K.E. is conserved

$m_1 v_1 = m_2 v_2$.



If atomic weight is close to light nuclei of a target, target moves in direction of incident $_0n^1$ and incident $_0n^1$ recoils in opp. direction.

If Atomic wt. very much larger than incident particle, incident particle recoils in opp. direction of target, with no loss of energy, target stays in place.

3. Macroscopic Cross Section : (Con't)

where: Σ_f = Macroscopic Cross Section for fission

4. <u>Total Microscopic Cross Section (σ_t)</u>

:

$$\sigma_t = \sigma_s + \sigma_{n,\gamma} + \sigma_{c.p.c.} + \sigma_f$$

D. <u>Cross Section Vrs Energy</u>

1. The magnitude of a cross section is also dependent upon the energy of the neutron.

2. For elements of mass number $>$ 100 their exist three important neutron energy regions for absorption cross sections.

(A.) <u>Low - energy region (<0.1ev)</u> Thermal

the absorption cross section decreases in magnitude as the neutron energy increases between 0 and 0.1ev.

$- \therefore \sigma_a \propto 1/v$

(B.) <u>Resonance region (0.1ev to 1kev)</u>

the absorption cross section within this region is characterized by the occurrence of peaks where the absorption cross section rises fairly sharply to high values for certain neutron energies and then falls again.

(C.) <u>High Energy Region (> 1 kev)</u> Fast

is characterized by a steady decrease in cross section as neutron energy increases beyond 1 kev. This region is not of much importance when compared to the other two regions and the role they play in nuclear physics.

E. Classification of Neutron Energies (NOTE: ranges vary for textbook to textbook)

Test Q

1. Slow Neutron : Neutrons of energy < 1ev.

2. Intermediate Neutrons : Neutrons of energy between 1ev to 10kev.

3. Fast Neutrons : Are Neutrons greater than 10kev.

4. Thermal Neutrons : are Neutrons which are in thermal equilibrium with their surroundings. at 72°F, v = 2200 meters/sec For Test Q.
.025ev

n_{th}

F. Neutron Flux (Ø)

1. Neutron Flux is defined as the total distance traveled by all of the neutrons in one cubic centimeter of material in one second.

$$\text{Ø} = nv$$

where : n = neutron density (n/cm^3)

v = neutron velocity (cm/sec)

$$\therefore \text{Ø} = \frac{n}{cm^3} \times \frac{cm}{sec} = n/cm^2 - sec$$

G. Neutron Reaction Rate

1. Reaction Rate = Ø Nσ or ΣØ
(reaction/cm^3 - sec)

where : Ø = neutron flux

N = target atom density

σ = microscopic cross section

Σ = macroscopic cross section

2. Reactions per second = $\phi N \sigma V$ or $\Sigma \phi V$

where : V = volume of target material

VII. Nuclear Fission

A. Fission Cross Section

1. For U - 235 the fission cross section follows a 1/v characteristic. Cross section varies from 10^3 barns at 10^{-2}ev to approximately .3 barns at 10^6ev.

2. Typical thermal cross section for Uranium isotopes and Pu-239 (barns)

Cross Section		U-235	U-238	U-233	Pu-239
σ_f	838	580	0	533	750
$\sigma(n,\gamma)$	15.437	107	2.75	52	315
σ_s	13.3	9.0	8.3	--	9.6
σ_{total}		696	11.05	--	1074.6

T-P

B. Fertile Material : Non-fissionable material to thermal neutrons but which can be made into fissionable material by neutron bombardment.

Be able to use σ vrs. Energy Curve in Relation to σ_f to explain Fertile Material non fissioning with thermal neutrons

1. U-238 to Pu-239

$$_{92}U^{238} + {}_0n^1 \longrightarrow {}_{92}U^{239} + \gamma$$

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1. U-239 to Pu-239 (Con't)

$$_{92}U^{239} \xrightarrow{23.5\ min} {}_{93}Np^{239} + B^- + \gamma$$

$$_{93}Np^{239} \xrightarrow{2.3\ d} {}_{94}Pu^{239} + B^- + \gamma$$

2. Th-232 to U-233

$$_{90}Th^{232} + {}_{0}n^{1} \longrightarrow {}_{90}Tn^{233} + \gamma$$

$$_{90}Th^{233} \xrightarrow{23.3\ min} {}_{91}Pa^{233} + B^- + \gamma$$

$$_{91}Pa^{233} \xrightarrow{27.4\ d} {}_{92}U^{233} + B^- + \gamma$$

C. Fission Reactions and Energy Release

1. Typical Process :

$$_{92}U^{235} + {}_{0}n^{1} \longrightarrow {}_{z}X^{a} + {}_{z}Y^{a} + \overset{\approx 2.46\ n\text{'s/fission}}{(?)}\ {}_{0}n^{1}$$

where : ${}_{z}X^{a}$ and ${}_{z}Y^{a}$ are two isotopes produced in the process and are called " Fission Products ".

Ex. $${}_{92}U^{235} + {}_{0}n^{1} \longrightarrow {}_{36}Kr^{95} + {}_{56}Ba^{139} + 2\,{}_{0}n^{1}$$

2. Mass defect

Before Fission		After Fission	
U-235	= 235.124 amu	Kr95	= 94.945 amu
n	= 1.009 amu	Ba-139	= 138.955 amu
Total	236.133 amu	2n	= 2.018 amu
		Total	235.918 amu

exothermic

(a) Mass defect = 235.918 - 236.133 = 0.215amu

(b) Energy release per fission = $\frac{931 \text{ mev}}{\text{amu}} \times 0.215$ = 198 mev

3. Distribution of Fission Energy

Instantaneous fission E

K.E. of fission fragments	168 ± 5 mev.
Instantaneous gamma photon energy	5 ± 1 mev.
K.E. of fission neutrons	5 ± 0.5 mev.
Beta particles from F.P.	7 ± 1 mev.
Gamma from F.P.	6 ± 1 mev.
Antinentrinos	10 mev.
Total	201 ± 6

Main Source

Lost

Delayed Energy Release (Speed)

allows for control of Reactors

Due to radiative capture, the γ's released give off ≈ 10 mev which can be added to the above to make up for lost neutrino energy.

(a) Rate of Production of Decay Heat Following shutdown from full power.

Time after Shutdown	% of Full Power
1 sec.	6.0

all the fission products fall into two broad groups, a "light" group, with mass numbers from 80 to 110, and a "heavy" group, with mass numbers from 125 to 155. The most probable type of fission, comprising nearly 6.4% of the total, gives products with mass numbers 95 and 139. Therefore, it is apparent that in the great majority of cases, the fission of U-235 is unsymmetrical. Pu-239 and U-233 have similar fission product yield curves which are also shown on the figure.

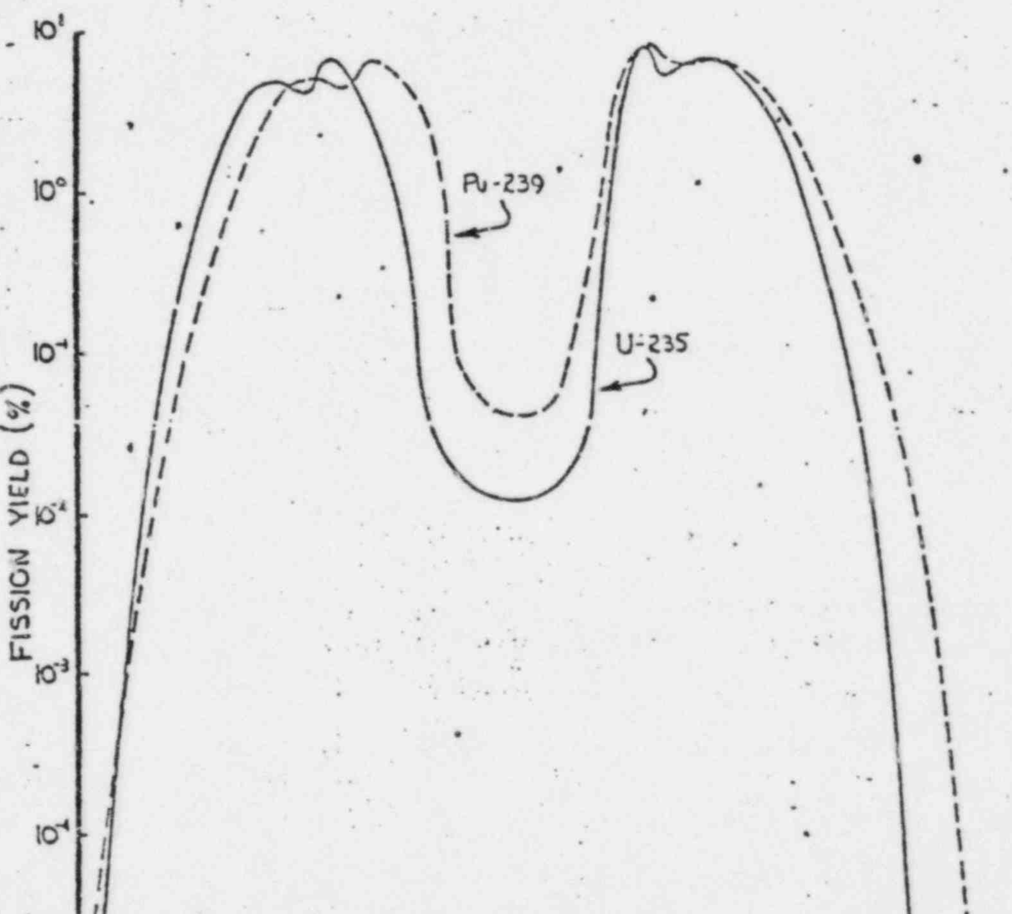
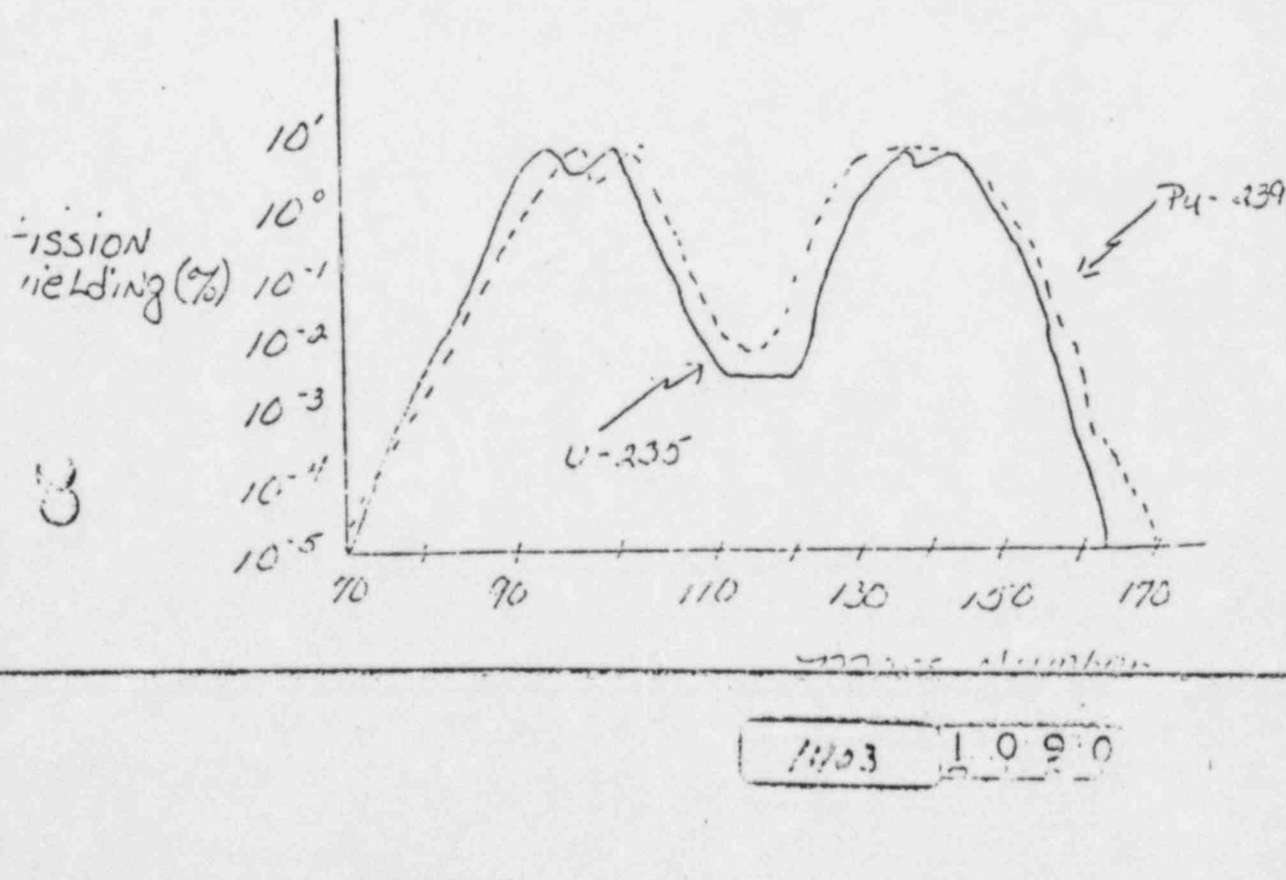


Figure 8-6: Thermal Fission Yield versus Mass Number

Time After Shutdown	% of Full Power
1 min.	4.5
30 min.	2.0
1 hr.	1.6
8 hr.	1.0
24 hr.	0.7
48 hr.	0.6

E. Fission Products

1. The thermal fission of U-235 has shown possibility of splitting up 40 different ways yielding over 80 primary fission products.
2. Range of fission products from A = > 2' to 160.
3. Most probable type of fission comprises approximately 6.4 % of the total produced and gives rise to products with mass numbers 95 and 139.
4. Yielding Curve :



F. Radioactivity of Fission Products

1. Nearly all fission products are radioactive and emit negative beta particles.

2. Fission products passes as excess of neutron. Therefore high n/P ratio. Ratio is so high that it usually requires three Beta emissions to reach stability.

3. Approximately 80 different isotopes produced in fission. Each produces an average of two additional radioisotopes. Result is a production of over 200 radioisotopes within a short time after fission.

4. Certain isotopes such as Xe-135 and Sm-149 are very important fission products and play a major role during reactor operation.

G. Fission Neutrons

1. Fission fragments eject their neutrons in aprox. 10^{-14} sec. after fission. These neutrons are called "prompt neutrons".

2. For U-235 prompt neutrons constitute aprox. 99.367% of all fission neutrons.

3. The remaining 0.64% of fission neutrons are called "delayed neutrons".

4. Delayed neutrons are produced from the decay of certain fission products.

5. For U-235 , U-233 and Pu-239 the delayed neutrons fall into 6 groups.

DELAYED NEUTRON GROUPS

Group	$T_{1/2}$ (sec)	Energy (mev)	Fraction of Total Fission Neutrons (%) U-233	U-235	Pu-239
1	55.6	0.25	0.022	[illegible]	0.007
2	22.7	0.46	0.078	[illegible]	[illegible]
3	[illegible]	0.41	0.006	[illegible]	[illegible]
4	[illegible]	0.45	[illegible]	0.25	[illegible]
5	0.6	0.42	[illegible]	[illegible]	[illegible]

(Con't)

Group	$T_{1/2}$ (sec)	Energy (mev)	Fraction of total Fission Neutrons (%) U-233	U-235	Pu-239
6	0.2	----	0.009	0.03	0.009
		Total	0.26	0.64	021

Delayed N: are used to control Reaction Rate of the Rx.

6. The 55.6 sec half-life is associated with the decay of Bromine-87. The 22.7 sec half-life is associated with Iodine-137.
7. Prompt neutrons have a continuous energy spectrum with a peak at 1mev and an average of 2mev.
8. The energies of prompt neutrons are considerably higher than the energies of the various delayed neutron groups.

H. Spontaneous Fission

1. All isotopes with atomic numbers equal to or greater than 90 are unstable and have a definite probability of undergoing spontaneous fission (fission without neutron bombardment)
2. For Uranium and Plutonium isotopes the probability is small $T_{1/2} \approx 10^{15} - 10^{17}$ years.

Isotope	Spontaneous Fission Half-Life (years	SPR per ton Of Material (fiss/sec-ton)
U-235	1.8×10^{17}	280
U-238	810×10^{15}	6300
Pu-239	5.5×10^{15}	9100

REACTOR OPERATIONS

I. <u>Introduction to Reactors</u>

A. <u>Chain Reactions</u>

1. Approximately 2.5 neutrons are born on the average in the fissioning of U-235.

2. If each fission causes at least one more fission, the process will be self sustaining and will constitute a "chain reation".

3. The problem is to assure that at least one of the 2.5 neutrons produced causes fission.

4. A reactor is a device which initiates a chain reaction and then controls it.

B. <u>Types of Fuel</u>

1. <u>Natural uranium</u> : Contains 99.3% of the isotope U-238 and 0.7% U-235. Natural uranium by itself cannot be used for the construction of chain reaction. Reason : It is impossible to establish a chain reaction with fast neutrons in natural uranium because, at the average fission neutron energy of 2mev, the fission cross section of U-238 is only about 0.5 barns as compared with the scattering cross section of about 6 barns.

(a) A chain reaction with thermal neutrons is also impossible because U-238 has very high resonance peaks in the intermediate range. Therefore, most neutrons would be absorbed before reaching thermal energies.

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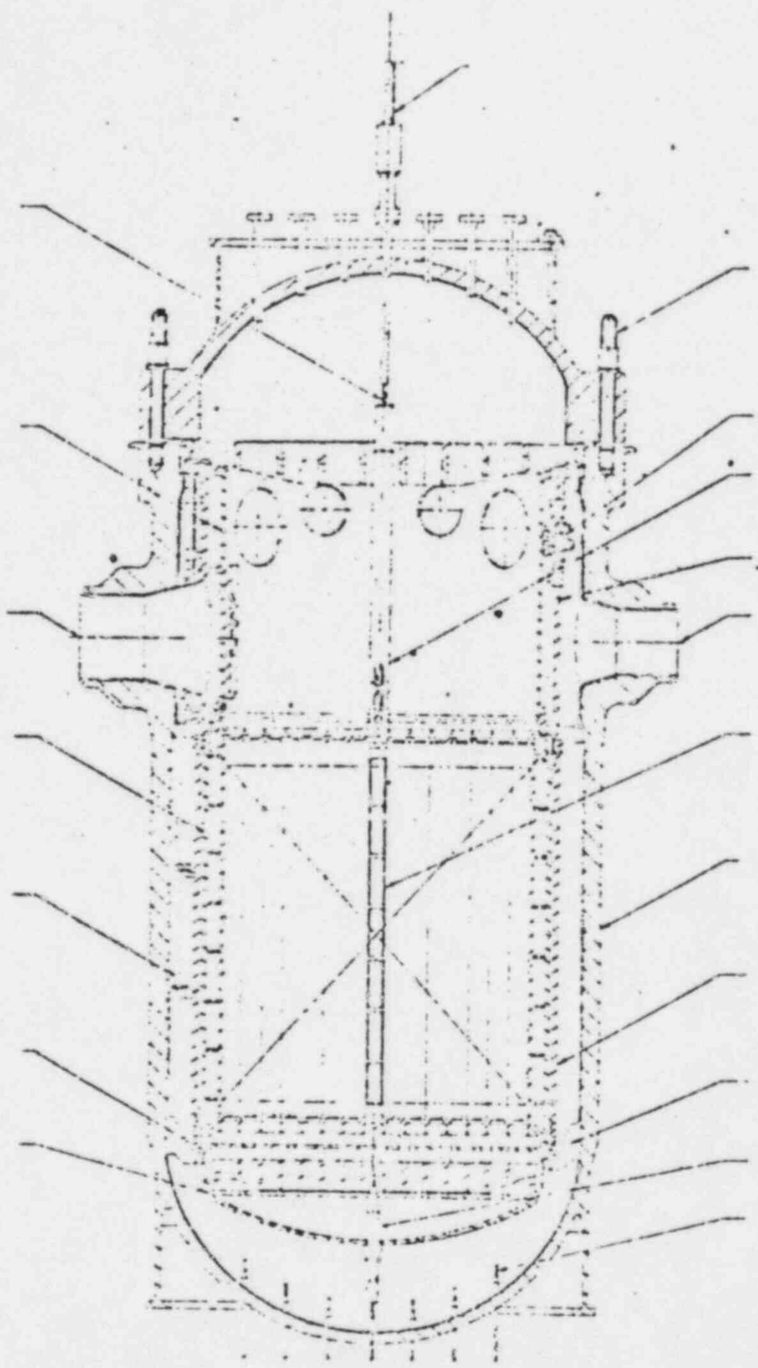

REACTOR VESSEL AND INTERNALS
GENERAL ARRANGEMENT

THREE MILE ISLAND NUCLEAR STATION UNIT 1

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FINAL EXAM [illegible]

NUCLEAR PHYS[illegible] $\frac{58}{60}$

PART I

931 Mev/amu

1. The half life of I^{135} is 6.7 hrs. What is the decay constant for I^{135}? (5 pts)

$$\lambda = \frac{.693}{T_{1/2}\ hrs.} = \frac{.693}{6.7\ hrs} = .1032\ hrs^{-1}\ \text{or}$$

$$1.032 \times 10^{-1} \frac{}{hrs} \times \frac{hrs}{3600\ sec} = .287 \times 10^{-4}\ sec^{-1}$$

2. If you have a 2 curie source of I^{135} at some time t, how many atoms of I^{135} were present at time t₀ which is 10 hrs. before time t? (5 pts)

$1c = 3.7 \times 10^{10}$ atoms/sec

$N = N_0 e^{-\lambda t}$

$R = N\lambda$

$N = R/\lambda$

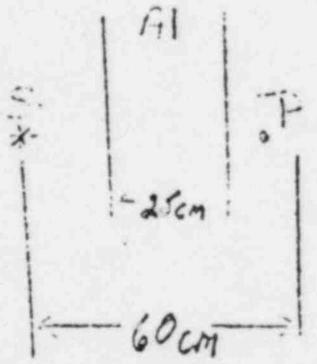
$$\frac{N}{e^{-\lambda t}} = N_0 = \frac{2(3.7 \times 10^{10})}{e^{-(.1032 \times 10\ hrs)}} = \frac{7.4 \times 10^{10}}{.356} = 2.08 \times 10^{11}\ atoms$$

$e^{-1.032}$

$.356$

3. Describe the phenomenon of a pair production. a γ photon of 1.02 mev or > energy, passes close to the nucleus of an atom (or into the nucleus), disappears, giving rise to a positron and electron, which collide with other oppositely charged particles & electron themselves producing two low energy gammas of .51 mev ea. or greater depending on initial energy of γ. (5 pts)

4. Consider the following sketch;



$\rho_{Al} = 2.7\ gm/cm^3$ (10 pts)

$D_T = (KE)(E)(\phi_u)B$ fixed

$$\phi_u = \frac{S_0 e^{-\mu x}}{4\pi x^2} \qquad S_0 = \frac{4\pi x^2 \phi_u}{e^{-\mu x}}$$

$$\phi_u = \frac{D_T}{(KE)(E)(B)}$$

$25\ mrem/hr \approx .025\ R/hr$

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

al $\frac{\mu}{\rho} = .04\ cm^2/gm$

$\mu = .04 \times 2.7 = .108$

The total dose rate at point P is 25 mrem/hr and the source emits 2 Mev gammas. How many γ/sec are being emitted from S_0?

$\mu x = 25 \times .108 = 2.6$ $\quad z = 13$ $\quad B \approx 3$

$$\phi_u = \frac{2.5 \times 10^{-2}\ R/hr}{(1.6 \times 10^{-6})(2)(3)} = 2.6 \times 10^{3}\ \frac{\gamma}{cm^2 \cdot sec}$$

9.6×10^{-6}

$$S_0 = \frac{4\pi(60)^2\, 2.6 \times 10^{3}}{e^{-2.6}} = \frac{12.64(3.6 \times 10^{3})(2.6 \times 10^{3})}{.074} \approx 16 \times 10^{8}\ \gamma/sec$$

or $1.6 \times 10^{9}\ \gamma/sec$

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Part I
Page 2

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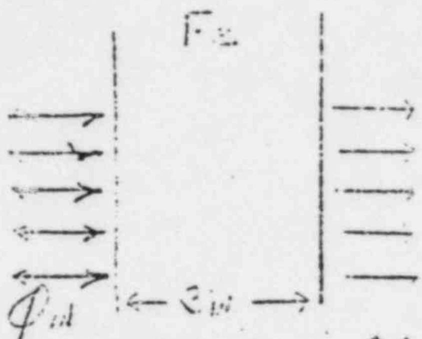
5. Calculate the flux entering the slab in the following sketch:



$\rho_{Fe} = 7.86\ gm/cm^3$

ϕ out = 5×10^{20} /cm^2sec; 4 mev (5 pts)

$2.54\,cm/in \times 3\,in = 7.62\,cm$

$Q_{out} = Q_{in}\, e^{-\mu x}$

Fe. $\frac{\mu}{\rho} = .032$ 3.2×10^{-2}

$\mu = \left(\frac{\mu}{\rho}\right)\rho = (.032)(7.86) = .252$ $.1$

$Q_{in} = \dfrac{5\times10^{20}}{e^{-(.252)(7.62)}} = \underline{3.4\times10^{21}}$ $\frac{\quad}{cm^2\text{-sec}}$

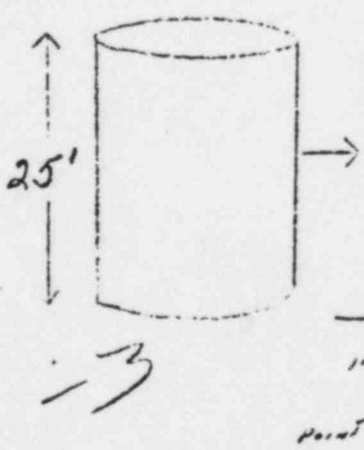
$.252\times10^{-1}$

$e^{-1.92}$

$.147$

1.47×10^{-1}

6. Consider the following sketch:

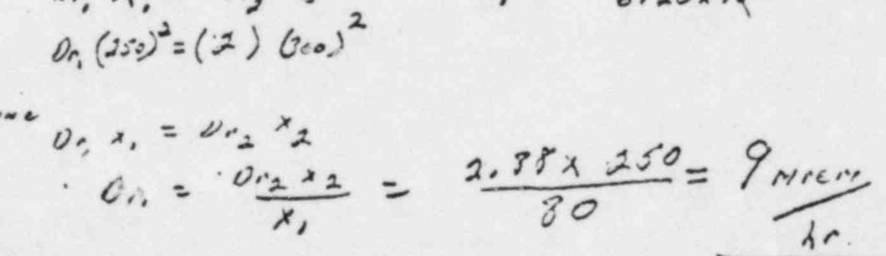


The dose rate 300 ft. from the side of this tank is measured to be 2 mrem/hr. Calculate the dose rate a) 80' from the tank and b) 200' from the tank.

(10 pts)

$3(25) = 75'$

$10(25) = 250'$



Point $Dr_1 X_1^2 = Dr_2 X_2^2$

$Dr_1 (250)^2 = (2)(300)^2$

$Dr_1 = \dfrac{18\times10^4}{6.25\times10^4} = 2.88$ mr/hr

Line $Dr_1 x_1 = Dr_2 x_2$

$Dr_1 = \dfrac{Dr_2 x_2}{x_1} = \dfrac{2.88\times250}{80} = 9$ $\underline{mrem/hr}$

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PART II

(-2)

1. How many $\frac{\text{Watts}}{\text{cc}}$ are produced from a flux of 6×10^{20} $\frac{\text{neutrons}}{\text{Sec-cm}^2}$?

$RR = \Sigma\phi$

Given: $\Sigma_f = .041 \text{ cm}^{-1}$

1 watt = 3.1×10^{10} $\frac{\text{Fissions}}{\text{Sec}}$

(1 neutron = 1 fission) (5 pts)

$RR = \Sigma_f \phi = .041 \text{cm}^{-1} \; 6 \times 10^{20} \frac{\text{neutrons}}{\text{cm}^2 \text{sec}}$

$= 24.6 \times 10^{18} \frac{n}{\text{cm}^3\text{sec}}$

$\frac{2.46 \times 10^{18}}{3.1 \times 10^{10}} = \underline{7.94 \times 10^{8} \frac{\text{watt}}{\text{cc}}}$ ✓

2. What is the macroscopic cross-section for fission of a 4.3% enriched U^{235} Fuel Assembly.
Given: Density of U^{235} = 18.7 $\frac{\text{grams}}{\text{cc}}$ $\Sigma_f = \frac{\%\rho N}{m}\sigma_f$ $1b = 10^{-24} \text{cm}^2$ (5pts) ✓

$\Sigma_f = (.043)(18.7)\left(\frac{6.02 \times 10^{23}}{235.0439}\right)(580) = \underline{1.195 \text{ cm}^{-1}}$

3. What is the binding energy per nucleon of O^{18}? (5pts) ✓

18.14916
17.999160 amu $\quad e - 5.5 \times 10^{-4}$ amu $\times 8 = 44.0 \times 10^{-4}$ amu

Mass Defect → $.15000$ amu $\quad p - 1.00727 \times 8 = 8.05816$ amu

B.E. ⟶ $931 \times .15 = 139.8$ mev $\quad n - 1.00866 \times 10 = 10.08660$ amu

18.14916 ← Total mass of O^{18}

B.E./nucleon ⟶ $\frac{139.8 \text{ mev}}{18 \text{ nucleons}} \approx 7.76$ mev/nucleon

4. Find the energy released from the following reaction:

$_{92}U^{235} + {_0n^1} \longrightarrow {_{42}Mo^{100}} + {_{50}Sn^{128}} + 8\,{_0n^1}$

Given: Sn^{128} = 127.85461 (5pts) ✓

1.00866
$\times 8$
8.06928

235.0439 amu
1.00866 amu
236.05256
-235.83236
Δ Mass → $.22020$

127.85461
99.90747
8.06928
235.83236

$B.E = (931)(.2202) \approx \underline{205.2 \text{ MeV}}$

$9.31 \times 10^2 \times 2.202 \times 10^{-1}$

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5. Finish the following:

A. $_{92}U^{235} + _{0}n^{1} \longrightarrow _{36}Kr^{95} + \underline{_{56}Ba^{138}} + 3_{0}n^{1}$ (2pt)

B. $\underline{_{0}N^{1}} \longrightarrow _{1}P^{1} + _{-1}e^{0} + \underline{\bar{\nu}}$ (2pt)

C. $_{18}Ar^{40} + \underline{_{2}He^{4}} \longrightarrow [_{2}Ca^{40}]^{*} \longrightarrow _{20}Ca^{43} + _{0}n^{1}$ (2pt)

D. Fission fragments usually decay by means of $\underline{\beta^{-}}$ decay. (2pt)

E. $_{90}Th^{232} + \underline{_{0}N^{1}} \longrightarrow \underline{Th^{233}} + \gamma$ (2pt)

$\xrightarrow{22.2\ min} {}_{91}Pa^{233} \underline{\qquad} + \beta^{-} + \gamma$

$\xrightarrow{27\ days} \underline{_{92}U^{233}} + \beta^{-} + \gamma$

6. Using the chart of the nuclides. Take the fip Xe^{142} to its final stable state. T1/2 are not necessary (5pts)

$_{54}Xe^{142} \xrightarrow{\beta^-} {}_{55}Cs^{142} \xrightarrow{\beta^-} {}_{56}Ba^{142} \xrightarrow{\beta^-} {}_{57}La^{142} \xrightarrow{\beta^-} {}_{58}Ce^{142}$

$T_{1/2} > 5 \times 10^{16}$ y (therefore stable)

Foust

7. True - Flase (If False "Why") (Each count 2 pts).

-2 A. Slow neutrons will cause no naturally occuring U^{233} to fission. True

B. The longest delayed neutron group has a half life of 55.6 seconds. True

C. On the average U^{235} releases about 3.4 neutrons per fission. False, it releases ≈ 2.5 neutrons per fission.

D. The Fips ranges are usually found near 95 for the light nuclei and 200 for the heavy fission product. False
1.39 for heavy fission products

E. Production of Plutonium 239 can be used to lengthen the life expectancy of low enriched fuels. True.

F. Nuclear reactions are harder to produce than atomic reaction because the repulsive force of the nucleus. True

G. Endothermic reactions create highly energetic anti ~~neutrons~~ netrinos which take the newly created energy from the system. False
exothermic reactions

H. Uranium 238 could be a good moderator, because heavy nuclei cause elastic collisions in which the neutrons slow down. False, first off u238 has a small σ_f causing loss of neutrons from the system and enters into inelastic collisions.

I. Most of the energy released by fission comes off as instantaneous and delayed gammas. False comes off as in KE of fission fragments mostly.

J. The region where high peaks are usually found is known as the resonance region. True

1103 1099

8. What is the main purpose for delayed neutrons being significant in reactors? ([illegible]pts)

Delayed neutrons Provide stability in Reactors in allowing its Reaction Rate to be Controled.

(12)

LECTURE : REACTOR OPERATIONS & KINETICS

DIFFERENT TYPES OF COMMERCIAL REACTORS

WATER REACTOR

1. PRESSURIZED
2. Boiling
3. Homogeneous
4. Heavy WATER

GAS-COOLED REACTOR

LIQUID-METAL REACTOR

PRESSURIZED WATER

ADVANTAGES

1. Common MATERIALS ARE USED (H_2O, STEEL, ETC)
2. MINIMIZES ADDITIONAL DEVELOPMENT OF EQUIPMENT (MOST OF IT ALREADY IN USE)
3. SEPARATE PRIMARY AND SECONDARY SYSTEMS MEANS NO RADIATION IN SECONDARY UNIT FOR EASIER ACCESSIBILITY
4. HIGHER CORE DENSITY ($\frac{MW}{FT^3}$) THAN BWR WHICH MEANS A SMALLER FUEL INVENTORY. IT OFF-BALANCES THAN GREATER ENRICHMENT NEEDED.

PWR DISADVANTAGES:

1. High Pressure Equipment NEEDED
2. HIGHER COST DUE TO BORIC ACID CONTROL AND STEAM GENERATOR
3. TO PREVENT BULK CORE BOILING A MAXIMUM COOLANT TEMP OF 600°F LIMITED
4. THIS CAUSES POORER CYCLE EFFICIENCY
5. THE HIGHER POWER DENSITY MEANS RATHER SENSITIVE SYSTEM TO COOLANT LOSS OR PUMP FAILURE.

Boiling WATER REACTOR

ARE DIRECT CYCLE PLANTS, WHICH MEANS NO STEAM GENERATOR. THIS MEANS LOWER PRESSURES NEEDED (≈ 1000 Psig). FORCED CIRCULATION USED

THE STEAM MUST BE DRIED AND RADIOACTIVITY CONTROL FOR THE TURBINE SYSTEM.

THE SHUTDOWN OF BWR's COMES DIRECTLY FROM CONTROL RODS only.

Advantages: 1. CONVENTIONAL MATERIALS -

2. Simple CYCLE

NO CHEMICAL SHIMS

DISADVANTAGES:

1. RADIOACTIVE MATERIALS REACH TURBINES. "SHIELDING REQUIRED & LIMITED ACCESS".
2. LOWER POWER DENSITIES
3.

<u>HOMOGENEOUS</u>

THE FUEL IS MIXED WITH THE MODERATOR. THERE ARE NO FUEL ELEMENTS AND NO CONTROL RODS (GENERALLY). AGAIN HEAT IS REMOVED FROM THE SYSTEM BY STEAM GENERATORS. STILL, THE FISSION PRODUCTS IN SOLUTION MAKE IT HIGH RADIOACTIVE. THE SYSTEM USES HIGH PRESSURE (AS PWR).

PRESENTLY, THERE ARE <u>NONE</u> IN USA; HOWEVER THERE HAVE BEEN 4 RESEARCH MODELS.

1. THE URANYL SULFATE SYSTEM (HRE-2) OPERATED AT 570°F AND 1800 PSIG.

ADVANTAGES:

1. MOBILITY OF FUEL
2. SIMPLE CORE DESIGN - HIGH POWER DENSITY
3. NO NEED FOR FUEL FABRICATION
4. EASE OF CONTROL
5. VERY SENSITIVE - TEMP. COEFF (FUEL ρ LOAD FOLLOWER)

DISADVANTAGES.

1. CORROSION
2. FLOW IMPORTANT (STAGNANT - OVERHEAT AREAS)
3. RADIATION HAZARD (LEAKS)
4. MUST PREVENT CRITICAL MASS FROM FORMING IN OTHER PARTS - OTHER THAN CORE.

1103 1 1 0 3

⑮

D_2O REACTOR

NATURAL URANIUM CAN BE USED AS A REACTOR FUEL. D_2O IS SUPERIOR MODERATOR (EXTREMELY LOW σ_A). IT'S COST, HOWEVER, IS THE DRAWBACK. → \$28/lb. ALSO TRITIUM IS PRODUCED DIRECTLY FROM THE D_2O AND IS A RADIATION HAZARD

(USE PICTURE)

GAS COOLED

He, CO_2

CYCLE EFFICIENCY > PWR, BWR

CYCLE EFFICIENCY ≅ FOSSIL

USE OF GRAPHITE MODERATOR

CAN BE ADOPTED FOR EITHER THERMAL OR FAST BREEDER PLANTS

LIQUID METAL (SODIUM)

THEY ALLOW HIGH OPERATING TEMPERATURES WITHOUT THE NEED FOR HIGH PRESSURES.

($T = 1300°F$ AND ONLY ~100 psig)

PLANT EFFICIENCIES ARE > 35%.

THEIR WIDESPREAD USE WILL BE AS FAST BREEDERS

1) $Na^{23} \longrightarrow Na^{24}$ (RADIOACTIVE MUST BE CONTROLLED)

2) SODIUM REACTS VIOLENTLY WITH H_2O.

NEED FOR INTERMEDIATE LOOP.

2. Enriched uranium : Uranium in which by artificial means the percentage or U-235 content is increased. Commonly used in power reactors today.

   * For TMI our fuel consists of low-enriched UO_2 pellets of cylindrical shape. Pellets are contained in hollow rods in each fuel assembly. Total number of assemblies is 177. Average enrichment is 2.62%.

C. Critical Mass

1. Some neutrons born from fission will escape or leak out of reactor. Some will be absorbed in non-fissionable material.
2. The larger the reactor, the smaller the fraction of neutrons which leak out.
3. By assembling a core until just enough fuel is added to overcome the percentage leakage and absorption, a chain reaction is sustained. The minimum amount of fuel is called a "critical mass".
4. When a reactor is just critical the production of neutrons through fission is exactly balanced by the loss of neutrons through leakage and absorption.

$\therefore$ Neutron Production = leakage + absorption

D. Reactor Control

1. For a reactor to operate at any appreciable power, more fuel must be added than just that to create a critical mass.
2. More fuel is necessary to overcome the effects of temperature, the buildup of fission products poisons (non-fissionable neutron absorbers) and fuel burnup.

3. To assemble this additional fuel initially would create a supercritical reactor. That is, a reactor that could be increasing power at a rate which would be uncontrollable. Therefore, some means of controlling the chain reaction is necessary.
4. Methods of Control
   (a) Poison control rods.
   (b) Soluble poison in coolant.

E. <u>Classification of Reactors</u>

1. <u>By physical arrangement of moderator and fuel</u>
   (a) Homogeneous Reactor : where both fuel and moderator are mixed together in a uniform mixture.
   (b) Hetergeneous Reactor : where fuel and moderator are separate bodies. TMI - type.
2. <u>By Type of Coolant used</u>
   (a) Water cooled (TMI - type)
   (b) Sodium cooled
   (c) Gas cooled
3. <u>Neutron Energy Spectrum</u>
   (a) Fast Reactors : rely on fast neutrons fissions.
   (b) Thermal Reactors (TMI -types) : rely on moderated neutrons (Thermal neutrons fissions).
4. <u>Use of fuel</u>
   (a) Breeder Reactor : Produces more fissile material than it utilizes.
   (b) Converter Reactor : Produces some new fissile material but less than that consumed. (TMI-types).
   (c) Burner Reactor : Produces no new fissile material.

F. Moderators

1. Recall: That natural Ur could not fission because as neutrons slowed down they were absorbed in U-238 resonance peaks, therefore no U-235 fission.

2. Natural Ur can create a chain reaction if mixed with certain materials which could rapidly slow down the high energy neutrons.

3. A substance which rapidly slows down high energy neutrons is called a "moderator".

4. Slowing down mechanism is a moderator is "scattering".

5. Characteristics of a good moderator are:

   (a) Allows the neutron to lose a large part of its energy in a single collision. Therefore dictates a light material.

   (b) Low absorption cross section

   (c) High scattering cross section

   (d) Can withstand high temperature enviroments.

   (e) Can withstand intense radiation.

6. Common moderators are:

   (a) Water (light) - (TMI) ← Good σ_s but also high σ_a used due to cost (cheap)

   (b) Heavy water ← High σ_s, small σ_a

   (Carbon) (c) Graphite ← Bulky and cooling problem

   (d) Beryllium ← very good But Cost's high

G. Reflector : A material which will reduce neutron leakage from a reactor core by reflecting neutrons back into the core.

1. Advantages :

   (1) Reduces critical mass

   (2) Lower fuel cost

   (3) Smaller reactor components

2. Characteristics of a good scatter : is that it have a high scattering cross section and a low absorption cross section.
3. Since its characteristics are about the same as a moderator, most reactors consider the moderator the reflector as well. (TMI)

H. Coolant :

1. Since reactors are used as heat sources, coolant is used to remove or transfer the heat.
2. Coolant Characteristics
   (a) Suitable heat transfer properties
   (b) Low absorption cross section
   (c) Compatable with reactor enviroment
3. Types of Coolants :
   (a) Light water (TMI)
   (b) Heavy water
   (c) Gases (helium, CO_2)
   (d) Liquid metals (Sodium)
   (e) Organic materials
   (f) Molten salts

I. Thermal Shield

1. A steel shield separating the reactor core and the reactor vessel.
2. Purpose : To protect the reactor vessel from possible damage from the heat liberated upon absorption of radiation. It is effective for absorbing γ photons and scattering fast neutrons. 5/7 e./

J. Fuel Rods

1. Usually UO_2 in modern reactors (discussed earlier).

K. Control Rods :

1. TMI : "See handout"

II. Multiplication and Reactivity Considerations in a water moderated Reactor

A. Multiplication Factor (K)

1. Multiplication Factor (K) definition: is the ratio of the number of neutrons in one generation to the number of neutrons in the previous generation.

$$K = \frac{\#\text{ of neutrons in the } (n + 1) \text{ generation}}{\#\text{ of neutrons in the nth generation}}$$

2. Neutron Generation definition : is the period of time between successive fissions.

3. When K = 1 : The number of neutrons in the (n + 1) generation equals the number of neutrons in the n^{th} generation. In this condition, the reactor is considered to be "just critical".

4. When K > 1 : The neutron population (or reactor power) will continue to increase with time. The Reactor is said to be "supercritical".

5. When K < 1 : The neutron population (or reactor power) will continue to decrease with time. The reactor is said to be "subcritical".

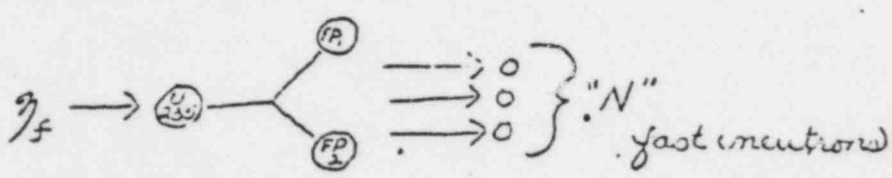
B. Neutron Cycle: is the sequence of major events which neutrons undergo from their "birth" through fission to their "death" through absorption

3. Neutron Cycle : (Con't)

in some material or leakage from the reactor.

1. Example Steps :

A. Consider "N" fast (2mev) released from fission



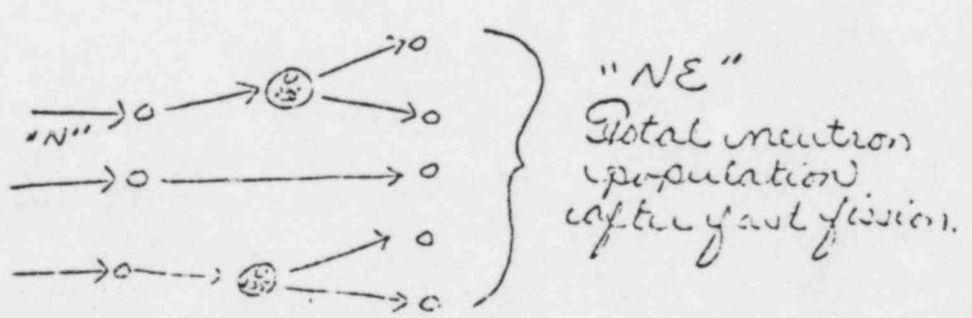
B. Fast Fission

(1) Since the fuel is general lumped together in the form of rods, there is a definite probability that some of the original "N" neutrons will strike other uranium atoms before they have a chance to reach the moderator and slow down.

(2) Some neutrons which strike U-238 will have energies above the threshold fission energy of 1mev and therefore cause fission.

(3) Also note that U-235 has a small fission cross section for fast neutrons however, some fissions will occur.

(4) Total result: Fast neutron population will increase.



(5) Let "E" equal probability of fast fission. "E" is called the "fast fission factor . Note: "E" is a value > 1.

Total fast neutron population = "NE".

C. Fast Leakage (1 - $\mathcal{L}_f$)

1. "NE" fast neutrons are ready to enter the moderator and begin to slow down.

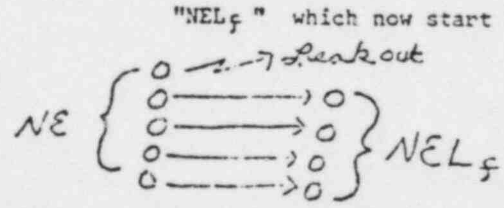
2. However, before they do there is some definite chance that some of theirs (particular those born near the edge of the core) with leak of the reactor and be lost.

3. The probability that a fast neutron will not leak out of the core is called "the fast non - leakage probability" (L_f). Therefore, the probability of leakage then equals $1-\mathcal{L}_f$.

4. L_f then represents the fraction of the total number of fast neutrons which remains in the core.

5. The total now remaining equals:

"NEL_f " which now start to slow down.



D. Resonance Escape Probability (P)

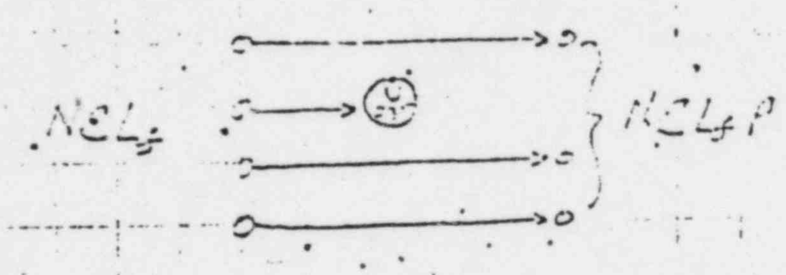
1. As the fast neutrons slow down, the encouncer U-238 absorption renosances.

2. The probability that a neutron will not be captured in the U-238 resonance escape probability (P).

3. "P" represents the fraction of the total number of neutrons which remains in the core.

4. Therefore, the total remaining in the core equals:

   "NEL_fP" which represents the total number of neutrons which reach thermal energies.



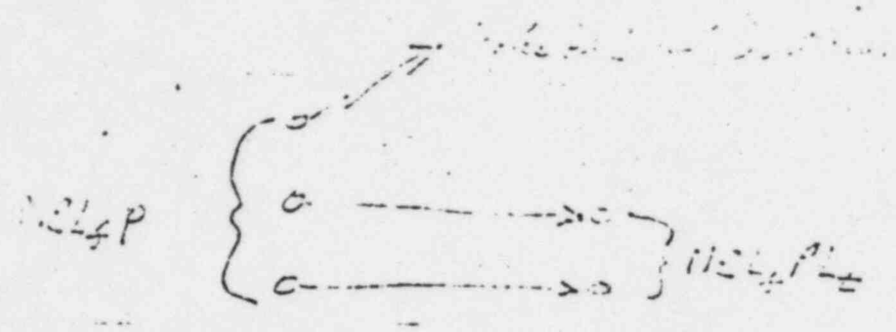
E. Thermal Leakage ($1-L_t$)

1. As the thermal neutrons move around in the reactor, a certain percentage of them will leak out while thermal.

2. The probability that a thermal neutron will <u>not</u> leak out of the core is called "the thermal non-leakage probability (L_t).

3. Therefore, the total number of thermal neutrons remaining in the core equals:

   "NEL_fPL_t" represents the total number of thermal neutrons which do not leak out from the core and will end up being absorbed in one of the various materials in the core.
   <u>Note</u>: The only neutrons which will be of any use in furthering the chain reaction will be those which are absorbed in fissionable materials.



F. Thermal utilization factor (f)

1. The probability that a thermal neutron will be absorbed in fissionable material as apposed to all other materials is termed "the thermal utilization factor "f".

2. Therefore, the total number of neutrons absorbed in fissionable material is:

$NEL_f PL_t f$

Note: Of the thermal neutrons absorbed in fissionable material, some will come fissions and some will be wasted in (n,γ) reactions.

G. Reproduction Factor (η)

1. The number of fission neutrons produced per thermal neutrons absorbed in fuel is called the reproduction factor "η".

2. Therefore, the total number of new fission neutrons equals:

$NEL_f PL_t \eta$ represents the number of fast neutrons available at the start of the next generation.

H. Multiplication Factor (K_{eff}) (Finite Reactor)

1. The ratio of the number of neutrons in the (n+1) generation to that

H. Multiplication Factor (Keff) (Con't)

in the n[th] generation is then:

$$K_{eff} = \frac{N E L_f P L_t \eta}{N} = E L_f P L_t \eta$$

Note : $K_{eff} = E L_f P L_t \eta$ is referred to as the "Six factor Formula".

2. Multiplication Factor (K_∞)

" Although leakage can never be completely eliminated in a real reactor, certain solutions to certain problems can be found conveniently by considering a reactor with no leakage. Such a reactor would have to be of infinite size. The multiplication factor for this type of reactor would be referred to as the "infinite multiplication factor (K_∞).

where:

$K_\infty = \eta E P f$ also referred to as the "four factor formula".

Note: $K_{eff} = K_\infty L_f L_t$

I. Six Factor Formula defined

$$K_{eff} = \frac{\text{\# of neutrons in any generation of a finite reactor}}{\text{Number of neutrons in the previous generation}}$$

$$K_\infty = \frac{\text{Number of neutrons in any generation of an infinite Reactor}}{\text{Number of neutrons in the previous generation}}$$

PREVIOUS
EXPOSURE
REFILMED

MICROFILM

Epi thermal - above-Therma lneutrons

1103 111F

H. Multiplication Factor (Keff) (Con't)

in the n^{th} generation is then:

$$Keff = \frac{N \epsilon L_f p L_t \eta}{N} = \epsilon L_f p L_t \eta$$

Note : $Keff = \epsilon L_f p L_t \eta$ is referred to as the "Six factor Formula".

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$$K_\infty = \frac{\text{Number of neutrons in any generation of an infinite Reactor}}{\text{Number of neutrons in the previous generation}}$$

1103

I (n) Life Cycle (Generation)

1. Birth from thermal fission (U-235)
2. Birth from fast fission (U-238)
3. Fast neutron leakage
4. U^{238} Resonance absorbtion
5. moderated (thermalized)
6. Thermal neutron leakage
7. Thermal neutron absorbtion (any thing)

<u>Generation</u> is period of time between successive fissions.

Multiplication Factor

$$K_{eff} = \frac{\text{Production}}{\text{absorbtion} + \text{Leakage}}$$

$$K_\infty = \frac{\text{Production}}{\text{absorbtion}}$$

$$K_\infty = \epsilon\, p\, f\, \eta$$

$$K_{eff} = \epsilon\, p\, f\, \eta\, L_f\, L_t$$

Average core Life (BOL)

BOL values for Unit 1 Rx.

η - 1.67
* ϵ - 1.25
p = .65
L_f = .996
L_t = .998
f = .80

(P.S.) Power Fractions For This Unit I Core

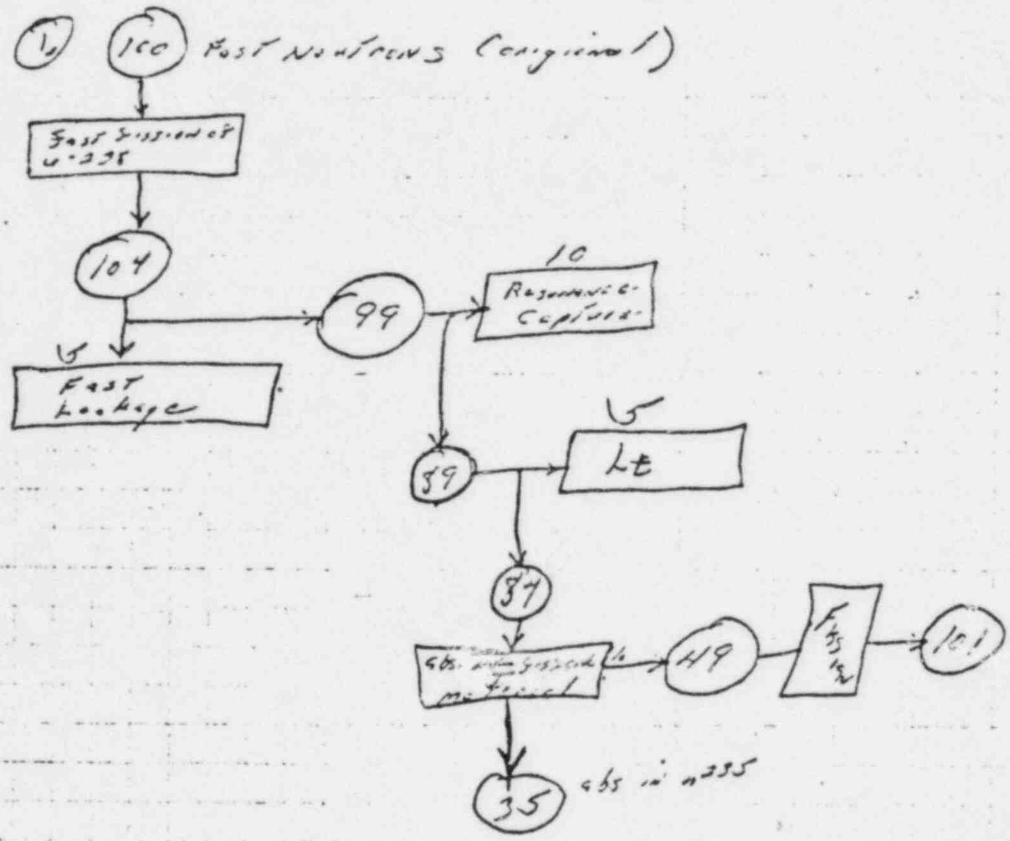
BOL		EOL	
U-235	P.S. .93	U-235	P.S. .49
U-238	P.S. .07	U-238	P.S. .07
		P-239	P.S. .38
		P-241	P.S. .055

* (TMI) we ~~[illegible]~~ are dependent more on fast fission than the High enriched core like the Navy.

P-240 is also produced over core life which captures Neutrons in the Resonance absorbtion region (which widens as core Temp. increases) as well as U-238.

Problems on

Six Factor Formula



What is Keff and the value of each factor?

(2)

1000 Fast neutrons

Picked up 30 bonus neutrons

Next

980 Loss of 50 neutrons from core

110 neutrons lost in U-238

870

loss of 60 neutrons from core

810

300 absorbed in moderator, poisons, etc.

32 absorbed in U-235

475

75 non fiss capture in U-235

403 - cause fission in U-235

1006 Find Keff & value of 6 factors?

FAUST

1. $K_{eff} = \cancel{\frac{104}{100}} \times \frac{99}{104} \times \frac{89}{99} \times \frac{84}{89} \times \frac{84}{35} \times \frac{104}{84} = 1.04$

$E = \frac{104}{100} = 104\%$

$L_f = \frac{99}{104} = 95.3\%$

$p = \frac{89}{99} = 90\%$

$L_t = \frac{84}{89} = 94.5\%$

$f = \frac{35+49}{84} = 100\%$

$\eta = \frac{104}{84} = 1.24\%$

2. $E = \frac{1030}{1000} = 103\%$

$L_f = \frac{980}{1030} = 95.1\%$

$p = \frac{870}{980} = 88.8\%$

$L_t = \frac{810}{870} = 92\%$

$f = \frac{478}{510} = 59.\%$

$\eta = \frac{1006}{478} = 210\%$

$K_{eff} = \frac{1030}{1000} \times \frac{980}{1030} \times \frac{870}{980} \times \frac{810}{870} \times \frac{478}{510} \times \frac{1006}{478}$

$K_{eff} = 1.006$ or 100.6%

1103 1120

(3.) all same as (2) except

from

810 → 300 abs. in Mod, fission, ect.

↓

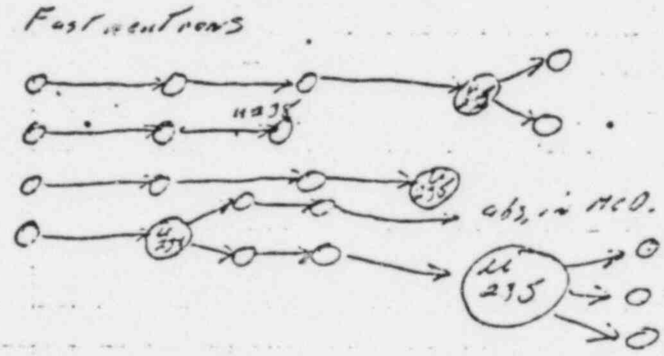
510 → 107 abs in U-238 and U-235 (new neutrons)

403 cause fission in U-235

↓

1000 . . same question

(4) Fast neutrons



3. $\xi = \frac{510}{810} = 63\%$

η ~~[illegible]~~

$\frac{1006}{403}$ ← wrong $= 249\%$ Error due to follow through on ξ.

$\frac{1006}{510}$ should have been

$\frac{\cancel{1030}}{1000} \times \frac{\cancel{990}}{\cancel{1030}} \times \frac{\cancel{890}}{\cancel{990}} \times \frac{\cancel{870}}{\cancel{890}} \times \frac{510}{\cancel{870}} \times \frac{1006}{403}$

$K_{eff} = .51 \times 2.49 = 1.271$

10^{-1}

4. $K_\infty = \epsilon p \xi \eta$

$K_\infty = \frac{5}{4} \times \frac{4}{3} \times \frac{3}{4} \times \frac{5}{3} = 1.25$

Beta
U235 = .0064
U235 = .0148
Pu 239 = .0021

$\overline{\text{Beta}}$ more effective
U235

β ... $\bar{\beta}$ ... core

Beta	
BOL	EOL
$\beta = .0071$	$\beta = ?$
$\bar{\beta} = .0069$	$\bar{\beta} = .00513$
	decrease due to Pu239 build in.

Known deviations

I. Six Factor Formula defined (Con't)

~~Eta~~ ~~Eta~~ ETa

.33% for producing neutrons from thermal fission

$$\eta^{4} = \frac{\text{Number of fast neutrons produced by thermal fission}}{\text{Number of thermal neutrons absorbed in fission material}}$$

Thermal utilization factor
$f < 1$

$$f^{3} = \frac{\text{Number of thermal neutrons absorbed in fission material}}{\text{Number of thermal neutrons absorbed in all reactor materials}}$$

Fast fission factor
for individual Rx (U235) value ... 1.0415

$E > 1$

Fast fission factor (Epsilon)

$$E^{1} = \frac{\text{Number of fast neutrons produced by fissions of all energies}}{\text{Number of fast neutrons produced by thermal fission}}$$

Non Resonance escape probability
$p < 1$

$$P^{2} = \frac{\text{Number of fast neutrons which slow down to thermal}}{\text{Number of fast neutrons which start to slow down}}$$

Probability of fast NON leakage
$l_f < 1$

$$\mathcal{L}_f^{1.5} = \frac{\text{Number of fast neutrons which remain in the core after leakage}}{\text{Total number of fast neutrons in the core before leakage}}$$

Probability of thermal NON-leakage
$L_t < 1$

$$L_t^{2.5} = \frac{\text{Number of thermal neutrons which remain in the core after leakage}}{\text{Number of thermal neutrons in the core before leakage}}$$

2. How various parameters affect each of the six factors.

(A) Reproduction factor "η" (eta)

(1) Is calculated for a particular type of fuel.

(2) For each type of fuel there is released "ν" (Mev) (gms) number of neutrons.

$$\therefore \frac{\text{neutrons produced}}{\text{absorption}} = \frac{\text{neutrons produced}}{\text{number of fissions}} \times \frac{\text{Number of fissions}}{\text{number of absorptions}}$$

Beta in ~~the~~ TMI Rx is more effective than $\bar{\beta}$ due the the Epithermic ~~...~~ area of Spectrum. The Navy was just the opposite.

1103 1123

IV Four Factor Formula (K_∞)

3. Production

A. Reaction Rate (Fission) $\longrightarrow nv = \phi$

$\Sigma_f = N\sigma_f$ $\quad FRR = \underbrace{N \sigma_f}_{} \; n \; v$ ← velocity

N: $\frac{\text{atoms}}{cm^3}$ $\quad n$: neutron density

$FRR = \Sigma_f \phi \quad$ (units) $\frac{\text{fissions}}{cm^3\text{-sec}}$

B. Neutron Production Rate (PR)

		ν (units) neutrons/fission
Known Experimental values	U235	2.43
	P239	2.89
	U233	2.51
	U natural	2.50

$PR_{(neutron)} = \nu \Sigma_f \phi \quad$ (units) $\frac{\text{neutron}}{cm^3\text{-sec}}$

C. absorption Reaction Rate (AR)

$AR = N \sigma_a^{\text{fuel \& structural materials}} \cdot nv$

$AR = \Sigma_a \phi$

For steady state ($K_\infty = 1$) conditions

$\Sigma_a \phi = \nu \Sigma_f \phi$

$PR = AR$

1103 1124

$$\eta = (\nu)\frac{\sigma_f}{\underbrace{(\sigma_a + \sigma_{n,\gamma})}_{\sigma_a}} + \eta \qquad \sigma_f$$

3. Example : Our fuel consists soley of 2.62% enriched U-235 and U-238 in oxide form, the oxygen has a negligible cross section and can be neglected. Then:

$$\eta = \frac{\nu^{235}\,\Sigma_f^{235}}{\Sigma_a^{235} + \quad + \Sigma_a^{238}}$$

ν for U-235 = 2.43
PU-239 = 2.89

$$\eta = \frac{\nu^{235}\,N^5\,\sigma_f^{235}}{\frac{N}{n}\,\sigma_a^{235} + \quad + \frac{N}{n}\,\sigma_a^{238}}$$

$$\eta = \frac{\nu^{235}\,\sigma_f^{235}}{\sigma_a^{235} + \cancel{\sigma_f^{235}} + \left(\frac{0.9738}{0.0262}\right)\sigma_a^{238}}$$

Included in σ_a Term

37.1

on Back

D. K_∞ Neutron Reaction cycle

$K_\infty = E\,P\,f\,\eta$

when dealing with entire area of core ϕ is considered same, so it can cancel out
ϕ_u^{235} same as ϕ_u^{235}

$$\eta = \frac{\Sigma_f \phi_{u235}\, \nu}{\Sigma_a \phi_u^{235} + \Sigma_a \phi_u^{238}}$$

$$\eta = \frac{\Sigma_f^{235}\, \nu}{\Sigma_{a_{u235}} + \Sigma_{a_{others}}}$$ u^{238} in this case

$$\eta = \frac{N_{235}\, \sigma_a^{235}\, \nu}{N_{235}\, \sigma_a^{235} + N_{238}\, \sigma_a^{238}}$$

$$\eta = \frac{\sigma_a^{235}\, \nu}{\sigma_a^{235} + \left(\frac{N_{238}}{N_{235}}\right)\sigma_a^{238}}$$

Enrichment ratio terms such as 2.62 % u^{235} which leaves 97.38 % u^{238}

$N = \frac{\rho N_a}{M}$

$\Sigma = N\sigma$

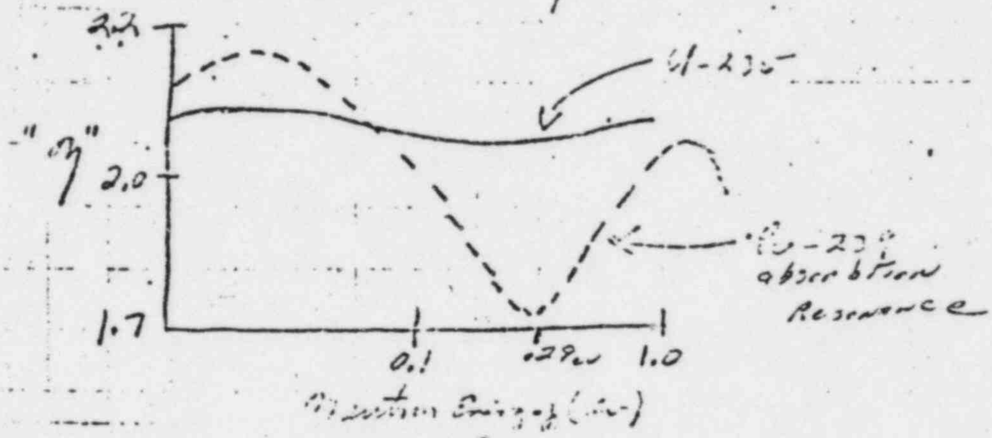
1. More mod to fuel
2. Lower temp.

4. <u>Items Affecting "η"</u>

(a) Enrichment, the greater the enrichment the larger "η".

(b) Neutron energy also affects "η"

(1) Since neutron enrgy is affected by the moderator temp the the following variation in "η" can be seen.



(2) Result: For U-235, "η" is essentially constant. For PU-239 there is a definite drop at approximately 0.29 ev.

(c) "ν" for both U-235 and PU-239 increases slightly with increase in neutron energy but for our purposes we consider it a constant.

(d) Also considered constant for our core over core age. (PU-239 compensates for U-235 burnup).

(e) "η" value for TMI core 1.67.

$$\mathfrak{f} = \frac{V_u \Sigma_u \Phi_u}{\Sigma V \Sigma \Phi} = \frac{V_u \Sigma_u \Phi_u}{V_u \Sigma_u \Phi_u + V_m \Sigma_m \Phi_m} =$$

$$= \frac{1}{1 + \frac{V_m \Sigma_m \Phi_m}{V_u \Sigma_u \Phi_u}}$$

$$\Sigma = N \sigma$$

$$N = \mathfrak{f} \rho \left(\frac{N_0}{m} \right. \quad N_0 = 6.02 \times 10^{23}$$

$g/_{AMU}$

E. Thermal Utilization Factor "f"

1. The thermal utilization factor refers to the thermal neutron absorption in fuel to the total absorption everywhere. It is expressed as:

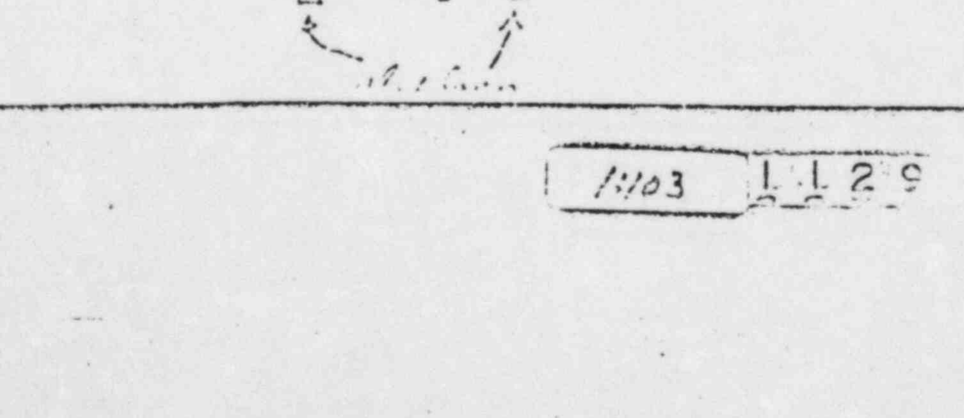
$$f = \frac{\phi_F \Sigma_a^{235} V_F}{\phi_F \Sigma_a^{235} V_F + \phi^{MOD} \Sigma_a^{MOD} V^{MOD} + \phi_0 \Sigma_a^{other} V_0 + \phi_p \Sigma_p V_p}$$

Where: ϕ_F = flux in fuel
ϕ^{mod} = flux in moderator
ϕ_0 = flux in other material
V_F = volume of fuel
V^{mod} = volume of moderator
V_0 = volume of other materials

Poison are taken into account in this term in the denominator.

$$f = \frac{\Sigma_a^{235}}{\Sigma_a^{235} V_F + (\phi^{MOD}/\phi_F)\Sigma_a^{mod} V^{mod} + (\phi_0/\phi_F)\Sigma_a^{other} V_0}$$

(a) The ratio ØMod/ØF is called the thermal disadvantage factor. It has a value of from 1.1 to 1.2 in water moderated UO_2 lattices.



2. Parameters affecting the thermal utilization factor.

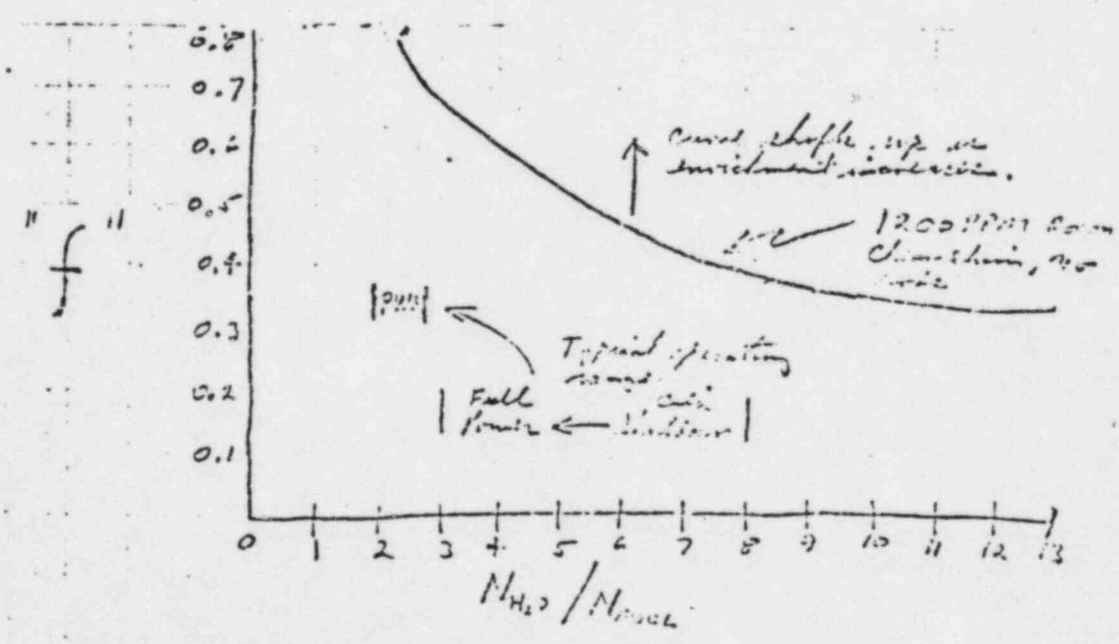
   (a) Is strongly dependent upon the relative amounts of various reactor materials. Particularly fission product poisons.

   (b) Movement of control rods directly affect "f". It affects "Σ_a others" in denominator of above equation.

   (c) Ratio of moderator molecules to fuel molecules affect "f" as the ratio increases "f" decreases. The ratio decreases as the temperature of the moderator increases.

   (d) "f" Decreases with cqre age because of the burnup of fuel and the buildup of fission product poisons.

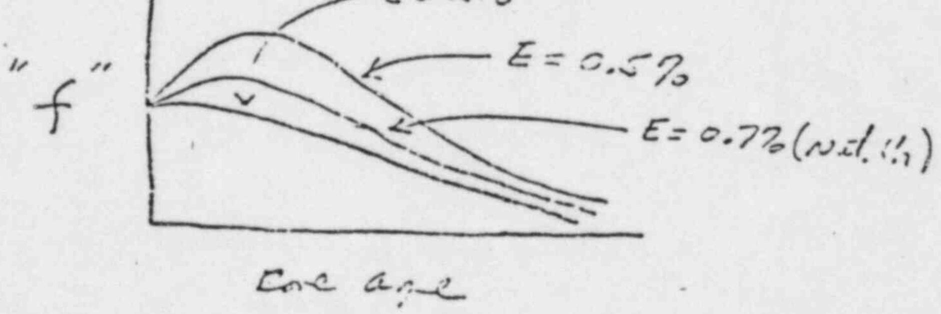
   (e) As fuel enrichment increases the thermal utilization factor also increases.



3. Value of "f" for TMI core = 0.8

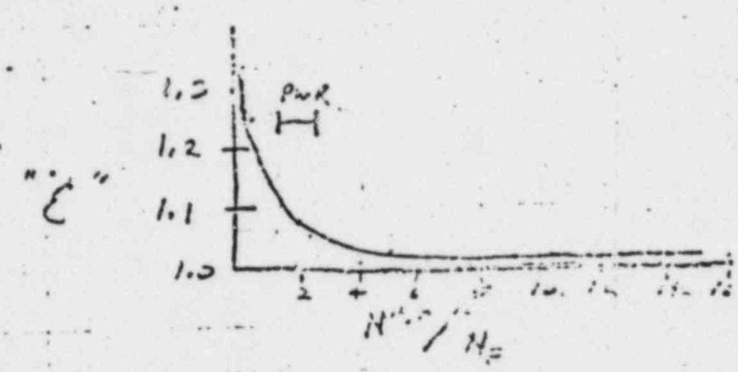
4. Effect of Pu-239 buildup over core age.

   (a) Pu-239 has a higher microscopic thermal absorption cross section (1029b) then does U-235 (683b) therefore, direct substitution of Pu-239 for U-235 would result in an increase in "f". However, in practice Pu-239 is only produced as the reactor is operated and thus appears at the expense of U-235 depletion. In converter reactors (TMI) "f" could increase as the fuels burns if for about the first 1000mwd/ton of UO_2 the initial production rate of Pu-239 per U-235 atom destroyed is about 0.8. However, the enrichment must be less than ≈ 2% (as enrichment is increased, there is less U-238 in the core and hense less Pu-239). Since TMI avg. enrichment is 2.6% "f" would decrease continuously throughout core life. However, even in this case, "f" decreases more slowly throughout core life than it would without the contribution from Pu-239 Buildup.



C. Fast Fission Factor "ε"

1. For water moderated reactors, the fast fission factor varies from about 1.02 to about 1.1. For TMI however, ε = 1.2. This large value is due to the fact that the B&W reactor depends more heavily upon fast fission of U-238 than does most water moderated reactors.
2. "ε" is strongly affected by the moderator to fuel ratio.



As the amount of moderator is reduced, the fast fission factor increases due to the fact that neutrons are less likely to encounter a moderator molecule and be slowed down before they can cause a fast fission.

3. In the range of enrichments (1.57 to 3.5 %) the fast fission factor is essentially independent of enrichment (because the fast σ_f for U-235 (1 b) and U-238 (o.45) do not differ greatly.
4. The buildup of plutonium as the core ages also does not effect "ε".
5. As the temperature of the moderator increases, "ε" increases for the same reason as #2.
6. Therefore, changes over the core life are insignificant.

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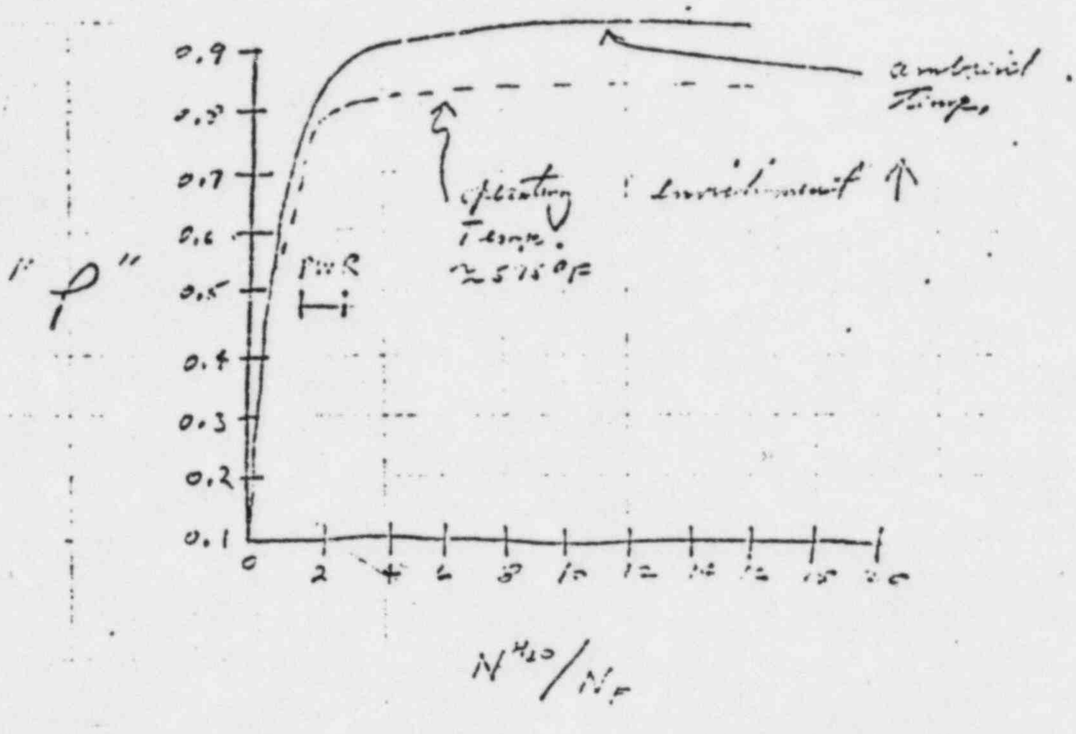
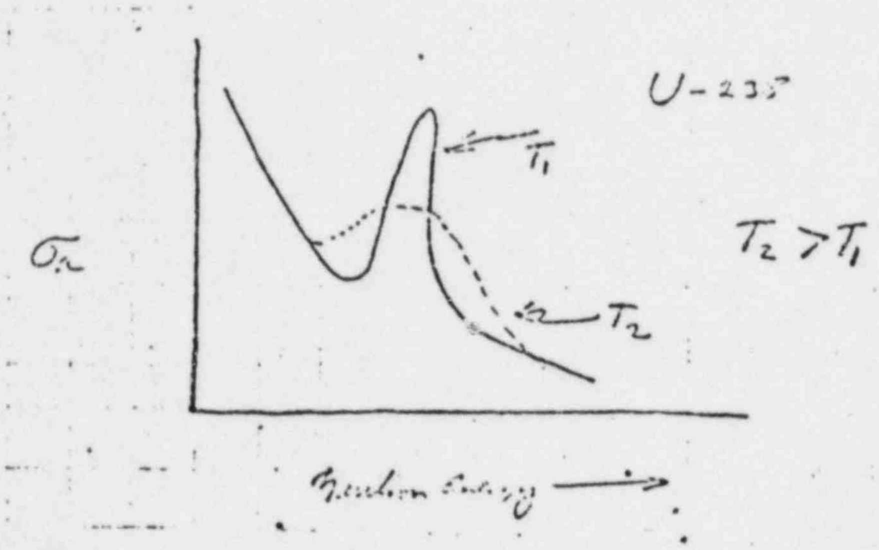
D. Probability of Resonance Escape (P)

1. Is strongly dependent upon the moderator to fuel ratio. As the amount of moderator in the core is reduced, the fast neutrons are not slowed as efficiently and therefore, spend more time with energies in and about the resonance region. As a result, more of them are captured in the U-238 resonances and the escape probability decreases.

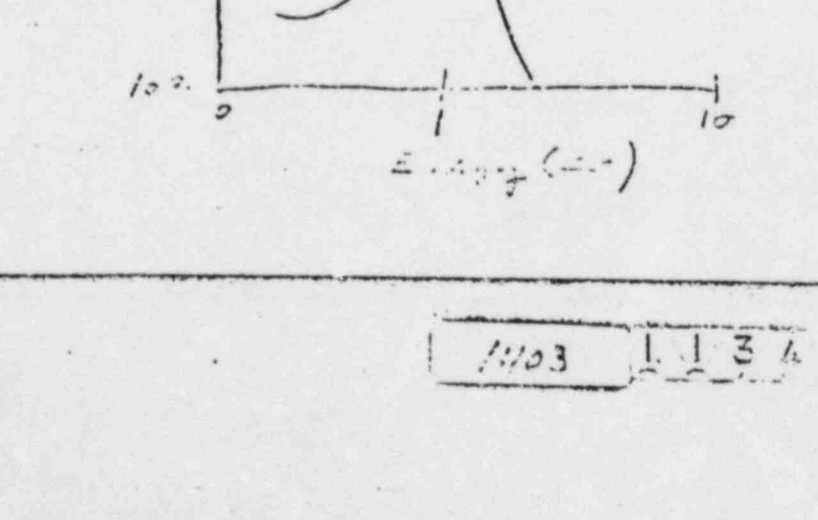


2. "P" is strongly affected by the fuel temperature. As the fuel temperature increases, U-238 resonance absorption increases and the escape probability decreases.



3. The resonance escape probability is also dependent upon the composition of the fuel. The higher the enrichment, the greater the "P".
4. As the core ages, "P" decreases due to the buildup of fission products. Principally the isotope Pu-240 which has a large resonance absorption.

Peak at 1.025 ev.



Measure of curvature of the flux to the

$$\mathcal{L}_{fast} = e^{-B^2 \tau}$$ τ = Fermi age, or L_f^2 (same) (measure of the dis. traveled by N' slowing down to thermal energies

$$\mathcal{L}_{therm} = \frac{1}{1 + B^2 L_t^2}$$

* $L_f^2 = \tau$

M^2 = migration length

$$M^2 = \tau + L_t^2$$

$$M = \sqrt{\tau + L_t^2}$$

* L_t for thermal diffusion length in cm, to use in equation the term is squared, as is the L_f to give the so designated Fermi age of neutrons

Diffusion Coefficient (D)

$$\frac{1}{\Sigma_s} = \lambda_s^{(cm)} \text{ (mean free path)}$$

$$D_{fast} \cong \frac{1}{3 \Sigma_s} \cong \frac{\lambda_s}{3}$$

$$L_t^2 = \frac{D}{\Sigma_a} \qquad \underline{D_{therm} = L_t^2 \Sigma_a}$$

$L_T = L_t L_f$

5. Value of "P" TMI core = 0.65

E. Thermal and Fast Non-Leakage (L_t and L_f)

1. L_t and L_f depend upon two quantities:

(a) The average distance the neutrons travel in the reactor.

(b) The size of the reactor.

2. Average distance traveled is determined by two parameters.

(a) The thermal diffusion length L_{1t} which is 41 % of the average crow flight path of a thermal neutron in the core between thermalization and absorption.

(b) The fast diffusion length L_f, which is 41% of the average crow flight path that neutrons travel between birth and thermalization. The square of the fast diffusion length is referred to as the fermi age.

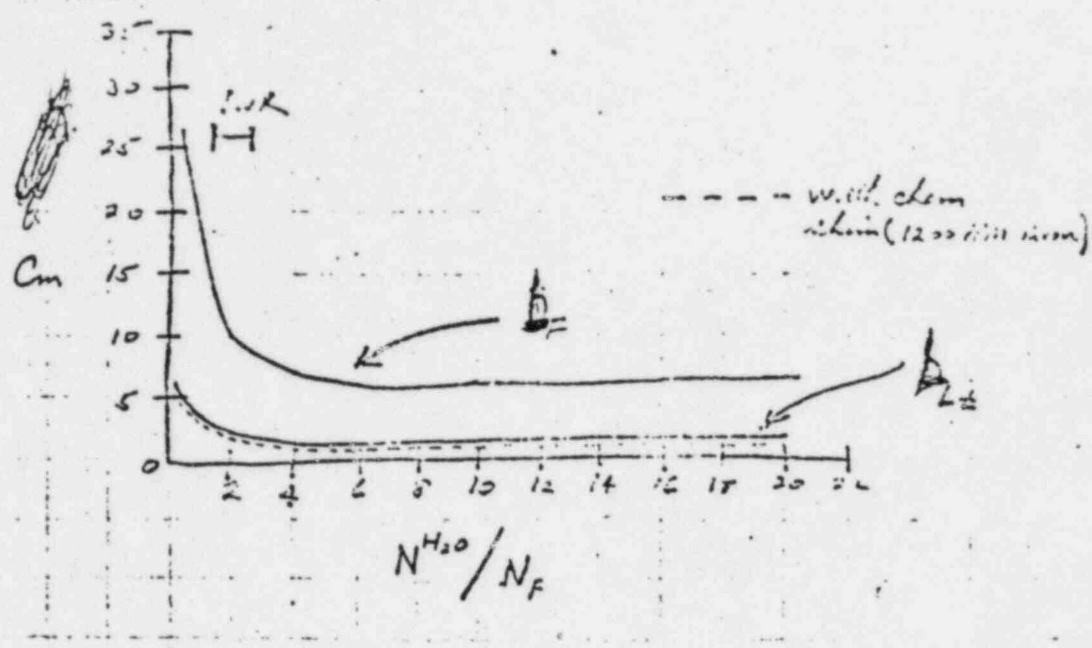
3. The size of the reactor is measured by:

(a) The geometric buckling (Bg^2) which is a measure of the reactor size.

(1) As the reactor size increases, the geometric buckling decreases. Therefore as Bg^2 increases, leakage increases.

4. $L_f + L_{1t}$ is very sensitive to the moderator to fuel ratio.

(1) As the ratio decreases, L_f increase because the neutrons are less likely to encounter moderator atoms and be slowed down, so the slowing process takes longer and the neutron travel farther.



2. Fast neutron removal from the core is mainly by thermalization rather than absorption. Therefore L_f depends strongly on the moderating ability of the core. In addition, the fast neutrons are heavly subjected to scattering reactions therefore travel a crooked path as they slow down. This path tends to reduce their crow flight path length. Therefore, reducing the fuel moderator to fuel ratio results in an increase of L_f, ~~becams~~ because the neutron path is straighter.

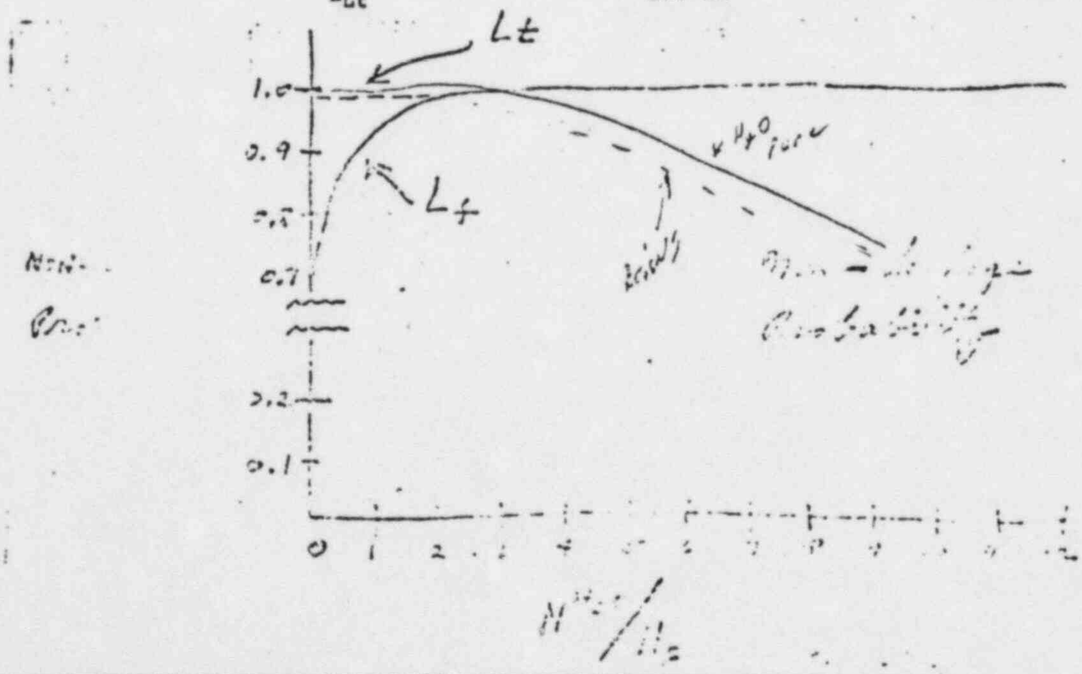
Principle removal of thermal neutron is by absorption.

As the moderator to fuel ratio decreases, the moderator absorbs fewer neutrons and the neutrons travel a straighter path. Therefore, L_{Lt} tends to increase.

The isotope composition of the core can have a significant effect upon L_{Lt}. For example a strong absorber such as chem shim tends to reduce L_{Lt}.

Typical values of crow flight and L_F and L_{Lt} are:

	Room Temp.	Oper. Temp.
crow flight (Th)	2.9 cm	3.6 cm
crow flight (F)	16 cm	19.2 cm
L_F	6.6 cm	7.9 cm
L_{Lt}	1.2 cm	1.5 cm



Note : (a) Fast Non-leakage probability is not greatly affected by changes in fuel composition or burnup since the distance traveled by a fast neutron is principally determined by scattering properties of the moderator.

(b) Fast Non-leakage is dependent upon the power distribution in the core. The more power that is generated on the edges of the core the greater the leakage. Since near end of core life the power shifts to the edges of the core, leakage will increase.

(c) Thermal Non-leakage is affected by both power distribution and the buildup of fission product absorbers. However, from a practical standpoint thermal leakage is so low it has very little effect over core age.

5. Value of non-leakage terms for TMI reactor are:

$$L_f = 0.996$$

$$L_t = 0.998$$

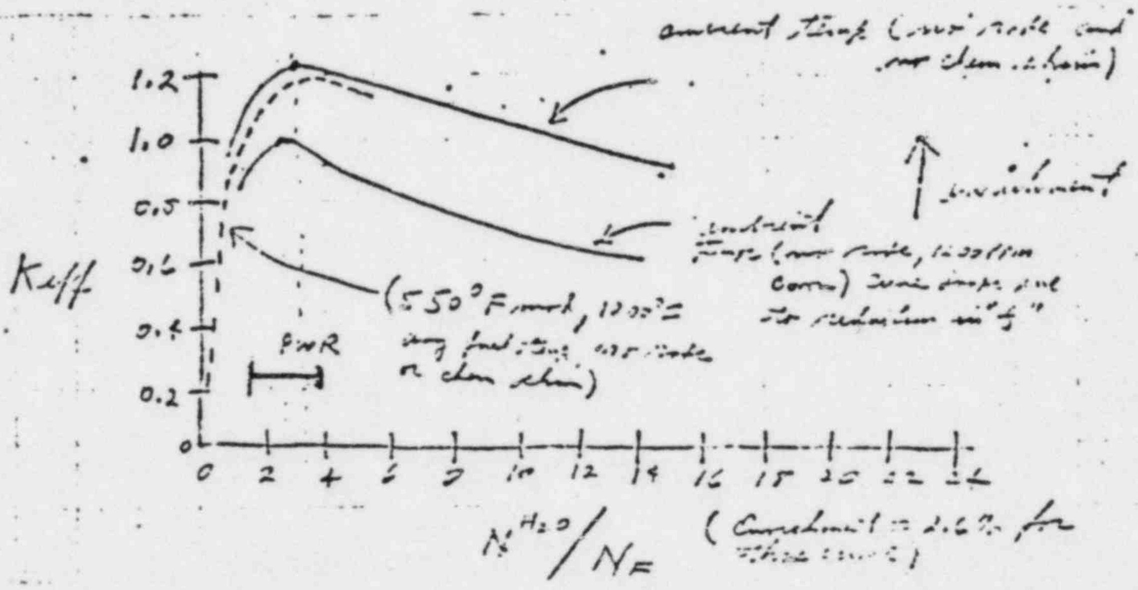
J. Water to Fuel Ratio

1. Keff is the product of the six factors.
2. It can be seen by the previous discussed curves that for low mod/fuel ratios Keff decreases rapidly due to the rapid decrease in "P".
3. For high values of mod/fuel ratios Keff also decreases due primarily to the decrease of " f " (See previous discussed curve.)

4. There is an intermediate point where Keff is a maximum for any given reactor and set of operating conditions. This point is the optimum design for a maximum Keff.

Note: (a) When the mod/fuel ratio is less than optimum value, the the core is said to be undermoderated. (It has too little moderator to acheive the maximum Keff).

(b) When the mod/fuel ratio is greater than the optimum point, the core is said to be overmoderated.



Note: (a) For a core with chem shims, the optimum point occurs at a lower value of water/mod ratio.

Explanation: When the moderator is pure water, adding some of it to the fuel improves "P" without significantly hurting "f" (because water has a low absorption cross section). The result is an increase in Keff which would continue as we add water until we get "P" close to 1.

In chemical shim core, the addition of water helps "P" but this time the high absorption cross section of the poison (Boron) starts killing "f" ("f" falls off alot faster with chem shim than without). Thus, the benefit we get form increasing "P" is much more quickly offset by the reduction in "f" so the optimum point occurs at a lower water to fuel ratio.

K. Epithermal Contributions to Keff:

1. Resonance capture in fissle material:

When fuel rods are closely packed as in our reactor, the relativity low mod/fuel ratio results in incomplete thermalization of many neutrons before they are absorbed. As a result, resonance fission will represent a significant fraction of the total power (10 to 20%) Usually this is accounted forby adjusting "f" however, B&W consider this as fast fission and therefore adjust "$\mathcal{E}$" .

.2. Fast and resonance absorption in the moderator and core structure also occur and is accounted for by adjusting the resonance escape probability "P".

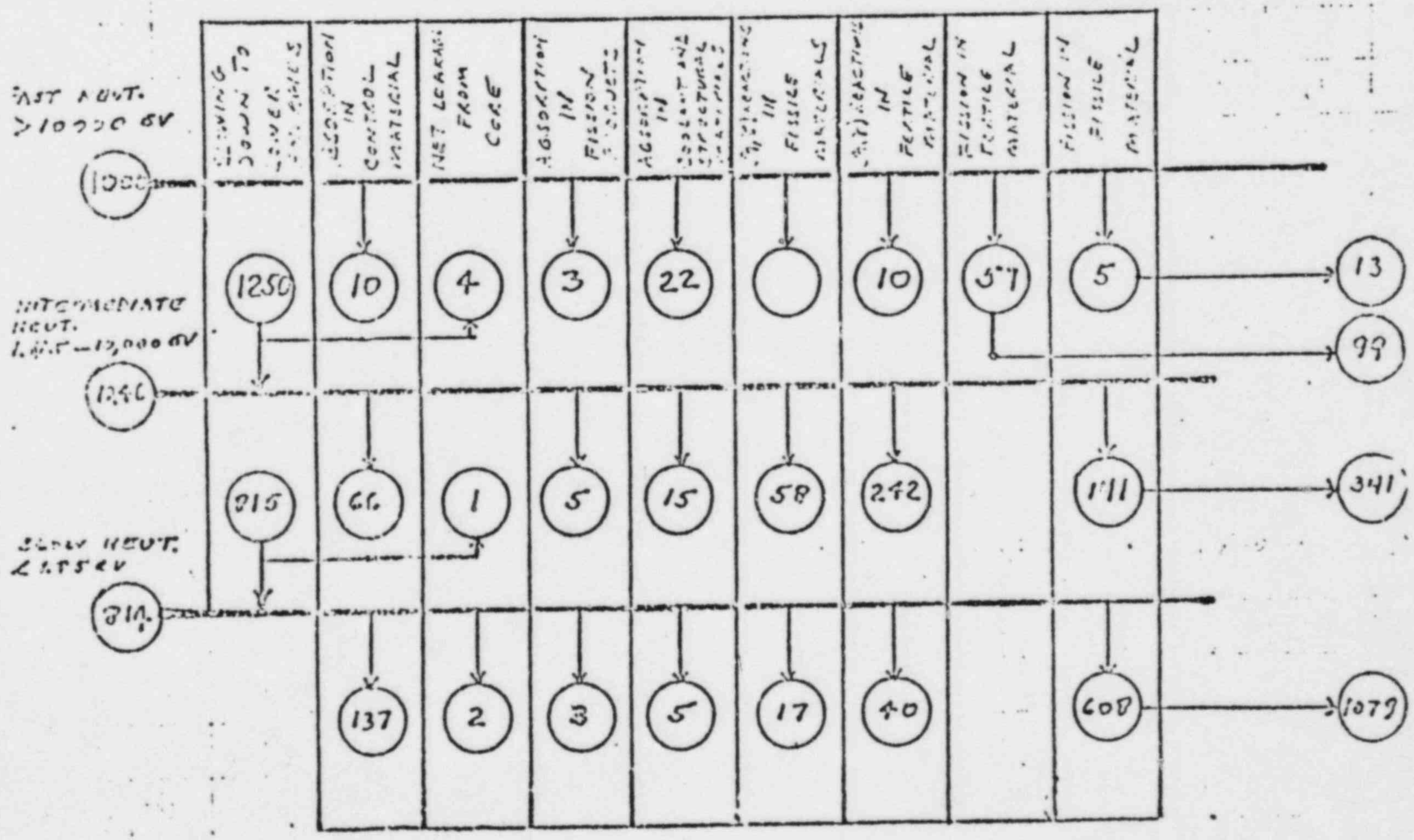
3. Boric Acid also results in a small amount of epithermal absorption. This is accounted for by a correction to "f".

Examples:

I

DETAILED NEUTRON BALANCE IN TMI CORE (BOL)

REVISION 1
1/2/74



NOTE: TOTAL FISSIONS = 811

(65)

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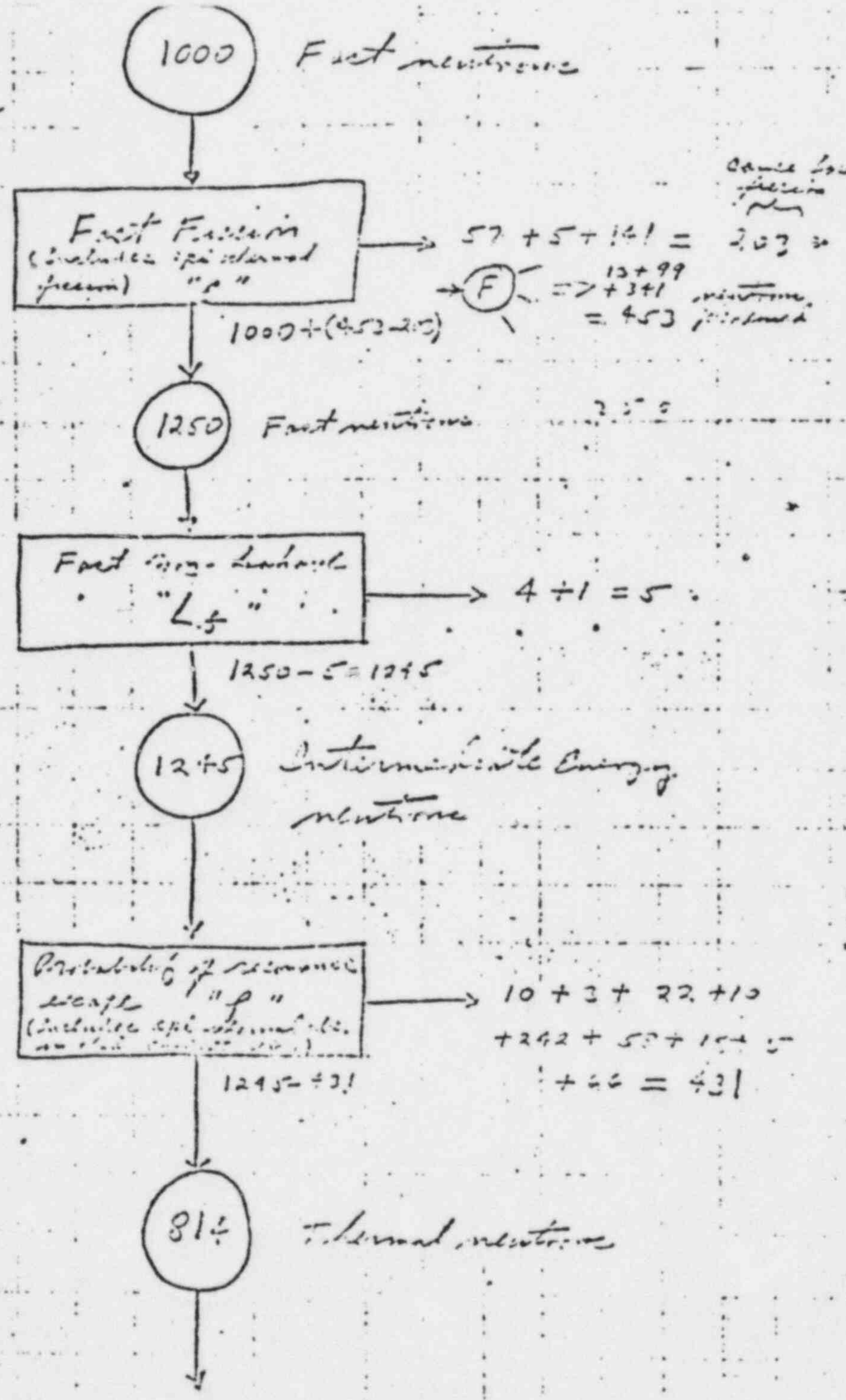
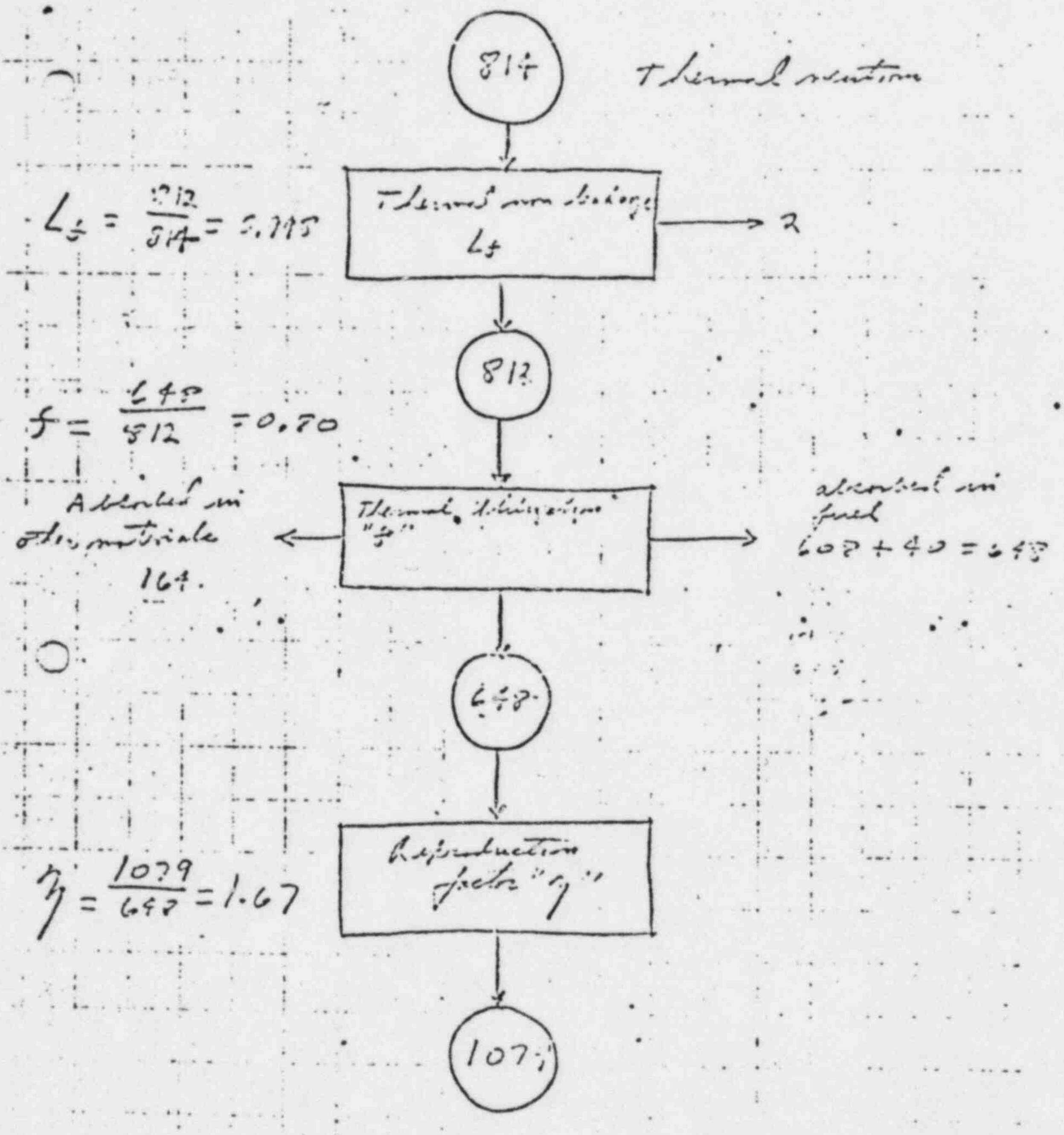
TMI - 1301



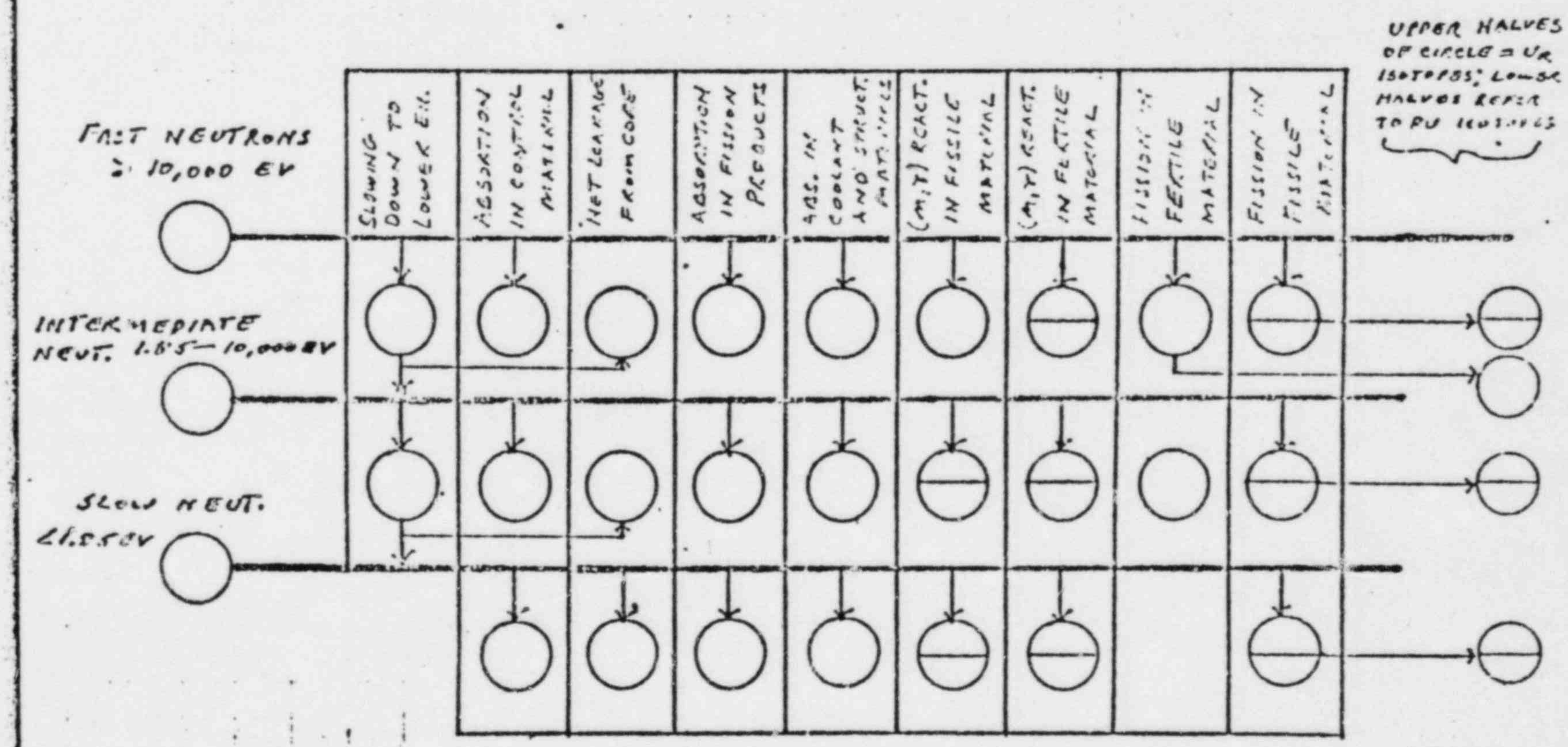
$$\varepsilon = \frac{1250}{1000} = 1.25$$

$$L_f = \frac{1245}{1250} = 0.996$$

$$p = \frac{814}{1245} = 0.65$$


814
Thermal neutrons
$L_t = \frac{812}{814} = 0.998$
Thermal non-leakage L_t
2
812
$f = \frac{648}{812} = 0.80$
Absorbed in other materials 164
Thermal utilization "f"
absorbed in fuel 608 + 40 = 648
648
$\eta = \frac{1079}{648} = 1.67$
Reproduction factor "η"
1079

Detailed Neutron Balance
In TMI Core (EOL)



(14)

III. Reactivity

A. Reactivity (P)

1. Definition : Approximate fractional change in the neutron population per generation.

Reactivity

$$P = \frac{K_{eff} - 1}{K_{eff}} = \frac{\Delta K}{K}$$

Note: When Keff = 1, P = 0. This means that when the reactor is just critical the reactor has zero reactivity.

2. Super critical condition

ex: If K = 1.003

$$P = \frac{1.003-1}{1.003} = +0.00299 \ \Delta K/K$$

or +0.299% $\Delta K/K$

∴ If in the first generation of neutrons we had 1000 neutrons, the second generation would yield (1000) (Keff) Or (1000)(1.003) = 1003 neutrons.

3. Subcritical Condition

ex: If K = 0.997

$$P = \frac{0.997-1}{0.997} = 0.00301 \ \Delta K/K$$

or -0.301% $\Delta K/K$

∴ If in the first generation of neutrons we had 1000 neutrons the second generation would yield (1000)(Keff) or (1000)(0.997) = 997 neutrons.

(adjusted)

our (TMI) ϵ is changed from 1.1 (about) to 1.2 due to epi-thermal fission.

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4. Summary

Rx Condition	Keff	P
Subcritical	< 1.00	Negative (< 0)
Critical	1.00	0
Supercritical	> 1.00	Positive (> 0)

5. Δ K vrs $\Delta K/K$

It should be noted that Keff-1 approximately equals $\frac{\text{Keff-1}}{\text{Keff}}$

when Keff is close to 1.00. Therefore, when Keff $\approx$ 1.00, $\Delta K = \Delta K/K$.

B. Controls Rods

1. Used in a reactor to control:

   a. reactivity within operating limits.

   b. reactor power

   c. power distribution

   d. rapid reactor shutdown

silver cadmium + (very expensive) cadmium alloy used at TMI provides a more complete shut down due to the epithermal fissions. i.e. [illegible] fuel atoms

2. Control rods contain materials having high cross sections for thermal and in the core of TMI epithermal neutrons as well.

3. The principle effect of control rod insertion is the reduction in the thermal utilization factor.

4. Control rods are localized poisons, therefore have certain important effects upon reactor operations.

5. TMI Control Rods (See Handout)

6. The potential strength of a control rod depends upon the percentage of total core thermal and epithermal neutron population.

7. Factors which affect the absorbability of reactivity worth of control rod.

epithermal / neutrons are those neutrons abs in fuel resonance absorption region causing fast fission

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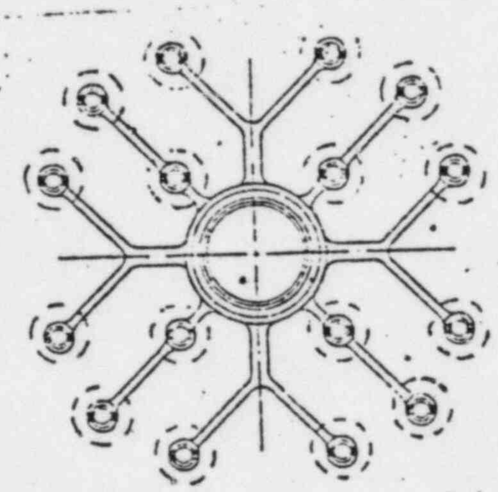
(a). Absorption Envelope

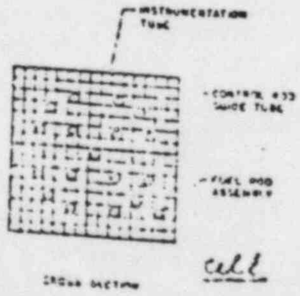
(1) Control rods have a good chance of absorbing thermal neutrons from within a diffusion length of the rod. Since crow flight is approximately 1 to 2 inches and since $b_{lt} \approx 41\%$ of crow flight, $b_{lt} \approx 0.8$ inches.



Dotted lines show area of influence.

The % of the total cell which would be under the influence of an inserted control rod is given by

$$\frac{\text{Volume of Absorption Area Envelope Surrounding Rod}}{\text{Total Volume of Cell}} = \frac{b_{lt}\, Sa}{V_{cell}}$$



Where: b_{lt} = thermal diffusion length

Sa = surface area

Vc = volume of lattice cell

(2) Volume of core under the influence of control rods.

Since the real core contains many cells, some which are uncontrolled and some which are uncontrolled, we generalize and say that the % of core under the influence of any control rods is given by:

$$\frac{\text{Vol. of Abs. area Envelope Surrounding all rods of interest } (L_{1t} Sa)}{\text{Total Volume of core } (V_{core})}$$

where: Sa = total surface area of all rods of interest.

V_{core} = volume of core.

(b) <u>Percentage of core neutron inventory likely to be found within volume of interest.</u>

(1) <u>Observations:</u> In general the neutrons density near the edge of a reactor is somewhat lower than in the center at BOL and opposite at EOL.

(2) The number of neutrons which are potentially influenced by the insertion of a rod is proportional to the product of the value of core influenced by the rod and the neutron concentration within in this volume.

% of neutrons influenced $(L_{1t})\ (Sa)\ (\phi tcr)$

where: L_{1t} = diffusion length

Sa = total surface area of all Control rods in area of interest.

ϕ_{tcr} = thermal neutron flux at the location of the rods (prior to insertion).

Note: The total number of thermal neutrons in the core would be proportional to (Vcore) x (Øt avg)

∴ The % of the total core neutron population which a control rod or group of rods is capable of absorbing is then

$$\frac{(D_{cr})(S_c)(\emptyset_{tcr})}{(V_{core})(\emptyset_{t\,avg})} = \frac{(D_{cr})(S_c)}{V_{core}} \times \frac{(\emptyset_{tcr})}{(\emptyset_{t\,avg})}$$

(c) <u>The importance of neutrons which are absorbed.</u>

(1) To rods which absorb equal fractions of the total neutron population may have different reactivity worths depending upon the importance of the neutrons they capture.

(2) Will assure that regions of the core having high neutron contribution must be where a lot of fission are taking place and therefore ar important regions.

(3) Will assume that regions of the core where there is a low concentration of neutrons must be regions where many non-productive processes such as leakage and absorption are occurring. Therefore these regions are relatively unimportant.

(4) Therefore importance is related to the ratio of thermal flux in the region of interest to the average throughout the core.

∴ To complete our determination of a control rods worth, we multiply the number of neutrons absorbed by their relative importance factor $\emptyset_{tcr}/\emptyset_{t\,avg}$.

(1.)

$\% \text{ influence} = \dfrac{\text{Volume of abs Area Envelope}}{\text{Total volume of Cell}} = \dfrac{L_t\, S_a}{V_{cell}}$

Surface area of 1 control Rod $= \pi D L = 134'' \times \pi \times .44''$

$= 185\ in^2$

$16\ rod \times 185\ in^2 = S_{a\ \text{control rod}} = 2.96 \times 10^3\ in^2$

assume the use active fuel length

one cell $= L \times W \times H = 9'' \times 9'' \times 144''$ $= 11.68 \times 10^3\ in^3$

$\% \text{ influence} = \dfrac{.5\ in \times 2.96 \times 10^3\ in^2}{11.68 \times 10^3\ in^3} = 1.268 \times 10^{-1}$

or $\boxed{12.68\%}$ influence of C.R.

(2.)

$\text{Volume of fuel} = 177 \times 11.68 \times 10^3\ in^3 = 2.065 \times 10^6\ in^3$

$\% \text{ influence} = \dfrac{.5 \times 2.96 \times 10^3\ in^3}{2.065 \times 10^6\ in^3} = ~~1.432 \times 10^{-3}~~$

or $\boxed{~~.001432~~}$ % influence of control rod.

7.16×10^{-4} or $.0716\%$

3. $61 \times 2.96 \times 10^3\ in^2 = 1.807 \times 10^5\ in^2$ Total Surface area of stick.

$\% \text{ influence} = \dfrac{.5 \times 1.807 \times 10^5}{2.065 \times 10^6} = 4.37 \times 10^{-2}$ or

4.37 % influence

$R.W. = (1.3)^2(4.37 \times 10^{-2}) = ~~5.67 \times 10^{-2}\ \rho~~$ Total

$= (1.0)^2(4.37 \times 10^{-2}) = ~~4.37 \times 10^{-2}\ \rho~~$

$= (.7)^2(4.37 \times 10^{-2}) = ~~2.09 \times 10^{-3}\ \rho~~$

61 Control Assemblies

16 Control Rods per assembly

.44" outer dia. of Control Rod

134" Length of Posion

144" Active fuel Length

.57" outer dia of fuel Pellet

165 3/4" entire Length of fuel assembly

.5" L_t

.7" Hr L_t

9" by 9" each assembly

15 assemblies across dia. of Core

Surface area = PL $A = \pi r^2$

(2πr)L $= \pi \left(\frac{D}{2}\right)^2$

(πD)L

ϕ_{ind} = 1. % influence of 16CR

Slux Center

assume 1.3 ← ~~center~~

1.0 ← average

.7 ← end

2. $\frac{\% \text{ of } 16CR}{V_{core}}$

3. $\frac{\% \text{ of } CRA}{V_{core}}$

4. RW. at $\approx \frac{\phi_r}{\phi_{ave}}$

$\therefore$ Rod worth $(\Delta k/k) \approx (L_{lt})(S_{cr}/V_{core})(\phi_{tcr}/\phi_{t\,avg})^2$

where: L_{lt} = diffusion length

Scr = surface area of control rod

Vcore = value of the core

Øtcr = thermal neutron flux in area of interest. (Control Rod)

Øt avg = average neutron flux in the core.

(d) <u>Summary</u>: Control rod worth is a function of:

1. The volume of core under the influence of control rods.

2. Percentage of core neutron inventory within volume of interest. control Rod

3. Relative importance of the neutrons.

(e) Worth of control rods vrs hot or cold core condition.

Control rods are worth more under hot conditions than cold conditions due to the increase of the thermal diffusion length "L_{lt}".

IV. <u>Neutron Sources</u>

A. Where neutrons are found to start up a reactor. Both natural and man made sources are available.

1. <u>Natural Sources</u>

a) <u>Cosmic Neutrons</u> : nuclear reactions from the sum produce approximately 50 neutrons/cm^2-hr at sea level.

b) <u>Spontaneous Fission</u>: Isotopes of uranium have a small probability of fusioning spontaneously.

(1) For U-235 is aprx. 2.60 fission/sec-ton

(2) For U-238 is aprox. 6300 fission/sec-ton

ex. For TMI aprox. 91 tons of U-238 would produce.

$$91 \text{ tons} \times \frac{6300 \text{ fission}}{\text{sec-ton}} \quad 5.733 \times 10^5 \text{ fission/sec}$$

On the average, 2.19 neutrons are emitted after each spontaneous fission of U-238, therefore the spontaneous fission source strength is $\approx 1.26 \times 10^6$ n/sec

(c) <u>Photonuclear Reactions</u>: After a reactor is operated for some time there is a large inventory of fission products built up in the core. Many of them are high energy gamma emitters and react with heavy hydrogen nuclei to produce neutrons:

$$\gamma + {}_1D^2 \longrightarrow {}_1H^1 + {}_0n^1$$

$$\gamma \geqq 2.21 \text{ mev}$$

however, note in light water reactors there is only one atom of heavy hydrogen per 6,500 atoms of ordinary hydrogen.

2. <u>Artificial Sources :</u> Since a natural supply of neutrons are available one could start up a reactor with them. However, nearly all reactors built to date use a self-contained neutron source installed in the core. The reason is "SAFETY". There are two main purposes of using an artifical source, they are:

1) It will supply a steady neutron flux which will be observable on the most sensitive neutron detector in the core. Therefore, the operator will have a indication to observe. Natural sources are of such a low intensity that they are not readily used.

(2) By having a reliable and reasonable strength source, one can observe the small changes in neutrons multiplication as control rods are slowly removed for criticality. Therefore yielding a safe approach to criticality.

(a) Examples of Artificial neutron sources

(1) Antimony - Beryllium source

$${}_{51}Sb^{123} + {}_{0}n^{1} \longrightarrow {}_{51}Sb^{124} + \gamma$$

$${}_{51}Sb^{124} \xrightarrow{T_{1/2}=60d} {}_{52}Te^{124} + {}_{-1}\beta^{0} + \gamma \text{ ([illegible])}$$

$$\gamma + {}_{4}Be^{9} \longrightarrow {}_{4}Be^{8} + {}_{0}n^{1} \text{ (Source)}$$

(2) Neutron source used at TMI is a

Americium-curium-Beryllium source.

neutron flux to initially irradiate the source

$${}_{95}Am^{241} + 10^{14}\frac{n}{cm^2\text{-sec}} \longrightarrow {}_{95}Am^{242} \xrightarrow{16\,hr} {}_{96}Cm^{242} + {}_{-1}\beta^{0}$$

$${}_{96}Cm^{242} \xrightarrow[163]{163D} {}_{94}Pu^{238} + {}_{2}\alpha^{4}$$

$${}_{2}\alpha^{4} + {}_{4}Be^{9} \longrightarrow {}_{6}C^{12} + {}_{0}n^{1} \text{ (Source)}$$

$$\approx 10^{10}\ n/sec\text{-}gm$$

B. Subcritical Multiplication: All Subcritical

4a

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Constant Source - large photo-neutron, ect., that produces a given amount of neutrons per generation.

All subcritical multiplication problems involve one of four general cases. We will review each case.

MODELS

1. Case I - Burst of neutrons in water
2. Case 2 - Burst of neutrons in fuel & water
3. Case 3 - Constant source of neutrons in water
4. Case 4 - Constant source of neutrons in fuel & water.

CASE #1 :

1. Neutron multiplication (No Source)

Consider tank of water containing no fuel:

1. One million neutrons are introduced into the water.
2. Neutrons will travel until either
   (a) absorbed by water ($n = p + {}_{-1}^{0}e$)
   (b) leak out

$\therefore$ In a short time no neutrons are left in water

2. CASE #2

Same tank of water with just enough fuel to make Keff. = 0.5 Source out (Removed after initial Generation)

1. Again if 10^6 neuts. added they will die by leakage and absorption. BUT some absorptions will produce new neuts. due to fission.

CASE #3

3. Neutron Multiplication (with a constant source) Source in

   (a) Assume source (So) emits 10^6 neutrons so timed to coincide with beginning of each generation

   Neutrons present as a function of each generation =

   - "SEE FIGURE 1" -

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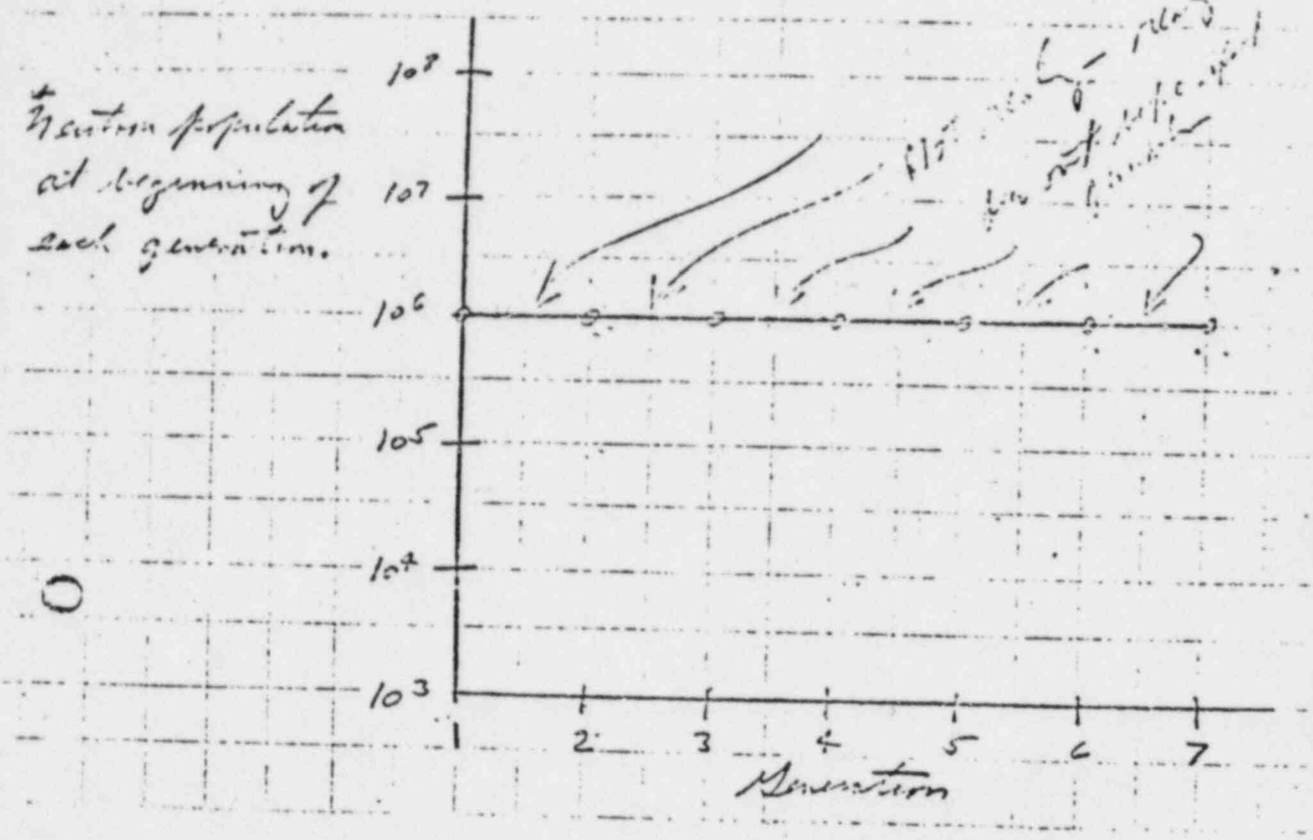
Figure 1



* NOTE: we have assumed that all absorption and leakage has taken place during generation.

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Note: (1) Neutron population does not drop to zero because at the beginning of the second generation the source is still present to emit another burst of 10^6 neutrons.

$\therefore$ Neut. population is constant at beginning of each generation.

(2) Keff $\neq$ 1 since Keff is intended to determine what happens to the neutrons from any single burst = THAT IS - ARE THEY SELF SUSTAINING OR NOT.

Since we have artificially sustained this neut population Keff $\neq$ 1. Source out $\Rightarrow$ neut die out

CASE #4. Source in

4. Same tank of water but with some fuel to make Keff = 0.5

- " SEE FIGURES # 2 & 3 & 4 " -

Fig. [illegible]

$K_{eff} = 0.5$

Generation	Contributing Factors	Reduced Equation	Total No. of n
1	S_0 (Source)		10^6
2	$S_0 + S_0K$	$S_0(1+K) = 10^6 + (10^6)(0.5)$	1.5×10^6
3	$S_0 + S_0K + S_0K^2$	$S_0(1+K+K^2)$	1.75×10^6
4	$S_0 + S_0K + S_0K^2 + S_0K^3$	$S_0(1+K+K^2+K^3)$	1.875×10^6
5	$S_0 + S_0K + S_0K^2 + S_0K^3 + S_0K^4$	$S_0(1+K+K^2+K^3+K^4)$	1.9375×10^6
6	$S_0 + S_0K + S_0K^2 + S_0K^3 + S_0K^4 + S_0K^5$	$S_0(1+K+K^2+K^3+K^4+K^5)$	1.97×10^6
7	$S_0 + S_0K + S_0K^2 + S_0K^3 + S_0K^4 + S_0K^5 + S_0K^6$	$S_0(1+K+K^2+K^3+K^4+K^5)$	1.98×10^6 [illegible]
nth	$S_0 + S_0K + S_0K^2 + S_0K^3 + S_0K^4 + S_0K^5 + S_0K^6 \ldots S_0K^{n-1}$	$S_0(1+K+K^2+\ldots+K^{n-1})$	

* Will assume 1st generation to be equal to source only

(4)

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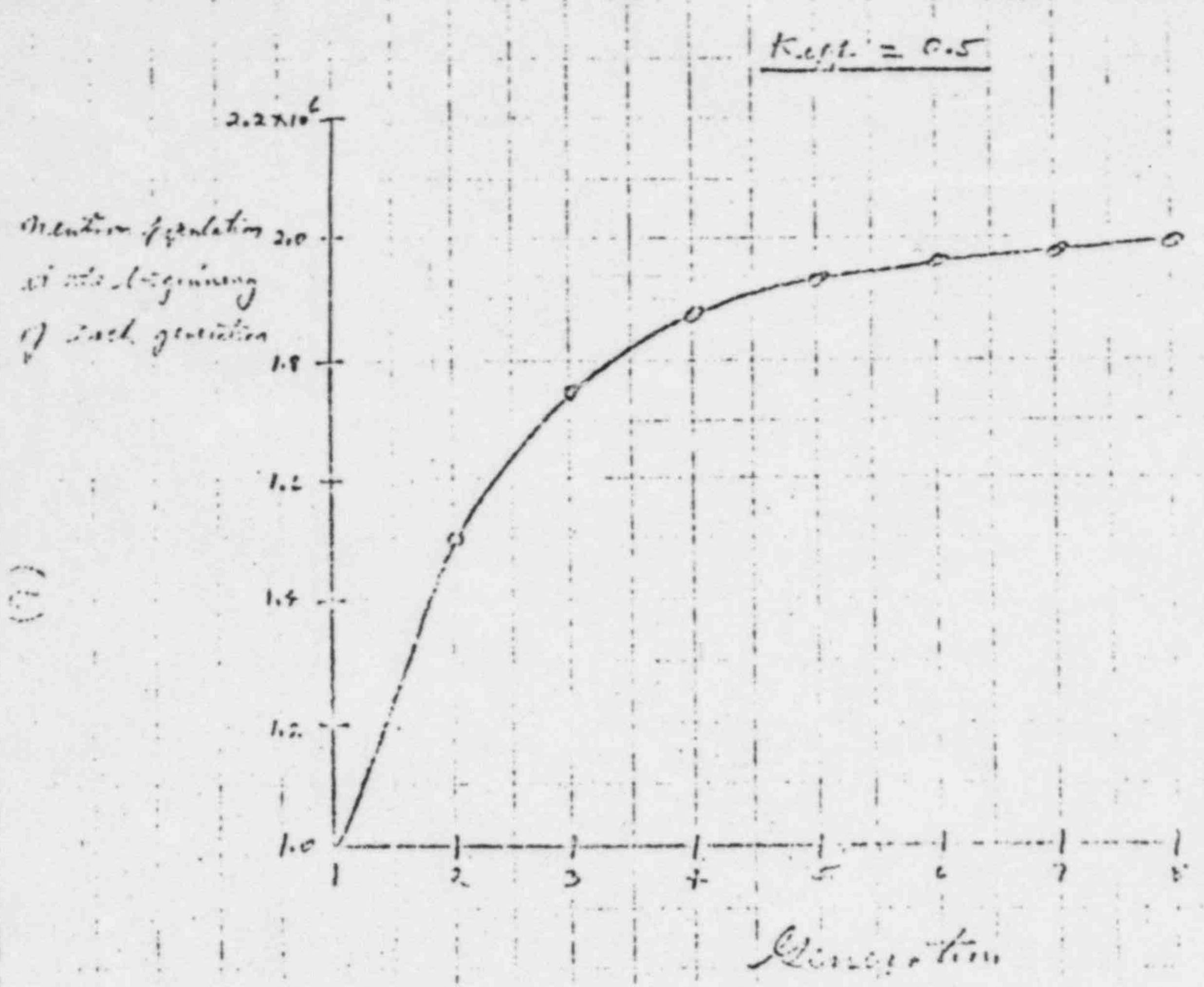
$K_{eff} = 0.5$

Source Burst	Generation 1	2	3	4	5	6	7	
1	10^6	0.5×10^6	0.25×10^6	0.125×10^6	0.063×10^6	0.032×10^6	0.016×10^6	[illegible]
2		10^6	0.5×10^6	0.25×10^6	0.125×10^6	0.063×10^6	0.032×10^6	0.[illegible]
3			10^6	0.5×10^6	0.25×10^6	0.125×10^6	0.063×10^6	0.[illegible]
4				10^6	0.5×10^6	0.25×10^6	0.125×10^6	0.[illegible]
5					10^6	0.5×10^6	0.25×10^6	0.[illegible]
6						10^6	0.5×10^6	0.[illegible]
7							10^6	0.[illegible]
8								[illegible]
Total	10^6	1.5×10^6	1.75×10^6	1.875×10^6	1.938×10^6	1.920×10^6	1.750×10^6	1.[illegible]

Fig 14


Keff = 0.5
Neutron population at the beginning of each generation
2.2x10^6
2.0
1.8
1.2
1.4
1.2
1.0
1
2
3
4
5
6
7
8
Generation

$\mu = \dot{m}\,\dot{s}$

Note:

1. An equilibrium valve for neutron population is gradually being approached. Equil $\approx$ 8th gen

   * In theory however - would take an infinite number of generations to reach equil.

2. If K were to equal 0.99 the number of generation necessary to reach equil would be much greater therefore, the time to level out will also take longer as on approaches K=1.

5. Subcritical neutron equilibrium population:

   A. Equilibrium level found from:

So= neut/generation

$$P = \frac{So}{1-Keff}$$

P = equilibrium population

So = Source strength

6. Subcritical Multiplication

   A. Neut. population measured in a reactor by noting indication on source range count rate meter which receives its signal from a neutron sensitive detector.

   (a) CR meter does not indicate all neutrons present in core but just a percentage which is proportional to total.

$\therefore$ Count rate (CR) $\propto$ actual neutron population (P)

or CR = N P   N = proportional constant.

   (b) By substitution we find:

eq(1) $P = \frac{So}{1-K}$

Subcritical Multiplication Factor – The amount by which source neutron population is multiplied by as result of presents of fuel in subcritical Rx.

as $1/M$ approach zero

Keff approaches 1

eg(2) $CR = N\,p$ or $P = \frac{CR}{N}$

Substitution:

eg(3) $\frac{CR}{N} = \frac{S_0}{1-K}$ or $\boxed{CR = \frac{N\,S_0}{1-K}}$

Not needed for solving for rest. Constant of indicator

Subcritical

7. Multiplication Factor (M)

Looking at $\frac{CR_2}{CR_1}$ we realize that this is nothing more than the multiplication factor

Know

$$M = \frac{CR_2}{CR_1} = \frac{1-K_1}{1-K_2}$$

$P = \frac{S_0}{1-K}$ ← source strength; equil. population.

$CR = \frac{N\,S_0}{1-K}$

∴ Note: (1) as CR_2 becomes larger K2 approaches 1.

(2) if we invert the equation:

$$\frac{1}{M} = \frac{CR_1}{CR_2} = \boxed{\frac{1-K_2}{1-K_1}}$$

$CR_2 \to \infty$; K_2 approaches 1 as $CR_2 \to \infty$; whole Term → 0

note that as CR_2 now increases K_2 again approaches 1, but $\frac{1}{M}$ approaches 0. ← zero

8. Examples: Assume a reactor with all control rods in core and a Keff = 0.9. Find new Keff for this reactor if rods are pulled such that the count rate increases 20 times the count rate observed for Keff = 0.9

$$\frac{C_2}{C_1} = \frac{1-K_1}{1-K_2}$$

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B. Examples (con't)

$$C_2 = 20C_1 \Rightarrow \frac{20\cancel{C_1}}{\cancel{C_1}} = 20$$

$K_1 = 0.9$

$K_2 = ?$

$$20 = \frac{1-0.9}{1-K_2}$$

$$1-K_2 = \frac{0.1}{20}$$

$$K_2 = 1-.005$$

$$\underline{\underline{K_2 = .995}}$$

(2) Prior to pulling control rods the count rate meter indicates 10CPS. After withdrawing three rods the count rate is 20CPS. The next three control rods are withdrawn and the count rate is now 40CPS. The shutdown Keff of the core was 0.9 when all the rods were inserted. Which rod group withdrawn inserted the greatest amount of reactivity?

<u>1st - 3 rod group</u>

$$\frac{C_2}{C_1} = \frac{1-K_1}{1-K_2}$$

C_1 = 10CPS

C_2 = 20CPS

K_1 = 0.9

$$K_2 = 1- \frac{C_1\ (1-K)}{C_2}$$

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$K_2 = 1 - \frac{(10)(.1)}{20}$

$K_2 = 1 - .05$

$K_2 = .95$ $\Delta K = .95 - .9 = \underline{.05}$

<u>2nd - 3 rod group</u>

$\frac{C_3}{C_1} = \frac{1-K_1}{C_3}$ $C_3 = 40$

$C_1 = 10$

$K_1 = .9$

$K_3 = 1 - \frac{C_1\ (1-K_1)}{C_3}$ $K_3 = 1 - \frac{10\ (.1)}{40}$

$K_3 = 1 - .025$ $K_3 = .975 \therefore \Delta K = .975 - .95 = \underline{.025}$

$\therefore$ First group was worth the most

A shutdown reactor has a source range reading of 270 CPM. The operator begins to pull one of the three supposedly equal rods. The count rate rises to 450 CPM. The operator now pulls the second of the three rods an equal amount and the count rate rises to 1350 CPM. Does the much greater count rate show that the rods are not equal or is this to be expected?

Explain:

Ans: This is to be expected. The closer the reactor is to the critical, the more the countrate will increase for an equal reactivity insertion.

Proof

Remember: $CR = \dfrac{So}{1-K}$

Steps: (1) Since we are only interested in finding ΔK we can assume in this problem any value of So and then use this value to find value of K for each CR.

(2) Determine ΔK for each step. If they are equal, then rods had equal value.

Step 1:

$C_1 = \dfrac{So}{1-K_1}$ Let So = 10 CPM, C_1 = 270 CPM

then $K_1 = 1 - \dfrac{So}{C}$

$K_1 = 1 - \dfrac{10}{270} = 1-0.0372$

$K_1 = .9628$

Step 2:

Let So = 10
$C_2 = 450$

$K_2 = 1- \dfrac{10}{450} = 1-0.0222$

$K_2 = 0.9773$

Step 3:

Let $So = 10$
$C_3 = 1350$

$$K_3 = 1- \frac{10}{1350} = 1-0.0074$$

$$K_3 = 0.9926$$

Step 4:

Determine ΔK between withdrawals

$$\Delta K_1 = K_2 - K_1 = 0.9778 - 0.962 = \underline{0.0148}$$

$$\Delta K_2 = K_3 - K_2 = 0.9926 - 0.9778 = \underline{.0148}$$

$\therefore \quad \Delta K_1 = \Delta K_2$ and rod groups are equal.

4. During fuel load the Keff of the core is determined to be 0.98 with the count rate at 45 CPS. After several additional elements are added a count rate of 180 CPS is observed. What is Keff?

$$\frac{C_2}{C_1} = \frac{1- K_1}{1- K_2}$$

$$\frac{180}{45} = \frac{1-.98}{1-K_2}$$

$$4 = \frac{.02}{1-K_2}$$

$$1-K_2 = \frac{.02}{4}$$

$$1-K_2 = .005$$

$$K_2 = \underline{\underline{.995}}$$

5. A reactor shows 10CPS during startups with a Keff of 0.9. What will the counts per second be when Keff is 0.975?

$$\frac{C_2}{C_1} = \frac{1-K_1}{1-K_2}$$

$$C_2 = \frac{(10)\ (.1)}{.025} = \frac{1}{.025} = \underline{\underline{40\ CPS}}$$

V. Reactor Period vrs. Power Level

A. Concept

1. A subcritical reactor increases power eponetially. Therefore, power increase with time in a subcritical reactor can be expressed as:

$$P = Po\ e^{\Delta t/\tau}$$

where:

Po = initial power

P = power at some later time (t)

Δt = change in time between initial power level "Po" and final power level "P".

τ = reactor period (Sec)

$$\phi_x = \phi_0 e^{\Delta t/\tau}$$

2. Reactor Period (τ): is defined as the length of time for the reactor power to increase by a factor of "e" (2.717)

3. Example : If the reactor is initially at 100 watts when control rods are withdrawn such that the reactor is increasing power on a 30 second period. Determine at what power the reactor will be at 1 minute later.

$$P = Po\ e^{\Delta t/\tau}$$

Po = 100 watts

Δt = 1 min. or 60 seconds

τ = 30 seconds

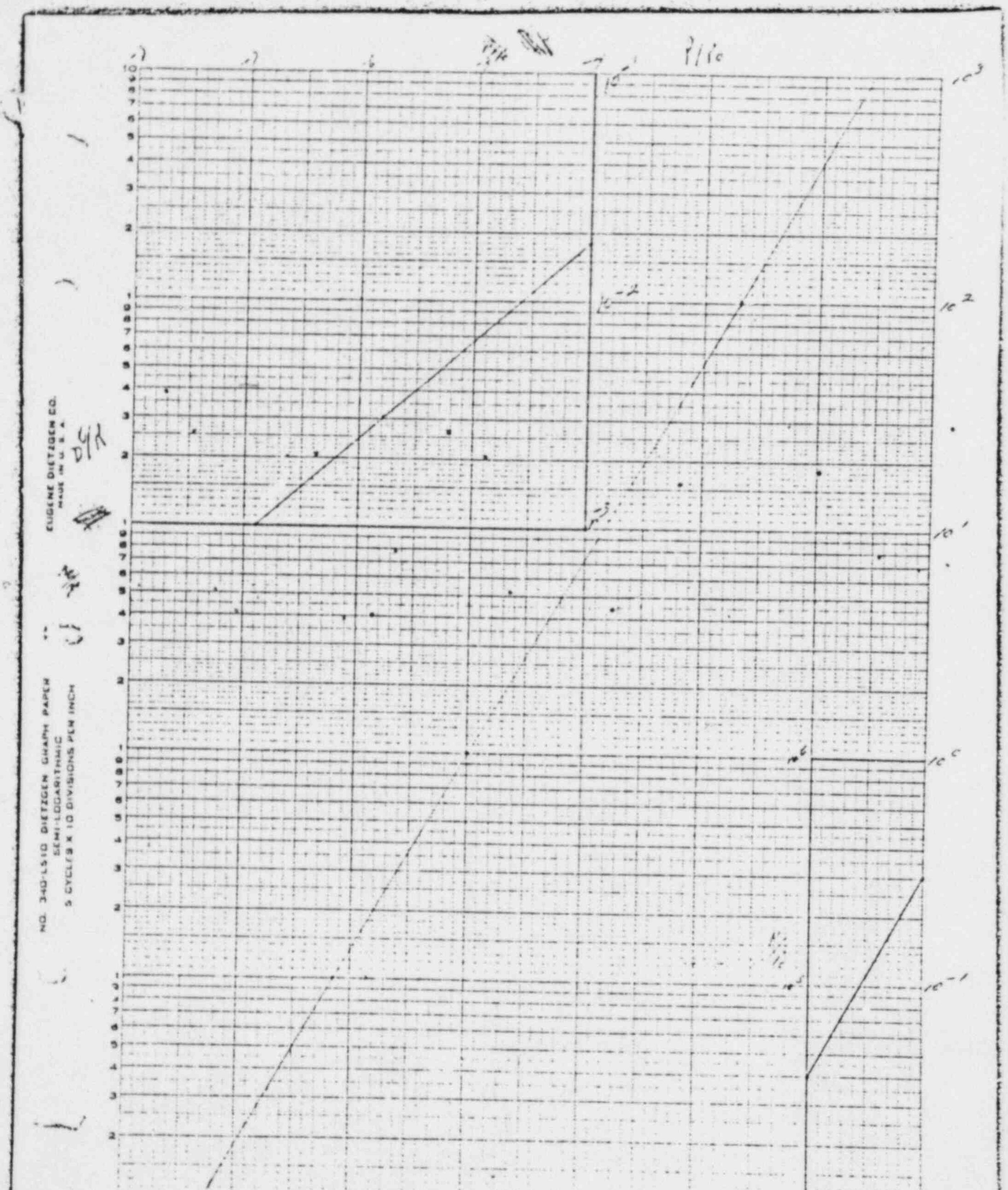
$$e^{\,\cdots} \qquad \ln \frac{P}{P_0} = e^{\left(\frac{\Delta t}{\tau}\right)}$$

$$\ln x = y \qquad P = P_0\, e^{\frac{\Delta t}{\tau}} \qquad P = P_0 \ln \frac{\Delta t}{\tau}$$

$$\frac{P}{P_0} = e^{\frac{\Delta t}{\tau}}$$

$$\ln \frac{P}{P_0} = \frac{\Delta t}{\tau} \qquad = e^{\frac{\Delta t}{\tau}}$$

$$\ln \Delta t/\tau = e^{\Delta t/\tau}$$


NO. 340-L510 DIETZGEN GRAPH PAPER
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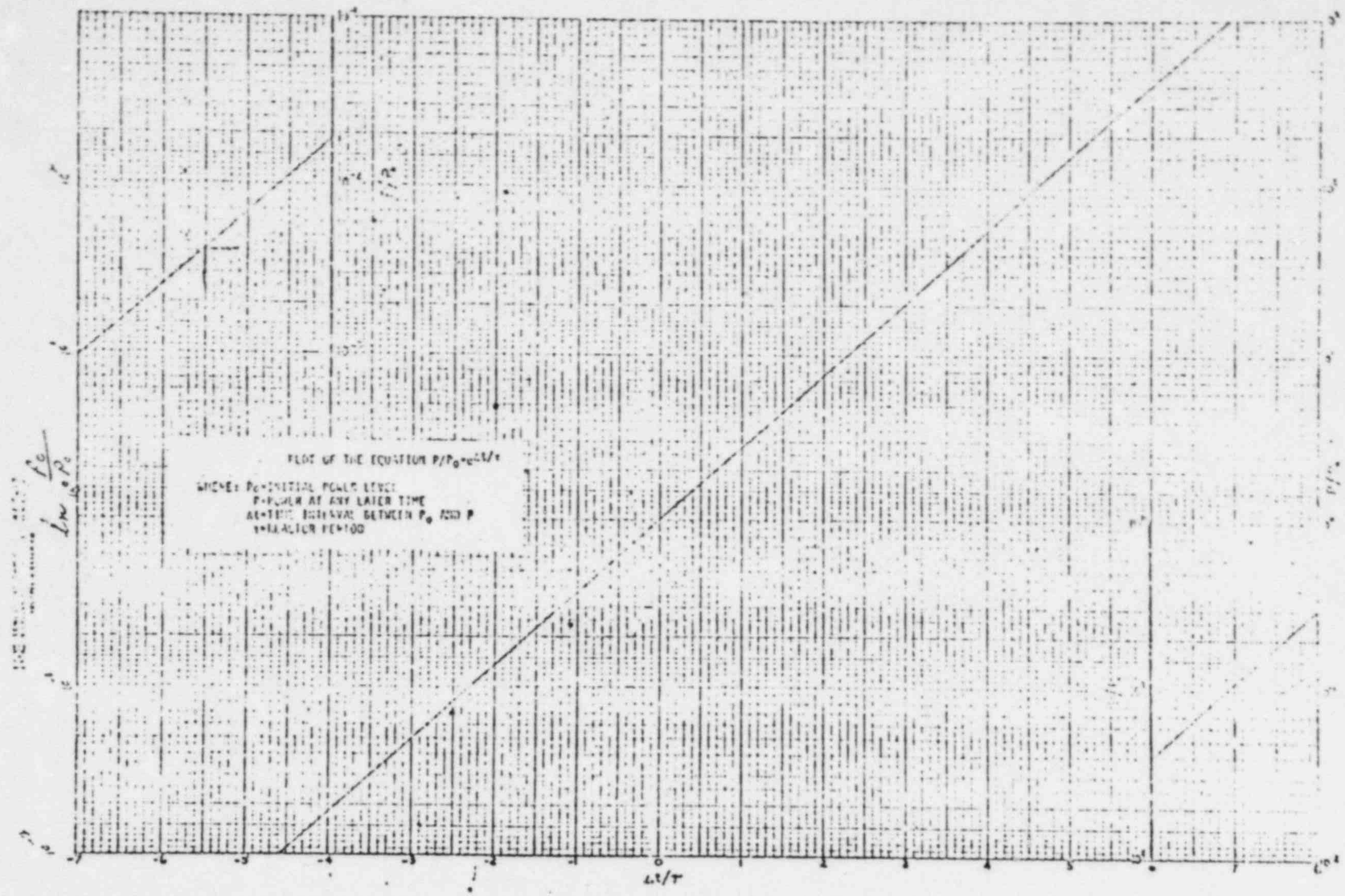
PLOT OF THE EQUATION P/P0=e^Δt/τ
WHERE: P0=INITIAL POWER LEVEL
P=POWER AT ANY LATER TIME
Δt=TIME INTERVAL BETWEEN P0 AND P
τ=REACTOR PERIOD
Δt/τ
-7
-6
-5
-4
-3
-2
-1
0
1
2
3
4
5
6
7

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[illegible]

[illegible]

[illegible]

4. Since a table of exponentials or logarithms is required, it is sometimes convenient to use graphs. One finds that a plot of P/po vrs $\Delta t/\tau$ yields a straight line on semi-log paper and is therefore convenient to me.

5. Doubling Time (DT) : Is defined as the time required to increase or decrease reactor power by a factor of 2. The relationship between reactor period " τ " and DT is as follows:

$$\tau = \frac{DT}{0.693}$$

derived from $P = P_0 e^{\frac{\Delta t}{\tau}}$

$\frac{P}{P_0} = e^{\frac{\Delta t}{\tau}}$

doubles $2 = e^{\frac{\Delta t}{\tau}}$

$\ln 2 = \frac{\Delta t}{\tau}$

$.693 = \frac{\Delta t}{\tau}$

Ex: If reactor power doubles in 50 seconds what is the reactor period?

$$\tau = \frac{50}{0.693} = 72 \text{ seconds}$$

6. "Rule of thumb" for power changes less than 20%.

$$\tau = \frac{\Delta t}{(P/po) - 1}$$

B. <u>Startup Rate (SUR)</u> : is defined as the rate of changes of reactor power expressed in decades per minute (DPM), where a decade is a factor of 10.

$$\tau = \frac{(ld) \text{ life time neutron}}{K_{eff} - 1}$$

average
life time of neutron
is 5×10^{-5} sec for
prompt neutrons

.13 sec average
total life time of neutron
for thermal neutrons

total average of two is
.1 sec

% yield (β_{eff}) average is .65%.

% yield at TMI (β_{eff}) is .69% to be used in calculations.

When ever ρ addition is greater than β_{eff} the Rx is Prompt Critical. (Critical on fast neutrons alone) incase of TMI $\rho_{addition}$ > .0069 the Rx will go or be Prompt Critical.

(a) The relationship between an increase or decrease of reactor power and SUR is given as follows:

$$P = (Po)(10)^{(SUR)(\Delta\tau)}$$

where: SUR = startup rate in decades per minute

$\Delta\tau$ = time interval between Po and P in minutes.

Po = initial power

P = final power

(b) The relationship between SUR and period "τ" is also given as:

$$\tau = \frac{26}{SUR}.$$

where: SUR = startup rate in DPM

τ = reactor period in seconds.

C. Prompt neutron generation time

1. Recall: Generation lifetime is the average length of time between successive neutron "births".

2. Total prompt neutron generation time is composed of three parts:

(a) Avg. time required to slow newly born fission neutrons down to thermal energies (approx. 7×10^{-5} sec)

(b) thermal lifetime between thermalization and death by absorption or leakage (approx. 3.5×10^{-5} sec)

(c) Avg. time between absorption of thermal neutrons and the birth of a new fast neutron (approx. 10^{-12} sec)

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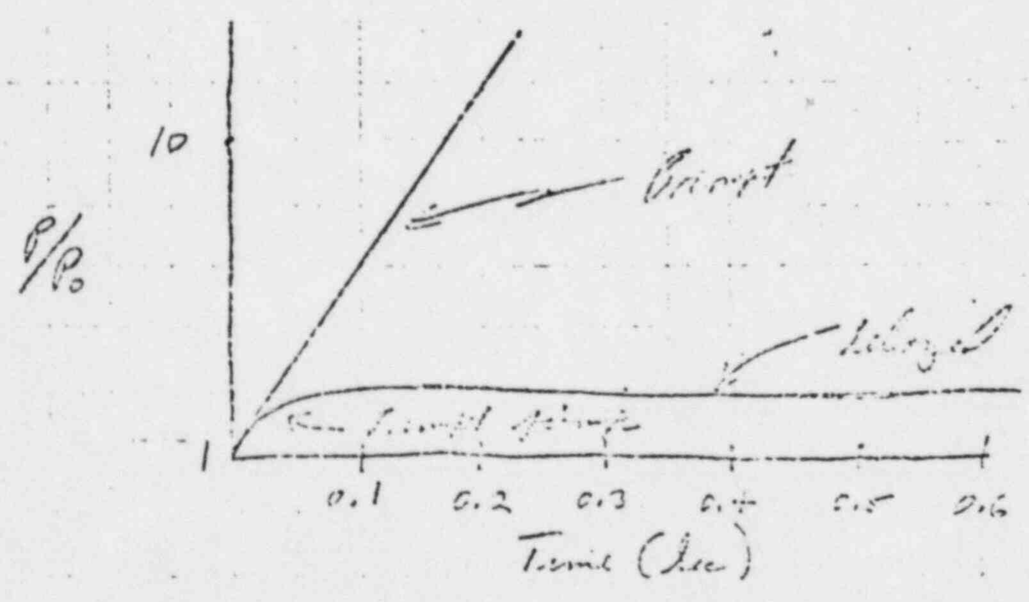
∴ Total prompt neutron generation time approx. 4.2 x 10^{-5} sec.

D. Effect of Delayed neutrons upon reactor behavior

1. Since delayed neutrons are produced from the decay of fission fragments, they will be produced sometime later than that of the prompt neutrons.
2. Delayed neutrons markedly increase the average generation time due to the long time delay involved in the birth of delayed neutrons. Therefore they increase the core average generation time from about 4 x 10^{-5} sec (which is the value in a prompt reactor) to about 0.08 - 0.10 seconds. This is enough time delay to allow an operator and the inherent control mechanism to safely control the reactor.



These equations relate τ & ρ

$$\boxed{\overset{\text{Short}}{\tau < .1\,\text{sec}} \qquad \rho = \frac{l}{k_{eff}\,\tau} + \beta}$$

β prompt neutrons most important.

$$\boxed{\tau > 1\,\text{sec} \qquad \rho = \frac{\bar{\beta}}{1+\lambda\tau}}$$

delay neutrons most important (applies to a shut down condition)

l^* average of Generation Time

$$\boxed{\rho = \frac{l^*}{\tau} + \frac{\beta}{1+\lambda\tau}}$$

$$\boxed{\rho = \frac{l}{k_{eff}\,\tau} + \frac{\bar{\beta}}{1+\lambda\tau}}$$

Good for every case

Takes into account the effect of Delayed Neutrons.

$\bar{\beta}$ same as β_{eff}

.0835 sec

$\lambda = .08\,\text{sec}^{-1}$ (Critical Rx)

$$\boxed{\tau = \frac{l + (\bar{\beta} - \rho)\tau}{\rho}}$$ when $\frac{\beta}{2} < \rho < \beta$

(average life time of delayed Neutron Precursors.)

$\tau = 13$ sec

$\lambda = .0125\,\text{sec}^{-1}$ (Shut down)

$l = 5 \times 10^{-5}$ sec

$$\boxed{\tau = \frac{\bar{\beta} - \rho}{\rho\lambda}}$$

$\rho < \frac{\beta}{2}$ Negative ρ (Shut Down)

$\tau = \frac{1}{\lambda}$

$$\boxed{\tau = \frac{l}{\rho}}$$

$\rho > \bar{\beta}$ prompt critical

l^* Average Generation Time changes, depending on media used or ...

l Prompt neutron life time is same always set value through experimentation of approx. 5×10^{-5} sec.

3. tial rapid rate of rise of neutron power due to oduction.

E. Prompt Critical defined as reactor criticality on prompt neutrons alone.

Condition:

when P_{excess} B

F. Relationship between Reactivity and Period :

1. The inhour equation : an equation relating reactivity and reactor period.

$$\boxed{P = \frac{\ell}{K_{eff}\,T} + \frac{\beta_{eff}}{1 + \lambda T}}$$

Condition

All

where: ℓ = Prompt neutron lifetime for U^{235}

T = Reactor Period

Beff = effective delayed neutron fraction

λ = Weighted decay constant of 6 delayed neutron groups. (.08 sec^{-1})

2. Rearranged for period :

Condition

$$T = \boxed{\frac{\ell}{P} + \frac{(B - P)}{\lambda P}}$$ when → $\frac{B}{2}$ < P < B

where: T = Weighted extreme (12.6 sec) P = reactivity ($\Delta K/K$

Condition

$$T = \frac{(\bar{B} - P)}{P\,\lambda}$$ P < B/2

$$T = \frac{\ell}{P}$$ P > B

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2. Example : (1) Determine the reactor period associated with a reactivity insertion of 0.001 Δ K/K to a just critical reactor

Given : at BOL , Beff = .00589 , ℓ* = 23 Us

$$T = \frac{\ell^* + (\bar{B} - P)\tau}{P}$$

$$T = \frac{23 \times 10^{-6} + (.00689 - .001)12.6}{.001}$$

$$T = \frac{23 \times 10^{-6} + (.00589)(12.6)}{10^{-3}}$$

$$T = \frac{-6 + 7.42 \times 10^{-2}}{10^{-3}} = \underline{74.2 \text{ sec}}$$

2. If a reactor is on a 10 second period, determine the excess reactivity of the core. Given : ℓ * = 23Us, Beff = 0.00689

$$T = \frac{\ell^* + (B-P)\tau}{P}$$

$$PT = \ell^* + (B-P)\tau$$

$$PT = \ell^* + B - P\tau$$

$$PT = \ell^* + B\tau$$

$$P(T + \tau) = \ell^* + B\tau$$

$$P = \frac{\ell^* + B\tau}{T + \tau} = \frac{23 \times 10^{-6} + (.00689)(12.6)}{10 + 12.6}$$

$$P = \frac{8.66 \times 10^{-2}}{22.6} = 0.383 \times 10^{-2}$$

$$= 3.83 \times 10^{-3} \ \Delta K/K$$

3. Since Beff = 0.00156 at EOL due to Pu-239 buildup, determine what the reactor period in example #1 would be at EOL for the same reactivity insertions.

$$T = \frac{(B-P)\tau}{P}$$

$$T = \frac{(.00156 - .001)(12.6)}{.001}$$

$$T = >1 \text{ sec}$$

NOTE: the shorter period that results at EOL for the same reactivity insertion.

4. Determine the reactor period for a reactivity insertion of .006 K/K to just critical reactor.

Given · $\ell^* = 10^{-6}$, Beff = .005.

$$T = \frac{\ell^*}{P}$$

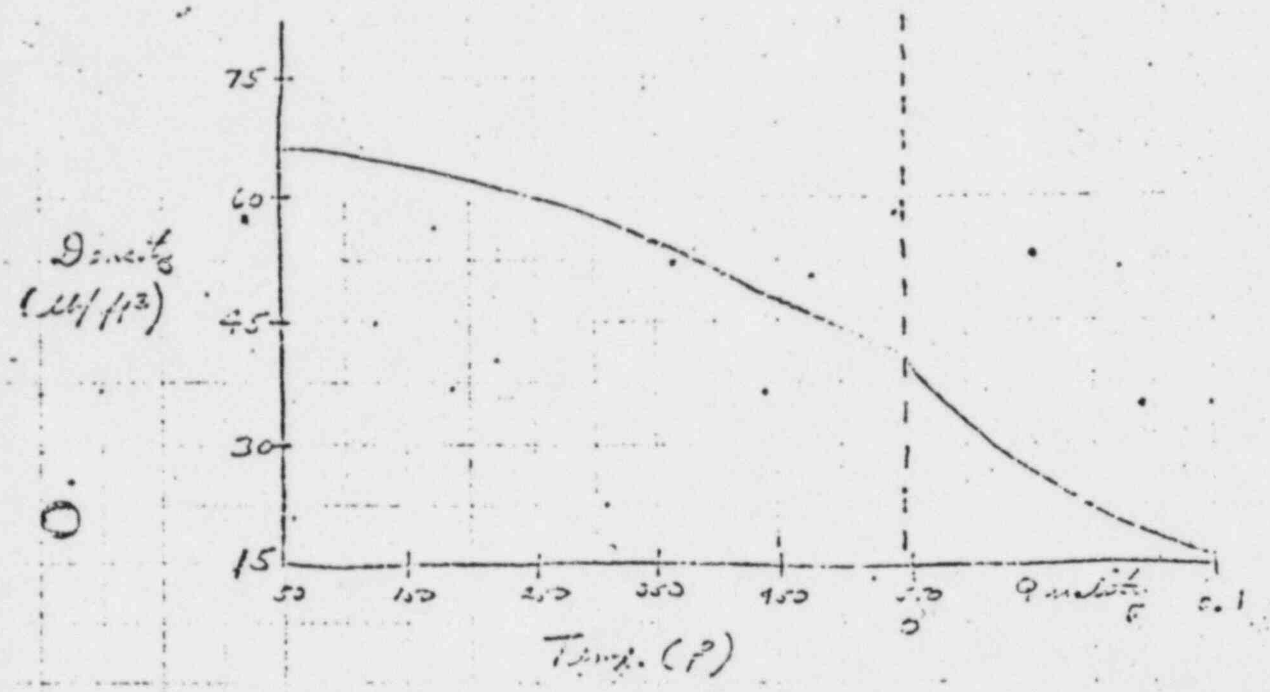
$$T = \frac{10^{-6} \text{ sec}}{.005} = 1.67 \times 10^{-4} \text{sec.}$$

note: P > Beff ∴ Prompt critical

VI. Reactivity Coefficients

A. Moderator Temperature Coef.

1. As the temp. of the water moderator is increased, its density decreases.



2. Since the volume of the core which is occupied by the moderator remains constant, a reduction in moderator density means that there is a reduction in the number of moderator molecules in the core.
3. Note: The number of fuel molecules in the core remain constant as the temperature increases therefore, <u>the next result of a temperature increase is that the moderator to fuel ratio</u> (on a weight or molecule basis) <u>decreases.</u>

4. Summary of the effect of a decrease in moderator density.

FACTOR	EFFECT	IMPORTANCE: BOL	EOL
η	Neutrons are not thermalized as efficiently because there are fewer moderator molecules available and the molecules have a higher vibrational energy. Thus average energy of slow neutrons is increased.	Negl'gable negative effect (η for U-235 nearly independent of neutron energy).	Small negative effect (η for Pu-239 decreases faster than "η" for U-235 as neutron energy increases).
ε	Neutrons not slowed down as quickly because there are fewer moderator molecules available. Therefore, neutrons spend more time at high energies and more fast fissions occur.	← Medium Positive effect →	
p	Decreased moderator efficiency means that neutrons spend more time with resonance energies and resonance capture increases.	← Strong Negative effect →	
f	1. Removal of water from core reduces capture in water.	← Medium Positive effect →	
	2. Avg. slow neutron energy is increased. More neutrons are captured in Pu-239 resonance at 0.29ev.	← Small Positive effect →	
	3. Slow neutrons travel farther and more are able to reach the surface of inserted control rods. Worth of inserted rods increases.	← Medium Negative effect. Throughout core lifetime nearly all control rods are withdrawn when at full power. →	

FACTOR	EFFECT	IMPORTANCE BOL	EOL (Consistent)
	4. As moderator expands dissolved boric acid is removed from the core so there is less neutrons absorption in chemical shim.	If chem shim in moderator is high, positive effect.	Negligible negative effect. Nearly all chem shim has been removed from moderator at EOL.
$\mathcal{L}_f$	Less moderator means neutrons travel farther during slowing down process. Leakage increases.	← Small to Medium negative effect →	
$\mathcal{L}_t$	Reflector ability of water is reduced so slow neutrons travel farther. Leakage increases.	← Small negative effect →	

5. <u>Definition of moderator temperature coefficient:</u> (α_m)

"A change in reactivity per degree change in moderator temperature."

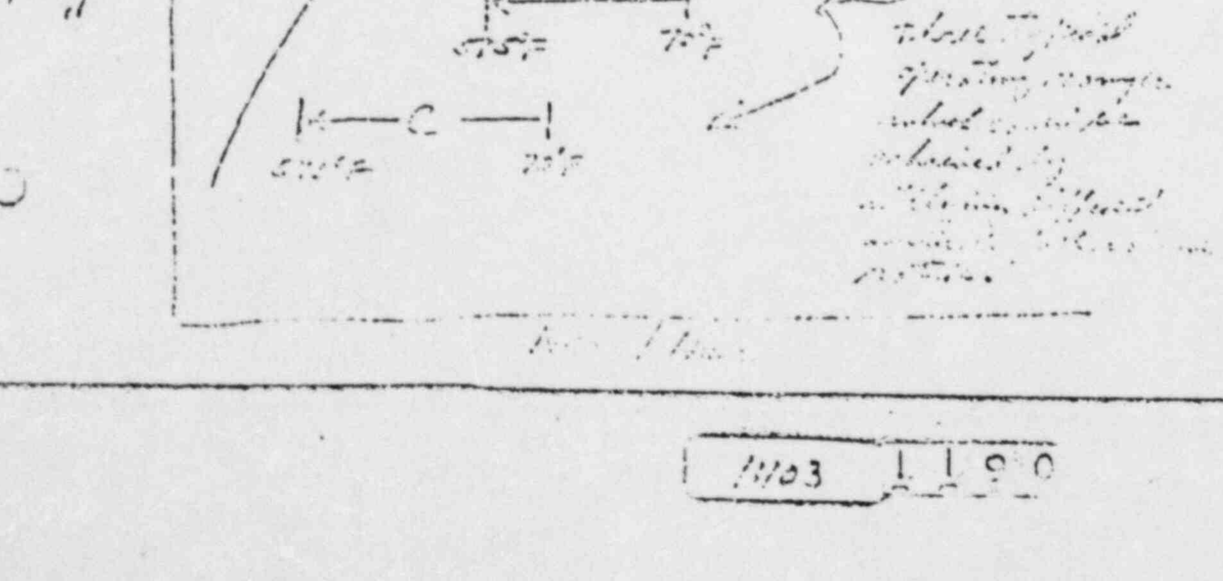
6. <u>Example:</u>

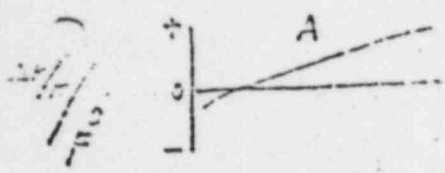
Assume Keff = 1.000 (ρ = 0 Δ K/K) and moderator temperature = 400°F. Suppose the moderator temperature is raised to 450°F without making any other change in the reactor. If Keff now decreases to 0.995 then the moderator temp. coefficient is negative and has the following value:

$$\alpha_m = \frac{\Delta\rho}{\Delta T} = \frac{0.005 - 0}{450 - 400}$$

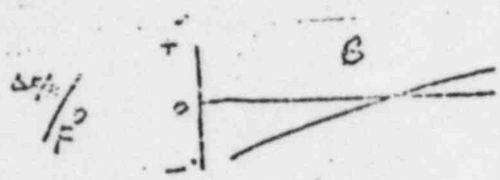
$$\alpha_m = \frac{-5 \times 10^{-3}}{5 \times 10^{1}} = -1 \times 10^{-4} \ \Delta K/K \Big/ {}^{\circ}F_m$$

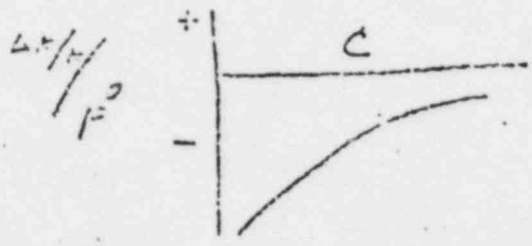
7. Moderator Temp. coefficient over operating ranges.





Corresponding Moderator temperature coefficients





a). PWR Considerations

1. At BOL considerable concentrations of boric acid is in the coolant.

   NOTE Any mechanism which removes moderator containing boron from the core will result in the simultaneous removal of an equivalent amount of poison. The removal of poison will tend to increase Keff and will represent a positive contribution to the temperature coefficient. The greater the concentration of boric acid, the more important is this effect and the more the moderator temp coeff. tends to be positive.

2. The moderator coefficient gets more negative over core age as chemical shim is removed from the core.

- See diagram on next page -

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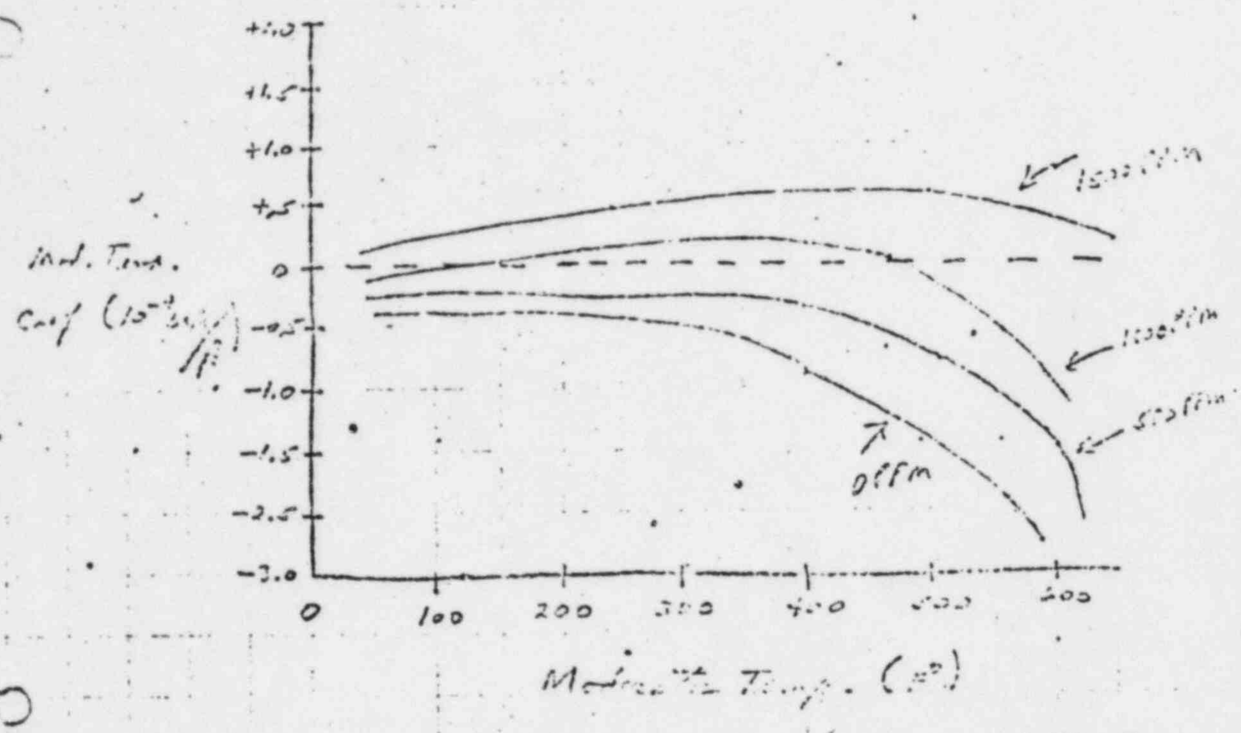


B. <u>Moderator Void Coefficient</u>

1. Moderator Coefficient is defined as: the change in reactivity associated with a unit change in the moderator void.

$$\alpha_V = \frac{\Delta \rho}{\%\ \text{void}}$$

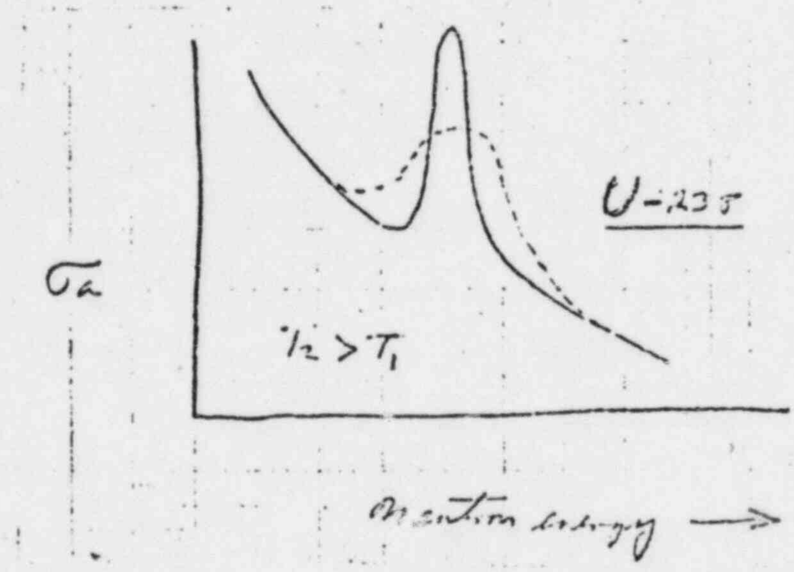
2. Since the moderator in a PWR reactor is operated in the subcooled region due to operations at high pressure, a high degree of boiling is impossible.
3. Surface boiling does occur in a PWR and at full power may result in a void volume fraction of approx. 0.5% in the coolant.
4. Void coefficient at BOL is slightly negative. Range over core life is approx. -0.03 to -0.10% $\Delta K/K$ / % void

1403 1192

C. Fuel Temperature Coefficient

1. The fuel temperature coefficient is referred to as the "doppler coefficient". $\alpha_F = \frac{\Delta P}{\Delta T_F}$

2. Doppler Effect: is the broadening and flattening of the U-238 absorption cross section resonance peaks as U-238 temperature increases.

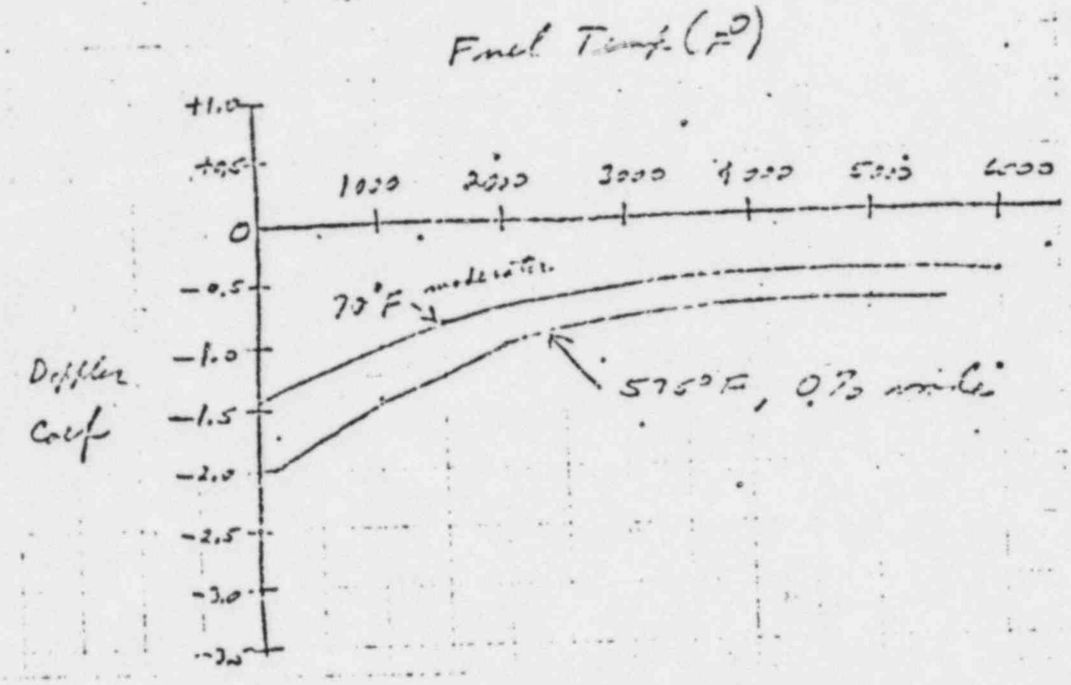


3. In a PWR (large amount of U-238) this effect is great yielding a fuel temp. coefficient in the order of -10^{-5} K/K $_F{}^{o}$.

4. The fuel temp. coef. acts immediately upon a change in fuel temp. and is the shutdown mechanism for a fast power rise transient. Therefore, called a prompt coefficient as compared with moderator or void coefficients.

5. Influence of moderator temp. on fuel temp. coef.

   (a) As mod. temp. increases the average energy of the slow neutrons increases.



   (b) The absorption cross section of the surface of the fuel rod decreases which allows more slow neutrons to penetrate into the centers of the fuel rods. Therefore, the U-238 nuclei in the center of the rod gets a greater chance to contribute to the droppler coef. making it more negative.

6. As burnup of fuel occur over core age, the fuel temp. coef. becomes more negative. This is because of the buildup of resonances.

1195 11103

$$\rho = \frac{k_{o\,ss} - 1}{k_{o\,ss}} = \frac{\Delta k_\infty}{k_\infty} = \frac{F' - F}{F'}$$

$$F = \frac{V \Sigma_o \Phi_o}{V \Sigma_o \Phi_o + V \Sigma_m \Phi_m}$$

$$F' = \frac{V \Sigma_o \Phi_o}{V \Sigma_o \Phi_o + V \Sigma_a \Phi_m + V \Sigma_a \Phi_r}$$

6. Con't

in the fuel which were not originally present. The most important contributions to these resonances is that from Pu-240 which has a resonance of $\simeq 10^5$ barns at 1.03 ev.

VII. Reactor Poisons

A. Xenon Poison

1. Xe-135 is one of 200 possible fission products.
2. Xe-135 has a thermal microscopic absorption cross section σ_a = 2,700,000 barns.
3. Production of Xe-135 is by two methods:
   (a) Directly as a fission product (0.3%) the is the number of Xe-135 nuclei formed per 100 fissions.
   (b) Through the decay (by successive B⁻ emissions) from Te-135 (formed in 5.9% of all fissions) to Xe-135.
4. Te-135 decay scheme

$$_{52}Te^{135} \xrightarrow[11\ sec]{B^-} {_{53}I^{135}} \xrightarrow[6.7\ hr]{B^-} {_{54}Xe^{135}} \xrightarrow[9.2\ hr]{B^-}$$

$$_{55}Cs^{135} \xrightarrow[2 \times 10^6\ yr]{B^-} {_{56}Ba^{135}} \text{ (stable)}$$

NOTE: Since Te-135 has a short half-life as compared to Iodine-135, we say that for all practical purposes, I-135 appears as the only direct fission product. Thus we can say that Xe-135 production comes from two major sources:

(i) Direct Fission $\frac{0.3\%}{5.9\%}$ = 5%

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$$\frac{dX_e}{dt} = \lambda_{Xe} X_e - \sigma_{Xe}\,\phi X_e + \lambda_I I + \gamma_{Xe}\,\Sigma_f\,\phi$$

(2) Decay of I-135 $\frac{5.6\%}{5.9\%} = 95\%$

5. <u>Removal</u> of Xenon-135 occurs through two methods:

   (a) By radioactive decay of Xe-135 to Cs-135

   (b) By capture of a neutron resulting in the formation of stable Xe-136.

6. Net Amount of Xenon in the core is dependent upon the relative rates of formation and removal.

$$\text{Rate of change of Xenon level} = \left[\begin{array}{l}\text{Rate of Prod. from direct}\\ \text{fission and from decay of}\\ \text{iodine}\end{array}\right] - \left[\begin{array}{l}\text{Rate of removal}\\ \text{by decay and by}\\ \text{neutron capture}\end{array}\right]$$

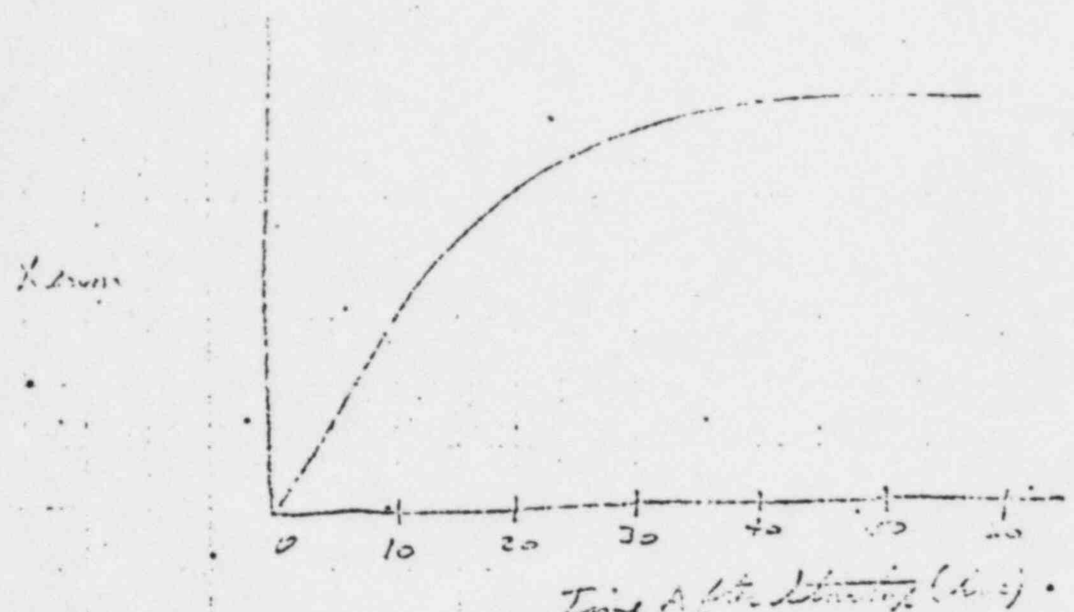
OR

$$\frac{\Delta Xe}{\Delta T} = \underbrace{\left[\lambda_I N_I + \Sigma_f \phi \, \gamma_{xe}\right]}_{\text{PRODUCTION}} - \underbrace{\left[N_{xe} \sigma_a^{xe} \phi + \lambda_{xe} N_{xe}\right]}_{\text{DECAY}}$$

γ_{xe} = fission yield = 5.9%

7. <u>Reactor Startup from cold clean</u> <u>(no initial Xenon-135 present)</u> <u>Conditions</u> to equilibrium.

- See Diagram -



a) Time to reach equilibrium is dependent upon the power density and therefore varies for various reactors. However, in practice one usually considers equilibrium to be reached in aprox. 40 hrs.

b) Explanation of equilibrium buildup:

(1) As soon as reactor reaches power Xe begins to form due to direct fission. In addition a large amount of I-135 begins to be formed and Xe begins to appear at a gradual rate from I-135 decay.

(2) At the beginning there is little Xe in the core (N_{Xe} is small) therefore, the removal mechanisms are too small to balance the production of Xenon and so Xe concentration increases. However, as N_{Xe} gets larger, removal by both decay and burnup increases, tending to cancel the production and causing the curve to start leveling out. When the removal completely catches up to the production, equilibrium is reached.

greater fuel inventory needed due to xenon build up & peaking. the greater fuel inventory allows overriding this poison addition at its peak.

1200 11/03

c) At equilibrium the rate of change of Xe is zero therefore:

$$0 = \lambda_I N_I + \Sigma_f \phi \gamma_{xe} - N_{xe} \sigma_a^{xe} \phi + \lambda_{xe} N_{xe}$$

or at equilibrium

$$N_{xe}\ (\text{at equil}) = \frac{\lambda_I N_I + \Sigma_f \phi \gamma_{xe}}{\sigma_a^{xe} \phi + \lambda_{xe}} = \frac{(\gamma_{xe} + \gamma_I)\Sigma_f \phi}{\lambda_{xe} + \sigma_{a_{xe}} \phi}$$

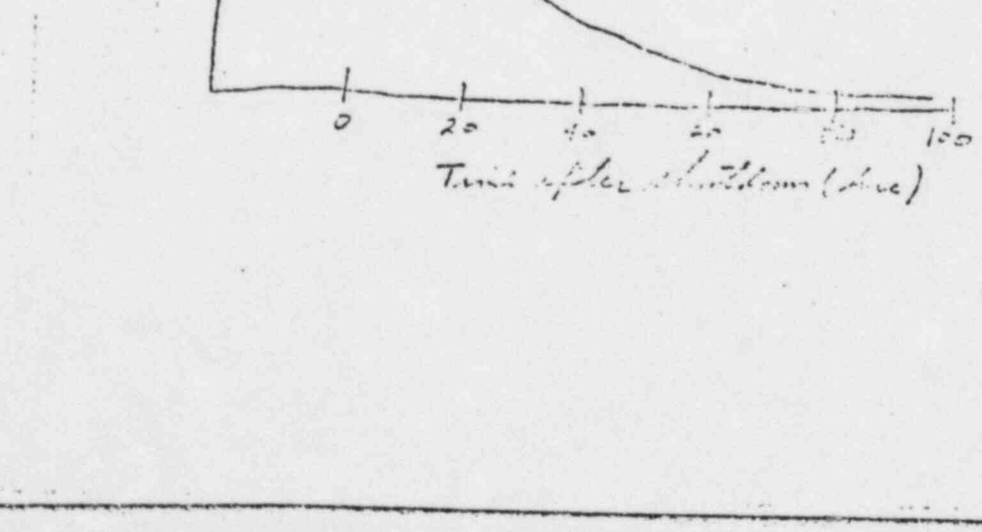
.3 ; 5.9 ; 6.3

$2.7 \times 10^{6} b = \sigma_{a_{xe}}$ $\lambda_{xe} = 2.1 \times 10^{-5}\ sec^{-1}$

(1) Note: Xenon equilibrium concentration depends upon neutron flux (Ø).

d) Full power equilibrium Xenon reactivity worth is aprox. -2.7% $\Delta k/k$

8. Reactor Shutdown from equilibrium Xenon:



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a) After shutdown it takes 8 to 10 hours for Xenon to peak in core. Total Xenon worth is aprox. 5.3% at peak. Xenon reaches peak and decays off to approx. zero in 100 hrs.

b) Explanation of Xenon peak.

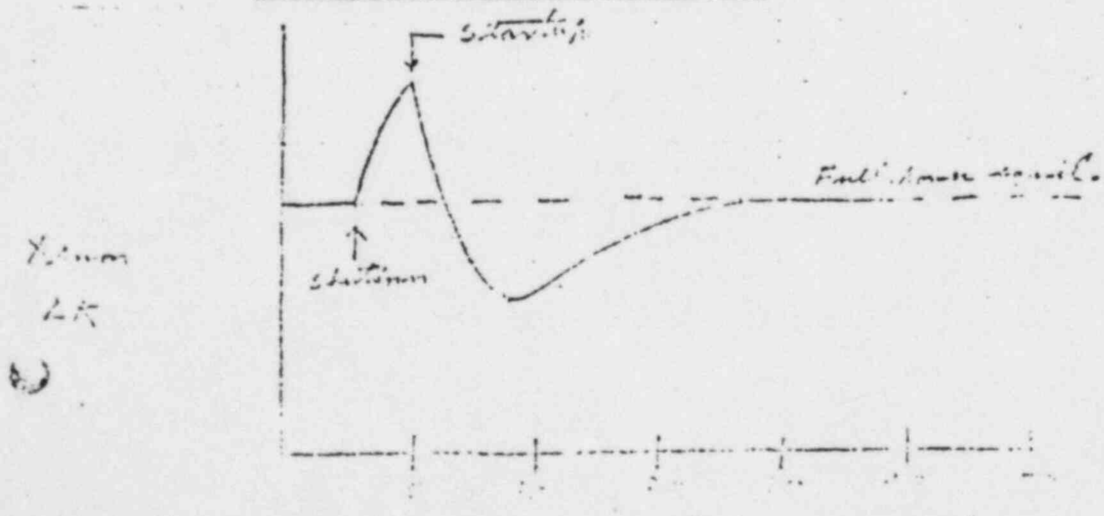
(1) At time of reactor shutdown equil. of I135 and Xe-135 are present.

(2) At shutdown, the production of I-135 and Xe-135 by direct fission stops.

(3) I-135 already in the core then decays to Xe-135. Since the reactor is shutdown, burnup no longer occurs therefore Xe-135 builds up and then decays out.

c) Note: For a reactor to start up at peak Xenon, there must be sufficient reactivity in the control rods to overide peak Xenon. Toward end of core life it may not be possible to overide peak. Xenon therefore, startup after a trip must take place within the first two hours or the reactor must wait to achieve criticality in a day or so.

9. Reactor startup with Xenon in the core:

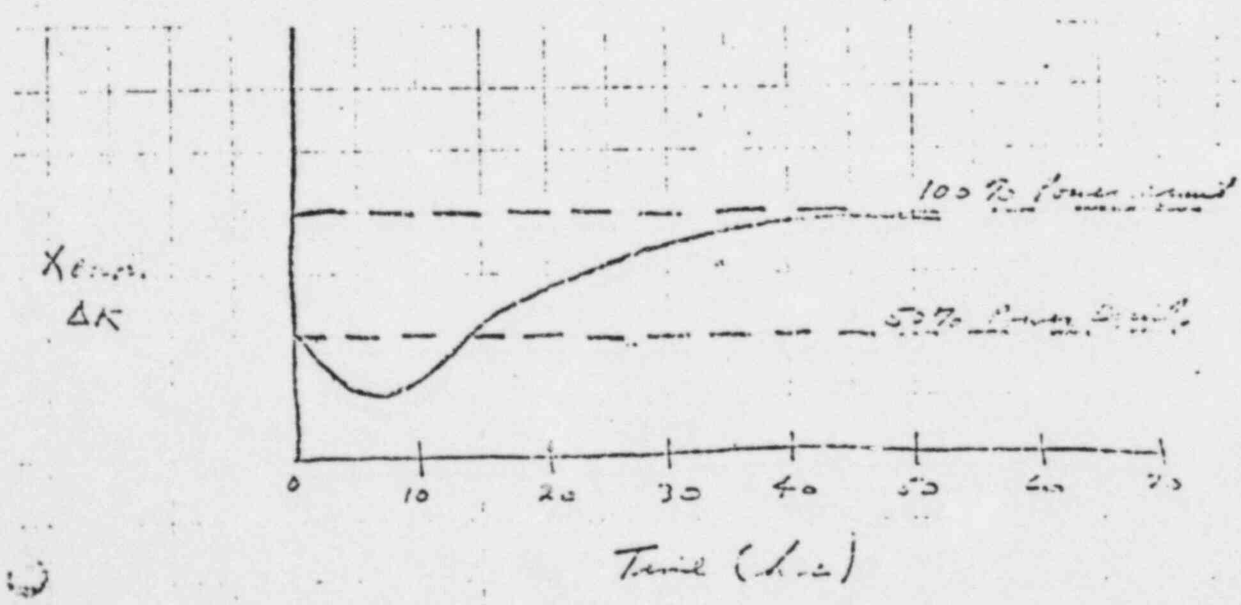


a) Once reactor reaches power, Xenon begins to drop because of neutron capture (as well as decay).

b) Xe-135 rate of production at time of reactor startup has fallen off from its equilibrium value because decay during the shutdown time has depleted the I-135 inventory. Then the rate of Xenon removal at this time exceeds the rate of production.

c) Now the I-135 inventory starts to recover immediately upon the return to power. Its half-life however, introduces a lag of several hours before the Xenon begins to recover. Eventually, the Xenon also returns to equilibrium for the obtained power level.

10. Xenon behavior on load changes



a) Explanation is self evident with respect to information already given.

B. Samarium-149

1. Samarium-149 is another fission product having a large thermal absorption cross section (σ_a = 53,000 barns)
2. It is a stable rather than a radioactive isotope.
3. It is not a direct or primary fission product.
4. It is produced in 1.4% of all fissions by decay of neodiniums-149.

yield

$$\text{Fission} \rightarrow {}_{60}Nd^{149} \xrightarrow[1.7\ hr]{\beta^-} {}_{69}Pm^{149} \xrightarrow[47\ hr]{\beta^-} {}_{62}Sm^{149}\ \text{(stable)}$$

1.4%

5. Startup from a cold clean condition
   a) Sm-149 will build up to an equilibrium value in aprox. 30EFPD (effect full power days.)
   b) Equilibrium reactivity worth is aprox. - 1% $\Delta K/K$
   c) Equilibrium inventory is flux independent because it is a stable isotope. The reason is if because the reactor power doubles, the equilibrium production rate doubles but so does the equilibrium removal rate double without requiring any change in samarium concentration.

yield   production term   removal term

$$\frac{\Delta N_{sm}}{\Delta T} = \gamma_{sm}\,\phi\,\Sigma_f - \phi\,\sigma_a^{sm}\,N_{sm}$$

at equilibrium : $\frac{\Delta N_{sm}}{\Delta T} = 0$

$$0 = \gamma_{sm}\,\phi\,\Sigma_f - \phi\,\sigma_a^{sm}\,N_{sm}$$

$$\text{OR}\quad N_{sm} = \frac{\gamma_{sm}\,\phi\,\Sigma_f}{\phi\,\sigma_a^{sm}} = \frac{\gamma_{sm}\,\Sigma_f}{\sigma_a^{sm}}$$

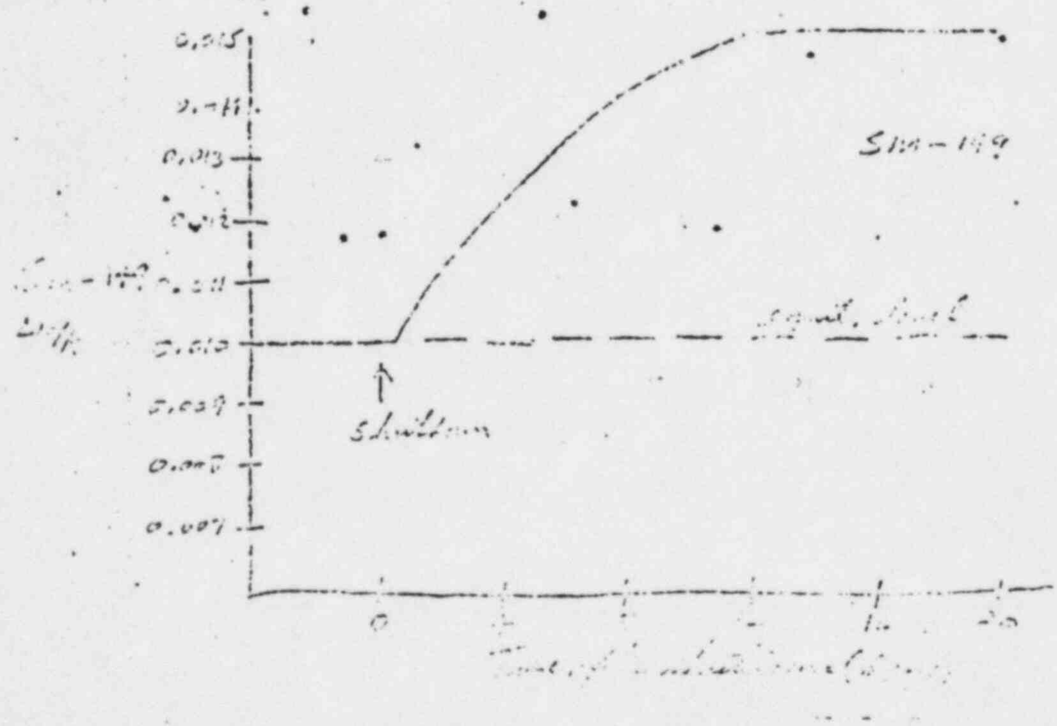
flux independent

1103   1204

6. Samarium-149 buildup after a reactor shutdown

   a) Upon a reactor shutdown Sm-149 builds up beyond its equilibrium value due to the decay of Pm-149 to Sm-149. It does not decay away after this buildup because Sm-149 is a stable isotope.

   b) Total worth in core after shutdown is aprox. 1.5% ΔK/K. (0.5% from buildup from equilibrium).



VIII. Fuel Burnup

   A. Fuel Exposure

      1. Is expressed in megawatt days per ton of uranium (MWD/T).

Thermal Diffusion Equation

$\frac{dN}{dt} = D\nabla^2\phi - \Sigma_a\phi + S$

leakage absorption Source

$\frac{K_\infty}{f} = \Sigma_a\phi$

D - Diffusion term

$S = \eta\epsilon f\Sigma_a\phi$ also $S = K_\infty\Sigma_a\phi$

[illegible] $\rightarrow \nabla^2$ - represents Shape of flux

ϕ - neutron flux

β - delayed neutron yield

$1-\beta$ - prompt " "

(S) Production of neutrons =

$(1-\beta)K_\infty\Sigma_a\phi + \lambda C$

prompt delayed

total no. neutrons changing with time

$$\frac{dN}{dt} = \underbrace{D\nabla^2\phi}_{\text{leakage}} - \underbrace{\Sigma_a\phi}_{\text{absorption}} + \underbrace{(1-\beta)\Sigma_a K_\infty\phi}_{\text{prompt}} + \underbrace{\lambda C}_{\text{delayed}}$$

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2. It describes the amount of energy released (megawatt days) per weight of reactor fuel.

   Note: The term MWD/T can be applied to a single fuel assembly, the average value in a particular region, or to the whole core.

3. Most cores today can achieve burnup on the order of 10,000 to 30,000 MWD/T. TMI burner is rated at 14,400 MWD/T.

B. Conversion Ratio

1. Refers to the number of Pu-239 nuclei which is produced per 100 U-235 nuclei consumed expressed in percent. "235 to Pu239"

2. Typical conversion ratios for a PWR is aprox. 50-70%.

C. Effects of burnup

1. Buildup of Plutonium reduces the average delayed neutron fraction for the core. Therefore, for a given reactivity addition, a shorter period results.

2. Effects the reactivity coefficients in the core. The fuel temp. coefficient becomes more negative. The void and moderator temp. coef. become more negative due to poison removal.

3. The power distribution throughout the core changes with burnup (To be discussed later). will effect core stability and heat transfer limits.

Thermal diffusion Equation.

$$\frac{1}{v}\frac{d\phi}{dt} = \frac{n}{l} = [\rho - \beta] + \lambda C$$

change of neutron precursors with time

$$\frac{1}{T} C_0 e^{t/T} = \beta k_\infty \Sigma_a \phi_0 e^{t/T} - \lambda C_0 e^{t/T}$$

$$C = \frac{\beta k_\infty \Sigma_a \phi_0}{\frac{1}{T} + \lambda}$$

$$\frac{l^*}{T} = \rho - \beta + \frac{\lambda \beta}{(\frac{1}{T} + \lambda)}$$

l prompt neutron life time.

l^* neutron Generation time.

$$\boxed{\rho = \frac{l^*}{T} + \frac{\beta}{1 + \lambda T}}$$

(for all situations). assuming $k_{eff} = 1$ or close

another form

$$\boxed{\rho = \frac{l}{T\, k_{eff}} + \frac{\beta_{eff}}{1 + \lambda T}}$$

General equation

For a short Period

$T = <.1$ sec

$$\rho = \frac{l}{(\rho - \beta_{eff}) k_{eff}}$$

1103 1208

CORE PERFORMANCE

WEEKLY EXAM #1

NAME: Faust

DATE: 14 June 74

MAXIMUM POINTS 70

70
-3

Points 67

Grade 96%

11103 1 2 1 0

REACTOR STARTUP TO FULL POWER

PRACTICAL EXERCISE

Directions:

1. Each step will be worked out by the student.

Initial Conditions:

1. Reactor coolant temperature at 530F
2. Reactor coolant pressure at 2200 psia
3. Reactor coolant boron concentration at 1800 ppm
4. Reactor is 15% subcritical

Data Sheet:

1. CRA Worths

   Group 1 = 3% $\Delta k/k$

   Group 2 = 3% $\Delta k/k$

   Group 3 = 3% $\Delta k/k$

   Group 4 = .5% $\Delta k/k$

   Group 5 = 1% $\Delta k/k$

   Group 6 = 1% $\Delta k/k$

   Group 7 = 1.5% $\Delta k/k$

2. Doppler Coefficient

   α_D = -.01% $\Delta k/k/\%$ power

3. Moderator Coefficient

   α_M = .005% $\Delta k/k/°F$

4. Boron Worth

   1ppm Boron =-.01% $\Delta k/k$

5. Pressure Coefficient

   α_P = -.01% $\Delta k/k$/100 psi.

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I. Assume initial conditions as outlined in handout for each problem.

(10) 1. Establish criticality under the following conditions:

1.1 average temperature 580°F
1.2 Pressure at 2200 psia
1.3 Reactor Power at ~~[illegible]~~ 75%
1.4 Boron concentration 1300 ppm

Q. Find the critical rod group positions at equilibrium Xenon conditions.

(10) 2. Establish criticality under the following conditions:

1.1 average temperature at 530°F
1.2 Pressure 2200 psia
1.3 Rod Positions:

Groups #1,2,3,4 at upper limit
Groups 6 and 7 at Lower limit
Group 5 at 40% withdrawn

1.4 Essentially Zero Power Level

Q. What is the critical boron concentration?

(10) 3. Establish criticality under the following conditions:

1.1 Average temperature at 580°F
1.2 Pressure at 2200 psia
1.3 Reactor Power at 85%
1.4 Rod group positions:

Group 1,2,3,4,5 at upper level
Group 7 at lower level
Group 6 at 30% withdrawn

Q. Find the boron concentration to maintain the above conditions.

II. Define the following and explain the basis in which it applies. (5 points each)

1. Isothermal temperature coefficient
2. Power Doppler Coefficient
3. Moderator Temperature Coefficient
4. Power Coefficient

1103 1 2 1 2

(10) III. From Figure 2-50 determine the value of the Doppler Coefficient for 1500 ppm boron

IV. Utilizing figure 2-54 determine the Reactivity deficient due to doppler when
(10) increasing power from 40 to 90% power.

Given: EFPD = 300

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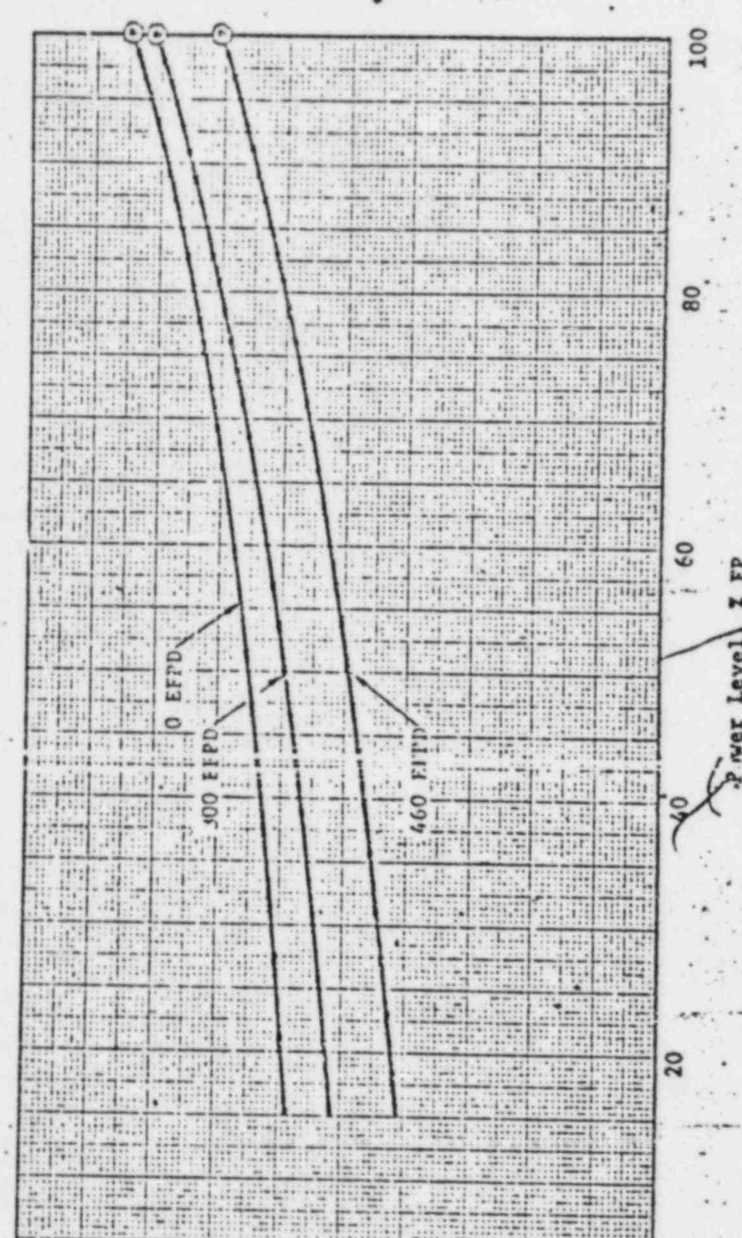
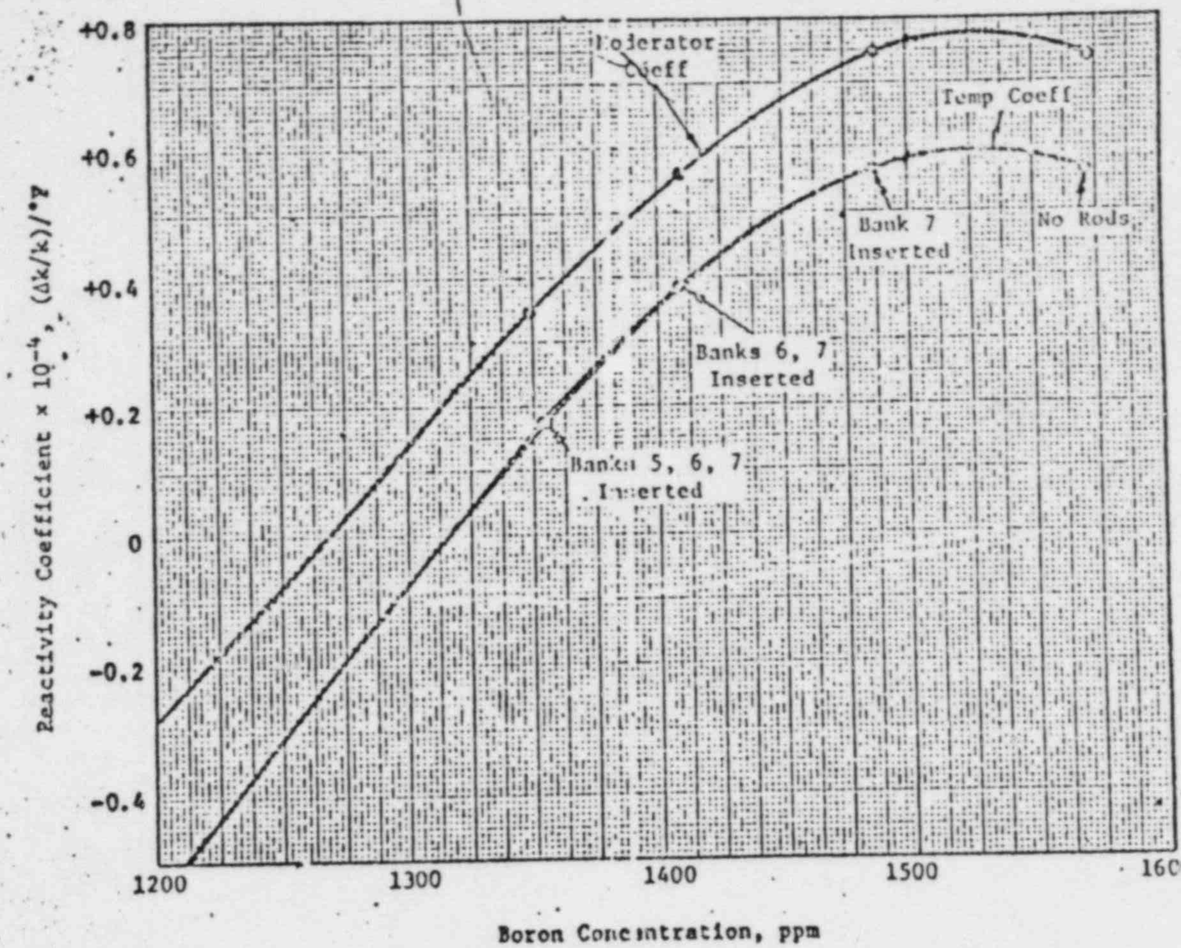


Figure 2-34. Power Doppler Cofficient Vs Power Level at 0, 300, and 460 EFPD

Figure 2-30. Moderator and Temperature Coefficients of Reactivity Vs Boron Concentration at 300F, 800 psi, 0 EFPD



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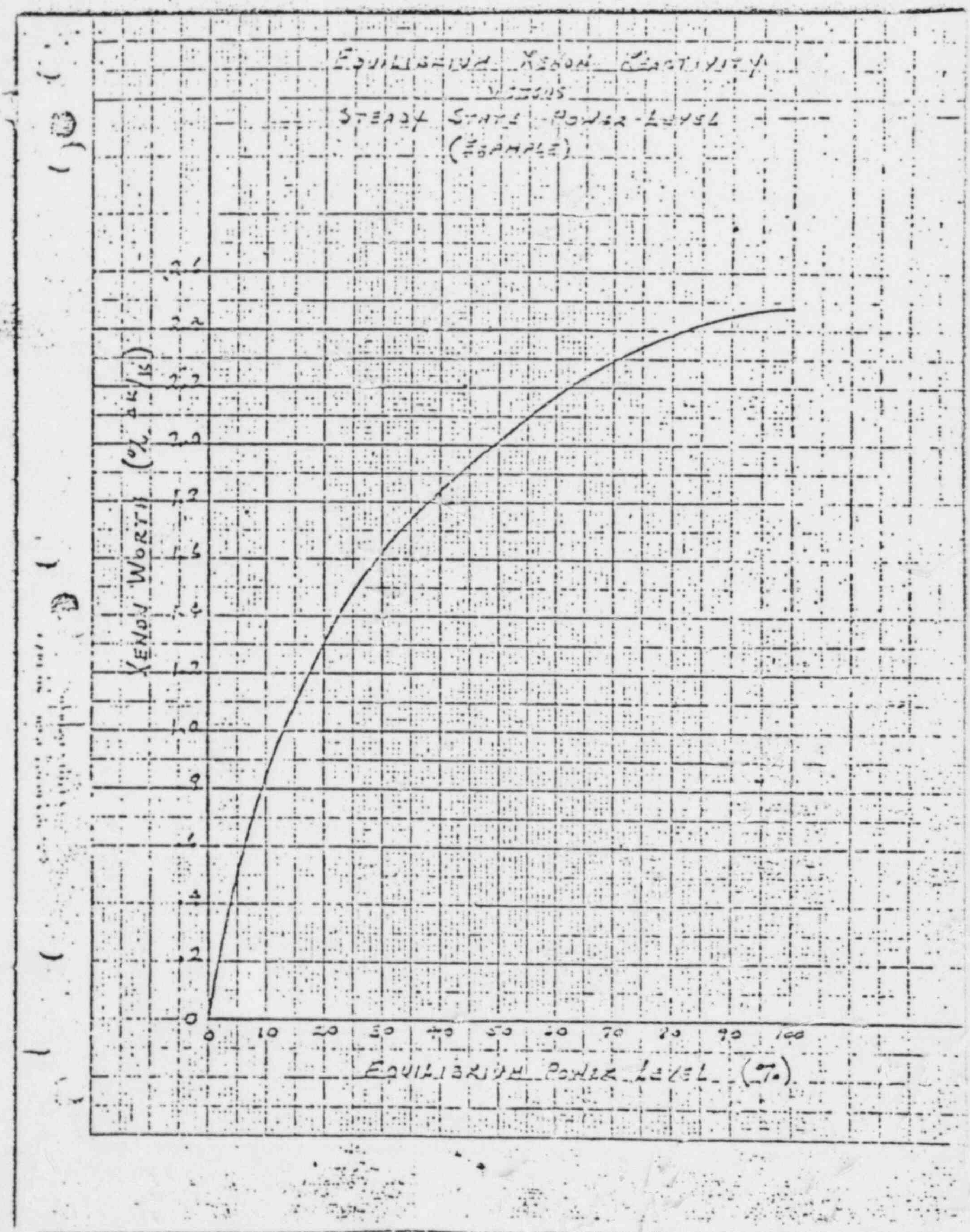

EQUILIBRIUM XENON REACTIVITY
VERSUS
STEADY STATE POWER LEVEL
(EXAMPLE)
XENON WORTH (% ΔK/K)
0 10 20 30 40 50 60 70 80 90 100
EQUILIBRIUM POWER LEVEL (%)

1103 1216

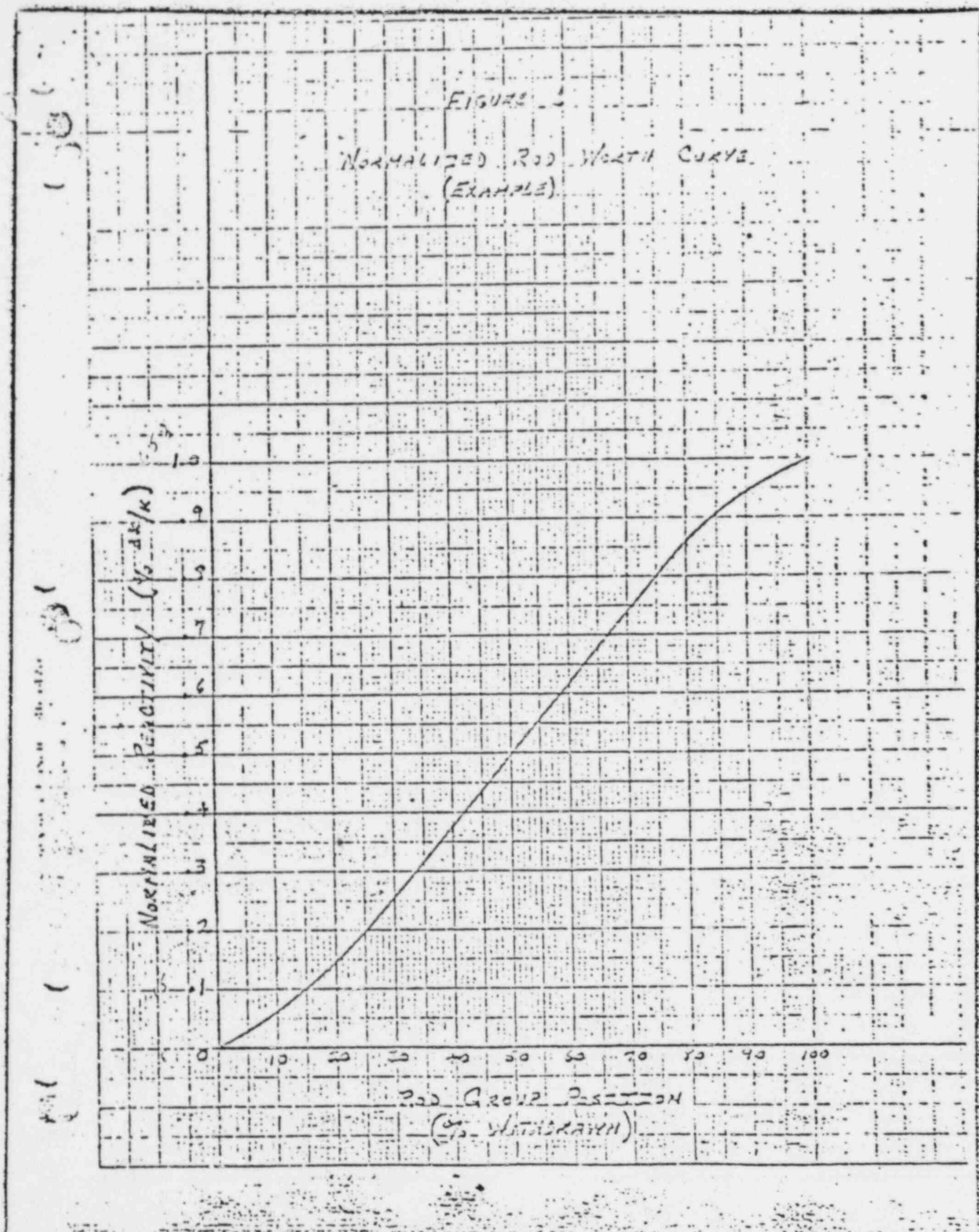
FIGURE 1
NORMALIZED ROD WORTH CURVE
(EXAMPLE)
NORMALIZED REACTIVITY (% Δk/k)
1.0
.9
.8
.7
.6
.5
.4
.3
.2
.1
0
10
20
30
40
50
60
70
80
90
100
ROD GROUP POSITION
(% WITHDRAWN)

I:

Fault
Reactivity Changes from boration changes

1. $\Delta T = 580 - 530°F = +50°F$ $\quad 50°F \times .005\% \Delta K/K/°F = +.250\% \Delta K/K$

$\Delta P = 75 - 0 = 75\%$ Pwr $\quad -.01\%/\% \times 75\% = -.75\% \Delta K/K$

Rx is $-.15 \Delta K/K$ shut down $\quad -15\% \Delta K/K$

Δ Boron $= -1500 + 1000 = -500$ ppm $\quad -500$ ppm $\times -.01\% \Delta K/K/$ppm $= +5\% \Delta K/K$

$-2.35\% \Delta K/K$ at 75% Pwr $\pm$ equilib Xe.

Total neg Rx overall in or added to core

$-15.\%$
$-.75\%$
-2.35%
$-18.10\% \Delta K/K$

Positive $\Delta K/K$ overall without Rod withdrawal.

$+.25\%$
$+5.00\%$
5.25%

$-18.10\% + 5.25\% = -12.85\% \Delta K/K$ Rods must be drawn this much to reach 75% Power

Groups 1 through 6 are all pulled To upper limit. gives $+11.5\% \Delta K/K$ added

Group 7 is pulled to add $+1.35\% \Delta K/K$ which corresponds to $\boxed{84.0\%}$ withdrawn (Group) Position

2. Rx Shut down $= -15\% \Delta K/K$

Gr 1-4 $= +9.5\% \Delta K/K$

Gr 5 at 48 wd. $= +.375\% \Delta K/K$

Remaining $= 9.875\% - 15\% = -5.125\% \Delta K/K$

$$\frac{-5.125\% \Delta K/K}{.01\% \Delta K/K/\text{ppm}} = -512.5 \text{ ppm}$$

Critical Boron concentration $=$ 1500 $-$ 512.5 $= \boxed{1257.5 \text{ ppm}}$

1103 121 8

Faust

3. + .25% ΔK/K from 50°F change

$85\% \cdot 0 = 35\%_x - .01\% \Delta K/K/°F = -.35\% \Delta K/K$

- 15% ΔK/K for subcritical condition

+ 10.5% ΔK/K Gr 1-5 ω

+ .25% ΔK/K Gr 6 at 30% WD

$$\begin{array}{r} 10.5 \\ .25 \\ .25 \\ \hline +11.00\% \Delta K/K \end{array} \qquad \begin{array}{r} -15. \\ -.85 \\ \hline -15.85\% \Delta K/K \end{array}$$

$11 - 15.85 = -4.85\% \Delta K/K$

$$-\frac{4.85\% \Delta K/K}{.01\% \Delta K/K/ppm} = -485 \text{ ppm}$$

Boron concentration = 1800 - 485 = $\boxed{1315 \text{ ppm}}$

No Xe considered in →

II

1. Isothermal temp. coefficient – is the fractional change in Keff per unit change in core temp. applies to that period of time in Rx history before any EFPH is seen by the core, or just temp. equilibrium in the core & moderator.

2. Power Doppler Coefficient – is the fractional change in Keff per unit change in power exclusive of moderator coefficient. Applies to periods when the average temp. of moderator does not change.

Fast

3. Moderator Temp Coefficient - the fractional change in Keff per unit change in moderator temp. When a change in the core's moderator temp occurs this applies.

4. Power Coefficient - The fraction change in Keff per unit change in Power. Applies any time you have a power change for it is equal to $\alpha_m \frac{\Delta T_m}{\Delta \% Pwr} + \alpha_D \frac{\Delta T_{fuel}}{\Delta \% Pwr}$. ½
(any time ramping Pwr 0 to 100% or vise-verso.

III. $\alpha_D = \alpha_T - \alpha_m$

$= .6 - .76$

$\alpha_D = -.16 \times 10^{-4}\ \Delta K/K$ or $-.0016\%\ \Delta K/K$

IV. $-1.43\ \times 10^{-4}\ \% @ 40\% = -.000572\ \Delta K/K$

$-1.25\ \times 10^{-4}\ \% @ 90\% = -.001125\ \Delta K/K$

~~.1=2 ... 900 ... /K~~

$-.001125\ \Delta K/K - (-.000572\ \Delta K/K) = -.000553\ \Delta K/K$

or $-.0553\%\ \Delta K/K$

ok for this trial - however, since the power coef is not a linear function you must take an average to determine its value.

11103 1221

$$\Delta\rho = \rho_2 - \rho_1 = \frac{K_2 - 1}{K_2} - \frac{K_1 - 1}{K_1} = \frac{K_2 - K_1}{K_2 K_1}$$

$$\frac{CR_2}{CR_1} = \frac{1-K_1}{1-K_2}$$

$$SUR = \frac{\partial 6}{T}$$

$$\Sigma = f \rho \frac{N_0}{m} \sigma \qquad N = f\rho \frac{N_0}{m}$$

% abundance of isot. — 6.02×10^{23} — atomic mass no.

$$f = \frac{V_f \Sigma_a \phi_f}{V_f \Sigma_a \phi_f + V_M \Sigma_a \phi_M}$$

$$f = \frac{1}{1 + \frac{V_M \Sigma_a \phi_M}{V_f \Sigma_a \phi_f}}$$

$$\nu = 2.43 \frac{\text{neutrons}}{\text{fission}} \text{ in } U^{235}$$

$$\eta = \frac{\Sigma_f \phi_{U^{35}} \nu}{\Sigma_a \phi_{U^{235}} + \Sigma_a \phi_{U^{238}}}$$

$$= \frac{\Sigma_f \nu}{\Sigma_a^{235} + \Sigma_a^{238}} = \frac{\sigma_f^{35} \nu}{\sigma_a^{35} + \frac{N_{238}}{N_{235}} \sigma_a^{238}}$$

% of ~~total cell~~ influence $= \frac{L_{IC} S_a}{V_{cell}}$ $\qquad L_{IC} = .8$ in.

% of neutrons influenced $= \frac{L_{IC} S_a \phi_{rod}}{V_{core} \phi_{avg}}$

Rod worth $\left(\frac{\Delta K}{K}\right) = \frac{L_{IC}(S_a)}{V_{core}} \left(\frac{\phi_{rod}}{\phi_{avg}}\right)^2$

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$P = P_0 \, 10^{(SUR)(t)}$

$P = P_0 \, e^{\frac{\Delta t}{T}}$

$$\boxed{\rho = \frac{l}{k_{eff} T} + \frac{\bar{\beta}}{1+\lambda T}}$$

+ period $\lambda = .08 \, sec^{-1}$

− period $\lambda = .0125 \, sec^{-1}$

Short $T < .1 \, sec$ $\rho = \frac{l}{k_{eff} T} + \beta$

Long $T > 1 \, sec$ $\rho = \frac{\bar{\beta}}{1+\lambda T}$

$\tau = \frac{1}{\lambda}$

$T = \frac{l + (\bar{\beta} - \rho) \cdot \tau}{\rho}$ for $\frac{\beta}{2} < \rho < \beta$; τ = avg. life of delayed neutron precursors 13 sec or 12.6 sec

$T = \frac{\bar{\beta} - \rho}{\rho}$ for $\rho < \frac{\beta}{2}$

$T = \frac{l}{\rho}$ for $\rho > \beta$

The Reactivity Coefficients.

Reactivity coef. is a measure of Δ in ρ that results from a specific change in Rx parameter. Such as Pwr, Temp, Press; ect. Alpha Symbol represents this (α)

$$\alpha = {}^{\Delta K/K}/_{\Delta\ \text{parameter}}$$

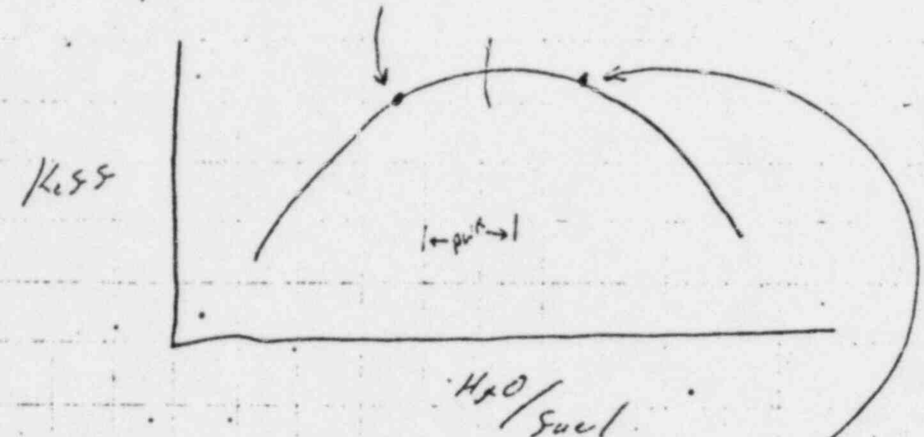
<u>Temp. Coef</u> (α_T) - The fractional change in the effective multiplication factor of the core per unit change in the core temp.

$\alpha_T = {}^{\Delta k/k}/_{\Delta Tc}$ ΔTc - core temp. change.

(special case) when the core (fuel) and the moderator are in temp. equilibrium <u>does this Coef. apply</u>. this is why it is refered to as the <u>ISOthermal Temp. Coef.</u> this condition can really only be obtained at initial criticality before any EFPH has occurred or eqilibrum occurs between the two

The difference between the MODERATOR COEFF. & TEMP. COEFF. (roughly) is the DOPPLER COEFF.

The core is an under moderated Heterogeneous core (Just moderator added.



Boron is added to moderator which shifts us to other side of curve.

Average temp. of Moderator is defined as

$$\frac{T_{in} + T_{out}}{2} = \overline{T}m$$

Moderator Coeff. (α_m) - fractional change in effective multiplication factor for unit change in moderator temp. $\alpha_m = {}^{\Delta k/k}/_{\Delta T_m}$

This coeff. cannot be directly measured from the core, ~~as is the~~

Tech Spec. limit - at Pwr > 95% the α_m shall not be positive to assure that the max clad temp. will not exceed acceptance criteria based on LOCA analysis. (most severe small break)

1103 1.225

$\alpha_{T_c} = \alpha_m + \alpha_D$ ← Doppler

$\therefore \quad \underset{\text{mod. coeff.}}{\alpha_m} = \underset{\text{temp coeff}}{\alpha_{T_c}} - \alpha_D$

Doppler Coeff. is the fractional change in the effective multiplication factor corresponding to a change in the fuel temperature. $\alpha_D = {}^{\Delta k/k}/_{\Delta T_{fuel\,(ave)}}$

Becomes more neg. over core age Due to Pu 240 Buildup. Because

Not Directly mesurable. (Can not measure - the fuel directly from core a operating core.)

figured from Pwr Doppler Coeff.

T changes from POWER Changes

Pwr Coeff (α_p) the fractional change in the effective multiplication factor corresponding to a change in core power level.

$$\alpha_p = {}^{\Delta k/k}/_{\Delta Power}$$

Contributing factors to α_p

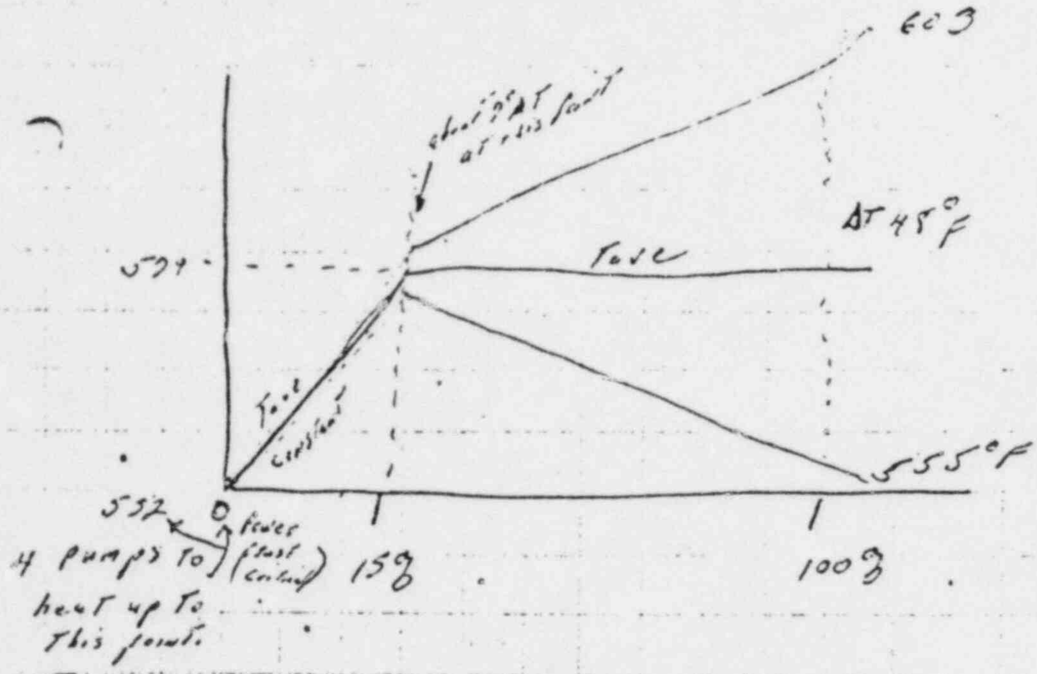
1. Ave. moderator Temp 2. Ave. fuel Temp.

$$\alpha_p = \frac{\Delta k/k}{\Delta T_m} \times \frac{\Delta T_m}{\Delta power} + \frac{\Delta k/k}{\Delta T_{fuel}} \times \frac{\Delta T_{fuel}}{\Delta power}$$

$$\alpha_p = \alpha_m \frac{\Delta T_m}{\Delta pwr} + \alpha_D \frac{\Delta T_{fuel}}{\Delta pwr}$$

if mod. temp is held constant while increasing fuel temp. then

$$\alpha_p = \alpha_D \frac{\Delta T_{fuel}}{\Delta pwr} \qquad \left(\frac{\Delta T_m}{\Delta pwr} = 0.0\right) \text{ no change}$$



Power Doppler Coef. α_{PD}.

$$\alpha_{PD} = \alpha_P \text{ when } \frac{\Delta T_m}{\Delta P_{wr}} = 0.0$$

$\alpha_{PD} = \alpha_D \left(\frac{\Delta T_F}{\Delta Power}\right)$ for Refueler between 15% → 100% Power.

α_{PD} is a measure of the change in the effective multiplication factor per change in core power (exclusive of the moderator temp. effect.

$$\alpha_D = \frac{\Delta k/k}{\Delta °F_{fuel\,(Ave)}}$$

Moderator Pressure Coeff. (α_{press})

The fractional change in the ~~moderator~~ effective multiplication factor per unit change in the moderator density resulting from a change in RC Pressure

BOL	EOL
$-3 \times 10^{-7}\ \Delta k/k/psi$	$+3 \times 10^{-6}\ \Delta k/k/psi$

effect is so small at this time we don't use it.

BOL	EOL
$+\alpha_m$ at BOL. an increase in Press. gives neg feed back gives a neg α_{press}	$-\alpha_m$ just the opposite occurs.

Moderator Void Coeff.

The fractional change in the ρ per % void in moderator.

BOL		EOL
-3×10^{-4}	to	$-19 \times 10^{-4}\%\ \Delta k/k/\%\ void$

To small! we don't use it.

The Moderator ~~Temp~~ Coef. becomes more neg over Core age due to removal of Poison (Boron Shim.)

Doppler coeff becomes more neg due to Build up of Poison especially P_u240.

Know average fuel Temp. at 100% Power. (~1250°F)

Figure 2-50. Moderator and Temperature Coefficients of Reactivity Vs Boron Concentration at 300F, 800 psi, 0 EFPD

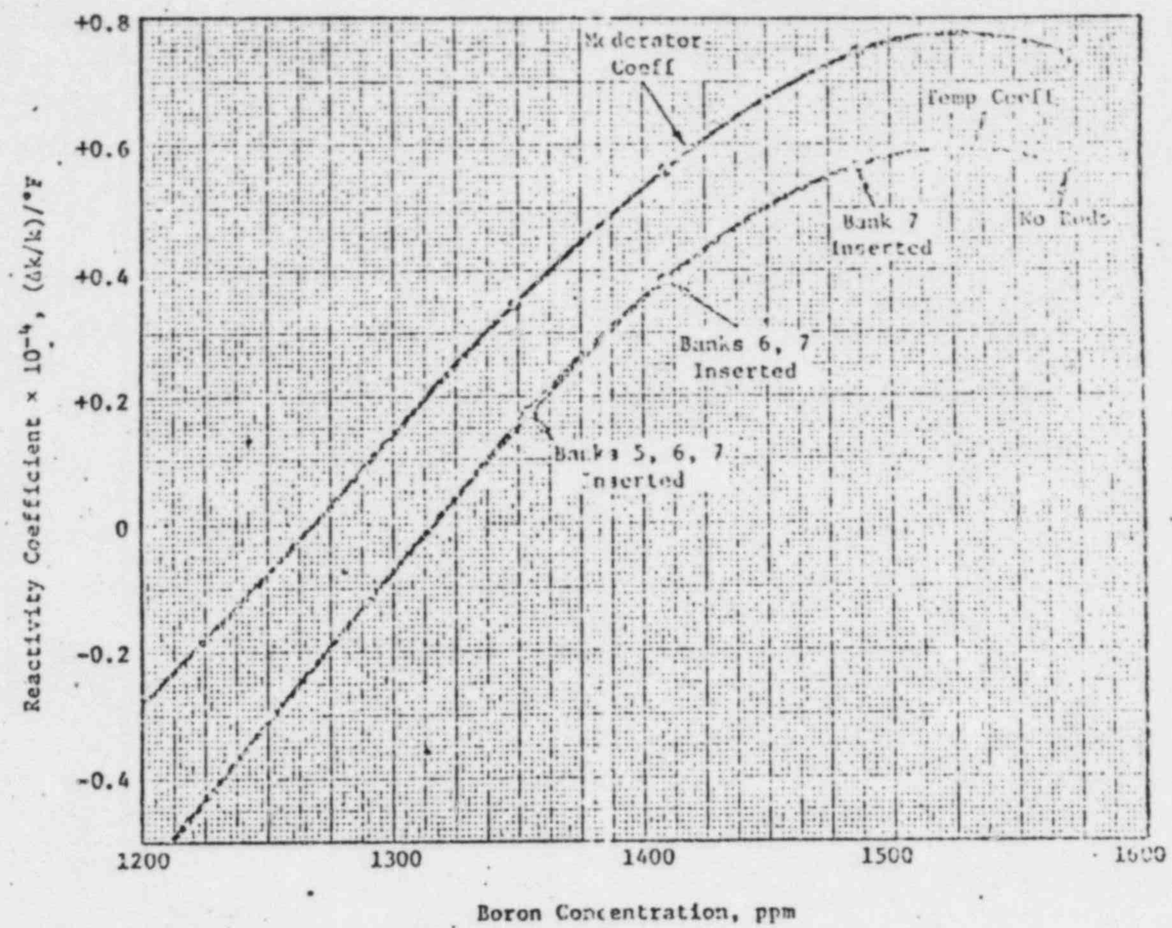


Figure 2-51. Moderator and Temperature Coefficients of Reactivity Vs Boron Concentration at 532F, 2155 psi, 0 EFPD

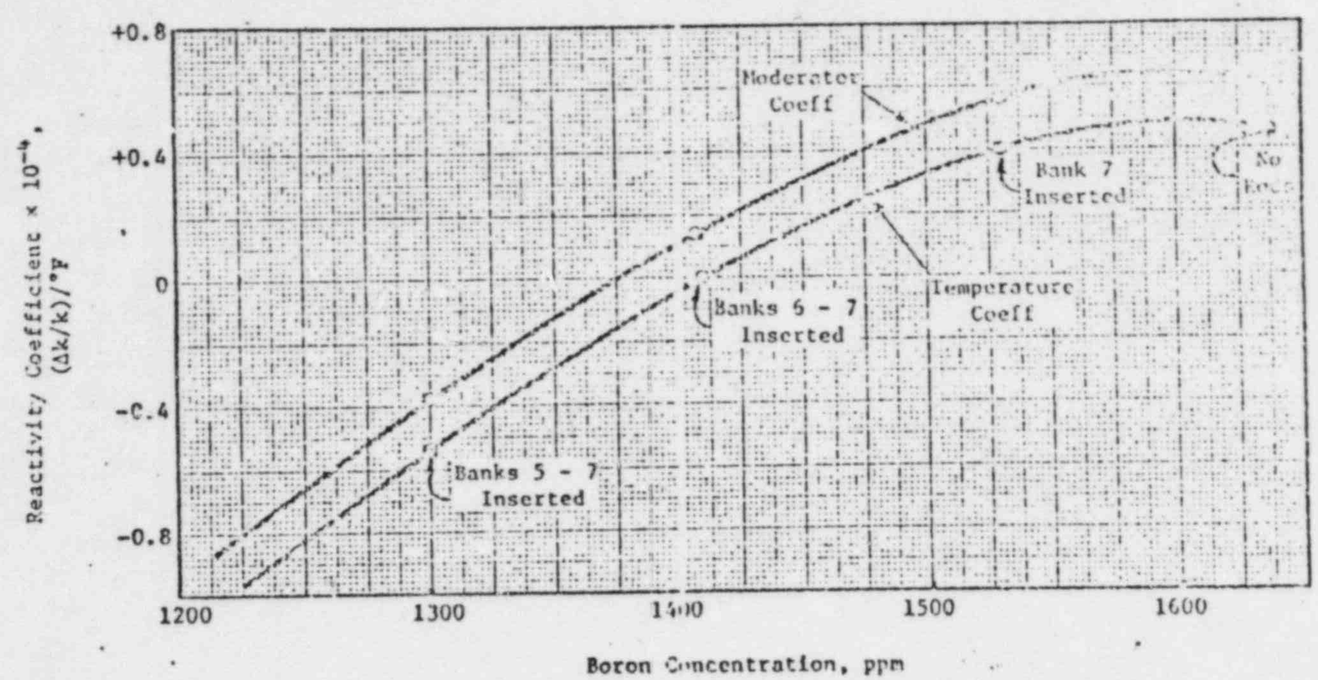


Figure 2-32. Moderator Coefficient of Reactivity Vs Boron Concentration for Critical Boron and Rod Worth Conditions, 0 EFPD

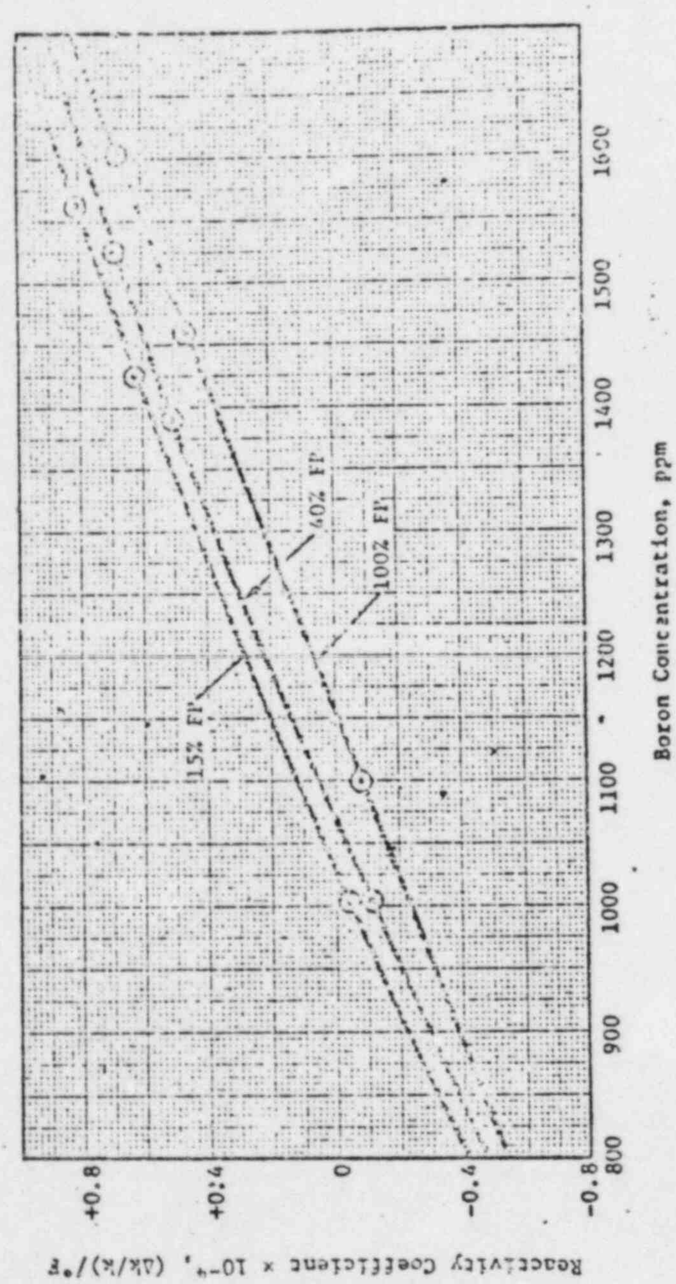


Figure 2-53. Temperature Coefficient of Reactivity Vs Boron Concentration for Critical Boron and Rod Worth Conditions, 0 EFPD

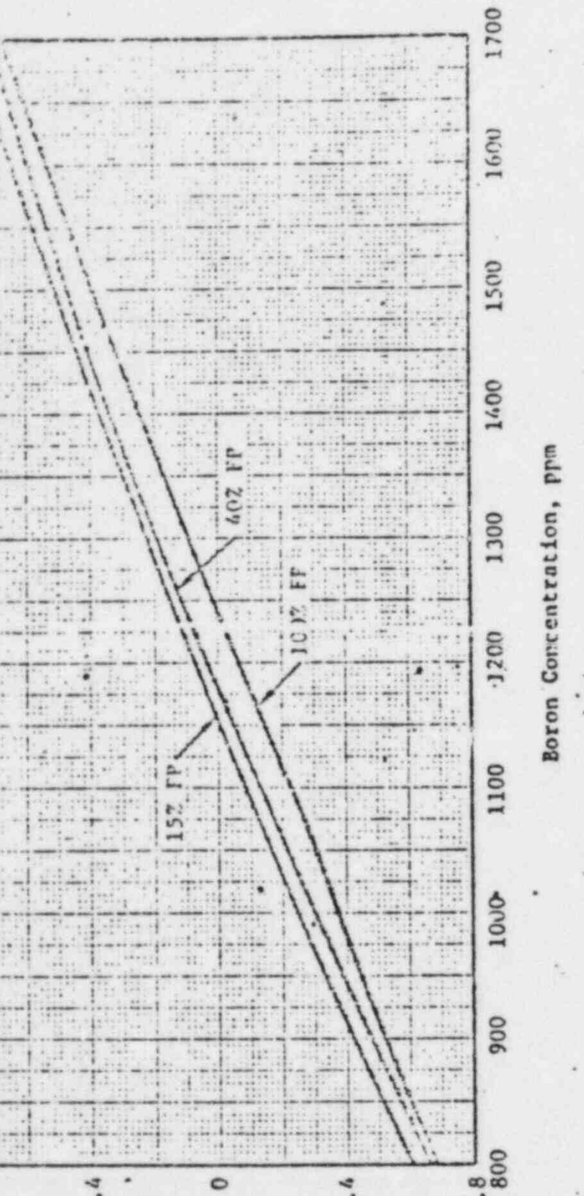
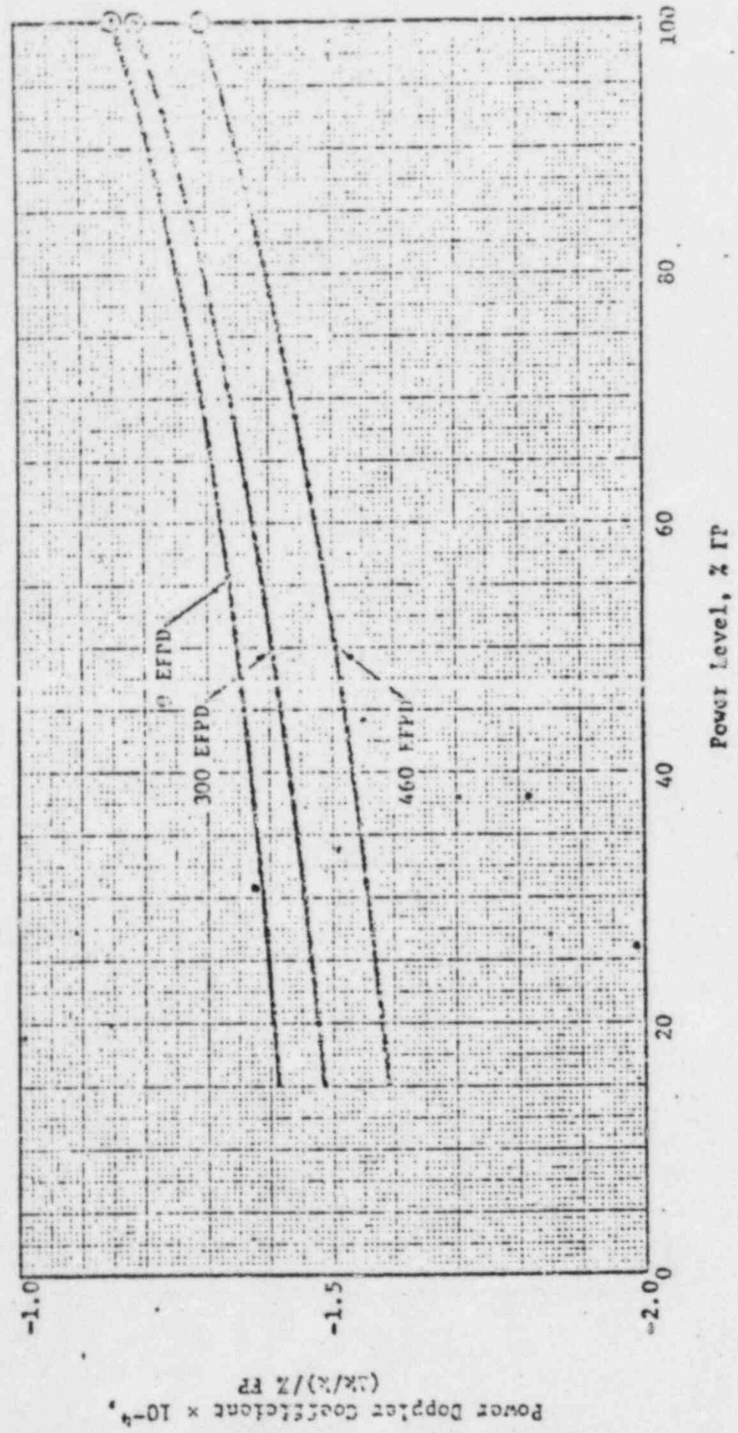


Figure 2-54. Power Doppler Coefficient Vs Power Level at 0, 300, and 460 EFPD



Build up of fission (mainly Pu241) over core age makes is less. more neg.

Figure 2-55. Power Doppler Reactivity Deficit Vs Power Level at 0 and 466 EFPD

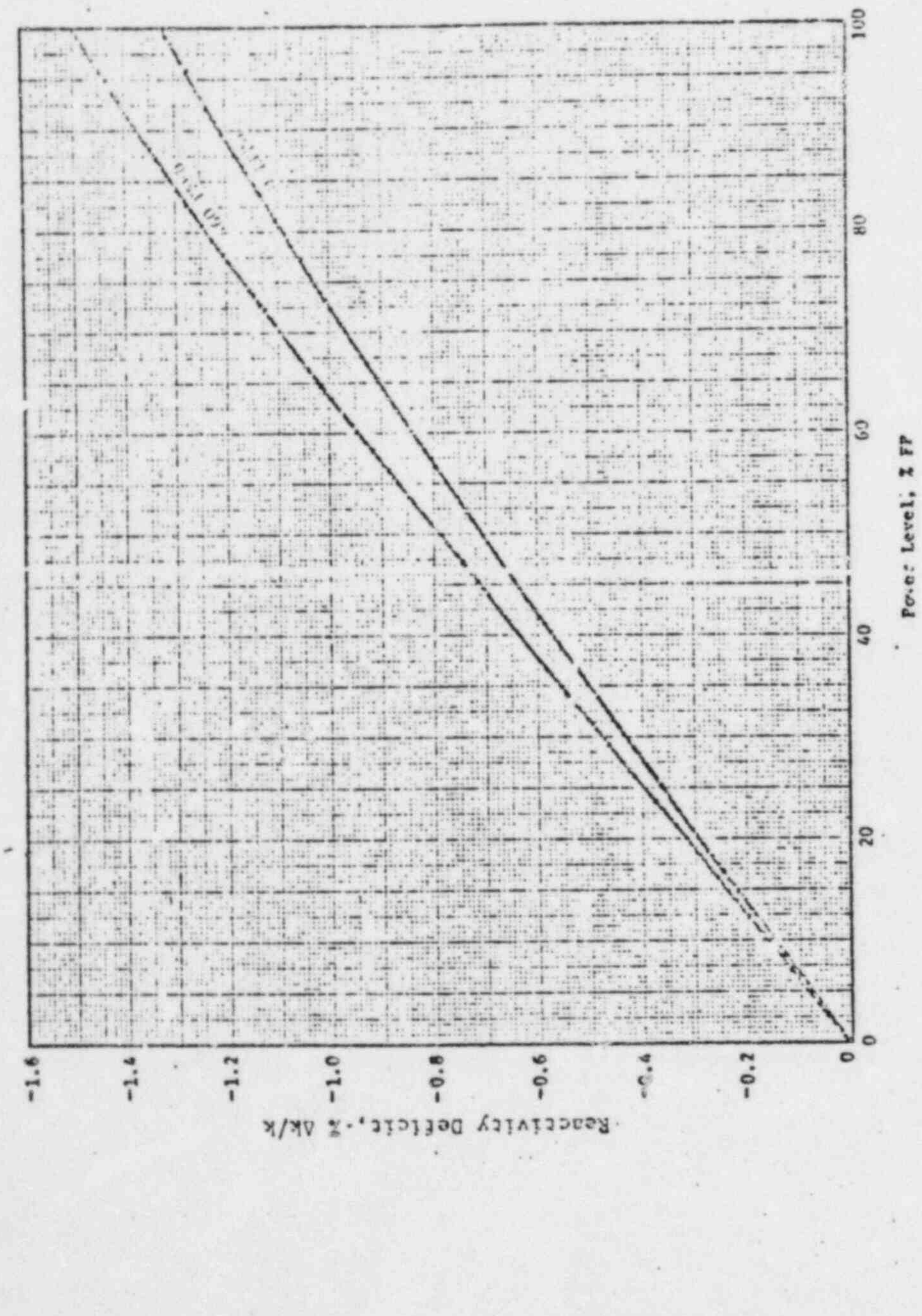


Figure 2-56. Average Fuel Temperature Vs Power Level at 0 EFPD

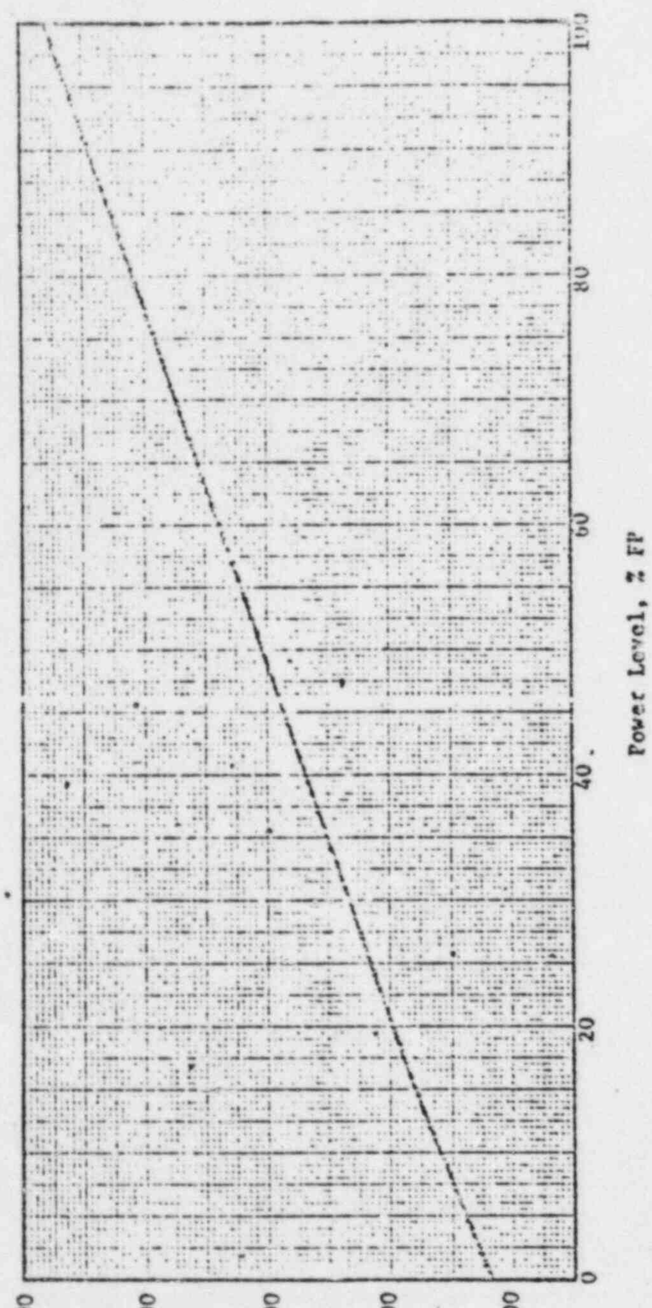


Figure 2-57. Moderator Coefficient Vs Cycle 1 Lifetime, 100% FP, Operating Control Rod Patterns, Critical Boron

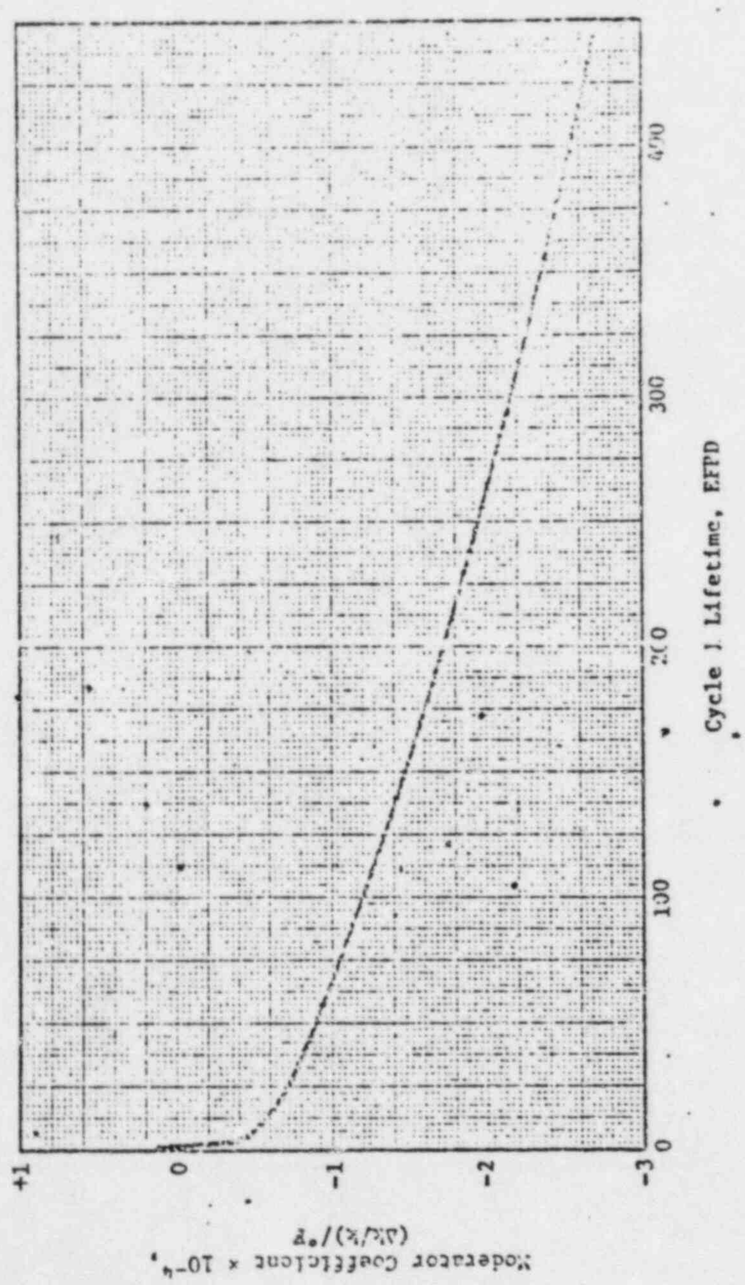


Figure 2-38. Temperature Coefficient Vs Cycle 1 Lifetime, 100% FP. Operating Control Rod Patterns, Critical Boron

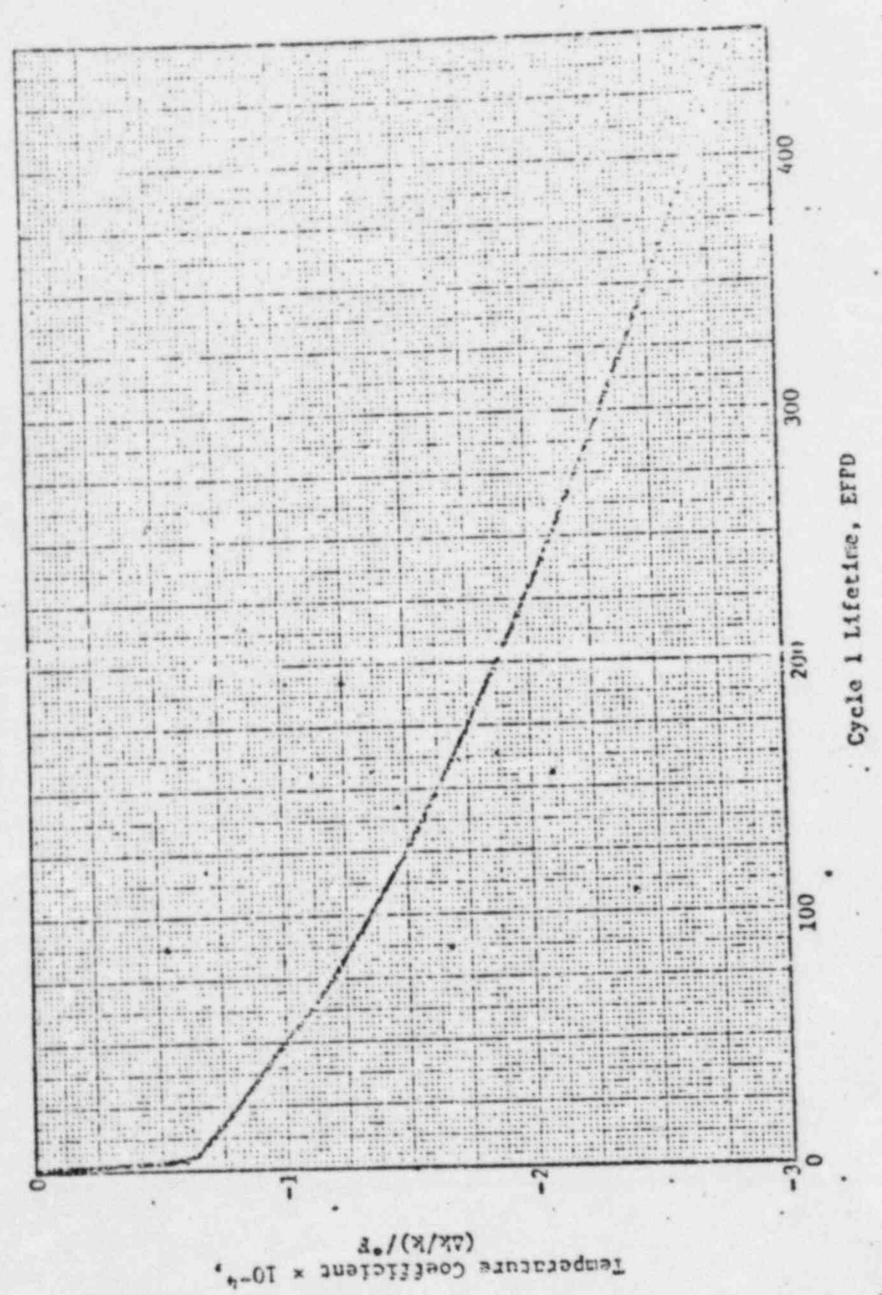
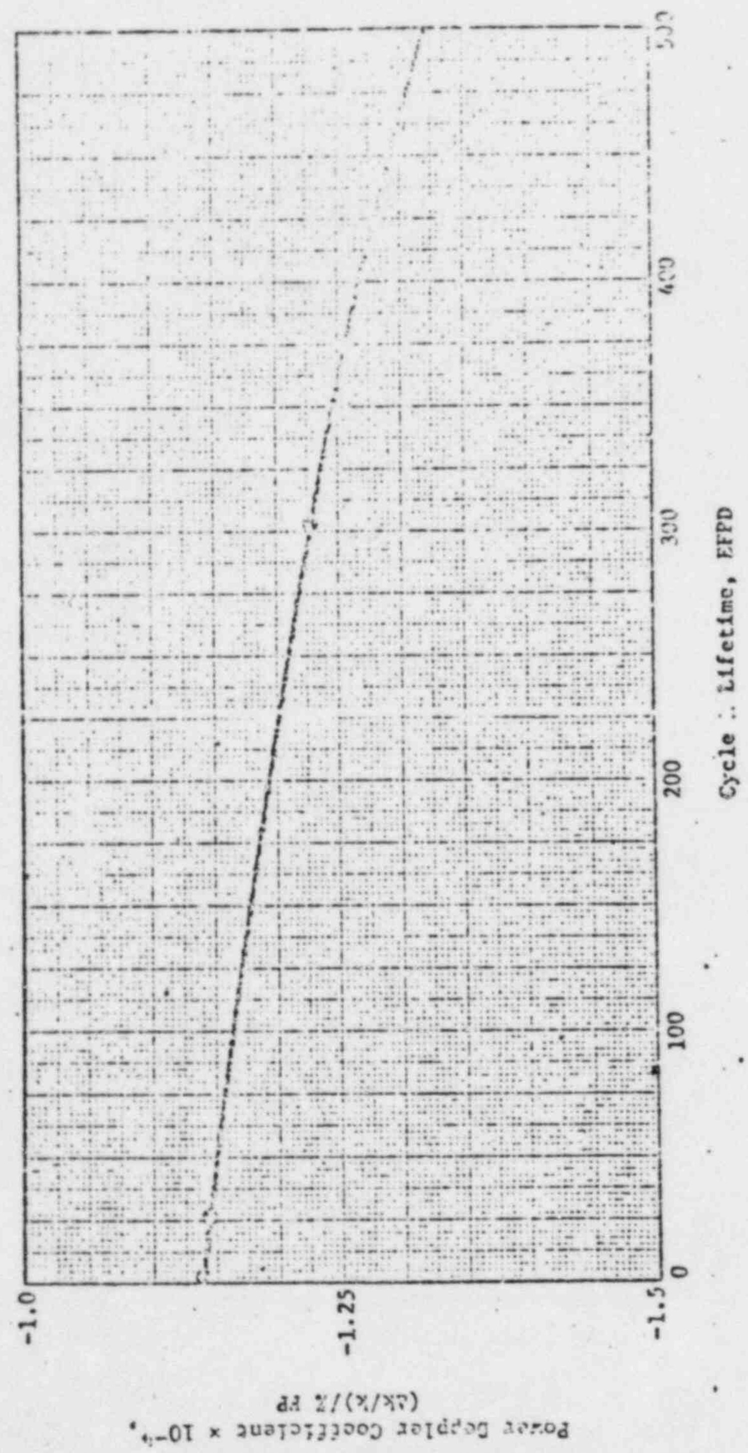


Figure 2-59. Power Doppler Coefficient of Reactivity Vs Cycle 1 Lifetime at 100% FP Conditions



4

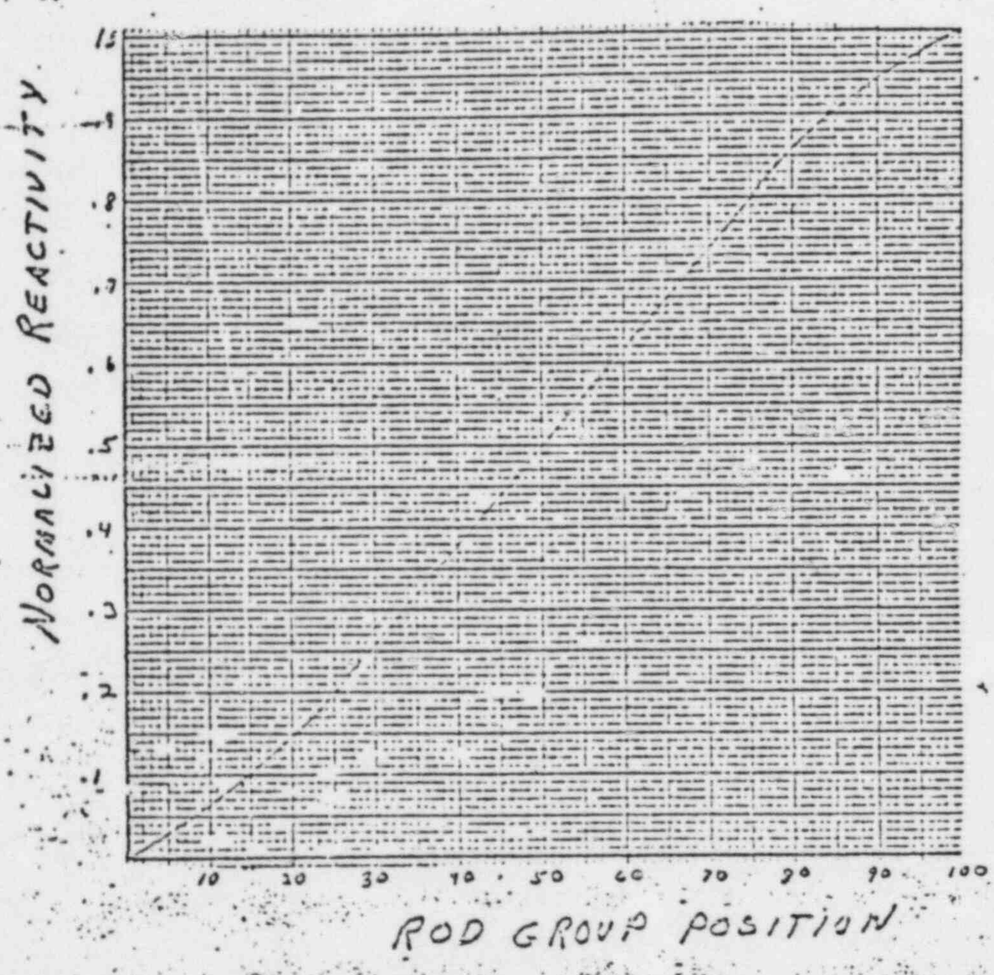
FORM -123-

THE BABCOCK & WILCOX CO.

GENERAL CALCULATIONS

FIGURE 1

NORMALIZED ROD WORTH CURVE



CUSTOMER	PROD. NO.	JOB NO.
SUBJECT	DWG. NO.	EST. NO.
	SUMM. SHEET NO.	
CALC. BY / DATE / CHK'D BY / DATE	ANAL. NO. / SCHEME NO.	SHEET NO.

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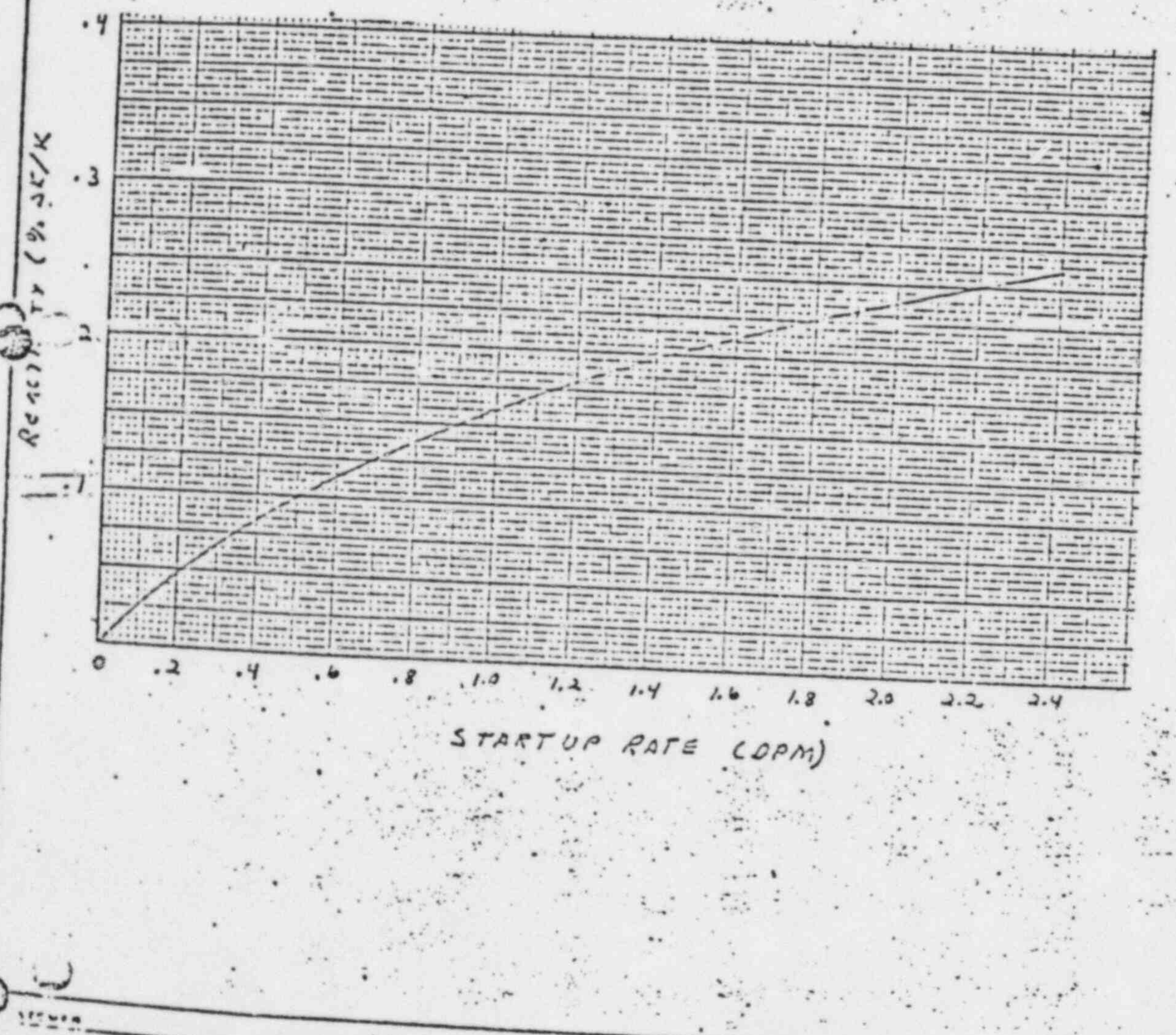
FORM -129-

THE BABCOCK & WILCOX CO.
GENERAL CALCULATIONS

FIGURE 3

REACTIVITY VERSUS STARTUP RATE



SUBJECT

PROD NO
JOB NO
DWG NO
EST NO
SUMM SHEET NO
ANAL NO
DATE
CHK'D BY
DATE

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F β u235 - .0065

β Pu239 - .0020

TMI's effectiveness is 106%

to find β_{eff} multiply β by 106%

0% 50% 100%

pu239 $\bar{\beta}$ = .002120

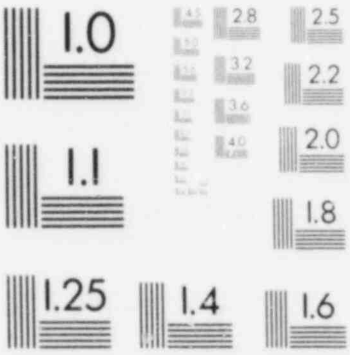
5

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

IMAGE EVALUATION
TEST TARGET (MT-3)



150mm

6″

[illegible]

I. Heat Addition to a material (no phase change).

A. BTU[s], Heat Capacity (C_p)

1. Definition of BTU : 1 the amount of heat required to raise the temperature of one pound of "water" 1°F.

2. Heat Capacity (C_p) definition :

a) Heat capacity, sometimes referred to as specific heat, is the heat required to raise the temperature of one pound of "material" 1°F. By definition, the specific heat of water is 1. All other specific heats are relative to this.

b) Unit of heat capacity (C_p) is BTU/lb/°F.

$$C_p = \frac{Q}{M \Delta T} \; \frac{BTU}{lb\text{-}°F}$$

where: C_p = heat capacity of a material (BTU/lb - °F)

Q = total heat added or subtracted from the body (BTU)

M = mass of body (lb)

ΔT = (T final - T initial) = temperature change of body °F.

Example: Determine how much heat must be supplied to 100 lb of a substance with a heat capacity of 0.5 BTU/lb/°F to change its temperature from [illegible]° to 212° F.

[illegible]

[illegible]

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$$\boxed{Q = MC_p \Delta T}$$

only applies in a uni. state condition

$Q = (100)(0.5)(212-90)$

$Q = 50 \times 122$

$Q = 6100$ BTU

C. Heat capacities of various materials

$\frac{dQ}{dt} = C_p \cdot \frac{dm}{dt} \Delta T$

Same (change in Time)

$\dot{Q} = C_p \dot{m} \Delta T$

MATERIAL	C_p(BTU/lb - F°)
water (70°F)	1.0
water (575°F)	1.3
steam	0.48
UO_2	0.03
steel	0.13

3. Fundamental heat Equation

a) The fundamental heat equation is given as:

$$\boxed{Q = MC_p \Delta T}$$

it describes the effect of heat addition to a body under the condition that the body undergoes no change in phase (boiling, melting, or freezing). It applies equally to solids, liquids or gases.

4. <u>Heat applies to flowing fluids</u>:

In our plant, applications of the fundamental equation can be made by first changing the equation to consider heat and mass flow rates as follows:

$$\dot{Q} = \dot{M}\, C_p \Delta T$$

where:

$\dot{Q}$ = heat addition (or subtraction) rate (BTU/hr)

$\dot{M}$ = mass flow rate (lb/hr)

5. <u>Example</u>:

In the TMI reactor the coolant enters the bottom of the core at 555°F (full power) and leaves the top at 603°F. The coolant flow rate is approx. 131,000,000 lb/hr. In the range of 555 to 603°F, the heat capacity of water is aprox. 1.3 BTU/lb - °F. At what power is the reactor operating?

Ans:

$$\dot{Q} = \dot{M}\, C_p \Delta T$$

$$\dot{Q} = \left(\frac{1.31 \times 10^{8}\ \text{lb}}{\text{hr}}\right)\left(\frac{1.3\ \text{BTU}}{\text{lb}\ {}^{\circ}\text{F}}\right)(603 - 555\ {}^{\circ}\text{F})$$

$$\dot{Q} = 8.17 \times 10^{9}\ \text{BTU/hr}$$

Since, 1MW = 3.413 × 10^{6} BTU/hr

$$\dot{Q} = \frac{8.17 \times 10^{9}}{3.413 \times 10^{6}} = [illegible] \times 10^{3}\ \text{MW (thermal)}$$

[illegible]

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(λ_{vap}) heat of vaporization (Btu/lb)
at a given pressure, the heat of vaporization (latent)
is the amount of heat to vaporize 1lb of a liquid

$Q = mC_p\Delta T$

(saturated water in stm) $Q = m_{vap}\ \lambda_{vap}$

Quality

$X = \dfrac{m_{vap}}{m}$ $\quad mx = m_{vap}$

$Q = X\, m\, \lambda_{vap}$

$\boxed{\dot{Q} = X \dot{m}\, \lambda_{vap}}$ rate of change equation, dot indicates this difference.

heat transfer for a change in phase from liquid to vapor

$\boxed{\dot{Q} = \dot{m}\, c_p \Delta T + \underset{\text{Quality}}{X} \dot{m}\, \lambda_{vap}}$ heat of vaporization

$\dot{Q} = \dot{m}_{water}\, c_{p_{water}}\, \Delta T_{water} + \underset{\text{vapor stage}}{X \dot{m}\, \lambda_{vap}} + \underset{\text{100% stm}}{\dot{m}_{stm}\, c_{p_{stm}}\, \Delta T_{stm}}$

II. Heat Transfer Fundamentals

A. Relationship between Flow Rate, Driving Force and Resistance

1. The flow or transfer of an quantity, such as matter, electricity, or heat, is governed by three basic laws:
   1) There must be a driving force which causes the quantity to flow.
   2) The rate of flow increases in direct proportion to the magnitude of the driving force. If we double the driving force, we will double the rate of flow.
   3) There will always be some resistive effect which tends to impede the flow. Flow rate varies inversely with resistance.

$$\therefore \text{Flow Rate} = \frac{\text{Driving Force}}{\text{Resistance}}$$

B. Heat Transfer Rate Equation across a boundary

a) In the case of fluid flow:
   a) The driving force = Pressure drop
   b) The Resistance = frictional effects of piping, orifices, valves, etc.
   c) Flow = flowing fluid

b) Equation

$$\frac{Q}{A} = \frac{\Delta T}{R}$$

where: Q = [illegible] heat transfer (BTU, [illegible])

A = heat [illegible] (ft²)

[illegible] = [illegible] ([illegible] ft²)

[illegible]

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$$R = \frac{A \Delta T}{\varphi}$$

R = heat transfer resistance

$$\frac{[(\Delta^\circ F)(hr)(ft^2)]}{BTU}$$

C. Illustrative Examples

1. An insulated steam line: An insulated 1000°F steam line.

   a) Has large driving force = the difference between 1000°F and room temperature.

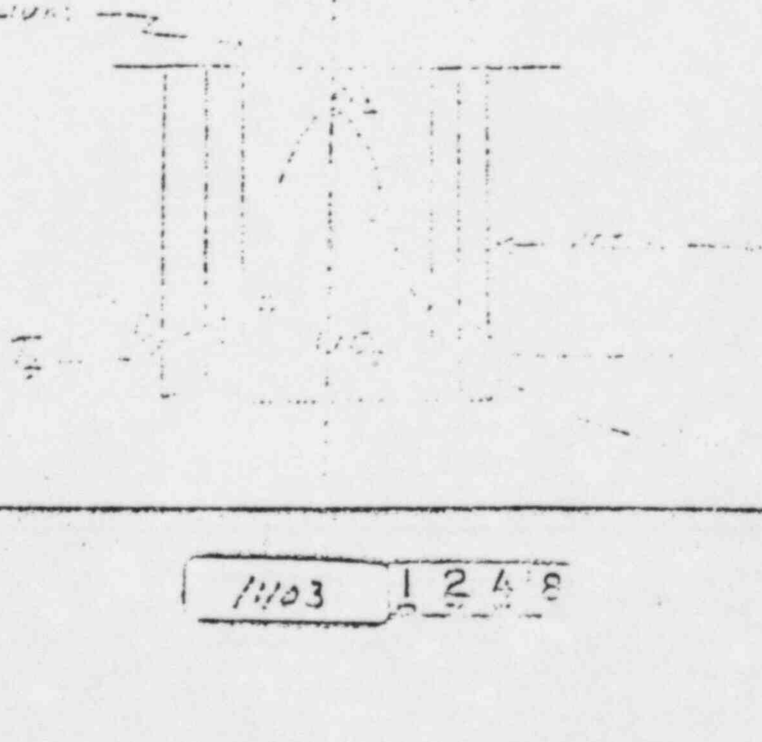
   b) Heat flux to room = quite small due to large resistance (Insulation) to heat transfer.

2. A Condenser: heat from the condensing steam is transferred through the condenser tube to the cooling water.

   a) Driving force = low due to relatively low temp. difference.

   b) Heat flux from the condenser tubes to the cooling water = large heat transfer due to low resistance. NOTE: Resistance in this case included the resistance of the film of steam and condensate on the outside of the tube, the resistance of the tube material, and the resistance of the stagnant film of water on the inside surface of the tube.

3. Typical Fuel Rod (Drawing):



where: T_c = Fuel Center Temp.
T_1 = Fuel Surface Temp.
T_2 = Interior Cladding Surface Temp.
T_s = Exterior Cladding Surface Temp.
T_f = Bulk Fluid Temp.

then: $$\frac{Q}{A} = \frac{\Delta T}{R}$$

where:
$$T_c - T_1 = (Q/A)(R_{UO_2})$$
$$T_1 - T_2 = (Q/A)(R_{gas})$$
$$T_2 - T_s = (Q/A)(R_{clad})$$
$$T_s - T_f = (Q/A)(R_{H_2O})$$

Note: $Q/A \approx$ a constant

Therefore:

$$(T_c - T_1) + (T_1 - T_2) + (T_2 - T_s) + (T_s - T_f) =$$
$$(Q/A)R_{UO_2} + (Q/A)R_{gas} + (Q/A)R_{clad} + (Q/A)R_{H_2O}$$

OR

$$(T_c - T_f) = Q/A \left[R_{UO_2} + R_{gas} + R_{clad} + R_{H_2O}\right]$$

D. Thermal Conductivity

a) Definition : The ability of a material to conduct heat. The higher the thermal conductivity, the better the material conducts heat and the lower is its resistance to heat transfer.

Examples: [illegible]

1. Metals.
2. Dense non-porous crystals.
3. Dry porous non-metals (ex asbestos, dry wood, cork).

E. Conduction:

a) Definition: The transfer of heat by atomic, molecular or electronic action.

b) Is the primary heat transfer mechanism in solids.

c) In a fuel rod, conduction is the primary heat transfer mechanism in the UO_2 and the cladding.

F. Convection:

a) Definition: Is the transfer of heat by the mixing of hot and cold bodies.

b) Generally limited to heat transfer in fluids (liquids or gases).

G. Radiation:

a) Definition: The emission of electromagnetic radiation by a body which possesses heat energy.

b) Every body of matter which possesses heat energy continuously emits and absorbs energy in the form of electromagnetic radiation.

H. [illegible] coolant water

1. [illegible] heat transfer [illegible] heat [illegible] carry a [illegible] of [illegible] coolant [illegible]

There are two kinds of boiling: nucleate boiling and film boiling.

2. Nucleate Boiling

In nucleate boiling there is a generation of small spherical bubbles of steam, distinct and separate. These bubbles then break away from the heated surface and are carried into the coolant where they join or cluster with other bubbles.

3. Film Boiling

In this kind of boiling, a thin film of vapor completely covers the heated surface. The primary mode of heat transfer is radiation through the vapor. Surface temperature increases greatly since the vapor film acts as a barrier to conduction or convection heat transfer.

4. Conditions for which Nucleate or Film boiling can occur.

These conditions are called subcooled boiling and bulk boiling.

a) Subcooled Boiling:

When either nucleate or film boiling is occurring and the average fluid temperature in which the boiling is taking place is less than the saturation temp., the condition is called subcooled boiling. There is no net steam generation since all the steam is condensed in the cooler water when it breaks away from the heated surface.

b) Bulk Boiling

When either nucleate or film boiling is occurring in a fluid that is at saturation temp., the condition is called bulk boiling. In this condition, net steam generation is realized.

* NOTE: [illegible]

[illegible]

[illegible]

[illegible]

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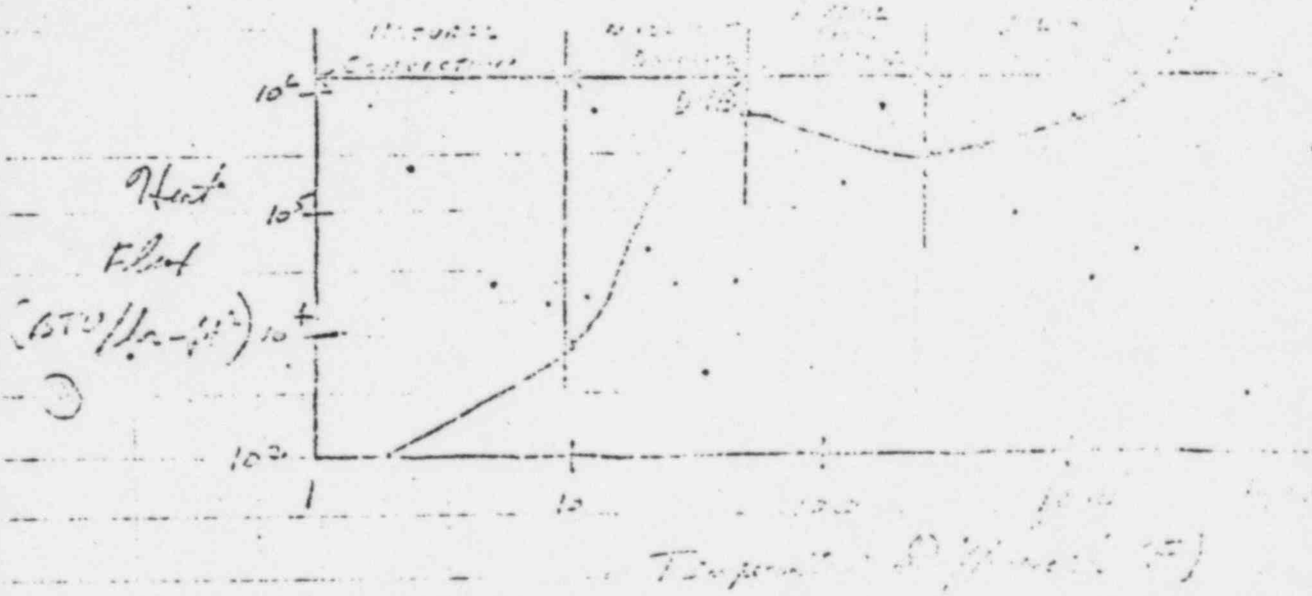
5. <u>Transition from [illegible] boiling to [illegible] and burnout</u>

If the heat flux is increased [illegible] in the nucleate boiling region, there is a point reached [illegible] a transition from nucleate boiling to film boiling occurs suddenly and the temperature difference increases rapidly.



a) The point just before [illegible] from nucleate boiling to film boiling is called "the point of departure from nucleate boiling" commonly written as [illegible]

b) [illegible] heat flux [illegible] "[illegible] flux".

c) [illegible]

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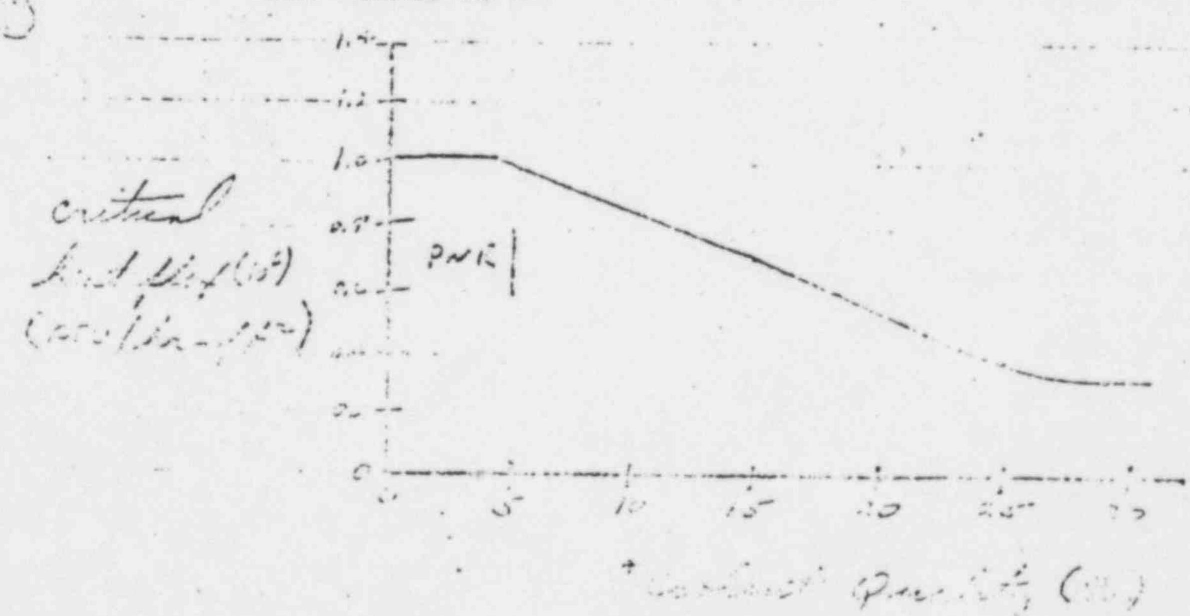
the heat being produced from the surface of the fuel rod to the reactor coolant increases greatly. Since the reactor coolant temp. is fixed, the temp. of the surface of the fuel rod increases greatly. Usually this temp. difference is high enough to cause the fuel rod cladding to exceed its melting point and a fuel failure would occur. This effect is called "burnout".

I. Critical Heat Flux (CHF)

1. Magnitude of the critical heat flux is dependent upon many variables however, the following remarks apply:

   a) The CHF increases with increasing coolant flow rate.

   b) If the coolant is subcooled (PWR), an increase in the subcooling will increase the CHF.



* coolant quality (x) equals: $x = M_{vap}/M$

where: M_{vap} = mass of liquid which is vaporized (m)

M = total mass of fluid

c) The typical operating range of a PWR would be 0-5% quality and a critical heat flux on the order of 10^6 BTU/hr-ft^2.

J. Critical Heat Flux Ratio (DNBR)

1. To avoid burnout, a reactor must be operated such that the critical heat flux is not exceeded at any point in the core. The limit is expressed as the Departure from Nucleate Boiling Ratio (DNBR) where:

$$DNBR = \frac{\text{critical heat flux at specific location}}{\text{Actual heat flux at same location}}$$

2. Since critical heat flux is a difficult parameter to measure under various operating conditions, a degree of Conservation is built into the operating DNBR limit. Most PWR reactors are designed to operate with DNBR ratios greater than 1.3 to 1.5 at all locations during steady state and transient operating conditions.

3. DNBR ratio for TMI is:

$$DNBR \geq 1.3$$

K. Peak Flux

1. Since the DNBR ratio for TMI is 1.3 and since this corresponds aprox. to a heat flux of 10^6 BTU/hr-ft^2, the peak flux must be limited to about $10^6/1.3$ or 7.7×10^5 BTU/hr-ft^2.

L. Critical Heat Flux and Actual Heat Flux vs. the Axial Position along a Typical Fuel Rod

- See Diagram on following Page -

$$\dot{Q} = UA\Delta T$$

$$U = \frac{1}{R_1 + R_2}$$

$$\frac{\dot{Q}}{A} = \frac{\Delta T}{R} = \frac{1 \cdot \Delta T}{R} = \left(\frac{1}{R}\right)(\Delta T)$$

$$\dot{Q} = UA\Delta T$$

$\frac{1}{R}$ $\frac{F^\circ \cdot hr \cdot ft^2}{BTU}$

$\frac{BTU}{F^\circ \cdot hr \cdot ft^2}$

$\frac{1}{1/3}$ $\frac{1}{1}$ $\frac{2}{1}$

$U =$

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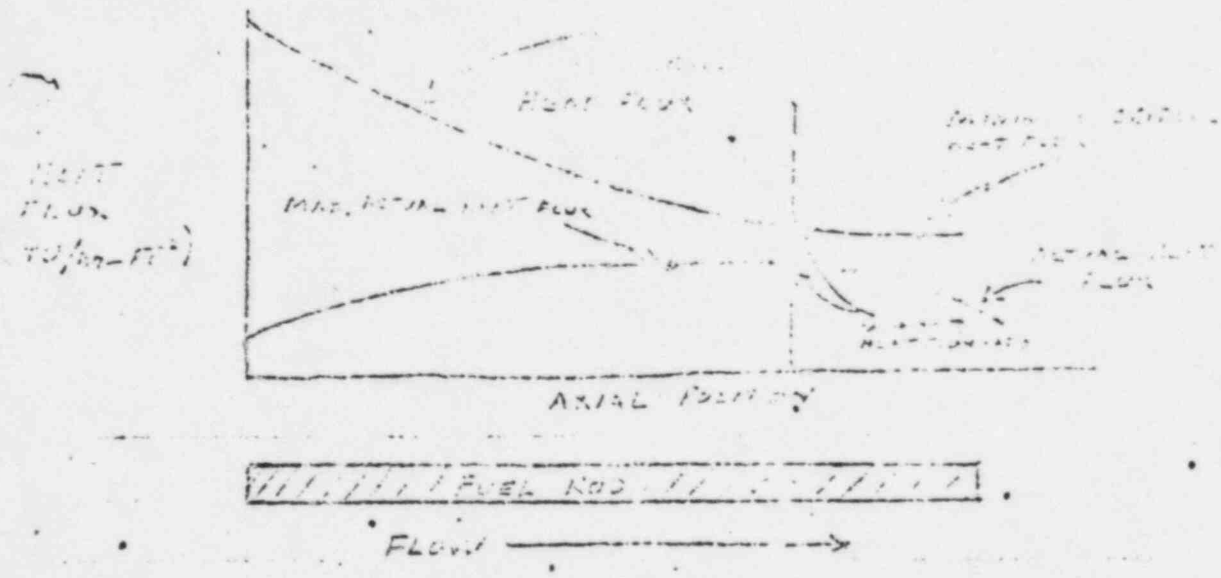

H. Fuel pellet melting considerations

1. There are two heat transfer limits which are required of water moderated reactors. They are:

   (1) Minimum critical heat flux ratio (DNB) : which is intended to prevent cladding failure.

   (2) Limit on fuel rod center temperature : which is intended to avoid fuel pellet melting during normal operations and expected transients.

2. Melting Point of UO_2 : Fuel pellet melting occurs at aprox. 5000°F. Since the fuel pellet temperature is not directly measured, the equation expressed earlier establishes the limit.

$(T_c - T_f) = (Q/A)(R_{UO_2} + R_{gap} + R_{clad} + R_{film})$ OR

$Q/A = \frac{1}{(R_1 + R_2 + \dots R_n)}$ $Q/A = U \Delta T$ $U = \frac{1}{(R_1 + R_2 + R_n)}$

$Q = AU\Delta T$

The limit is expressed in [illegible] or in units of kilowatts per foot (kW/ft) which would produce a fuel center temperature near the melting point.

(1) NOTE: In a PWR the peak fuel temperature at full power is aprox. 4,000°F. Under transient conditions, the value may approach aprox. [illegible]°F. Core avg. fuel temp. values are aprox. 1250°F.

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CHAPTER 6

OPERATION CHARACTERISTICS

INITIAL FUEL LOADING

I. A typical fuel bundle

A. $K_\infty = 1.3$

B. Leakage very high because bundle size is small, therefore keff < 1.0

1. Increasing fuel, increasing active core dimensions.

2. Decrease in leakage with additional fuel bundles

3. 10-20 fuel bundles are critical

4. Next 100 bundle add to the excess reactivity of core

II. Precautions against criticality while loading

A. Moderator is highly borated

B. Control Rods are in fuel bundles

III. Loading

A. Several neutron detectors are placed in fuel bundles.

B. The fuel bundles with the source are placed in core.

C. Neutron population is constantly monitored.

D. Fuel bundles are added continuously until core filled.

E. Only 1/2 the number of bundles, extrapolated to criticality ($\frac{1}{M}$ plot), can be loaded into core.

1. Subcritical equation;

$$CRR = \frac{CR_2}{CR_1} = \frac{E_2S_2/(1-K_2)}{E_1S_1/(1-K_1)}$$

a. Assume the source has a long half-life and is constant during count taking.

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b. Assume efficiency of both counts are the same.

2. Simplification

$$CRR = \frac{CR_2}{CR_1} = \frac{1/(1-K_2)}{1/(1-K_1)} = \frac{(1-K_1)}{(1-K_2)}$$

a. As K_2 gets greater, the CCR goes to ∞.

b. ∞ can not be plotted

c. invert ∞; becomes 0.

d. $\frac{1}{M} = \frac{1}{CRR} = \frac{1}{\infty} = 0$

e. As $\frac{1}{M} \rightarrow 0$ the core keff approaches criticality, K = 1.0

Approach to criticality in loaded core.

I. Procedure for PWR

A. Withdrawal of control rod groups called the shutdown bank.

B. Boron dilution

C. Criticality reached by withdrawal of additional control rods.

II. Keff is varying by the changing thermal utilization factor, whereas during loading Keff changed due to non-leakage factors.

III. Neutron Count

A. Control rod produces step changes

B. Boron dilution has gradual change

C. Varies due to geometry

D. Varies with position of core components

1. Higher if near fuel-source bundle
2. Lower if near control rod bundle

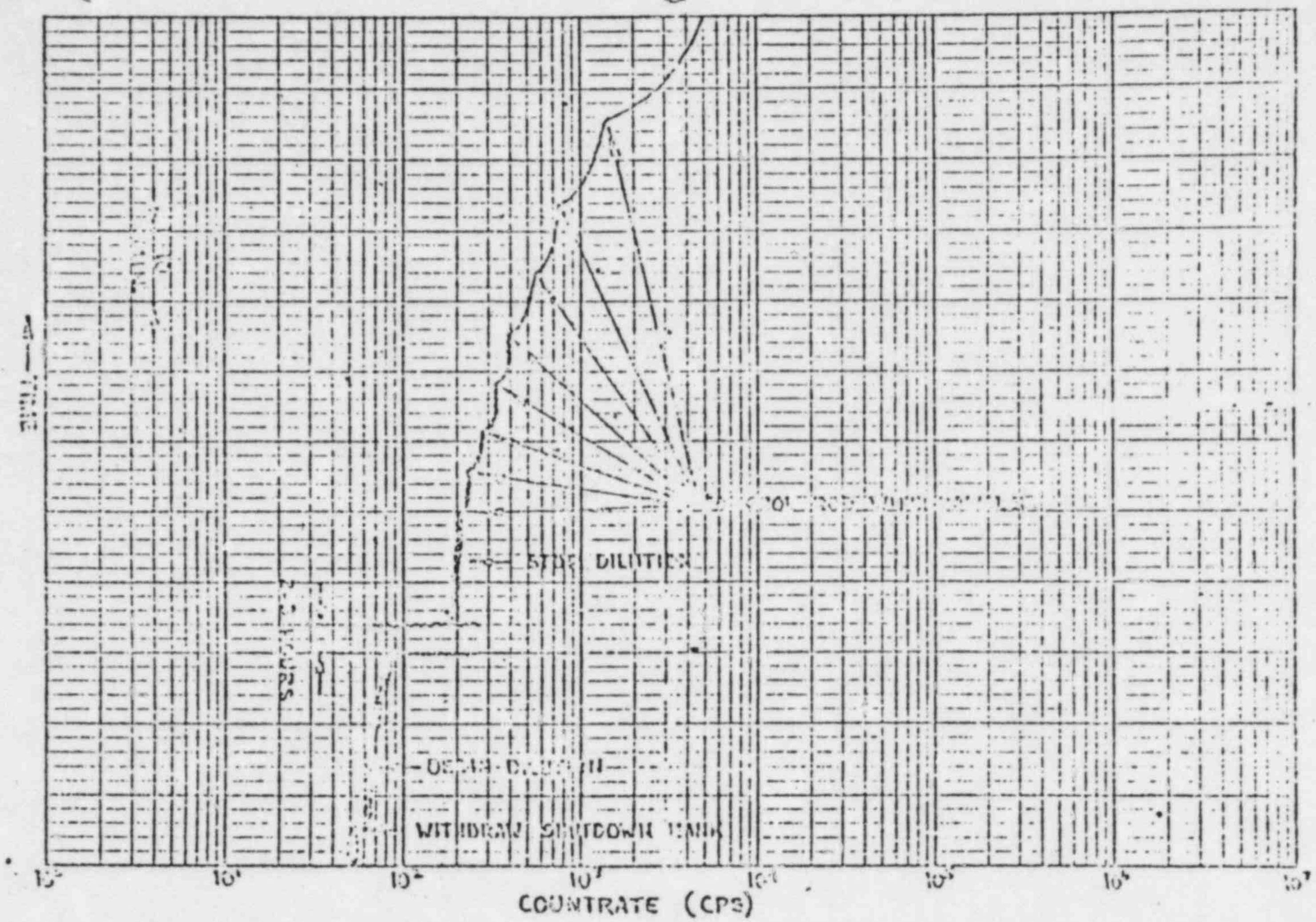


Figure 12-7: Idealized Out-of-Core Detector Response During Approach to Critical on a PWR

CONTROL ROD WORTH CHARACTERISTICS

I. Loaded Core

A. Subcritical by 4-5% $\frac{\Delta K}{K}$ (K = .94-.95).

B. Center control rod withdrawn

1. Still 1% $\Delta K/K$ till critical

2. Rod Worth 3-4% $\Delta K/K$

C. Ave. rod worth = .5% $\Delta K/K$

D. Rod worth is 10-20 times average.

1. Size of core; only those effected by control rod.

2. Effective size of core is effected strongly by rod making its worth high.

E. Second rod pull and still subcritical

1. Therefore rod worth < 1.0% $\Delta K/K$

2. Far away from first

3. Active core now the fuel effected by control rod plus the fuel between them.

F. More rods pulled increase the core size and rod worth decreases proportionly with area covered.

G. All of bank pulled

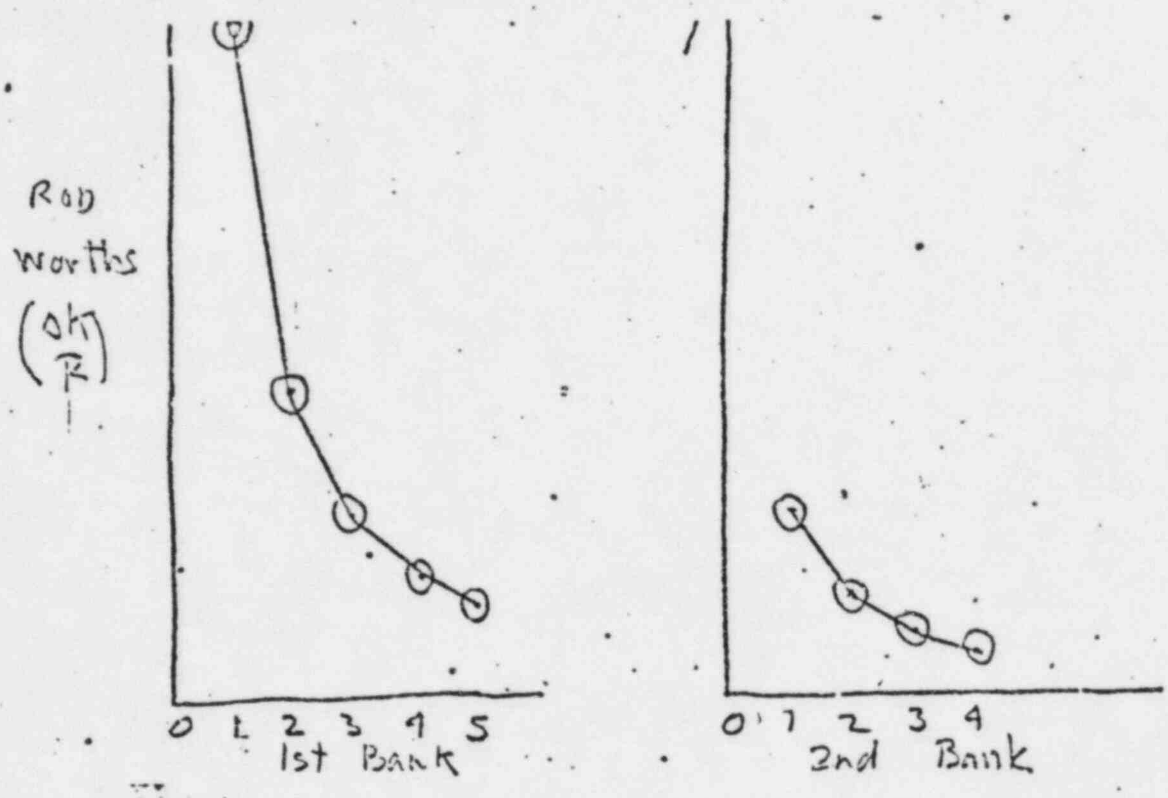
1. Even flux

2. No peaking

H. Next bank pulled

1. The 1st rod has most worth because it causes flux peak - due to uneven control rod distribution

2. Next rod less important worth due to the fact they, balance flux.



II. Undesirable patterns

A. Those which confine the effective critical volume.

B. Two-rod clump

1. Not critical on 1st rod withdrawn but Keff = .99

2. Second rod doubles core size, so it is worth 50% of first rod or 2% $\Delta K/K$. Now Keff ≥ 1.0

C. Hot, Standby condition

1. BWR shutdown with all rods inserted.

2. Is at rated temperature and pressure.

3. Exists immediately following scram.

4. Due to strong negative temperature coefficients means it is now possible to pull more rods before criticality than with cold start-up.

5. Lt larger due to "hot" condition.

INCREMENTAL CONTROL ROD WORTHS

I. The total rod worth will be the sum of the worths associated with each increment of movement.

A. Relative incremental worths can be found by:

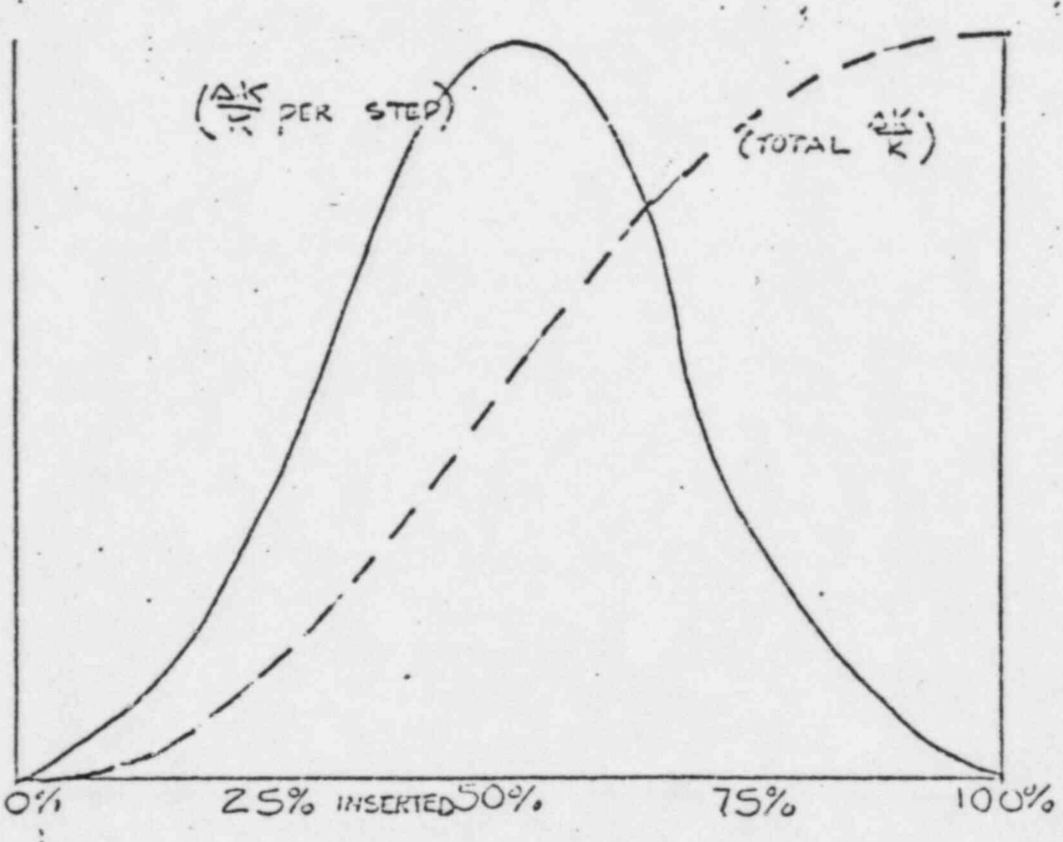
$$\text{Rod Worth}\left(\frac{\Delta K}{K}\right) \simeq Lt \times \frac{Scr}{V\ Core} \times \left(\frac{\emptyset t,cr}{\emptyset t,ave}\right)^2$$

1. Physical size of all control rod increments is uniform.

2. Physical size of the core is also constant.

3. Variations in worths, therefore, must come from variations in Lt and Øt.

II.



Percentage Control Rod Insertion

A. The control rod incremental worth follow the same rule with $\left(\frac{\phi_T}{\phi_{ave}}\right)^2$ and the axial flux is a cosine, therefore the curve is bell shaped.

B. The total worth after each increment follows the "s" shape curve. The larger change coming from the center, where the flux is greatest.

C. From the plot, it appears the strongest increment would come from the center, however, this is usually false and it is moved about due to certain factors.

1. PWR, the strongest increments are below the core centerline (Skewed to the right)

   a. Rods are partially inserted from the top.

   b. Warmer water in the top of core

2. Hotter water near the top causes increase in the thermal diffusion, Lt.

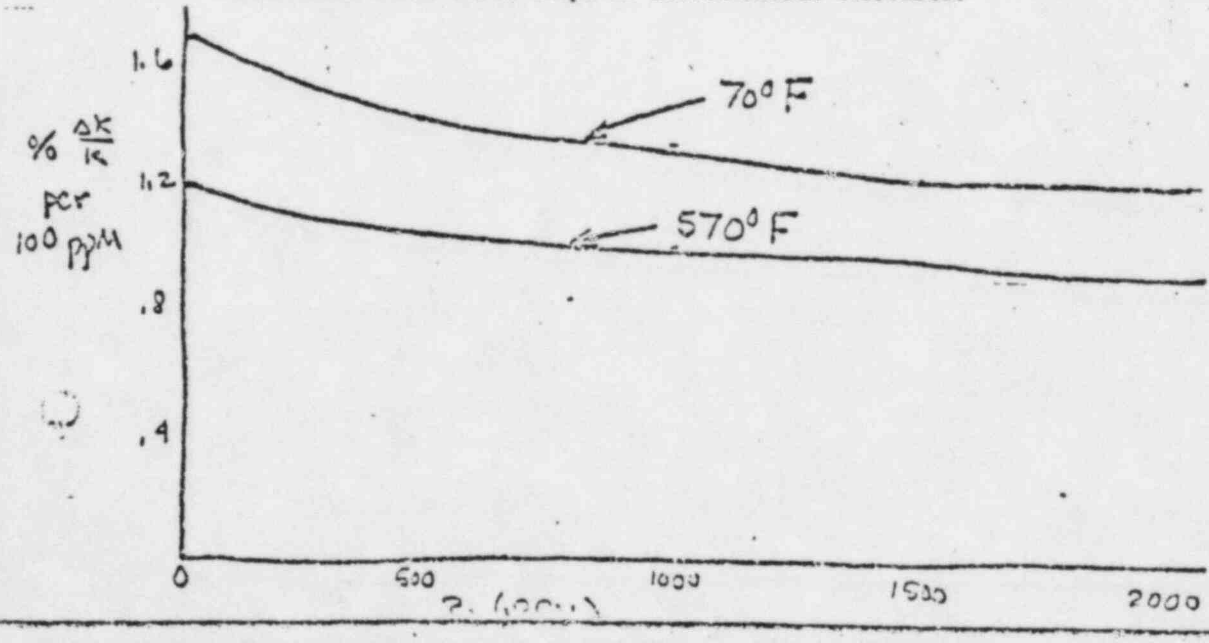
III. Incremental boron worth refers to the reactivity effect of each increment of dissolved chem shim added to the core.

A. Approximation: 100 ppm B to reduce $\frac{\Delta K}{K}$ by 1%.

1. 100 ppm added to free core (0-100 ppm) will have more effect than adding to a highly borated core (1000-1100 ppm).

2. The newly added boron has to compete with the old 1000 ppm, while the first boron in the coolant has the field to itself.

B. Incremental boron worth drops as concentration increases.



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C. Incremental worth decreases as moderator temperature increases, due to decrease in moderator density.

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