A Lecture Series in Nuclear Power Plant Operations Training

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Catherine Cook CC
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NUCLEAR REACTOR FACILITY
THE PENNSYLVANIA STATE UNIVERSITY
UNIVERSITY PARK, PENNSYLVANIA

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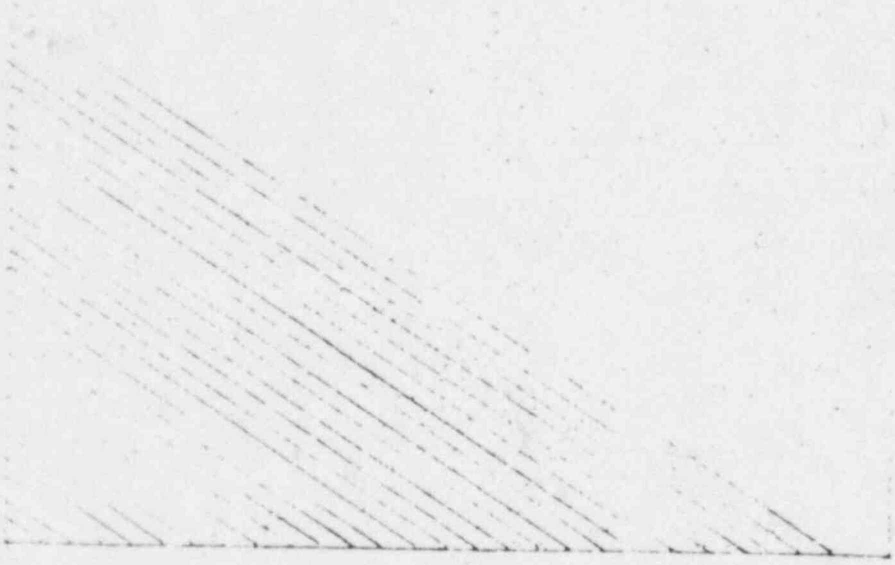

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ERRATA FOR THE METROPOLITAN
EDISON CO. EDITION OF THE TRAINING MANUAL

PAGE #	LINE #	CORRECTION
1-3	18	${}^{A}_{Z}X \to {}^{A-4}_{Z-2}Y + {}^{4}_{2}He$; ${}^{210}_{84}Po$ ....
1-6	9	1 Ci = 3.7×10^{10} *sec*$^{-1}$ (exactly)
2-10	34	.... = (½) (3000 lb.) $\frac{(44 \text{ ft/sec})^2}{(32 \text{ ft/sec}^2)}$
3-6		Fig. 4B, L should be *l*
3-6	26	electrons in "*l*" shell.
3-8		Fig. **6**, L should be *l*
4-9	4	question which arises at this time *is*,
5-13		Fig. 9, radius vector r is not indicated.
5-17	40	$k_\infty = \eta \epsilon p f$
7-2	11	The major disadvantage is the *problem* ...
7-3		Fig. 2, As in Fig. 1, the ordinate of the left hand drawing should be labeled *Axial Length*, and the ordinate of the right hand drawing should be labeled φ with the abscissa labeled *Radial distribution*.
7-9	23	... let us consider what *determines* its
8-2	9	associated with adding and *rejecting* heat.
8-35	25	8.2.10 *Uncertainties* and Hot Channel Factors
9-3	20	The neutron is one of the basic *building* blocks
9-15	27	... it is perhaps *Less* of a problem ...
9-29	13	reassure oneself of the *adequacy* ....
9-40	4	delete *not*
9-40	27	of operation make *it* necessary .....
9-52	24	Once the rod is set in motion, *it* must .....

10-6	3	human body) absorbs thermal neutrons.
10-6	8	because cadmium has a high *affinity* .....
10-7	39	Each of the above methods *is* dependent ....
10-14	2	listed in Figure *8*
10-15	21 & 22	buildup factors as the *product* of the attenuation coefficient and the thickness.
10-16		Fig. 11, first column heading should be μx.
10-16	1	The product $\mu \cdot x$ is ....
10-21	15	is in the calibration of *portable* survey ......
10-23	6	I = 1.11 mR/hr
10-23	23	..... stainless steel. *Of* course, ...
10-25	5	gammas due to the *thermal* neutron ....
10-25	21	(barium sulfate) and has a density of 3.5 gm/cm^3
10-30	12	ususally below 1 mR/hr.

10^{9} = 7. 33 [illegible]

A LECTURE SERIES IN
NUCLEAR POWER PLANT OPERATIONS
TRAINING

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NOTICE

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HEALTH PHYSICS

1.0 Introduction

Health Physics is a profession devoted to the protection of man and his environment from unwarranted radiation exposure. The health physicist is a person engaged in the study of problems and practices of providing radiation protection; he is concerned with an understanding of the mechanisms of radiation damage, with the development and implementation of methods and procedures necessary to evaluate radiation hazards and with providing protection to man and his environment from unwanted radiation exposure.

Putting the above formal definition in more practical terms, the applied Health Physicist is concerned with one or more of the following aspects of radiation protection.

1. Personnel monitoring
2. Radiation control
3. Contamination control
4. Bioassay
5. Environmental radioactivity levels
6. Transportation of radioactive material
7. Waste disposal
8. Use of radiation producing machines
9. Shielding
10. Laboratory design
11. Experiment design
12. Dosimetry
13. Instrumentation
14. Decontamination
15. Protective equipment and clothing
16. Emergency planning
17. Biological effects of radiation
18. Ventilation control
19. Calibration of instruments

The following notes are intended to be used in conjunction with the programmed instruction book Radiation Monitoring by J. E. Wade and G. E. Cunningham, the text Principles of Radiation Protection by K. Z. Morgan and J. E. Turner, the series of National Bureau of Standards Handbooks, and the Radiological Health Handbook. Laboratory demonstrations and films will be used to supplement the information and demonstrate important techniques. Shielding and basic atomic and nuclear physics are covered in other sections of this manual and will be only briefly mentioned in this section.

1.1 The Atom

The atom is the basic building block of matter. It consists of a central nucleus around which electrons revolve in orbits. The nucleus is composed of protons and neutrons. The proton has a positive charge numerically equal to the charge on one electron (1.6×10^{-19} coulombs), and a mass of 1.007593 atomic mass units (1 amu = 1.65979×10^{-24} grams). The neutron is electrically neutral and has a mass of 1.008982 amu. The electron has a negative charge and a mass of 0.000549 amu. The nucleus is extremely small and very dense (i.e., a sphere with the dimensions of an ordinary marble and the same density as the nucleus would weigh about 40,000,000 tons). The orbit of the electron is very large compared to the size of the nucleus so that the volume of the atom is mostly empty space.

1.2 Molecules

Molecules are the smallest divisions of matter with the same chemical and physical characteristics. Molecules may be composed of one or more atoms. An element contains only atoms with the same chemical properties e.g., iron, lead, silicon, oxygen, hydrogen. Molecules are usually designated by chemical symbols with subscripts to designate the number of atoms of each element per molecule.

H_2O -- water molecule composed of 2 hydrogen and 1 oxygen atoms

$NaCl$ - salt molecule composed of 1 sodium and 1 chlorine atom

$CaCO_3$ - calcium carbonate composed of 1 calcium, 1 carbon, and 3 oxygen atoms

O_2 -- oxygen molecule composed of 2 atoms of oxygen

Ne -- neon molecule composed of 1 atom of neon

1.3 Isotopes

The mass of the electron may be neglected and the mass of an atom estimated in atomic mass units as simply the sum of the number of protons and neutrons. The mass number of the atom is designated by a superscript added to the left of the element symbol, i.e., ^{59}Fe (or a superscript to the right as in older usage, Fe^{59}). The normal atom has a corresponding orbital electron for every proton in the nucleus so the net electrical charge is zero. The chemical properties of an atom or molecule are determined by the number of electrons and their configuration. Thus, all atoms of a given element have the same number of electrons and protons. The number of neutrons may vary, but the chemical properties are not strongly affected by the weight of the atom. The number of protons or atomic number is indicated by a subscript added to the left of the chemical symbol, i.e., $^{59}_{26}Fe$ (iron, 26 protons and 33 neutrons for a total atomic weight of 59 amu). Atoms of the same element with different numbers of neutrons are called isotopes, i.e., $^{54}_{26}Fe$, $^{56}_{26}Fe$, $^{57}_{26}Fe$. Normal iron is composed of a mixture of isotopes with the following atom percentages, 5.82% ^{54}Fe, 91.66% ^{56}Fe, 2.19% ^{57}Fe, and 0.33% ^{58}Fe. Atoms with the same atomic mass number are called isobars, those with the same number of neutrons isotones.

1.4 Radioactivity

Some atoms are unstable and are transformed to a more stable state by the spontaneous disintegration of the nucleus with the emission of radiation. These atoms are called radioactive. In general, the process must have a measurable lifetime (10^{-18} sec to 10^{17} yr) for the atom to be considered radioactive. The transition to a stable atom may take place with one or a series of separate disintegrations.

1.4.1 Alpha Radiation

A radioactive atom may decay by emission, from the nucleus, of a particle consisting of 2 protons and 2 neutrons (helium nucleus $^{4}_{2}$He). The particle is called an alpha particle or alpha radiation (α). The process involves a decrease of 4 in the mass number of the nucleus and a decrease of 2 in the atomic number.

$$^{A}_{Z}X \longrightarrow {}^{A-4}_{Z-2}Y + {}^{4}_{2}He\,;\quad {}^{210}_{84}Po \longrightarrow {}^{206}_{82}Pb + \alpha$$

1.4.2 Beta Radiation

If a nucleus decays by emission of a negative or positive electron the process is called beta decay. The particles are called beta particles (β, β^- or β^+). The positive electron may also be called a positron.

$$^{A}_{Z}X \longrightarrow {}^{A}_{Z+1}Y + \beta^-$$

$$^{A}_{Z}X \longrightarrow {}^{A}_{Z-1}Y + \beta^+$$

1.4.3 Gamma Radiation

After the emission of particles such as alpha or beta radiation, from the nucleus, the daughter or product nucleus may be left in an excited state. The transition of the nucleus to a stable or ground state is accompanied by the emission of high energy photons called gamma rays. The gamma emission usually takes place immediately after the particulate radiation is emitted (within a time too short to measure). In some cases the excited state of the nucleus may have a relatively long life before the gamma ray is emitted, in which case the two energy levels of the nucleus are called *isomers*.

$$^{A}_{Z}X^{*} \longrightarrow {}^{A}_{Z}X + \gamma$$

1.4.4 Orbital Electron Capture

Another type of radioactive decay is absorption of an orbital electron by the nucleus or *K-capture*. This is usually accompanied by the emission of an X-ray.

$$^{A}_{Z}X + e^{-} \longrightarrow {}^{A}_{Z-1}X$$

1.4.5 Internal Conversion

Internal conversion may be described as the emission of a photon from the nucleus, which causes ejection of an orbital electron. It is probably best described as a type of gamma decay even though an electron is emitted from the atom.

1.5 Ionization

Charged particles passing through matter transfer energy to the matter and hence gradually lose energy and are "*absorbed*". The energy lost may appear as *excitation* of the electrons in the molecules. That is, the orbital electrons may be raised to higher energy levels and exist in an excited state. If enough energy is absorbed, an electron may even be ejected from the molecule. Any process such as this which causes the neutral atom to lose or gain electrons and acquire a net charge is called *ionization*. The negatively charged electron and the positively charged molecule are called an *ion pair*. The energy required to cause ionization is about 10-15 electron volts (ev) for most molecules. However, because of the energy lost in excitation and other processes the average energy required to form an ion pair, the *W value*, is considerably higher. For air the W value for electrons is 33.7 ev and for alpha particles 35 ev. The electrons which are ejected by the incident radiation have relatively high energies and can also cause ionization. They are sometimes called *delta rays* or *secondary radiation*. Radiation which is uncharged (neutrons and γ rays) produces ionization by indirect means. This usually involves a reaction or interaction in which the uncharged radiation produces a charged particle, such as an electron or proton, which then causes ionization.

1.5.1 Photoelectric Effect

In the *photoelectric effect* a gamma ray interacts with an electron and the photon disappears and the electron is ejected with a kinetic energy equal to the energy of the gamma ray, minus the binding energy of the electron. The probability for photoelectric type absorption is highest for low energy radiation and for high atomic number materials, varying about as E^{-3} and Z^4.

1.5.2 Compton Effect

In the Compton process a gamma ray interacts with an electron and is scattered. The scattered gamma ray has less energy than the incident gamma ray and the electron is ejected with the difference in energy. The scattered photon can undergo further Compton or photoelectric interactions. The probability for Compton interactions decreases slowly with increasing energy of the photon and is relatively independent of atomic number. It is the predominant absorption process for moderate energy photons and all but very high atomic number materials.

1.5.3 Pair Production

At photon energies in excess of 1.02 Mev pair production can take place. In this process a photon is converted to an electron and a positron. The photon must have an energy of at least 1.02 Mev as this is the combined rest-mass energy equivalent of the electron and positron formed. The energy in excess of 1.02 Mev appears as kinetic energy of the particles. The positron annihilates with another electron, usually after losing all kinetic energy, forming two 0.51 Mev photons. The probability for pair production increases slowly with energy but strongly with increasing atomic number (Z^2).

1.5.4 Neutron Interactions

Neutrons are also uncharged and cause ionization by secondary charged particles. At thermal energies (.025 ev) the most important reactions are the absorption of a neutron by hydrogen forming deuterium with the emission of a 2.2 Mev gamma ray [$^1H(n,\gamma)\,^2H$] and the absorption by nitrogen forming radioactive ^{14}C and emitting a 0.6 Mev proton [$^{14}N(n,p)\,^{14}C$]. At high energies the collision of neutrons with the nuclei of atoms, especially hydrogen, imparts kinetic energy to the nuclei which become charged particle radiation. These "recoil nuclei" cause a great deal of ionization. Many other types of interactions, producing a variety of ionizing radiations, are possible with neutrons.

1.6 Units

In 1925 the International Commission on Radiological Units and Measurement (ICRU) was organized to develop internationally acceptable quantities and units of radiation and radioactivity, procedures for measurement, and recommendations for application of procedures. The following units are taken from the ICRU report on units, number 10a, 1962 (also available as National Bureau of Standards Handbook 84).

1.6.1 Activity

The activity (A) of a quantity of a radioactive nuclide is the quotient of ΔN by Δt where ΔN is the number of nuclear transformations which occur in this quantity in time Δt.

$$A = \frac{\Delta N}{\Delta t}$$

The special unit of activity is the curie (Ci).

$$1\ \mathrm{Ci} = 3.7 \times 10^{10}\ \mathrm{s}^{-1}\ \text{(exactly)}$$

NOTE: In accordance with the former definition of the curie as a unit of quantity of radioactive nuclide, it was customary and correct to say: "X curies of P-32 were administered . . . . . ." It is still permissible to make such statements rather than use the longer form which is now correct: "A quantity of P-32 was administered whose activity was Y curies."

1.6.2 Exposure

The exposure (X) is the quotient of ΔQ by Δm, where ΔQ is the sum of the electrical charges on all the ions of one sign produced in air when all the electrons (negatrons and positrons), liberated by photons in a volume element of air whose mass is Δm, are completely stopped in air.

$$X = \frac{\Delta Q}{\Delta m}$$

The special unit of exposure is the roentgen (R).

$$1\mathrm{R} = 2.58 \times 10^{-4}\ \mathrm{C/kg}^{*}$$

*This unit is numerically identical with the old one defined as 1 e.s.u. of charge per .001293 gram of air. C is the abreviation for coulomb.

NOTES: (a) The words "charges on all the ions of one sign" should be interpreted in the mathematically absolute sense. (b) The ionization arising from the absorption of bremsstrahlung emitted by the secondary electrons is not to be included in ΔQ. Except for this small difference, significant only at high energies, the exposure as defined above is the ionization equivalent of the kerma in air. (c) With present techniques it is difficult to measure exposure when the photon energies involved lie above a few Mev or below a few kev. (d) As in the case of kerma (1.6.5), note (c), it may often be convenient to refer to a value of exposure or of exposure rate in free space or at a point inside a material different from air. In such case the value will be that which would be determined for a small quantity of air placed at the point of interest. It is, however, permissible to make a statement such as: "The exposure at the point P inside a water phantom is . . . ."

1.6.3 Absorbed Dose

The absorbed dose (D) is the quotient of ΔE_d by Δm, where ΔE_d is the energy imparted by ionizing radiation to the matter in a volume element, Δm is the mass of the matter in that volume element.

$$D = \frac{\Delta E_d}{\Delta m}$$

The special unit of absorbed dose is the rad.

$$1 \text{ rad} = 100 \text{ erg/g} = \frac{1}{100} \text{ J/kg}$$

NOTE: J is the abbreviation for Joule.

1.6.4 Dose Equivalent

1. For protection purposes it is useful to define a quantity which will be termed the "dose equivalent", (DE).

2. (DE) is defined as the product of absorbed dose, D, quality factor, (QF), dose distribution factor, (DF), and other necessary modifying factors. (DE) = D (QF) (DF) ....

3. The unit of dose equivalent is the "rem". The dose equivalent is numerically equal to the dose in rads multiplied by the appropriate modifying factors.

The term "RBE dose" has in past publications of the Commission not been included in the list of definitions but was merely presented as a "recognized symbol". In its 1959 report the Commission also expressed misgivings over the utilization of the same term, "RBE", in both radiobiology and radiation protection. It now recommends that the term RBE be used in radiobiology only and that another name be used for the linear-energy-transfer-dependent factor by which absorbed doses are to be multiplied to obtain, for purposes of radiation protection, a quantity that expresses, on a common scale for all ionizing radiations, the irradiation incurred by exposed persons. The name recommended for this factor is the quality factor, (QF). Provisions for other factors are also made. Thus a distribution factor, (DF), may be used to express the modification of biological effect due to non-uniform distribution of internally deposited isotopes.. The product of absorbed dose and modifying factors is termed the dose equivalent, (DE) As a result of discussions between ICRU and ICRP the above formulation has been agreed upon.

The following is taken from 10 CFR 20 and gives the "quality factors" or RBE values commonly used.

20.4 Units of radiation dose

(a) "Dose", as used in this part, is the quantity of radiation absorbed, per unit of mass, by the body or by any portion of the body. When the regulations in this part specify a dose during a period of time, the dose means the total quantity of radiation absorbed, per unit of mass, by the body or by any portion of the body during such period of time.. Several different units of dose are in current use. Definitions of units as used in this part are set forth in paragraphs (b) and (c) of this section.

(b) The rad, as used in this part, is a measure of the dose of any ionizing radiation to body tissues in terms of the energy absorbed per unit mass of the tissue. One rad is the dose corresponding to the absorption of 100 ergs per gram of tissue. (One millirad (mrad)=0.001 rad.)

(c) The rem, as used in this part, is a measure of the dose of any ionizing radiation to body tissue in terms of its estimated biological effect relative to a dose of one roentgen (r) of X-rays. (One millirem (mrem)=0.001 rem.) The relation of the rem to other dose units depends upon the biological effect under consideration and upon the conditions of irradiation. For the purpose of the regulations in this part, any of the following is considered to be equivalent to a dose of one rem:

1. A dose of 1 r due to X- or gamma radiation;
2. A dose of 1 rad due to X-, gamma, or beta radiation;
3. A dose of 0.1 rad due to neutrons or high energy protons;
4. A dose of 0.05 rad due to particles heavier than protons and with sufficient energy to reach the lens of the eye.

1.6.5 Kerma

The kerma (K) is the quotient of ΔE_k by Δm, where ΔE_k is the sum of the initial kinetic energies of all the charged particles liberated by indirectly ionizing particles in a volume element of the specified material, Δm is the mass of the matter in that volume element.

$$K = \frac{\Delta E_k}{\Delta m}$$

NOTES: (a) Since ΔE_k is the sum of the initial kinetic energies of the charged particles liberated by the indirectly ionizing particles, it includes not only the kinetic energy these charged particles expend in collisions but also the energy they radiate in bremsstrahlung. The energy of any charged particle is also included when these are produced in secondary processes occurring within the volume element. Thus the energy of Auger electrons is part of ΔE_k.

(b) In actual measurements Δm should be so small that its introduction does not appreciably disturb the radiation field. This is particularly necessary if the medium for which kerma is determined is different from the ambient medium; if the disturbance is appreciable an appropriate correction must be applied.

(c) It may often be convenient to refer to a value of kerma or of kerma rate for a specified material in free space or at a point inside a different material. In such a case the value will be that which would be obtained if a small quantity of the specified material were placed at the point of interest. It is, however, permissible to make a statement such as: "The kerma for air at the point P inside a water phantom is ...." recognizing that this is a shorthand version of the fuller description given above.

(d) A fundamental physical description of a radiation field is in the intensity (energy flux density) at all relevant points. For the purpose of dosimetry, however, it may be convenient to describe the field of indirectly ionizing particles in terms of the kerma rate for a specified material. A suitable material would be air for electromagnetic radiation of moderate energies, tissue for all radiations in medicine or biology, or any relevant material for studies of radiation effects.

Kerma can also be a useful quantity in dosimetry when charged particle equilibrium exists at the position and in the material of interest, and bremsstrahlung losses are negligible. It is then equal to the absorbed dose at that point. In beams of x or gamma rays or neutrons, whose energies are moderately high, transient charged-particle equilibrium can occur; in this condition the kerma is just slightly less than the absorbed dose. At very high energies the difference becomes appreciable. In general, if the range of directly ionizing particles becomes comparable with the mean free path of the indirectly ionizing particles, no equilibrium will exist.

1.5.6 The particle fluence or fluence (Φ) of particles is the quotient of ΔN by Δa, where ΔN is the number of particles which enter a sphere of cross-sectional area Δa.

$$\Phi = \frac{\Delta N}{\Delta a}$$

1.6.7 The particle flux density or flux density (ϕ) of particles is the quotient of $\Delta\Phi$ by Δt where $\Delta\Phi$ is the particle fluence in time Δt.

$$\phi = \frac{\Delta\Phi}{\Delta t}$$

NOTE: This quantity may also be referred to as particle fluence rate.

1.6.8 The energy fluence (F) of particles is the quotient of ΔE_f by Δa, where ΔE_f is the sum of the energies, exclusive of rest energies, of all the particles which enter a sphere of cross-sectional area Δa.

$$F = \frac{\Delta E_f}{\Delta a}$$

1.6.9 The energy flux density or intensity (I) is the quotient of ΔF by Δt where ΔF is the energy fluence in the time Δt.

$$I = \frac{\Delta F}{\Delta t}$$

NOTE: This quantity may also be referred to as energy fluence rate.

1.6.10 The mass stopping power S/ρ of a material for charged particles is the quotient of dE_s by the product of dl and ρ, where dE_s is the average energy lost by a charged particle of specified energy in traversing a path length dl, and ρ is the density of the medium.

$$\frac{S}{\rho} = \frac{1}{\rho}\frac{dE_s}{dl}$$

NOTE: dE_s denotes energy lost due to ionization, electronic excitation and radiation. For some purposes it is desirable to consider stopping power with the exclusion of bremsstrahlung losses. In this case S/ρ must be multiplied by an appropriate factor that is less than unity.

1.6.11 The linear energy transfer (L) of charged particles in a medium is the quotient of dE_L by dl where dE_L is the average energy locally imparted to the medium by a charged particle of specified energy in traversing a distance of dl.

$$L = \frac{dE_L}{dl}$$

NOTES: (a) The term "locally imparted" may refer either to a maximum distance from the track or to a maximum value of discrete energy loss by the particle beyond which losses are no longer considered as local. In either case the limits chosen should be specified.

(b) The concept of linear energy transfer is different from that of stopping power. The former refers to energy imparted within a limited volume, the latter to loss of energy regardless of where this energy is absorbed.

1.6.12 The average energy (W) expended in a gas per ion pair formed is the quotient of E by N_w, where N_w is the average number of ion pairs formed when a charged particle of initial energy E is completely stopped by the gas.

$$W = \frac{E}{N_w}$$

NOTES: (a) The ions arising from the absorption of bremsstrahlung emitted by the charged particles are not to be counted in N_w.

(b) In certain cases it may be necessary to consider the variation in W along the path of the particle, and a differential concept is then required, but is not specifically defined here.

1.6.13 A nuclide is a species of atom having specified numbers of neutrons in its nucleus.

1.6.14 The specific gamma ray constant (Γ) of a gamma-emitting nuclide is the quotient of $l^2\,\Delta X/\Delta t$ by A, where $\Delta X/\Delta t$ is the exposure rate at a distance l from a point source of this nuclide having an activity A.

$$\Gamma = \frac{l^2 \Delta X}{A \Delta t}$$

Special units of specific gamma ray constant are $Rm^2h^{-1}Ci^{-1}$ or any convenient multiple of this.

NOTE: It is assumed that the attenuation in the source and along l is negligible. However, in the case of radium the value of Γ is determined for a filter thickness of 0.5 mm of platinum and in this case the special units are $Rm^2h^{-1}g^{-1}$ or any convenient multiple of this.

1.6.15 Directly ionizing particles are charged particles (electrons, protons, α-particles, etc.) having sufficient kinetic energy to produce ionization by collision.

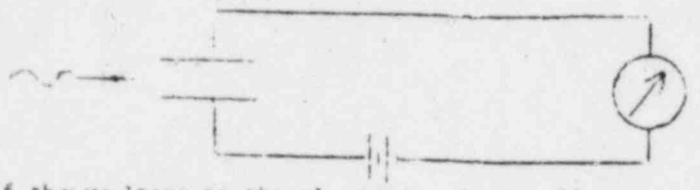
1.6.16 Indirectly ionizing particles are uncharged particles (nuetrons, photons, etc.) which can liberate directly ionizing particles or can initiate a nuclear transformation.

1.6.17 Ionizing radiation is any radiation consisting of directly or indirectly ionizing particles or a mixture of both.

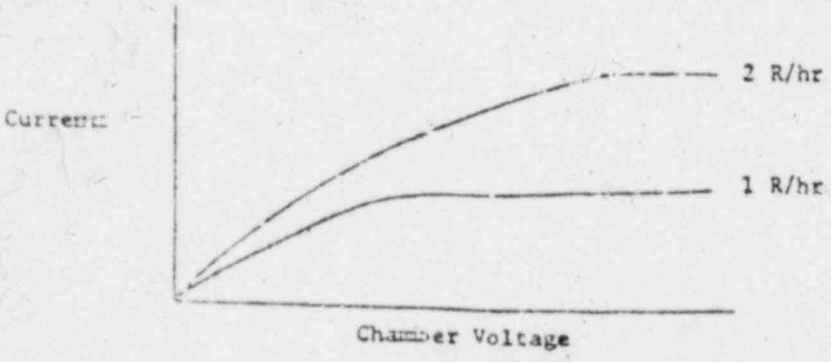
1.6.18 The energy imparted by ionizing radiation to the matter in a volume is the difference between the sum of the energies of all the directly and indirectly ionizing particles which have entered the volume and the sum of the energies of all those which have left it, minus the energy equivalent of any increase in rest mass that took place in nuclear or elementary particle reactions within the volume.

1.7 Ionization Chambers

The ionization of a gas is the method most commonly used for detecting and measuring radiation. An ionization chamber is an instrument for collecting the ions produced by radiation in a volume of gas, with no gas amplification. A typical ionization chamber consists of two electrodes with a potential difference so as to collect the ions formed in the gas between them. Common shapes for the chamber electrodes are parallel plate, cylindrical, or spherical. The current caused by the collection of the ions is measured to determine the amount of radiation.



If the voltage on the electrodes is insufficient, some of the ions may recombine before reaching the electrodes. As shown in the figure below the current increases as the voltage on the electrodes increases until all the ions are collected and saturation is reached. The voltage at which saturation occurs increases as the dose rate increases.



The effect of recombination is also dependent upon the size of the chamber and the design features. A smaller chamber requires less voltage to achieve saturation. Likewise, efficient design to reduce "dead spots" or areas of low electric field can also reduce the saturation voltage.

A variety of gases is used in ionization chambers. Air is commonly used to measure exposure in Roentgens because the Roentgen is defined in terms of ionization in air. Boron trifluoride may be used in chambers to measure thermal neutron radiation. Argon or argon mixtures are commonly used to increase the sensitivity of chambers. The same effect can be achieved by increasing the size of the chamber or by increasing the gas pressure.

The response of an ion chamber is very dependent upon the material of which the walls are made, as most of the electrons which cause ionization are produced in the wall material (because the wall material contains much more mass than the gas). Thus, chambers are designed with air-equivalent or tissue-equivalent walls depending on their intended use.

Because of the very small currents in ionization chambers, the resistance of the insulators must be very large. The high impedance makes ionization chambers sensitive to humidity and other effects and the response is slow. Amplifiers to measure the small currents are relatively complicated and expensive, which is reflected in the ionization chambers used for radiation measurements.

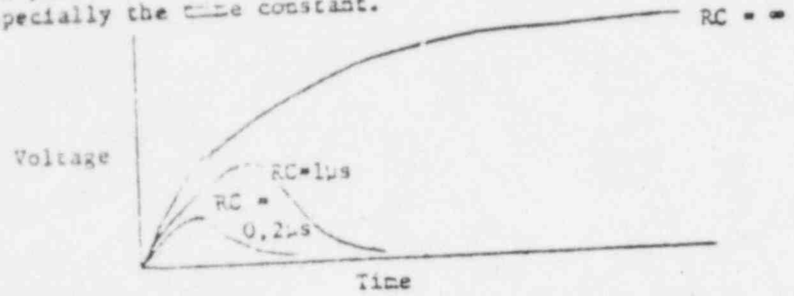
1.8 Proportional Counters

If the electric field strength is large enough, electrons can gain enough energy to cause further ionization. The electrons thus formed cause more ionization and an "electron avalanche" is initiated. This principle is used in proportional and Geiger-Mueller counters to "amplify" the number of electrons formed along the path of the ionizing particle. The gas gain or amplification in a proportional counter is commonly 10^3 or 10^4 (recall that the gas gain in an ionization chamber is unity). In order to achieve high electric field strengths a very small diameter anode wire is used. The field strength, Er, at radius r for a counter voltage Vo is given by

$$Er = \frac{Vo}{r \ln (b/a)}$$

where a is the radius of the wire anode and b is the radius of the collector or cathode. It is in the region very close to the center wire that the electric field strength becomes strong enough to initiate the electron avalanche. Each electron from the primary ionization initiates such an avalanche. In the proportional region the avalanches are small and do not overlap. Since the number of electrons formed in each avalanche is constant, the total number of electrons collected is proportional to the initial number of ions and thus to the energy absorbed in the counter gas. Discrimination between radiations which cause varying numbers of primary ions is thus possible. If all of the radiation is expended in the counter, the number of electrons collected is a measure of the energy of the radiation. As the voltage is increased the gas gain increases to a point where the avalanches overlap for radiations which create large numbers of primary ions. The counter is then said to be operating in the region of limited proportionality (proportional for lightly ionizing particles but not for densely ionizing ones).

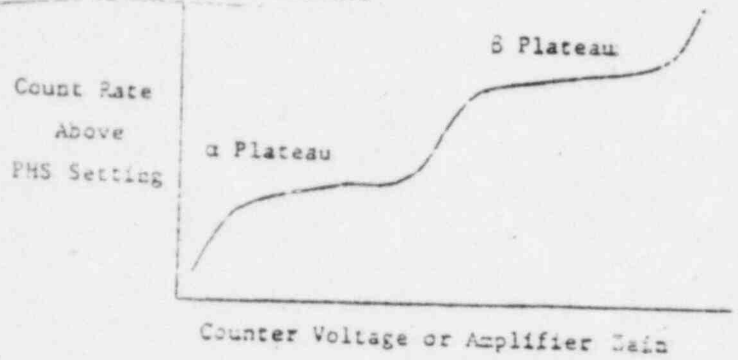
The electrons are usually collected very rapidly because of their proximity to the center wire and their high mobility. The positive ions are collected much more slowly. The pulse from the proportional counter therefore consists of a very sharply rising component from the electrons followed by the slowly rising, but major, part due to the positive ions. The actual shape and size of the pulse is very dependent upon the associated electronics, especially the time constant.



The avalanche is terminated when all the electrons have been collected in the proportional counter. The number of positive ions is insufficient to shield the center wire and prevent further operation of the counter. The excited atoms may emit ultraviolet radiation in returning to the ground state, which can cause the ejection of photo-electrons from the cathode. Electrons may also be ejected from the cathode during the neutralization of the positive ions. These electrons could then cause further counts. To prevent the above, "quenching" gases are added. The quenching gas molecules absorb the ultraviolet radiation and are dissociated instead of ionized. The quenching gas also has a lower ionization potential than the counting gas, so that ionization is all transferred to the quenching gas during the drift to the cathode. When neutralizing the cathode , the quenching gas dissociates instead of allowing electrons to be formed so the counting event is terminated. In the proportional counter quenching may be neglected but performance is usually better if a quenching gas such as methane is added to the usual noble gas, such as 10% CH_4 and 90% Ar. There are two general types of quenching gases, organic molecules and halogens. The halogen molecules recombine after dissociation and are not "used up" as is the case with the organic gases.

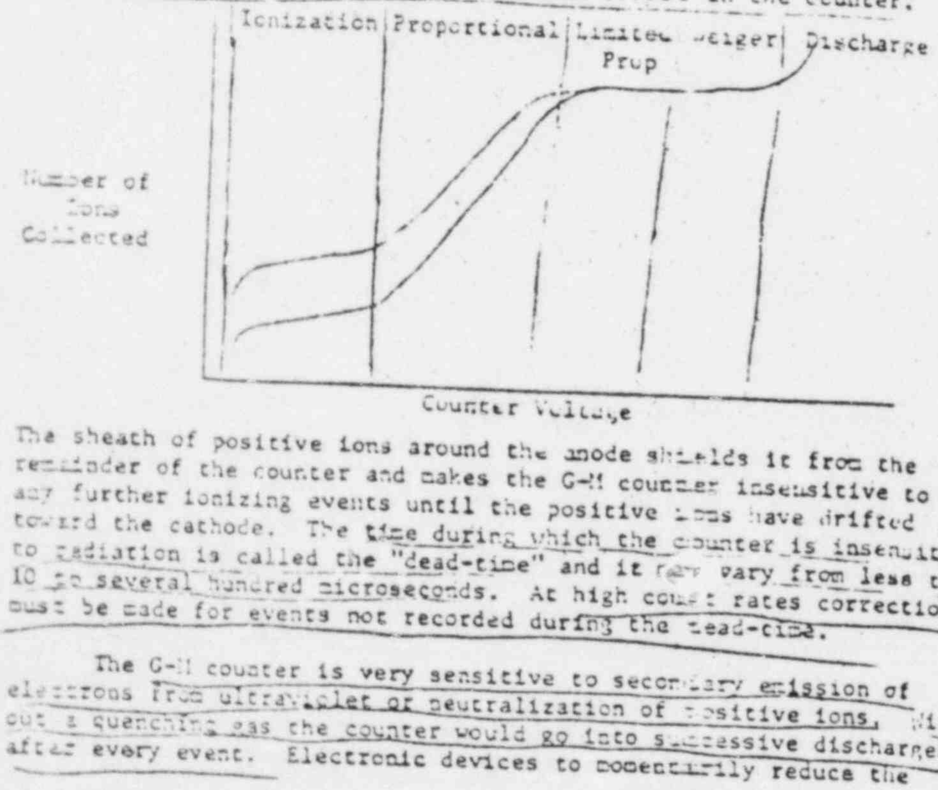
Most gases, with the exception of electronegative ones, can be used in proportional counters. Tissue equivalent organic gases are used in dosimeters to measure tissue dose and boron trifluoride gas enriched in ^{10}B is commonly used in thermal neutron counters. Oxygen and other electronegative gases should be excluded, although even air may be used in some applications where performance standards are not too rigid. The fact that the gas amplification takes place very close to the center wire makes the proportional counter relatively insensitive to cathode shape and counters can be tailored to fit a variety of uses. Cylindrical, hemispherical, bell and disk are some of the shapes commonly used, in addition to the sphere.

Using a proportional counter it is possible to discriminate against radiation of different energy or type. The discriminator on a scaler may be set to exclude all pulses due to beta radiation and count only those from high energy alpha particles. The instrument is then said to be operating on the "alpha plateau". Similarly the setting can be adjusted so that both beta and alpha are counted and a "beta plateau" is observed.



1.9 Geiger-Mueller Counters

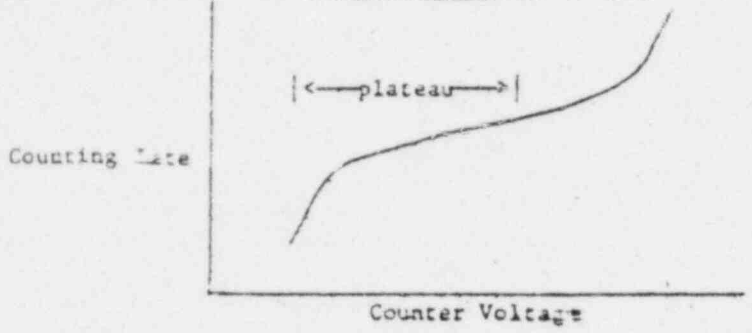
As the voltage in a proportional counter increases, the gas increases to the point where an electron avalanche from even a single electron spreads all along the anode wire. This is the area of maximum gas gain and all pulses are of approximately the same size, regardless of the number of initial ions formed in the counter.



The sheath of positive ions around the anode shields it from the remainder of the counter and makes the G-M counter insensitive to any further ionizing events until the positive ions have drifted toward the cathode. The time during which the counter is insensitive to radiation is called the "dead-time" and it may vary from less than 10 to several hundred microseconds. At high count rates correction must be made for events not recorded during the dead-time.

The G-M counter is very sensitive to secondary emission of electrons from ultraviolet or neutralization of positive ions. Without a quenching gas the counter would go into successive discharges after every event. Electronic devices to momentarily reduce the collection voltage may be used in place of or to supplement the quenching gas. This procedure is termed external quenching.

The gas gain remains relatively constant for several hundred volts above the threshold of the Geiger region. The count rate is thus constant and a "Geiger plateau" is observed. Above the plateau a continuous discharge takes place.



1.10 Scintillation Counters

In certain materials the absorption of ionizing radiation can produce "scintillations" or photons of light. This light can be detected by photomultiplier tubes and converted to an electric signal. The property of emitting light when exposed to ionizing radiations is called luminescence. If the emission of light is fast the process is called fluorescence. If the emission is delayed the process is called phosphorescence. The distinction between fluorescence and phosphorescence is not very clear but is usually given as a time of about 10^{-8} seconds. The quantity of light emitted is usually proportional to the total energy absorbed from the ionizing radiation. Thus, the light pulse is proportional to the energy absorbed or to the energy of the radiation, in the case of complete absorption. The emission of light should be of very short duration to give a sharp pulse of intense light. This also reduces interference between successive pulses. The scintillator should be transparent to the light emitted so there is minimum absorption of the light before it reaches the photomultiplier tube. It is also advantageous if the scintillator can be obtained in large sizes and of relatively high density to allow for maximum absorption.

The scintillation detector may be operated as a pulse counter or the total current can be integrated and used as a measure of the radiation exposure or dose, as in an ionization chamber. The pulse mode is used for counting or analysis of the energy of the incident radiation. The current mode is sometimes used in monitoring devices. The advantage of the scintillation detector over the ionization chamber lies in the higher density of the detector, allowing more energy absorption, and the high gain of the photomultiplier tube, about 10^6. These factors combine to produce a much greater electrical signal for the same intensity of incident radiation.

There is a general class of scintillators composed of organic crystals or powders. Of these anthracene is the best known. These detectors are very useful for beta radiation detection because in even thin layers beta particles are completely absorbed but gamma ray absorption is very low and causes little interference. Solutions of organic scintillators may be prepared in liquid or solid (plastic) form. The solutions can be poured into or cast in any shape and therefore find use in many special applications, especially those requiring a large volume. The use of liquid scintillators has become the most important method of radiation detection in the biological field. Samples may be dissolved right in the scintillator allowing for very high efficiency and negligible absorption of low energy beta particles. Because of the large hydrogen content plastic scintillators are sometimes used as fast neutron detectors, utilizing the recoil protons produced.

The inorganic crystals are the most efficient detectors of gamma radiation because of the high density and consequently the large absorption of quantum energy. In most cases "activators" must be added to the pure inorganic crystal to produce luminescent centers. An example is thallium added to sodium iodide, the most popular inorganic scintillator. Zinc sulfide is not transparent to its own radiation but can be used in very thin layers. In this form it can completely absorb alpha particles with insignificant absorption of beta or gamma radiation. It, therefore, is widely used in alpha detectors. Lithium iodide enriched in ^{6}Li is an excellent detector for thermal neutrons because of the ^{6}Li (n,α) ^{3}He reaction which produces a large pulse, which is easily distinguished from gamma radiation.

Scintillators can also be fabricated from various glasses composed of the oxides of boron, silicon, sodium, aluminum, lithium and magnesium with an activator such as cesium oxide. The glass scintillators can be formed in any shape and also closely match the refractive index of the photomultiplier tube. The boron or lithium content of the glass can be varied to make it sensitive to thermal neutrons for special applications.

The noble gases in a very pure state also exhibit the property of luminescence and can be used as scintillators. Their use, however, has been very limited. Because of the low density, gaseous scintillators are really suitable only for heavy charged particles such as alpha radiation.

1.11 Photographic Emulsions

Ionizing radiation, like visible light, can produce a latent image in a photographic emulsion. The path of charged particles passing through an emulsion is recorded as a number of silver grains when the film is developed. In some instances the actual path is examined microscopically to determine such things as charge, mass, and velocity of the particle. In the case of X- or gamma radiation the blackening of the film by a large number of individual tracks is used as an indication of the exposure or dose.

The photographic emulsion consists of silver halide crystals (AgBr) in gelatin. The silver halide grains vary in size for different emissions (0.3 microns nuclear track emulsions; 2 microns Xray emulsions) and the weight fraction of the silver halide in the emulsion may vary from 0.3 (photographic) to 0.8 (nuclear track). The emulsions are produced in a variety of thicknesses but the usual emulsion is 10-25 microns with a thin gelatin cover (T coat) of about 0.5 micron.

Radiation can excite electrons in the silver halide crystal and raise them to the conduction band. These electrons then travel to sensitivity centers in the crystal consisting of impurities or deformities. The electrons can then neutralize the small fraction of silver ions free to migrate in the crystal. Thus, a clump of silver atoms is formed in the silver halide grain. These clumps serve to catalyze the reduction of the silver in the crystal during the process of development.

The developer is a substance which will reduce the silver in the grains containing a latent image but does not effect the unexposed grains. Thus, the formation of a few silver atoms can catalyze a reaction leading to the reduction of a much larger number of silver ions giving a multiplication of about 10^{11}. The undeveloped silver halide is dissolved and removed from the emulsion leaving the grains of silver to form the image which serves as a measure of the radiation to which the emulsion had been exposed.

The nuclear emulsion is a very valuable radiation detector by virtue of its small size, simple construction, ruggedness, sensitivity, and long sensitive time (it is always "on" yet requires no power). Some difficulties encountered in the use of photographic emulsions are the large energy-dependence, fading of the latent image, fogging, and sensitivity to temperature and humidity in storage, use, and development.

The dependence on energy is due to the large amounts of high Z material (AgBr) in the emulsion. This is very important if a determination of dose in air, tissue, or other low Z material is being made. Usually a shield is chosen to preferentially absorb the low energy photons and smooth out the energy response curve.

For the case of mixed beta-gamma radiation a number of filters or shields is used, usually an "open window" or no filter, a plastic filter, and one or more metal filters. The beta radiation will be recorded behind the open window but not behind the plastic filter. However, the plastic filter will have little effect on the gammas. Thus, the difference between the plastic shield and open window density should be an indication of the beta dose. The density behind the metal shields will be an indication of the gamma dose.

The ratio of the film density behind the plastic filter to the film density behind the metal filters can be used as an indication of the average photon energy. However, it is very difficult to distinguish low energy X-rays from beta radiation.

Fast neutron film dosimetry is usually done by counting the tracks formed by recoil protons. This method is good only for high energy neutrons (>0.5 Mev) because the proton tracks are too short to distinguish from ordinary film fogging at low energy. The method is relatively insensitive and requires a tedious microscopic examination of the emulsion. Fading of the latent image is much more rapid than for beta-gamma emulsions. In 30 days at room temperature fading may decrease the number of tracks by 50%. Blackening of the film by gamma radiation may make it useless for neutron analysis.

1.12 Solid State Detectors

Solid state detectors include a variety of types. Only those which are or show promise of being important in the radiation protection field will be discussed. The semiconductor junction detectors have become very important in spectroscopy and heavy charged particle detection. A semiconductor junction detector is analogus to a solid state ionization chamber. The ionizing radiation produces electrons and "positive holes" in a depletion region created by a reverse bias. The electrons and positive holes then produce a detectable current. The amount of current is dependent upon the number of electron-hole pairs formed or the energy absorbed in the depleted region. If the ionizing particle is completely absorbed in the depletion region the pulse is proportional to the energy of the particle. The small amount of energy (3.5 ev) to produce an ion pair (34 ev in air) and the high density make the silicon device much more sensitive than a gaseous ionization chamber of the same size. The problem lies in producing a depletion region or sensitive volume which is very large. The silicon devices are used primarily for alpha detection and spectroscopy as the alpha particle can be completely absorbed in the small volume.

In order to increase the sensitive volume of semiconductor detectors, lithium is diffused into the silicon to increase the resistivity and thus the volume of the depleted region. This is being done with germanium diodes, and lithium drifted germanium detectors have become the preferred device for gamma spectroscopy. The energy resolution of such devices far exceeds that of scintillation detectors. They are, however, very expensive, insensitive because of the small volume and, in the case of lithium - drifted germanium, must be kept at low temperatures. Semiconductor devices find some use in therapy because they can be inserted inside probes and used inside the body. Semiconductors have found limited use as neutron detectors by applying a hydrogenous material to the surface and detecting proton recoils or by using a "lithium sandwich" and detecting the reaction products from the (n,α) reaction with lithium. Another type of solid state dosimeter uses the property of radiophotoluminescence in glass. Ionizing radiation produces in certain glasses stable luminescent centers. If the glass is then excited with ultraviolet light it will emit a light of certain wavelength. The intensity of the luminescence is a function of dose for constant excitation. Silver-activated phosphate glasses were first used for this type of dosimeter, but a variety of types is now available. The effective atomic number of the glass may be varied to make it energy independent or materials added to make it more or less neutron sensitive. Glass dosimeters are useful over a wide range (10^{-}[illegible] - 10^{6} rad) of doses and are stable and rugged. Glass dosimeters have been used in some instances as supplements to film for long-term personnel monitoring. They can also be obtained in very small sizes for medical measurements. Radiothermoluminescent dosimeters (TLD) show much promise, especially for personnel monitoring. In TLD devices electrons are raised by ionizing radiation to higher energy levels and trapped. Upon heating (or other excitation) the electrons can be freed from the traps and drop back to the ground state with the emission of light. The amount of light emitted is a function of the dose to the crystal of thermoluminescent material. The two phosphor materials most widely used are CaF_2 and LiF. Calcium fluoride is more sensitive to low doses than lithium fluoride but the low atomic number of lithium fluoride makes it more suitable for low-energy dosimetry. Both materials have very wide ranges (10^{-}[illegible] - 10^{5} rad) but LiF undergoes some damage at very high dose rates. Calcium sulfate activated with manganese is a very sensitive TLD and is sometimes used to measure very low doses. The process of "reading" the TLD destroys the luminescent centers and the device may be reused. The TLD devices are small, rugged, stable for long periods of time and are therefore well suited for personnel monitoring, although at this time such usage is not widespread.

Another phenomenon sometimes used for dosimetry is the change in color of optical density of materials with exposure to radiation. Glasses, such as silver-phosphate glass, some plastics, and dyes in waxes, gels, plastics or glasses are sometimes used in this type of dosimeter. In general the sensitivity is low and the major use is found in measuring rather large doses, (10^{3} - 10^{8} rad). The clear plastics are very useful in measuring large electron doses associated with accelerators.

1.13 Chemical Detectors

Chemical changes induced in solutions may be used for radiation dosimetry. In general these systems are useful only for large doses. Great care and skill is needed for accurate results because of the interference caused by small quantities of impurities. The Fricke dosimeter has been in use for about 40 years. It measures the oxidation of ferrous ion to ferric ion in acidic ferrous sulfate solution. It has a range of about 4×10^{3} to 4×10^{4} rads. For higher doses, the ceric sulfate dosimeter is used. This system utilizes the reduction of ceric ion to cerous ion and may be used for doses up to 10^{8} rad. Another class of chemical dosimeters utilizes the production of hydrochloric acid in chlorinated hydrocarbons. Other chemical reactions used are oxidation-reduction with color indicators, production of decomposition products, production or destruction of fluorescent substances, and polymerization.

1.14 Calorimetry

Calorimetric methods of dosimetry are used mainly for calibration purposes. The main advantage is that it is a direct measure of the energy absorbed. Other methods, such as ionization, depend upon a secondary process. However, the heat associated with even large doses of radiation is very small and the measurement must be very sensitive. Thus, this technique has been confined to only a few laborarories interested in primary standardization. More recently with improved temperature measuring devices it has been used in the medical field to measure relatively large doses in tissue. The method may be used to measure both radiation dose rates or total emission from a radioactive source to determine the activity.

1.15 Health Physics Instruments

In the following paragraphs brief descriptions are given of the instruments commonly used in the health physics profession. Operational characteristics are stressed with the purpose of assisting in the proper choice of an instrument for a given application. Measurements taken with the improper instrument or wrongly interpreted can be dangerously misleading.

1.15.1 Personnel Monitoring Instruments

Personnel monitors include any device which can be carried on the person to indicate or measure the amount of radiation. Such devices should be light in weight, rugged, stable, accurate, and allow the dose information to be obtained immediately. The film badge is the most widely used dosimeter, although it has some disadvantages such as limited range and the requirement for processing to obtain the dose information. The high atomic number materials used in film also make the response very much different from tissue. Filters can be used for energy dependence. Film can be used for beta-gamma or neutron radiation, but separate emulsions are required. For neutrons the sensitivity is low, the energy span detected small, and the dose range very limited. Alpha radiation or low-energy beta particles from radioisotopes such as ^{14}C, ^{35}S, or ^{3}H will not penetrate the paper wrapper around the film and will, therefore, not be detected. Typical film badge characteristics are:

Radiation	Energy Detected	Dose Range
Gamma	.02 - 20 Mev	.010 - 500 rem
Beta	> 1.5 Mev	.040 - 1000 rem
Neutron	I - 10 Mev	.015 - 10 rem

The pocket chamber is a pen-sized ionization chamber which requires an external charger and reader. The chamber is charged to a fixed voltage and ions caused by radiation are collected causing a current flow in the chamber and reducing the voltage. The drop in voltage is a measure of the ionization or radiation dose. Pocket chambers are an excellent supplement to film badges because they can be quickly and easily read at the end of an exposure period. However, the normal leakage current limit requires daily recharging and they are sensitive to shock and erratic readings. The usual range of the pocket chamber is 0.2R.

Pocket dosimeters are similar to pocket chambers except that they contain a built in electroscope which can be used to read the accumulated dose at any time. This is an advantage when the dose information is needed on the job site. A separate charger is required as with the pocket chamber. Pocket dosimeters come in a variety of ranges from 0.2 - 1000R and even in special types for fast and thermal neutrons.

Chemical or solid state dosimeters may be used in place of or as a supplement to film badges for personnel monitors. Glass and TLD dosimeters appear to be most promising in this respect. The wide range (up to 10^5 rem) of TLD and glass dosimeters, the stability, the low atomic number, and the ease of read-out are some of the advantages. The major drawback at the present time is the lack of commercial systems with the flexibility of film badges for measuring beta and gamma radiation of various energies.

A number of small electronic devices using G-M tube detectors are available as personnel monitors. These instruments may emit a variable tone or tone rate as an indication of dose rate or give an alarm when a pre-set dose has been received. They are very useful for work in areas where the dose rate may change with time or location.

Criticality detectors are sometimes used in conjunction with film badges or other personnel monitors at nuclear installations. These devices are designed to measure the large doses of neutrons normally associated with criticality accidents. They usually consist of a number of threshold detectors such as indium, bare gold, cadmium shielded gold, and sulfur.

1.15.2 Portable Survey Instruments

Radiation surveys for gamma radiation are usually performed using some type of ionization chamber. The range of these instruments (0.05-[illegible]) is most suited for such surveys. As noted in the preceding section the ion chamber is affected by humidity and has a slow response at low dose rates because of the high impedance. The use of vibrating-reed electrometers has greatly improved the performance of these instruments and eliminated the amplifier drift. Most chambers are equipped with a thin window with a removable cap for low energy gamma or beta radiation. The calibration is not accurate for beta radiation or narrow beam gamma radiation. Some instruments are capable of measuring an integrated dose in addition to dose rate. The condenser R-meter is a very accurate ionization-chamber using a separate charger and reader. It is used for very accurate measurements and calibration.

Scintillation detectors have become more common for gamma radiation surveys. Used in the pulse mode these instruments are very sensitive and can measure dose rates less than 10^{-5} R/hr. They can also be used in the current mode for very high dose rates. One portable instrument using a plastic scintillator has a useful range of 0.1 - 1000 R/hr. The energy response of such devices is, however, not very good at low energies.

Geiger-Mueller tubes can be used at low dose rates although the response is usually very energy dependent. Very small G-M tubes have recently been developed which can be used in simple circuits at dose rates in excess of 10^{3} R/hr. This type of tube is sometimes added as an auxiliary monitor to extend the range of a conventional survey meter. The usual instrument for measuring contamination is the Geiger-Mueller survey meter or GMSM with a range of about 10^{2} - 10^{5} c/m. The G-M tube may be either thin-walled (wall thickness ~30 mg/cm^2) or thin end-window (mica window 1.4 - 4 mg/cm^2). The thin end-window tube is needed if alpha or low energy beta radiation is to be detected. For large amounts of fixed contamination the ion chamber with window cover removed can be used. Proportional counters are available using either air or propane as the counting gas. The counters using air are somewhat unstable and sensitive to humidity because the operating voltage is close to the breakdown voltage. The counters using propane are very dependable but somewhat bulky because of the gas cylinder. Proportional counters have the advantage of large window area and thin windows (0.6 mg/cm^2 mylar) giving excellent sensitivity. While originally used only for alpha radiation, they have been adapted to beta detection and are the most sensitive beta survey meters available. Scintillation detectors are also used for contamination surveys. Sodium iodide is a very sensitive gamma radiation detector and zinc sulfide is used to detect alpha radiation. Plastic beta scintillators are available but are not as good as G-M or proportional counters because of the large background count rate and the low sensitivity.

The BF_3 proportional counter and the Li I scintillation counter are the most often used neutron survey meters. The BF_3 tube is sensitive to thermal neutrons. By surrounding it with a paraffin or plastic moderator it can be used for fast neutrons. The major disadvantage is the relative insensitivity to intermediate energy neutrons which compromise a large part of the leakage radiation from biological shields. Corrections also have to be made to convert the flux detected to rem dose using appropriate RBE values for the different energy neutrons. It has been found that a 10-12 inch polyethylene sphere will moderate neutrons in such a way that the count rate for a thermal neutron detector in the center of such a sphere will approximate the inverse of the RPG curve. Thus, the count rate for such an instrument can be converted directly to a dose in rems. The detector most often used is a Li I scintillator (^{6}Li (n,α) ^{3}He). One of the major problems with most neutron instruments is the gamma background. The Li I detector can discriminate against gamma radiation up to about 2 R/hr and the BF_3 tube up to about 10 R/hr.

1.15.3 Area Radiation Monitors

Area monitors are usually fixed instruments to continuously monitor the radiation levels in a certain location. They may have an audible indicator, recorder, remote indicator, and alarm attached. Scintillation detectors using the current mode and high-range G-M tubes have largely replaced ion chambers in these instruments because of the simple and more reliable electronics. The usual ranges are within the span of .001 - 10 R/hr depending on the application. Characteristics necessary for such instruments are reliability, low drift, rapid response, calibration stability and non-saturation in high radiation fields. Any alarm circuits should be fail-safe with respect to detector power or electronic failure.

1.15.4 Air Monitors

Air monitors may be designed to filter the particulate material from the air and detect any contamination in that form or the actual gas may be monitored. Normal background activity associated with Rn and Tn in the air is a problem in any air monitoring instrument. In some instruments the natural activity on the filter is allowed to decay before measurement. This, of course, delays the indication of high levels of activity. The ratio of beta and alpha activity in the background activity can be monitored and any deviation used as an indication of other contaminants. Where only one major contaminant is present some identifying characteristic such as beta or gamma energy can be used to distinguish it from background. For gaseous contaminants the simplest monitor is a radiation detector in a large volume of the gas, e.g., a G-M tube in a tank of the gas or in the exhaust stack of a building. The air in a facility can also be continuously flowed through an ionization chamber to detect low energy beta emitters which cannot be filtered, e.g., $^{14}CO_2$, $^3H_2$. Such instruments are, however, insensitive and are subject to interference from dust, smoke, etc.

1.15.5 Water Monitor

Water monitoring instruments are used to continuously monitor effluents from nuclear facilities or recirculating reactor coolant. In some instruments a G-M tube or other detector is placed in a small sample chamber to monitor the water directly. Such instruments are too insensitive for most uses, especially effluent streams. The sensitivity can be increased by concentrating the radioactive material by filtering or ion exchange and monitoring the filter or ion exchange bed. The detector is commonly a Na I scintillation crystal set to detect a specific energy gamma ray to further reduce background and improve sensitivity.

1.15.6 Laboratory Instruments

Health physics laboratory instruments consist mainly of sensitive detectors to measure the low levels of radioactivity associated with smears, wipes, water and air samples, bioassay samples, and environmental samples. Where discrimination against different energy beta particles or beta and alpha radiation is required, the gas-flow proportional counter is usually used. This may be in the form of an end-window, two Pi hemisphere, or four Pi spherical type. Liquid scintillation counters are very useful for liquid samples, ^3H, and samples where beta energy discrimination is necessary. For routine smear counting the single end-window G-M tube is usually sufficient. A low background counter is required for accurate determination of low activity samples such as are common in bioassay or environmental monitoring. Gaseous samples are usually analyzed in ionization chambers using very sensitive electrometers.

1.16 Biological Effects of Radiation

The absorption of radiation in biological systems and the associated ionization, excitation, heating, etc., can cause significant changes. The absorption of radiation can form chemical species such as ions, free radicals, and excited molecules which can cause extensive damage to the cell. Radiation can also do direct damage to cellular components by splitting molecules or damaging genetic material. Damage to genetic material may effectively be magnified by the exact reproduction of an abnormal cell for many generations. Chemical changes initiated by radiation damage may continue for relatively long periods of time after the initial event. The sensitivity of cells is variable, but in general, cells with a high metabolic rate and which reproduce rapidly are more sensitive to radiation. The sensitivity also varies with certain chemical concentrations in the cell. Large amounts of oxygen can increase the sensitivity to radiation by a factor of 2 or more. Many biological effects are dose rate dependent. At low dose rates the cells can undergo some repair of damage but large doses in a short time can kill the cell and eliminate any chance of repair. Thus, for a given dose the effect is usually less if the dose is delivered over a long period of time (months or years).

1.16.1 Acute Radiation Syndrome

For human exposure to large whole-body doses in a short period of time the symptoms of radiation sickness are characteristic and follow a definite pattern. The term LD_{50}

is used to express the dose which will result in death to 50% of a sample. The time span for animals is usually given as 30 days. For humans 60 days is probably equivalent to the same period. Without medical treatment the LD_{50} for man is probably 450 - 500 rem if the exposure is measured in air or 300 - 400 rem if measured at the center of the trunk. Below 200 rem the probability of death is low for normal persons and above about 500 rem death within a short period of time is probable, even with the best available medical care. With good medical care the LD_{50} is probably about 500 rem. The following table summarizes the general symptons of the radiation syndrome.

Symptons of Acute Radiation Syndrome

Time after Exposure	Dose		
	1000	500	200
First Week	Nausea and vomiting on first day		
Second Week	Nausea, vomiting, diarrhea, fever, inflammation of throat, prostration, dehydration emaciation leading to death		
Third Week		General malaise, loss of appetite,. loss of hair, hemorrhage, pallor, diarrhea, fever inflamation of throat, emaciation leading to death in 50% of victims	
Fourth Week			loss of appetit loss of hair, i flammation of throat, pallor, hemorrhage, diarrhea, Recovery begins

1.16.2 Effects on Individual Organs

The mature blood cells, with the exception of the lymphocytes, are relatively resistant to radiation damage. However, the progenetive cells which produce the blood cells are very radiosensitive and easily destroyed. As the blood cells are gradually used or eliminated there are no new cells to replace them. The red blood cells (erythrocytes) have a long life span (120 d) and do not decrease rapidly. Other blood cells such as granulocytes and platlets have shorter life spans and the decrease may be significant in a few days. These cells combat infection and aid in blood clotting so their destruction leads to infection and hemorrhaging. The mature lymphocyte cells are very sensitive to radiation and even small doses can cause a noticable reduction in the lymphocyte count. The vascular network can also be damaged by radiation leading to secondary effects caused by insufficient blood supply.

Transplantation of bone marrow has been used on several occasions to replace the blood forming tissues in persons who had received very large whole body exposures (400 - 600 rem). The rejection by the body of any foreign tissue makes this procedure very risky and it is used only where it is felt that recovery could not begin without it. For doses of 500 - 1000 rem death is usually caused by the damage to the blood forming organs (usually because of the resoluting infection or hemorrhaging).

The skin is also sensitive to radiation and will show erythema (reddening) after doses of 200-1000 rad, depending on the location, energy of radiation, complexion, etc. The skin may develop a blister, or in the case of large doses, the skin is destroyed, requiring a graft for replacement. Development of skin cancer is not uncommon in persons who have experienced skin "burns". Doses of 200 - 300 rad can damage the hair follicles and cause temporary epilation (loss of hair). Greater doses can cause permanent loss of the hair.

Radiation damage to the gastrointestinal tract destroys the progenitive cells which produce the lining of the intestines. The lining is continuously removed by the intestinal contents and without replacement cells the intestines become ulcerated and subject to dehydration, infection, and hemorrhaging. Doses in excess of 1000 rad lead to death within weeks because of damage to the GI tract.

The nervous system is relatively resistant to radiation. However, doses of the order of 10,000 rad cause immediate damage to the central nervous system leading to nausea, vomiting, convulsions, unconsciousness, and death within hours.

1.16.3 Other Effects

Radiation damage can increase the probability of cancer in organs such as the skin, blood-forming tissue, bone, lung, thyroid gland, and connective tissue. Radiation also has a degenerative effect on body tissues which have received large doses. This appears as a premature "aging" of the organ or tissue and loss of function. Another effect of radiation is on the growth and development of persons exposed to radiation at a very young age, especially in utero. Genetic effects have been observed in plants and animals at high doses. Complete

The above limits have been adopted as a federal regulation in 10 CFR 20 and as state regulations in most of those states with radiation protection regulations. Limits on concentration of radioactive material in air and water are based on the above limits for the organs involved. That is, continuously working at the maximum permissible concentration would give the same integrated dose to an organ as from external exposure at the dose limits above.

1.18 Population Exposure

Radiation exposure of the population-at-large is due to natural radiation, man-made sources, and environmental contamination. Natural sources producing external exposure are cosmic radiation, radioactive material in the soil, and radon and thoron in the air. Internal sources of radiation are radium and radium daughters, K-40 and C-14.

Natural Radiation

Sources	Annual Gonad Dose in mrem
External	
Cosmic	28
Terrestial	47
Atmospheric	2
Internal	
^{40}K	19
^{14}C	1.6
Rn-Tn	2
Ra	- (33 for bone)
	100

Man-made sources of radiation include occupational exposure, luminous dials, television receivers, etc., but are mostly medical. Estimates of man-made exposures are:

	mrem/year (gonad)
Medical-diagnostic	20 - 150
Medical-therapeutic	1 - 30
Medical-internal	<1
Occupational	<2

Environmental contamination is mainly from fallout. The average annual dose is probably less than 1 mrem.

1.19 Operating Procedures

The following are general suggestions for normal operations. These recommendations will not necessarily apply to special situations. However, if the reasons for such procedures are borne in mind, methods for dealing with almost any situation will be obvious.

1.19.1 Work in Contaminated or Radiation Areas

One of the first rules for working with radioactive material or in high-level radiation areas is to eliminate the unexpected. Plan operations ahead of time, make simulated or dummy runs, and predict the conditions of the operation to be performed so that adequate preparation can be made. Some allowance should always be made for unexpected occurrences such as leaks, breakage, etc. If work must be performed in a contaminated area adequate personal protection in the form of shoe covers, coveralls, gloves, respirators, etc., should be provided. The operation should also be conducted in such a way as to minimize the spread of contamination. Gloves should be worn when handling contaminated material and this is especially important if there are any breaks in the skin of the hands. Where radiation levels are high every means should be used to reduce exposure (time, distance, shielding). Efficient planning of operations can drastically reduce the exposure time, and use of all available shielding can reduce radiation levels. Remote handling will usually reduce personnel exposure significantly. Where the extremities can be exposed to higher radiation levels than the rest of the body, supplemental monitoring should be provided. When leaving contaminated areas always wash the hands and check for contamination on hands, feet, and clothing. Where working time limits have been set, they should be strictly followed. Any unusual events or suspected overexposures should be reported to the health physicist immediately.

1.19.2 Surveys

Surveys should be performed as often as is necessary to assure that radiation and contamination levels are within the working limits. Contamination surveys with a survey meter are greatly aided by earphones or other audible indicators of counting rate. The survey meter should, of course, be sensitive to and calibrated for the radiation present. For mixed fields of beta, gamma, and neutron radiation more than one instrument may be necessary. An instrument with sufficient range to cover the maximum dose rate expected should be used. Instruments should be checked to see that they are operating properly before entering the survey area. Normal GMSM instruments should not be used in high radiation areas because of the saturation effect. However, many of the newer GMSM instruments have circuits to prevent this effect.

1.19.3 Decontamination

Decontamination procedures should be designed to prevent the spread of contamination and to produce the least amount of waste material while effectively cleaning the contaminated area. The available methods for decontamination are too many to list. In most cases of serious contamination a method designed especially for that case is best. In general, cleaning is done from the areas of least contamination to the more highly contaminated areas to reduce the spread of radioactive material. Cleaning solutions and scrubbing agents must be changed often for the same reason.

Skin decontamination is sometimes very difficult, because the methods used must not injure the skin. The usual procedure is to scrub with mild soap solutions first and then progress to stronger cleaning agents as required. Hand lotion or oil should be frequently applied to the hands to prevent drying and irritation.

1.20 Regulations

Use of radioactive material, nuclear reactors, and radiation producing equipment is closely regulated by state and federal agencies. The basic regulation for persons employed in the atomic energy industry is Title 10, Code of Federal Regulations, Part 20 (10 CFR 20) entitled Standards for Protection Against Radiation. Other important parts of the federal regulations are 10 CFR 40 Licensing of Source Material; 10 CFR 50 Licensing of Production and Utilization Facilities; 10 CFR 55 Operator's Licneses; 10 CFR 70 Special Nuclear Material; 10 CFR 71 Packaging of Radioactive Material for Transport and 10 CFR 100 Reactor Site Criteria.

Part 20 includes the maximum permissible doses and concentration limits for occupational exposure and for the general public. Requirements for surveys, personnel monitoring, posting, labeling, storage, records, disposal and reporting of incidents are also given. No attempt will be made to summarize the regulations here as it would only result in duplication. The regulation is important enough to be studied in detail in its legal form and a copy of 10 CFR 20 is included for that purpose.

The shipment of radioactive material in interstate commerce is regulated by the Department of Transportation (DOT). The regulations are codified under Title 49 of the Code of Federal Regulations parts 171-190. Agent T. C. George's Tariff No. 19 lists the regulations for transportation of explosives and other dangerous articles and the specifications for shipping containers.

It should be noted that all the above regulations are subject to continual review and change and care should be used that decisions are based on updated copies of the regulations. Subscriptions including all supplements are available for 10 CFR and T. C. George's Tariff.

data on humans is lacking but changes in sex ratio were noted in the descendants of the survivors of the atomic bomb blasts in Japan.

1.17 Exposure Limits

Recommended limits for exposure to radiation have been reduced significantly over the past 40 years. The recommendation by the ICRP in 1934 of a limit of 0.2 R/day have decreased over the years to the present limit of 5 rem/year. None of the reductions has been made because of evidence of damage from working at any of the limits. However, as more data has been accumulated on the effects of radiation and as the number of persons exposed has increased it was thought wise to reduce the limits. The basic reasoning being that exposing a large number of people to a small dose is equivalent to exposing a small group to a large dose. The weighing of benefit and risk has also been considered. Thus, the population-at-large does not benefit as much from a radiation source (i.e., reactor as the radiation worker and should not be forced to take the same risk. Definite recommendations have not yet been made for exposure limits for whole populations. This will have to be considered in the near future in conjunction with the growing use of nuclear energy for electric power and other uses.

The groups usually considered responsible for establishing radiation exposure limits are the International Commission on Radiological Protection (ICRP), the National Council on Radiation Protection and Measurements (NCRP) and the Federal Radiation Council (FRC), which is an advisory body to the President of the United States. The present recommended limits of all three bodies are essentially identical.

Maximum permissible occupational doses and concentrations are set so that there is a negligible probability of serious radiation damage (somatic or genetic) for a person working an entire lifetime (50 yr.) at those limits. There will always, however, be a small probability that an individual will suffer serious damage. An attempt is made to place the risk on a level with or less than that associated with other occupations.

The present recommended permissible dose limits (ICRP) for occupational exposure are listed below:

Organ	Maximum Dose per quarter	Annual Permissible Dose	Accumulated Dose to Age N
Red bone marrow	3	5	5 (N-18)
Total body	3	5	5 (N-18)
Head and trunk	3	5	5 (N-18)
Gonads	3	5	5 (N-18)
Lenses of eyes	3	5	5 (N-18)
Skin	8	30	
Thyroid	8	30	
Feet, ankles	20	75	
Hands, forearms	20	75	

References

1. Radiation Monitoring, J. E. Wade and G. E. Cunningham, 1967, U.S. Atomic Energy Commission/Division of Technical Information.

2. Radiation Dosimetry, Volume II, F. H. Attix and W. C. Roesch, 1966, Academic Press, New York.

3. Principles of Radiation Protection, K. Z. Morgan and J. E. Turner, 1967, John Wiley and Sons, Inc., New York.

4. Atomic Radiation, Volume I and II, R C A Service Company, 1957, Camden, New Jersey.

5. Medical Aspects of Radiation Accidents, E. L. Saenger, U. S. Atomic Energy Commission, 1963, U. S. Government Printing Office.

6. National Bureau of Standards Handbook, Numbers 48, 51, 59, 63, 69, 80, 84, and 92, U. S. Superintendent of Documents.

7. Radiological Health Handbook, U. S. Department of Health, Education and Welfare, 1960.

8. Code of Federal Regulations, Title 10, Federal Register.

9. Interstate Commerce Commission Regulations, Agent T. C. Georges Tariff No. 19.

UNITED STATES ATOMIC ENERGY COMMISSION

RULES and REGULATIONS • TITLE 10 - ATOMIC ENERGY

PART 20

STANDARDS FOR PROTECTION AGAINST RADIATION

General Provisions

§ 20.1 Purpose.

(a) The regulations in this part establish standards for protection against radiation hazards arising out of activities under licenses issued by the Atomic Energy Commission [illegible] Atomic Energy Act of 1954 [illegible]

(b) The [illegible] radioactive material or other sources of radiation not licensed by the Commission is not subject to the regulations in this part. However, it is the purpose of the regulations in this part to control the possession, use, and transfer of licensed material by any licensee in such a manner that exposure [illegible] radiation from such material [illegible] exposure [illegible] material and to other [illegible] sources of radiation in the possession of the licensee, and to ensure [illegible] do not exceed the [illegible] radiation protection prescribed in the regulations in this part.

§ 20.2 Scope.

[illegible] in this part apply to [illegible] material [illegible] Commission [illegible] the regulations in Part [illegible] 70 of this chapter.

§ 20.3 Definitions.

(a) As used in this part:

(1) "Act" means the Atomic Energy Act of 1954 [illegible] including any amendments thereto;

(2) "Airborne radioactive material" means [illegible] material dispersed in the air in the form of dusts, fumes, mists, vapors, or gases;

(3) "Byproduct material" means any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material;

(4) "Calendar quarter" means any [illegible] to either of the following subdivisions:

[illegible]

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(i) The first period of any year may begin on any date in January [illegible] that the second, third and fourth [illegible] accordingly begin [illegible] April, July, and October [illegible] and that the fourth period extend [illegible] January of the succeeding year [illegible] to complete a [illegible] During the first year of [illegible] method of determination [illegible] the first period for that year shall include any additional days in January preceding the starting date for the [illegible] period.

(ii) The first period [illegible] 13 complete [illegible] weeks; the second [illegible] of 13 complete [illegible] weeks; the third [illegible] the fourth [illegible] year of 13 complete [illegible] calendar weeks.

Alternatively, the four periods [illegible] of the first 14 complete [illegible] calendar weeks, the next 12 [illegible] consecutive calendar weeks, the [illegible] complete consecutive calendar [illegible] and the last 12 complete [illegible] calendar weeks [illegible]

[illegible]

No licensee shall change the method [illegible] by him of determining [illegible] quarters for purposes of [illegible] at the beginning of a [illegible]

(5) "Commission" means the [illegible] Energy Commission or its duly [illegible] ized representatives;

(6) "Government agency" means [illegible] executive department, commission, independent establishment, [illegible] wholly or partly owned by the United States of America which is [illegible] mentality of the United States [illegible] board, bureau, division, [illegible] officer, authority, administration, [illegible] other establishment in the executive branch of the Government;

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(i) "Individual" means any human being.

(j) "Licensed material" means [illegible] material [illegible] byproduct material received, possessed, used or transferred under a general or specific license issued by the Commission pursuant to the regulations in this chapter.

(k) [illegible] under the regulations in Part [illegible] 40, or 70 of this chapter. "Licensee" means the holder of such license.

(l) [illegible] exposure to radiation [illegible] for the purpose of medical diagnosis or medical therapy of such individual.

(m) [illegible] partnership, firm, association, trust, estate, public or private institution, group, Government agency other than the Commission, any State [illegible] or nation or any political subdivision of any such government or nation, or other entity; and [illegible] any legal successor, representative, agent, or agency of the foregoing.

(n) "Radiation" means any or all of the following: alpha rays, beta rays, gamma rays, X-rays, neutrons, high-speed electrons, high-speed protons, and other atomic particles; but not sound or radio waves, or visible, infrared, or ultraviolet light.

(o) "Radioactive material" includes any such material whether or not subject to licensing control by the Commission.

~~(p) "Restricted area" means any [illegible] of [illegible] Restricted [illegible] any areas used as [illegible] although a separate [illegible] residential [illegible]~~

~~[illegible]~~

(q) "Special nuclear material" means [illegible] uranium [illegible] enriched in the isotope 233 or in the isotope 235, and any other material which the Commission, pursuant to the provisions of section 51 of the act, determines to be special nuclear material, but does not include source material; or [illegible] any material artificially enriched by any of the foregoing but does not include source material.

(r) "Unrestricted area" means any area access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, and any area used for residential quarters.

(b) Definitions of certain other words and phrases as used in this part are set forth in other sections, including:

(1) "Airborne radioactivity area" defined in § 20.203;

(2) "Radiation area" and "high radiation area" defined in § 20.202;

(3) "Personnel monitoring equipment" defined in § 20.202;

(4) "Survey" defined in § 20.201;

(5) Units of measurement of dose (rad, rem) defined in § 20.4;

(6) Units of measurement of radioactivity defined in § 20.5.

§ 20.4 Units of radiation dose.

(a) "Dose," as used in this part, is the quantity of radiation absorbed, per unit of mass, by the body or by any portion of the body. When the regulations in this part specify a dose during a period of time, the dose means the total quantity of radiation absorbed, per unit of mass, by the body or by any portion of the body during such period of time. Several different units of dose are in current use. Definitions of units as used in this part are set forth in paragraphs (b) and (c) of this section.

(b) The rad, as used in this part, is a measure of the dose of any ionizing radiation to body tissues in terms of the energy absorbed per unit mass of the tissue. One rad is the dose corresponding to the absorption of 100 ergs per gram of tissue. (One millirad (mrad) = 0.001 rad.)

(c) The rem, as used in this part, is a measure of the dose of any ionizing radiation to body tissue in terms of its estimated biological effect relative to a dose of one roentgen (r) of X-rays. (One millirem (mrem) = 0.001 rem.) The relation of the rem to other dose units depends upon the biological effect under consideration and upon the conditions of irradiation. For the purpose of the regulations in this part, any of the following is considered to be equivalent to a dose of one rem:

(1) A dose of 1 r due to X- or gamma radiation;

(2) A dose of 1 rad due to X-, gamma, or beta radiation;

(3) A dose of 0.1 rad due to neutrons or high energy protons;

(4) A dose of 0.05 rad due to particles heavier than protons and with sufficient energy to reach the lens of the eye;

If it is more convenient to measure the neutron flux, or equivalent, than to determine the neutron dose in rads, as provided in subparagraph (3) of this paragraph, one rem of neutron radiation may, for purposes of the regulations in this part, be assumed to be equivalent to 14 million neutrons per square centimeter incident upon the body; or, if there exists sufficient information to estimate with reasonable accuracy the approximate distribution in energy of the neutrons, the incident number of neutrons per square centimeter equivalent to the rem may be estimated from the following table:

NEUTRON FLUX DOSE EQUIVALENTS

Neutron energy (Mev)	[illegible]	[illegible]
Thermal	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]

(d) *For determining exposures to X or gamma rays up to 3 Mev, the dose limits specified in §§ 20.101 to 20.104, inclusive, may be assumed to be equivalent to the "air dose". For the purpose of this part "air dose" means that the dose is measured by a properly calibrated appropriate instrument in air at or near the body surface in the region of the highest dosage rate.*

§ 20.5 Units of radioactivity.

(a) Radioactivity is commonly, and for purposes of the regulations in this part shall be, measured in terms of disintegrations per unit time or in curies. One curie (Ci) = 3.7 × 10 [illegible] disintegrations per second (dps) = 2.2 × 10 [illegible] disintegrations per minute (dpm). A commonly used submultiple of the curie is the microcurie (µCi). One µCi = 0.000001 Ci = 3.7 × 10 [illegible] dps = 2.2 × 10 [illegible] dpm.

(b) For purposes of the regulations in this part, it may be assumed that the daughter activity concentrations in the following table are equivalent to an air concentration of [illegible] microcuries of Radon 222 per milliliter of air in equilibrium with the daughters RaA, RaB, RaC and RaC'.

[illegible]	[illegible]
[illegible]	[illegible]

The [illegible]

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(c) Natural uranium and natural thorium. (1) For purposes of the tabulation in the text of the table of natural uranium (U-natural) in Appendix B or C means the sum of 3.7 × 10 disintegrations per second from U-238 plus 3.7 × 10 dis/sec from U-234, plus 9 × 10 dis/sec from U-235. As to a curie of natural thorium (Th-natural) in Appendix B or C means the sum of 3.7 × 10 dis/sec from Th-232 plus 3.7 × 10 dis/sec from Th-228.

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(2) For the purposes of the regulations in this part one curie of natural uranium (U-natural) in Appendix B or C is equivalent to 3,000 kilograms or 6,615 pounds of natural uranium; and one curie of natural thorium (Th-natural) in Appendix B or C is equivalent to 9,000 kilograms or 19,850 pounds of natural thorium.

§ 20.5 Interpretations.

Except as specifically authorized by the Commission in writing, no interpretation of the meaning of the regulations in this part by any officer or employee of the Commission other than a written interpretation by the General Counsel will be recognized to be binding upon the Commission.

§ 20.6 Communications.

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Except where otherwise specified in this part, all communications and reports concerning the regulations in this part, and applications filed under them, should be addressed to the Director of Regulation, U.S. Atomic Energy Commission, Washington, D.C. 20545. Communications, reports, and applications may be delivered in person at the Commission's offices at 1717 H Street NW., Washington, D.C.; at 7920 Norfolk Avenue, Bethesda, Md.; or at Germantown, Md.

§ 20.101 Exposure of individuals to radiation in restricted areas.

(a) Except as provided in paragraph (b) of this section, no licensee shall possess, use or transfer licensed material in such a manner as to cause any individual in a restricted area to receive in any period of one calendar quarter from radioactive material and other sources of radiation in the licensee's possession a dose in excess of the limits specified in the following table:

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Rems per calendar quarter	
1. Whole body; head and trunk; active blood-forming organs; lens of eyes; or gonads	1¼
2. Hands and forearms; feet and ankles	18¾
3. Skin of whole body	7½

(b) A licensee may permit an individual in a restricted area to receive a dose to the whole body greater than that permitted under paragraph (a) of this section, provided:

(1) During any calendar quarter the dose to the whole body from radioactive material and other sources of radiation in the licensee's possession shall not exceed 3 rems; and

(2) The dose to the whole body, when added to the accumulated occupational dose to the whole body, shall not exceed 5 (N-18) rems where "N" equals the individual's age in years at his last birthday; and

(3) The licensee has determined the individual's accumulated occupational dose to the whole body on Form AEC-4, or on a clear and legible record containing all the information required in that form; and has otherwise complied with the requirements of § 20.102. As used in paragraph (b), "Dose to the whole body" shall be deemed to include any dose to the whole body, gonads, active blood-forming organs, head and trunk, or lens of eye.

§ 20.102 Determination of accumulated dose.

(a) This section contains requirements which must be satisfied by licensees who propose, pursuant to paragraph (b) of § 20.101, to permit individuals in a restricted area to receive exposure to radiation in excess of the limits specified in paragraph (a) of § 20.101.

(b) Before permitting, pursuant to paragraph (b) of § 20.101, any individual in a restricted area to receive exposure to radiation in excess of the limits specified in paragraph (a) of § 20.101, each licensee shall:

(1) Obtain a certificate on Form AEC-4, or on a clear and legible record containing all the information required in that form, signed by the individual showing each period of time after the individual attained the age of 18 in which the individual received an occupational dose of radiation; and

(2) Calculate on Form AEC-4 in accordance with the instructions appearing therein, or on a clear and legible record containing all the information required in that form, the previously accumulated occupational dose received by the individual and the additional dose allowed for that individual under § 20.101(b).

(c)(1) In the preparation of Form AEC-4, or a clear and legible record containing all the information required in that form, the licensee shall make a reasonable effort to obtain reports of the individual's previously accumulated occupational dose. For each period for which the licensee obtains such reports, the licensee shall use the dose shown in the report in preparing the form. In any case where a licensee is unable to obtain reports of the individual's occupational dose for a previous complete calendar quarter, it shall be assumed that the individual has received the occupational dose specified in whichever of the following columns apply:

Part of body	Column 1 Assumed exposure in rems for calendar quarters prior to Jan. 1, 1961	Column 2 Assumed exposure in rems for calendar quarters beginning on or after Jan. 1, 1961
Whole body, gonads, active blood-forming organs, head and trunk, lens of eye	3¾	1¼

Amended 34 FR 19546
December 11, 1969

(2) The licensee shall retain and preserve records used in preparing Form AEC-4. If calculation of the individual's accumulated occupational dose for all periods prior to January 1, 1961 yields a result higher than the applicable accumulated dose value for the individual as of that date, as specified in paragraph (b) of § 20.101, the excess may be disregarded.

§ 20.103 Exposure of individuals to concentrations of radioactive material in restricted areas.

(a) No licensee shall possess, use or transfer licensed material in such a manner as to permit any individual in a restricted area to be exposed to airborne radioactive material possessed by the licensee in an average concentration in excess of the limits specified in Appendix B, Table I, of this part. "Expose" as used in this section means that the individual is present in an airborne concentration. No allowance shall be made for the use of protective clothing or equipment, or particle size, except as authorized by the Commission pursuant to paragraph (c) of this section.

(2) The limits given in Appendix B, Table I, of this part are based upon exposure to the concentrations specified for forty hours in any period of seven consecutive days. In any such period where the number of hours of exposure is less than forty, the limits specified in the table may be increased proportionately. In any such period where the number of hours of exposure is greater than forty, the limits specified in the table shall be decreased proportionately.

(b)(1) Except as authorized by the Commission pursuant to this paragraph, no allowance shall be made for particle size or the use of protective clothing or equipment in determining whether an individual is exposed to an airborne concentration in excess of the limits specified in Appendix B, Table I.

(2) The Commission may authorize a licensee to expose an individual in a restricted area to airborne concentrations in excess of the limits specified in Appendix B, Table I, upon receipt of an application demonstrating that the concentration is composed in whole or in part of particles of such size that such particles are not respirable; and that the individual will not inhale the concentrations in excess of the limits established in Appendix B, Table I. Each application under this subparagraph shall include an analysis of particle sizes in the concentrations; and a description of the methods used in determining the particle sizes.

(3) The Commission may authorize a licensee to expose an individual in a restricted area to airborne concentrations in excess of the limits specified in Appendix B, Table I, upon receipt of an application demonstrating that the individual will wear appropriate protective equipment and that the individual will not inhale, ingest or absorb quantities

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use of radioactive material in excess of those which might otherwise be permitted under this part for employees in certain areas during a period of work. Such application under this subparagraph shall contain the following information:

(i) A description of the protective equipment to be employed, including the efficiency of the equipment for the material involved;

(ii) Procedures for the fitting, maintenance and cleaning of the protective equipment; and

(iii) Procedures governing the use of the protective equipment, including supervisory procedures and length of time the equipment will be used by the individuals in each work area. The proposed periods of use of the equipment in each work area should not be greater than the periods which are essential to the conduct of the proposed operation;

(iv) The average concentrations present in the areas occupied by employees.

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§ 20.104 Exposure of minors.

(a) No licensee shall possess, use or transfer licensed material in such a manner as to cause any individual within a restricted area who is under 18 years of age, to receive in any period of one calendar quarter from radioactive material and other sources of radiation in the licensee's possession a dose in excess of 10 percent of the limits specified in the table in paragraph (a) of § 20.101.

(b) No licensee shall possess, use or transfer licensed material in such a manner as to cause any individual within a restricted area, who is under 18 years of age, to be exposed to airborne radioactive material possessed by the licensee in an average concentration in excess of the limits specified in Appendix B, Table II of this part. For purposes of this paragraph, concentrations may be averaged over periods not greater than a week.

(c) The provisions of paragraph (c) of § 20.103 shall apply to exposures subject to paragraph (b) of this section.

§ 20.105 Permissible levels of radiation in unrestricted areas.

(a) There may be included in any application for a license or for amendment of a license proposed limits upon levels of radiation in unrestricted areas resulting from the applicant's possession or use of radioactive material and other sources of radiation. Such applications should include information as to anticipated average radiation levels and anticipated occupancy times for each unrestricted area involved. The Commission will approve the proposed limits if the applicant demonstrates that the proposed limits are not likely to cause any individual to receive a dose to the whole body in any period of one calendar year in excess of 0.5 rem.

(b) Except as authorized by the Commission pursuant to paragraph (a) of this section, no licensee shall possess, use or transfer licensed material in such a manner as to create in any unrestricted area from radioactive material and other sources of radiation in his possession:

(1) Radiation levels which, if an individual were continuously present in the area, could result in his receiving a dose in excess of two millirems in any one hour, or

(2) Radiation levels which, if an individual were continuously present in the area, could result in his receiving a dose in excess of 100 millirems in any seven consecutive days.

§ 20.106 Concentrations in effluents to unrestricted areas.

(a) A licensee shall not possess, use, or transfer licensed material so as to release to an unrestricted area radioactive material in concentrations which exceed the limits specified in Appendix "B", Table II of this part, except as authorized pursuant to § 20.302 or paragraph (b) of this section. For purposes of this section concentrations may be averaged over a period not greater than one year.

(b) An application for a license or amendment may include proposed limits higher than those specified in paragraph (a) of this section. The Commission will approve the proposed limits if the applicant demonstrates:

(1) That the applicant has made a reasonable effort to minimize the radioactivity contained in effluents to unrestricted areas; and

(2) That it is not likely that radioactive material discharged in the effluent would result in the exposure of an individual to concentrations of radioactive material in air or water exceeding the limits specified in Appendix "B", Table II of this part.

29 FR 14434

(c) An application for higher limits pursuant to paragraph (b) of this section shall include information demonstrating that the applicant has made a reasonable effort to minimize the radioactivity discharged in effluents to unrestricted areas, and shall include, as pertinent:

(1) Information as to flow rates, total volume of effluent, peak concentration of each radionuclide in the effluent, and concentration of each radionuclide in the effluent averaged over a period of one year at the point where the effluent leaves a stack, tube, pipe, or similar conduit;

(2) A description of the properties of the effluents, including:

(i) chemical composition;

(ii) physical characteristics, including suspended solids content in liquid effluents, and nature of gas or aerosol for air effluents;

(iii) the hydrogen ion concentrations (pH) of liquid effluents; and

(iv) the size range of particulates in effluents released into air.

(3) A description of the anticipated human occupancy in the unrestricted area where the highest concentration of radioactive material from the effluent is expected, and, in the case of a river or stream, a description of water uses downstream from the point of release of the effluent.

(4) Information as to the highest concentration of each radionuclide in an unrestricted area, including anticipated concentrations averaged over a period of one year:

(i) In air at any point of human occupancy; or

(ii) In water at points of use downstream from the point of release of the effluent.

(5) The background concentration of radionuclides in the receiving river or stream prior to the release of liquid effluent.

(6) A description of the environmental monitoring equipment, including sensitivity of the system, and procedures and calculations to determine concentrations of radionuclides in the unrestricted area and possible reconcentrations of radionuclides.

(7) A description of the waste treatment facilities and procedures used to reduce the concentration of radionuclides in effluents prior to their release.

(d) For the purposes of this section the concentration limits in Appendix "B", Table II of this part shall apply at the boundary of the restricted area. The concentration of radioactive material discharged through a stack, pipe or similar conduit may be determined with respect to the point where the material leaves the conduit. If the conduit discharges within the restricted area, the concentration at the boundary may be determined by applying appropriate factors for dilution, dispersion, or decay between the point of discharge and the boundary.

(e) In addition to limiting concentrations in effluent streams, the Commission may limit quantities of radioactive materials released in air or water during a specified period of time if it appears that the daily intake of radioactive material from air, water, or food by a suitable sample of an exposed population group, averaged over a period not exceeding one year, would otherwise exceed the daily intake resulting from continuous exposure to air or water containing one-third the concentration of radioactive materials specified in Appendix "B", Table II of this part.

(f) The provisions of this section do not apply to disposal of radioactive material into sanitary sewerage systems, which is governed by § 20.303.

25 FR 10914

§ 20.107 Medical diagnosis and therapy.

Nothing in the regulations in this part shall be interpreted as limiting the intentional exposure of patients to radiation for the purpose of medical diagnosis or medical therapy.

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§ 20.108 Orders requiring furnishing of bioassay services.

Where necessary or desirable in order to aid in determining the extent of an individual's exposure to concentrations of radioactive material, the Commission may incorporate appropriate provisions in any license, directing the licensee to make available to the individual appropriate bioassay services and to furnish a copy of the reports of such services to the Commission.

PRECAUTIONARY PROCEDURES

§ 20.201 Surveys.

(a) As used in the regulations in this part, "survey" means an evaluation of the radiation hazards incident to the production, use, release, disposal, or presence of radioactive materials or other sources of radiation under a specific set of conditions. When appropriate, such evaluation includes a physical survey of the location of materials and equipment, and measurements of levels of radiation or concentrations of radioactive material present.

(b) Each licensee shall make or cause to be made such surveys as may be necessary for him to comply with the regulations in this part.

§ 20.202 Personnel monitoring.

(a) Each licensee shall supply appropriate personnel monitoring equipment to, and shall require the use of such equipment by:

(1) Each individual who enters a restricted area under such circumstances that he receives, or is likely to receive, a dose in any calendar quarter in excess of 25 percent of the applicable value specified in paragraph (a) of § 20.101;

(2) Each individual under 18 years of age who enters a restricted area under such circumstances that he receives, or is likely to receive, a dose in any calendar quarter in excess of 5 percent of the applicable value specified in paragraph (a) of § 20.101;

(3) Each individual who enters a high radiation area.

(b) As used in this part,

(1) "Personnel monitoring equipment" means devices designed to be worn or carried by an individual for the purpose of measuring the dose received (e.g., film badges, pocket chambers, pocket dosimeters, film rings, etc.);

(2) "Radiation area" means any area, accessible to personnel, in which there exists radiation, originating in whole or in part within licensed material, at such levels that a major portion of the body could receive in any one hour a dose in excess of 5 millirem, or in any 5 consecutive days a dose in excess of 100 millirem;

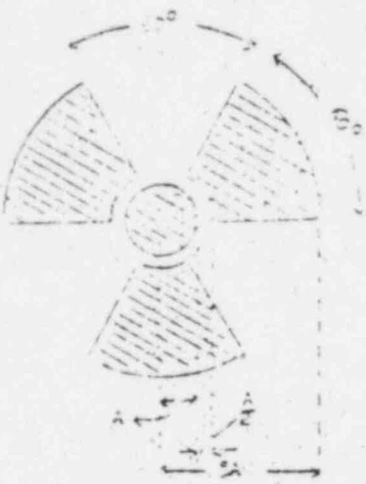
(3) "High radiation area" means any area, accessible to personnel, in which there exists radiation originating in whole or in part within licensed material at such levels that a major portion of the body could receive in any one hour a dose in excess of 100 millirem.

§ 20.203 Caution signs, labels, and signals.

(a) *General.* (1) Except as otherwise authorized by the Commission, symbols prescribed by this section shall use the conventional radiation caution colors (magenta or purple on yellow background). The symbol prescribed by this section is the conventional three-bladed design:

RADIATION SYMBOL

1. Cross-hatched area is to be magenta or purple.
2. Background is to be yellow.



(2) In addition to the contents of signs and labels prescribed in this section, licensees may provide on or near such signs and labels any additional information which may be appropriate in aiding individuals to minimize exposure to radiation or to radioactive material.

(b) *Radiation areas.* Each radiation area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION[1]
RADIATION AREA

(c) *High radiation areas.* (1) Each high radiation area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION[1]
HIGH RADIATION AREA

(2) Each high radiation area shall be equipped with a control device which shall either cause the level of radiation to be reduced below that at which an individual might receive a dose of 100 millirem in one hour upon entry into the area or shall energize a conspicuous visible or audible alarm signal in such a manner that the individual entering and the licensee or a supervisor of the activity are made aware of the entry. In the case of a high radiation area established for a period of 30 days or less, such control device is not required.

(d) *Airborne radioactivity areas.* (1) As used in the regulations in this part, "airborne radioactivity area" means (i) any room, enclosure, or operating area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations in excess of the amounts specified in Appendix B, Table I, Column 1 of this part; or (ii) any room, enclosure, or operating area in which airborne radioactive material composed wholly or partly of licensed material exists in concentrations which, averaged over the number of hours in any week during which individuals are in the area, exceed 25 percent of the amounts specified in Appendix B, Table I, Column 1 of this part.

(2) Each airborne radioactivity area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION[1]
AIRBORNE RADIOACTIVITY AREA

(e) *Additional requirements.* (1) Each area or room in which licensed material is used or stored and which contains any radioactive material (other than natural uranium or thorium) in an amount exceeding 10 times the quantity of such material specified in Appendix C of this part shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION[1]
RADIOACTIVE MATERIAL(S)

(2) Each area or room in which natural uranium or thorium is used or stored in an amount exceeding one hundred times the quantity specified in Appendix C of this part shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION[1]
RADIOACTIVE MATERIAL(S)

(f) *Containers.* (1) Except as provided in subparagraph (3) of this paragraph, each container of licensed material shall bear a durable, clearly visible label identifying the radioactive contents.

(2) A label required pursuant to subparagraph (1) of this paragraph shall bear the radiation caution symbol and the words "CAUTION, RADIOACTIVE MATERIAL" or "DANGER, RADIOACTIVE MATERIAL". It shall also provide sufficient information[1] to permit individuals handling or using the containers, or working in the vicinity thereof, to take precautions to avoid or minimize exposures.

31 F.R. 10514

(3) Notwithstanding the provisions of subparagraph (1) of this paragraph, labeling is not required:

(i) For containers that do not contain licensed materials in quantities greater than the applicable quantities listed in Appendix C of this part.

(ii) For containers containing only natural uranium or thorium in quantities no greater than 10 times the applicable quantities listed in Appendix C of this part.

(iii) For containers that do not contain licensed materials in concentrations

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[1] As appropriate, the information will include radiation levels, kinds of material, estimate of activity, date for which activity is estimated, mass enrichment, etc.

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greater than the applicable concentrations listed in Column 2, Table I, Appendix B of this part.

(iii) For periods of less than 8 hours when the container is attended by an individual who takes the precautions necessary to prevent the exposure of any individual to radiation or radioactive materials in excess of the limits established by the regulations in this part.

(iv) For containers when they are in transport and packaged and labeled in accordance with regulations of the Department of Transportation.

(v) For containers which are accessible only to individuals authorized to handle or use them, or to work in the vicinity thereof, provided that the contents are identified to these individuals by a readily available written record.

(vi) For manufacturing or process equipment such as nuclear reactors, reactor components, piping, and tanks.

§ 20.204 Exceptions from posting requirements.

Notwithstanding the provisions of § 20.203:

(a) A room or area is not required to be posted with a caution sign because of the presence of a sealed source provided the radiation level twelve inches from the surface of the source container or housing does not exceed five millirem per hour.

(b) Rooms or other areas in hospitals are not required to be posted with caution signs because of the presence of patients containing byproduct material provided that there are personnel in attendance who will take the precautions necessary to prevent the exposure of any individual to radiation or radioactive material in excess of the limits established in the regulations in this part.

(c) Caution signs are not required to be posted at areas or rooms containing radioactive materials for periods of less than eight hours provided that (1) the materials are constantly attended during such periods by an individual who shall take the precautions necessary to prevent the exposure of any individual to radiation or radioactive materials in excess of the limits established in the regulations in this part and; (2) such area or room is subject to the licensee's control.

(d) A room or other area is not required to be posted with a caution sign because of the presence of radioactive materials packaged and labeled in accordance with regulations of the Department of Transportation.

§ 20.206 Instruction of personnel; posting of notices to employees.

(a) All individuals working in or frequenting any portion of a restricted area shall be informed of the occurrence of radioactive materials or of radiation in such portions of the restricted area; shall be instructed in the safety problems associated with exposure to such materials or radiation and in precautions or procedures to minimize exposure; shall be instructed in the applicable provisions of the regulations in this part for the protection of personnel from exposures to radiation or radioactive materials occurring in such areas; and shall be advised of reports of radiation exposure which employees may request pursuant to the regulations in this part.

(b) Each licensee shall post a current copy of the regulations in this part, a copy of the license and a copy of operating procedures applicable to work under the license conspicuously in such locations as to ensure that employees working in or frequenting restricted areas will observe them on the way to and from their place of employment, or shall post a notice which describes these documents and states where they may be examined.

(c) Form AEC-3, "Notice to Employees" shall be posted in a sufficient number of places in every establishment where employees are employed in activities licensed by the Commission to permit employees working in or frequenting any portion of a restricted area to observe a copy on the way to or from their place of employment.

NOTE: Copies of Form AEC-3, "Notice to Employees" may be obtained by writing to the Director of the appropriate U.S. Atomic Energy Commission Regional Compliance Office listed in Appendix D or the Director, Division of Materials Licensing, U.S. Atomic Energy Commission, Washington, D.C. 20545.

§ 20.207 Storage of licensed materials.

Licensed materials stored in an unrestricted area shall be secured against unauthorized removal from the place of storage.

WASTE DISPOSAL

§ 20.301 General requirement.

No licensee shall dispose of licensed material except:

(a) By transfer to an authorized recipient as provided in the regulations in Part 30, 40, or 70 of this chapter, whichever may be applicable; or

(b) As authorized pursuant to § 20.302; or

(c) As provided in § 20.303 or § 20.304, applicable respectively to the disposal of licensed material by release into sanitary sewerage systems or burial in soil, or in § 20.106 (Concentrations in Effluents to Unrestricted Areas).

§ 20.302 Method for obtaining approval of proposed disposal procedures.

Any licensee or applicant for a license may apply to the Commission for approval of proposed procedures to dispose of licensed material in a manner not otherwise authorized in the regulations in this chapter. Each application should include a description of the licensed material and any other radioactive material involved, including the quantities and kinds of such material and the levels of radioactivity involved, and the proposed manner and conditions of disposal. The application should also include an analysis and evaluation of pertinent information as to the nature of the environment, including topographical, geological, meteorological and hydrological characteristics; usage of ground and surface waters in the general area; the nature and location of other potentially affected facilities; and procedures to be observed to minimize the risk of unexpected or hazardous exposures.

The Commission will not approve any application for a license to receive licensed material from other persons for disposal on land not owned by the Federal Government or by a State government.

§ 20.303 Disposal by release into sanitary sewerage systems.

No licensee shall discharge licensed material into a sanitary sewerage system unless:

(a) It is readily soluble or dispersible in water; and

(b) The quantity of any licensed or other radioactive material released into the system by the licensee in any one day does not exceed the larger of subparagraphs (1) or (2) of this paragraph:

(1) The quantity which, if diluted by the average daily quantity of sewage released into the sewer by the licensee, will result in an average concentration equal to the limits specified in Appendix B, Table I, Column 2 of this part; or

(2) Ten times the quantity of such material specified in Appendix C of this part; and

(c) The quantity of any licensed or other radioactive material released in any one month, if diluted by the average monthly quantity of water released by the licensee, will not result in an average concentration exceeding the limits specified in Appendix B, Table I, Column 2 of this part; and

(d) The gross quantity of licensed and other radioactive material released into the sewerage system by the licensee does not exceed one curie per year.

Excreta from individuals undergoing medical diagnosis or therapy with radioactive material shall be exempt from any limitations contained in this section.

§ 20.304 Disposal by burial in soil.

No licensee shall dispose of licensed material by burial in soil unless:

(a) The total quantity of licensed and other radioactive materials buried at any one location and time does not exceed, at the time of burial, 1,000 times the amount specified in Appendix C of this part; and

(b) Burial is at a minimum depth of four feet; and

(c) Successive burials are separated by distances of at least six feet and not more than 12 burials are made in any year.

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[1] For example, containers in locations such as water-filled canals, storage vaults, or hot cells.

*Amended 34 FR [illegible]
December [illegible], 1969

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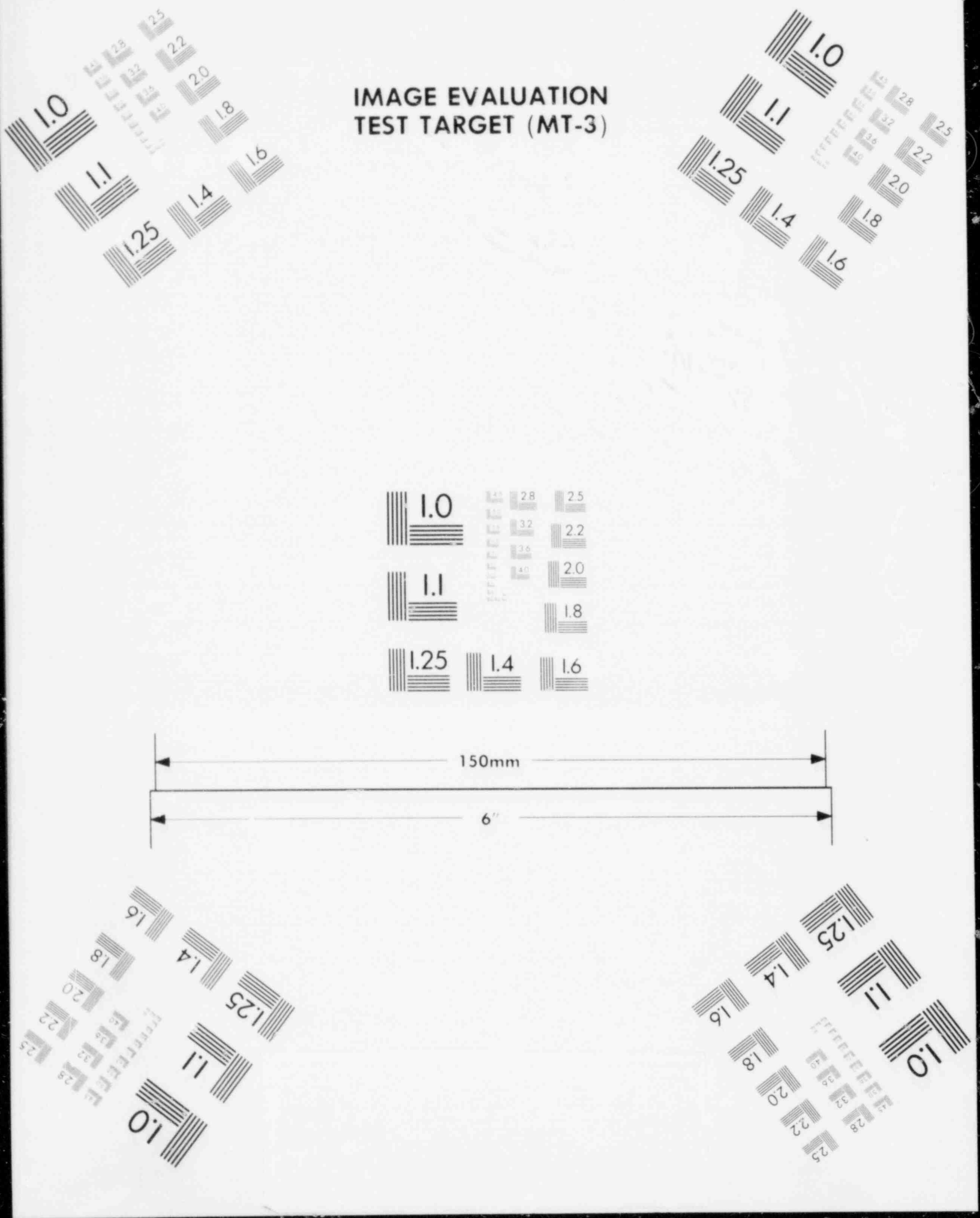
T-93-0622

IMAGE EVALUATION TEST TARGET (MT-3)



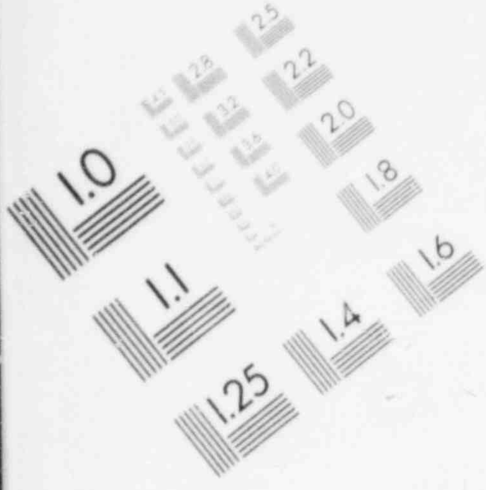
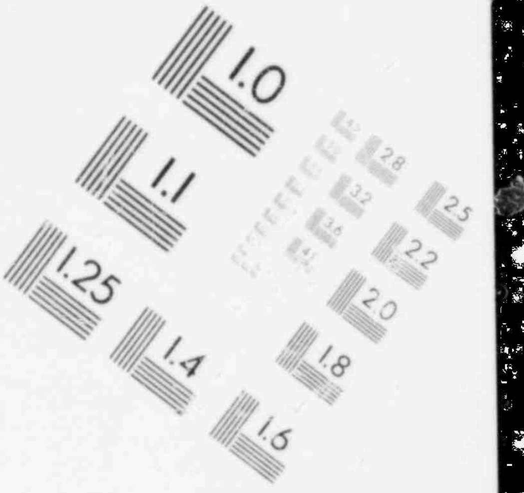
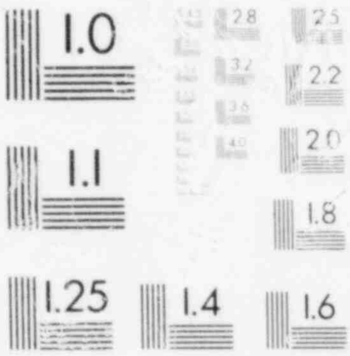
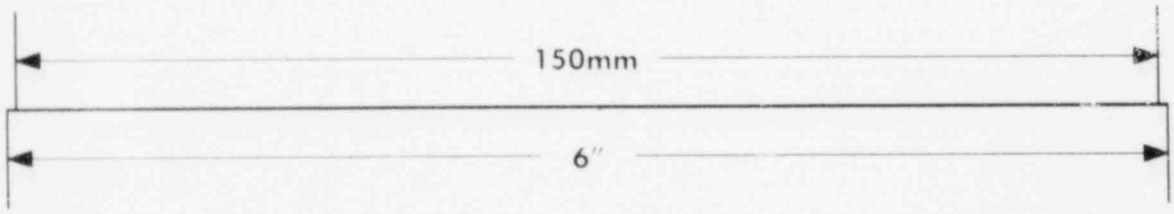
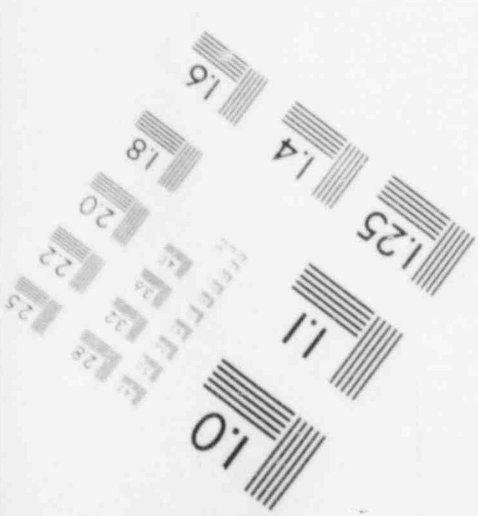


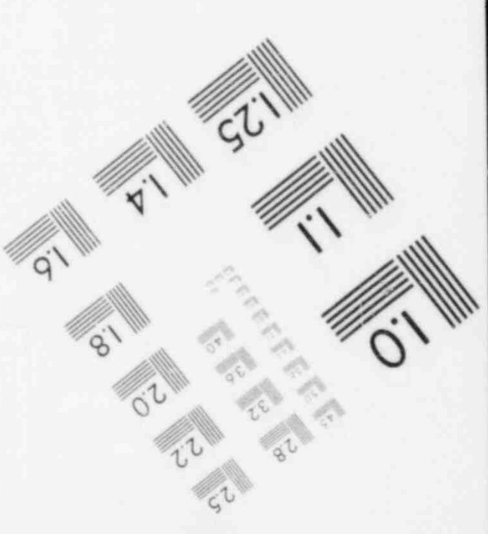
IMAGE EVALUATION
TEST TARGET (MT-3)











PART 20 - STANDARDS FOR PROTECTION AGAINST RADIATION

§ 20.305 Treatment or disposal by incineration.

No licensee shall treat or dispose of licensed material by incineration except as specifically approved by the Commission pursuant to §§ 20.106 (b) and 20.302.

RECORDS, REPORTS, AND NOTIFICATION

§ 20.401 Records of surveys, radiation monitoring, and disposal.

[illegible]

[illegible] shall maintain records [illegible] in the same units used in [illegible] this part showing the results of surveys required by § 20.201 [illegible] and disposals made under §§ 20.302, 20.303 and 20.304.

[illegible] Records of individual radiation exposure which must be maintained pursuant to the provisions of paragraph [illegible] of this section shall be preserved until December 31 [illegible] or until a date 5 years after termination of the individual's [illegible] whichever is later. Records which must be maintained pursuant to this part may be maintained in the form of microfilms.

[illegible] December 31 [illegible] the Commission [illegible] that amended this paragraph [illegible] the former preservation of records [illegible] should not be [illegible]

§ 20.402 Reports of theft or loss of licensed material.

[illegible] by telephone [illegible] the Director of [illegible] Safety Com[illegible] Office listed [illegible] after its occurrence [illegible] to the licensee [illegible] may result to [illegible]

[illegible] is required to [illegible] of this [illegible] days after he [illegible] make a report [illegible] the Director, Division of Compliance, U.S. Atomic Energy Commission, Washington, D.C. 20545, with a [illegible] the appropriate [illegible] Compliance Regional [illegible] in Appendix D [illegible]

[illegible] of the licensed [illegible]

[illegible] of the [illegible] theft [illegible]

[illegible] of disposition or [illegible] of the licensed material involved.

[illegible]

[illegible] Thermal [illegible] available [illegible]

[illegible] of the [illegible]

§ 20.403 [illegible]

[illegible]

Atomic Energy [illegible]

Commission [illegible]

[illegible]

(1) [illegible] of radio[illegible]

feet, [illegible] or [illegible] to 75 rems or more of radiation; or

(2) The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed [illegible] times the limit specified for such material in Appendix B, Table II; or

(3) A loss of one day or more of the operation of any facilities affected; or

(4) Damage to property in excess of $1,000.

[illegible] Any report filed with the Commission pursuant to this section shall be prepared so that names of individuals who have received exposure to radiation will be stated in a separate part of the report.

§ 20.404 Reports to former employees of exposure to radiation.

[illegible]

§ 20.405 Reports of overexposures and excessive levels and concentrations.

(a) In addition to any notification [illegible] each licensee shall [illegible] days [illegible] Director, Division of Compliance, U.S. Atomic Energy Commission, Washington, D.C. 20545 [illegible] with [illegible] of the appropriate [illegible]

[illegible]

[illegible] of any incident for which notification is required by § 20.403 [illegible] of radioactive material [illegible] exposure of any individual [illegible] in an unrestricted area in excess of ten times any applicable limit [illegible] in this part or in the licensee's [illegible] Each report [illegible] under this [illegible] shall describe the extent of exposure of persons to radiation or to radioactive material, levels of radiation and concentrations of radioactive material involved, the cause of the exposure, levels or concentrations, and corrective steps taken or planned to assure against a recurrence.

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PART 20 - STANDARDS FOR PROTECTION AGAINST RADIATION

(b) In any case where [illegible] required pursuant to the [illegible] this section to report to the Commission any exposure of an individual to radiation [illegible] the licensee [illegible] such individual [illegible] of exposure [illegible] writing and [illegible] statement.

The report [illegible] [illegible] regulations [illegible] You should preserve this report for future reference.

25 F.R. 10914

(c) Any report filed with the Commission pursuant to [illegible] shall be prepared so that [illegible] will be placed in a separate part of the report.

28 F.R. 6522

§ 20.406 [illegible] of [illegible]

[illegible]

(1) [illegible] Parts 30 and 34 of [illegible]

(2) Possession or use [illegible] material for purposes of [illegible] pursuant to Parts 30 and 34 of [illegible]

(3) Possession or use of [illegible] for purposes of [illegible] in a quantity exceeding [illegible] plutonium or any [illegible] pursuant to Part 70 of this chapter; or

(4) Possession or use at any one time for [illegible] pursuant to Parts 30, 32 or 33 of this chapter, byproduct material in quantities exceeding any one of the following quantities:

Radionuclide	Quantity in curies
[illegible]	[illegible]

(c) Each [illegible] within [illegible] first quarter of each calendar year [illegible] of the Director of [illegible] U.S. Atomic Energy Commission, Washington, D.C. 20545 [illegible]

[illegible]

Reports of personnel exposure on termination of employment or work

When a licensee is required [illegible] an individual subject to [illegible] the licensee [illegible] he terminates his work [illegible] the licensee shall furnish to such individual and to the Director of Regulation, U.S. Atomic Energy Commission, Washington, D.C. 20545 a report of the individual's exposure to radiation and radioactive material incurred during the period of employment or work assignment in the licensee's facility [illegible] the information [illegible] by the licensee pursuant to [illegible] and 20.108. Such report shall be furnished within 30 days after the exposure of the individual has been determined by the licensee or 90 days after the date of termination of employment or work assignment, whichever is earlier.

EXCEPTIONS AND ADDITIONAL REQUIREMENTS

§ 20.501 Applications for exemptions.

The Commission may [illegible] requirements of the regulations in this part [illegible] will not result in undue hazard to life or property.

§ 20.502 Additional requirements.

The Commission may [illegible] the protection of health or to minimize danger to life or property.

ENFORCEMENT

§ 20.601 Violations.

[illegible]

Appendix A [illegible]

[illegible] calendar year 1966 shall [illegible] within [illegible] days [illegible] the effective [illegible] licensee [illegible] of the year during [illegible] was in effect.

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APPENDIX B

Concentrations in Air and Water Above Natural Background

(See notes at end of appendix)

Element (atomic number)	Isotope[1]		Table I		Table II	
			Column 1	Column 2	Column 1	Column 2
			Air (μc/ml)	Water (μc/ml)	Air (μc/ml)	Water (μc/ml)
Actinium (89)	Ac 227	S	2×10^{-12}	6×10^{-5}	8×10^{-14}	2×10^{-6}
		I	3×10^{-11}	9×10^{-3}	9×10^{-13}	3×10^{-4}
	Ac 228	S	8×10^{-8}	3×10^{-3}	3×10^{-9}	9×10^{-5}
		I	2×10^{-8}	3×10^{-3}	6×10^{-10}	9×10^{-5}
Americium (95)	Am 241	S	6×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		I	1×10^{-10}	8×10^{-4}	4×10^{-12}	2×10^{-5}
	Am 242m	S	4×10^{-12}	1×10^{-4}	1×10^{-13}	4×10^{-6}
		I	3×10^{-10}	3×10^{-3}	9×10^{-12}	9×10^{-5}
	Am 242	S	4×10^{-8}	4×10^{-3}	1×10^{-9}	1×10^{-4}
		I	5×10^{-8}	4×10^{-3}	2×10^{-9}	1×10^{-4}
	Am 243	S	6×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		I	1×10^{-10}	8×10^{-4}	4×10^{-12}	3×10^{-5}
	Am 244	S	4×10^{-6}	1×10^{-1}	1×10^{-7}	5×10^{-3}
		I	2×10^{-5}	1×10^{-1}	8×10^{-7}	5×10^{-3}
Antimony (51)	Sb 122	S	2×10^{-7}	8×10^{-4}	6×10^{-9}	3×10^{-5}
		I	1×10^{-7}	8×10^{-4}	5×10^{-9}	3×10^{-5}
	Sb 124	S	2×10^{-7}	7×10^{-4}	5×10^{-9}	2×10^{-5}
		I	2×10^{-8}	7×10^{-4}	7×10^{-10}	2×10^{-5}
	Sb 125	S	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	3×10^{-8}	3×10^{-3}	9×10^{-10}	1×10^{-4}
Argon (18)	A 37	Sub[2]	6×10^{-3}		1×10^{-4}	
	A 41	Sub	2×10^{-6}		4×10^{-8}	
Arsenic (33)	As 73	S	2×10^{-6}	1×10^{-2}	7×10^{-8}	5×10^{-4}
		I	4×10^{-7}	1×10^{-2}	1×10^{-8}	5×10^{-4}
	As 74	S	3×10^{-7}	2×10^{-3}	1×10^{-8}	5×10^{-5}
		I	1×10^{-7}	2×10^{-3}	4×10^{-9}	5×10^{-5}
	As 76	S	1×10^{-7}	6×10^{-4}	4×10^{-9}	2×10^{-5}
		I	1×10^{-7}	6×10^{-4}	3×10^{-9}	2×10^{-5}
	As 77	S	5×10^{-7}	2×10^{-3}	2×10^{-8}	8×10^{-5}
		I	4×10^{-7}	2×10^{-3}	1×10^{-8}	8×10^{-5}
Astatine (85)	At 211	S	7×10^{-9}	5×10^{-5}	2×10^{-10}	2×10^{-6}
		I	3×10^{-8}	2×10^{-3}	1×10^{-9}	7×10^{-5}
Barium (56)	Ba 131	S	1×10^{-6}	5×10^{-3}	4×10^{-8}	2×10^{-4}
		I	4×10^{-7}	5×10^{-3}	1×10^{-8}	2×10^{-4}
	Ba 140	S	1×10^{-7}	8×10^{-4}	4×10^{-9}	3×10^{-5}
		I	4×10^{-8}	7×10^{-4}	1×10^{-9}	2×10^{-5}
Berkelium (97)	Bk 249	S	9×10^{-10}	2×10^{-2}	3×10^{-11}	6×10^{-4}
		I	1×10^{-7}	2×10^{-2}	4×10^{-9}	6×10^{-4}
	Bk 250	S	1×10^{-7}	6×10^{-3}	5×10^{-9}	2×10^{-4}
		I	1×10^{-6}	6×10^{-3}	4×10^{-8}	2×10^{-4}
Beryllium (4)	Be 7	S	6×10^{-6}	5×10^{-2}	2×10^{-7}	2×10^{-3}
		I	1×10^{-6}	5×10^{-2}	4×10^{-8}	2×10^{-3}
Bismuth (83)	Bi 206	S	2×10^{-7}	1×10^{-3}	6×10^{-9}	4×10^{-5}
		I	1×10^{-7}	1×10^{-3}	5×10^{-9}	4×10^{-5}
	Bi 207	S	2×10^{-7}	2×10^{-3}	6×10^{-9}	6×10^{-5}
		I	1×10^{-8}	2×10^{-3}	5×10^{-10}	6×10^{-5}
	Bi 210	S	6×10^{-9}	1×10^{-3}	2×10^{-10}	4×10^{-5}
		I	6×10^{-9}	1×10^{-3}	2×10^{-10}	4×10^{-5}
	Bi 212	S	1×10^{-7}	1×10^{-2}	3×10^{-9}	4×10^{-4}
		I	2×10^{-7}	1×10^{-2}	7×10^{-9}	4×10^{-4}

APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See notes at end of appendix)

Element (atomic number)	Isotope[1]		Table I		Table II	
			Column 1	Column 2	Column 1	Column 2
			Air (μc/ml)	Water (μc/ml)	Air (μc/ml)	Water (μc/ml)
Bromine (35)	Br 82	S	1×10^{-6}	8×10^{-3}	4×10^{-8}	3×10^{-4}
		I	2×10^{-7}	1×10^{-3}	6×10^{-9}	4×10^{-5}
Cadmium (48)	Cd 109	S	5×10^{-8}	5×10^{-3}	2×10^{-9}	2×10^{-4}
		I	7×10^{-8}	5×10^{-3}	3×10^{-9}	2×10^{-4}
	Cd 115m	S	4×10^{-8}	7×10^{-4}	1×10^{-9}	3×10^{-5}
		I	4×10^{-8}	7×10^{-4}	1×10^{-9}	3×10^{-5}
	Cd 115	S	2×10^{-7}	1×10^{-3}	8×10^{-9}	3×10^{-5}
		I	2×10^{-7}	1×10^{-3}	6×10^{-9}	4×10^{-5}
Calcium (20)	Ca 45	S	3×10^{-8}	3×10^{-4}	1×10^{-9}	9×10^{-6}
		I	1×10^{-7}	5×10^{-3}	4×10^{-9}	2×10^{-4}
	Ca 47	S	2×10^{-7}	1×10^{-3}	6×10^{-9}	5×10^{-5}
		I	2×10^{-7}	1×10^{-3}	6×10^{-9}	3×10^{-5}
Californium (98)	Cf 249	S	2×10^{-12}	1×10^{-4}	5×10^{-14}	4×10^{-6}
		I	1×10^{-10}	7×10^{-4}	3×10^{-12}	2×10^{-5}
	Cf 250	S	5×10^{-12}	4×10^{-4}	2×10^{-13}	1×10^{-5}
		I	1×10^{-10}	7×10^{-4}	3×10^{-12}	3×10^{-5}
	Cf 251	S	2×10^{-12}	1×10^{-4}	6×10^{-14}	4×10^{-6}
		I	1×10^{-10}	8×10^{-4}	3×10^{-12}	3×10^{-5}
	Cf 252	S	2×10^{-11}	7×10^{-4}	7×10^{-13}	2×10^{-5}
		I	1×10^{-10}	7×10^{-4}	4×10^{-12}	2×10^{-5}
	Cf 253	S	8×10^{-10}	4×10^{-3}	3×10^{-11}	1×10^{-4}
		I	8×10^{-10}	4×10^{-3}	3×10^{-11}	1×10^{-4}
	Cf 254	S	5×10^{-12}	4×10^{-6}	2×10^{-13}	1×10^{-7}
		I	5×10^{-12}	4×10^{-6}	2×10^{-13}	1×10^{-7}
Carbon (6)	C 14	S	4×10^{-6}	2×10^{-2}	1×10^{-7}	8×10^{-4}
	(CO_2)	Sub	5×10^{-5}		1×10^{-6}	
Cerium (58)	Ce 141	S	4×10^{-7}	3×10^{-3}	2×10^{-8}	9×10^{-5}
		I	2×10^{-7}	3×10^{-3}	5×10^{-9}	9×10^{-5}
	Ce 143	S	3×10^{-7}	1×10^{-3}	9×10^{-9}	4×10^{-5}
		I	2×10^{-7}	1×10^{-3}	7×10^{-9}	4×10^{-5}
	Ce 144	S	1×10^{-8}	3×10^{-4}	3×10^{-10}	1×10^{-5}
		I	6×10^{-9}	3×10^{-4}	2×10^{-10}	1×10^{-5}
Cesium (55)	Cs 131	S	1×10^{-5}	7×10^{-2}	4×10^{-7}	2×10^{-3}
		I	3×10^{-6}	3×10^{-2}	1×10^{-7}	9×10^{-4}
	Cs 134m	S	4×10^{-5}	2×10^{-1}	1×10^{-6}	6×10^{-3}
		I	6×10^{-6}	3×10^{-2}	2×10^{-7}	1×10^{-3}
	Cs 134	S	4×10^{-8}	3×10^{-4}	1×10^{-9}	9×10^{-6}
		I	1×10^{-8}	1×10^{-3}	4×10^{-10}	4×10^{-5}
	Cs 135	S	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	9×10^{-8}	7×10^{-3}	3×10^{-9}	2×10^{-4}
	Cs 136	S	4×10^{-7}	2×10^{-3}	1×10^{-8}	9×10^{-5}
		I	2×10^{-7}	2×10^{-3}	6×10^{-9}	6×10^{-5}
	Cs 137	S	6×10^{-8}	4×10^{-4}	2×10^{-9}	2×10^{-5}
		I	1×10^{-8}	1×10^{-3}	5×10^{-10}	4×10^{-5}
Chlorine (17)	Cl 36	S	4×10^{-7}	2×10^{-3}	1×10^{-8}	8×10^{-5}
		I	2×10^{-8}	2×10^{-3}	8×10^{-10}	6×10^{-5}
	Cl 38	S	3×10^{-6}	1×10^{-2}	9×10^{-8}	4×10^{-4}
		I	2×10^{-6}	1×10^{-2}	7×10^{-8}	4×10^{-4}
Chromium (24)	Cr 51	S	1×10^{-5}	5×10^{-2}	4×10^{-7}	2×10^{-3}
		I	2×10^{-6}	5×10^{-2}	8×10^{-8}	2×10^{-3}

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PART 20 - STANDARDS FOR PROTECTION AGAINST RADIATION

APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See notes at end of appendix)

Element (atomic number)	Isotope		Table I Column 1 Air (μc/ml)	Table I Column 2 Water (μc/ml)	Table II Column 1 Air (μc/ml)	Table II Column 2 Water (μc/ml)
Cobalt (27)	Co 57	S	3×10^{-6}	2×10^{-2}	1×10^{-7}	5×10^{-4}
		I	2×10^{-7}	1×10^{-2}	6×10^{-9}	4×10^{-4}
	Co 58m	S	2×10^{-5}	8×10^{-2}	6×10^{-7}	3×10^{-3}
		I	9×10^{-6}	6×10^{-2}	3×10^{-7}	2×10^{-3}
	Co 58	S	8×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
		I	5×10^{-8}	3×10^{-3}	2×10^{-9}	9×10^{-5}
	Co 60	S	3×10^{-7}	1×10^{-3}	1×10^{-8}	5×10^{-5}
		I	9×10^{-9}	1×10^{-3}	3×10^{-10}	3×10^{-5}
Copper (29)	Cu 64	S	2×10^{-6}	1×10^{-2}	7×10^{-8}	3×10^{-4}
		I	1×10^{-6}	6×10^{-3}	4×10^{-8}	2×10^{-4}
Curium (96)	Cm 242	S	1×10^{-10}	7×10^{-4}	4×10^{-12}	2×10^{-5}
		I	2×10^{-10}	7×10^{-4}	6×10^{-12}	3×10^{-5}
	Cm 243	S	6×10^{-12}	1×10^{-4}	2×10^{-13}	5×10^{-6}
		I	1×10^{-10}	7×10^{-4}	3×10^{-12}	2×10^{-5}
	Cm 244	S	9×10^{-12}	2×10^{-4}	3×10^{-13}	7×10^{-6}
		I	1×10^{-10}	8×10^{-4}	3×10^{-12}	3×10^{-5}
	Cm 245	S	5×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		I	1×10^{-10}	8×10^{-4}	4×10^{-12}	3×10^{-5}
	Cm 246	S	5×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		I	1×10^{-10}	8×10^{-4}	4×10^{-12}	3×10^{-5}
	Cm 247	S	5×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		I	1×10^{-10}	6×10^{-4}	4×10^{-12}	2×10^{-5}
	Cm 248	S	6×10^{-13}	1×10^{-5}	2×10^{-14}	4×10^{-7}
		I	1×10^{-11}	4×10^{-5}	4×10^{-13}	1×10^{-6}
	Cm 249	S	1×10^{-5}	6×10^{-2}	4×10^{-7}	2×10^{-3}
		I	1×10^{-5}	6×10^{-2}	4×10^{-7}	2×10^{-3}
Dysprosium (66)	Dy 165	S	3×10^{-6}	1×10^{-2}	9×10^{-8}	4×10^{-4}
		I	2×10^{-6}	1×10^{-2}	7×10^{-8}	4×10^{-4}
	Dy 166	S	2×10^{-7}	1×10^{-3}	8×10^{-9}	4×10^{-5}
		I	2×10^{-7}	1×10^{-3}	7×10^{-9}	4×10^{-5}
Einsteinium (99)	Es 253	S	8×10^{-10}	7×10^{-4}	3×10^{-11}	2×10^{-5}
		I	6×10^{-10}	7×10^{-4}	2×10^{-11}	2×10^{-5}
	Es 254m	S	5×10^{-9}	5×10^{-4}	2×10^{-10}	2×10^{-5}
		I	6×10^{-9}	5×10^{-4}	2×10^{-10}	2×10^{-5}
	Es 254	S	2×10^{-11}	4×10^{-4}	6×10^{-13}	1×10^{-5}
		I	1×10^{-10}	4×10^{-4}	4×10^{-12}	1×10^{-5}
	Es 255	S	5×10^{-10}	8×10^{-4}	2×10^{-11}	3×10^{-5}
		I	4×10^{-10}	8×10^{-4}	1×10^{-11}	3×10^{-5}
Erbium (68)	Er 169	S	6×10^{-7}	3×10^{-3}	2×10^{-8}	9×10^{-5}
		I	4×10^{-7}	3×10^{-3}	1×10^{-8}	9×10^{-5}
	Er 171	S	7×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	6×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
Europium (63)	Eu 152 (T/2 = 9.2 hrs)	S	4×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
		I	3×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
	Eu 152 (T/2 = 13 yrs)	S	1×10^{-8}	2×10^{-3}	4×10^{-10}	8×10^{-5}
		I	2×10^{-8}	2×10^{-3}	6×10^{-10}	8×10^{-5}
	Eu 154	S	4×10^{-9}	6×10^{-4}	1×10^{-10}	2×10^{-5}
		I	7×10^{-9}	6×10^{-4}	2×10^{-10}	2×10^{-5}
	Eu 155	S	9×10^{-8}	6×10^{-3}	3×10^{-9}	2×10^{-4}
		I	7×10^{-8}	6×10^{-3}	3×10^{-9}	2×10^{-4}

APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See notes at end of appendix)

Element (atomic number)	Isotope¹		Table I Column 1 Air (μc/ml)	Table I Column 2 Water (μc/ml)	Table II Column 1 Air (μc/ml)	Table II Column 2 Water (μc/ml)
Fermium (100)	Fm 254	S	6×10^{-8}	4×10^{-3}	2×10^{-9}	1×10^{-4}
		I	7×10^{-8}	4×10^{-3}	2×10^{-9}	1×10^{-4}
	Fm 255	S	2×10^{-8}	1×10^{-3}	6×10^{-10}	3×10^{-5}
		I	1×10^{-8}	1×10^{-3}	4×10^{-10}	3×10^{-5}
	Fm 256	S	3×10^{-9}	3×10^{-5}	1×10^{-10}	9×10^{-7}
		I	2×10^{-9}	3×10^{-5}	6×10^{-11}	9×10^{-7}
Fluorine (9)	F 18	S	5×10^{-6}	2×10^{-2}	2×10^{-7}	8×10^{-4}
		I	3×10^{-6}	1×10^{-2}	9×10^{-8}	5×10^{-4}
Gadolinium (64)	Gd 153	S	2×10^{-7}	6×10^{-3}	8×10^{-9}	2×10^{-4}
		I	9×10^{-8}	6×10^{-3}	3×10^{-9}	2×10^{-4}
	Gd 159	S	5×10^{-7}	2×10^{-3}	2×10^{-8}	8×10^{-5}
		I	4×10^{-7}	2×10^{-3}	1×10^{-8}	8×10^{-5}
Gallium (31)	Ga 72	S	2×10^{-7}	1×10^{-3}	8×10^{-9}	4×10^{-5}
		I	2×10^{-7}	1×10^{-3}	6×10^{-9}	4×10^{-5}
Germanium (32)	Ge 71	S	1×10^{-5}	5×10^{-2}	4×10^{-7}	2×10^{-3}
		I	6×10^{-6}	5×10^{-2}	2×10^{-7}	2×10^{-3}
Gold (79)	Au 196	S	1×10^{-6}	5×10^{-3}	4×10^{-8}	2×10^{-4}
		I	6×10^{-7}	4×10^{-3}	2×10^{-8}	1×10^{-4}
	Au 198	S	3×10^{-7}	2×10^{-3}	1×10^{-8}	5×10^{-5}
		I	2×10^{-7}	1×10^{-3}	8×10^{-9}	5×10^{-5}
	Au 199	S	1×10^{-6}	5×10^{-3}	4×10^{-8}	2×10^{-4}
		I	8×10^{-7}	4×10^{-3}	3×10^{-8}	2×10^{-4}
Hafnium (72)	Hf 181	S	4×10^{-8}	2×10^{-3}	1×10^{-9}	7×10^{-5}
		I	7×10^{-8}	2×10^{-3}	3×10^{-9}	7×10^{-5}
Holmium (67)	Ho 166	S	2×10^{-7}	9×10^{-4}	7×10^{-9}	3×10^{-5}
		I	2×10^{-7}	9×10^{-4}	6×10^{-9}	3×10^{-5}
Hydrogen (1)	H3	S	5×10^{-6}	1×10^{-1}	2×10^{-7}	3×10^{-3}
		I	5×10^{-6}	1×10^{-1}	2×10^{-7}	3×10^{-3}
		Sub	2×10^{-3}		4×10^{-5}	
Indium (49)	In 113m	S	8×10^{-6}	4×10^{-2}	3×10^{-7}	1×10^{-3}
		I	7×10^{-6}	4×10^{-2}	2×10^{-7}	1×10^{-3}
	In 114m	S	1×10^{-7}	5×10^{-4}	4×10^{-9}	2×10^{-5}
		I	2×10^{-8}	5×10^{-4}	7×10^{-10}	2×10^{-5}
	In 115m	S	2×10^{-6}	1×10^{-2}	8×10^{-8}	4×10^{-4}
		I	2×10^{-6}	1×10^{-2}	6×10^{-8}	4×10^{-4}
	In 115	S	2×10^{-7}	3×10^{-3}	9×10^{-9}	9×10^{-5}
		I	3×10^{-8}	3×10^{-3}	1×10^{-9}	9×10^{-5}
Iodine (53)	I 125	S	5×10^{-9}	4×10^{-5}	8×10^{-11}	2×10^{-7}
		I	2×10^{-7}	6×10^{-3}	6×10^{-9}	2×10^{-4}
	I 126	S	8×10^{-9}	5×10^{-5}	9×10^{-11}	3×10^{-7}
		I	3×10^{-7}	3×10^{-3}	1×10^{-8}	9×10^{-5}
	I 129	S	2×10^{-9}	1×10^{-5}	2×10^{-11}	6×10^{-8}
		I	7×10^{-8}	6×10^{-3}	2×10^{-9}	2×10^{-4}
	I 131	S	9×10^{-9}	6×10^{-5}	1×10^{-10}	3×10^{-7}
		I	3×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
	I 132	S	2×10^{-7}	2×10^{-3}	3×10^{-9}	8×10^{-6}
		I	9×10^{-7}	5×10^{-3}	3×10^{-8}	2×10^{-4}
	I 133	S	3×10^{-8}	2×10^{-4}	4×10^{-10}	1×10^{-6}
		I	2×10^{-7}	1×10^{-3}	7×10^{-9}	4×10^{-5}
	I 134	S	5×10^{-7}	4×10^{-3}	6×10^{-9}	2×10^{-5}

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APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See notes at end of appendix)

Element (atomic number)	Isotope[1]		Table I Column 1 Air (μc/ml)	Table I Column 2 Water (μc/ml)	Table II Column 1 Air (μc/ml)	Table II Column 2 Water (μc/ml)
Iodine (53)	I 134	I	3 × 10 [illegible]	2 × 10 [illegible]	1 × 10 [illegible]	6 × 10 [illegible]
	I 135	S	1 × 10 [illegible]	7 × 10 [illegible]	1 × 10 [illegible]	8 × 10 [illegible]
		I	4 × 10 [illegible]	2 × 10 [illegible]	1 × 10 [illegible]	7 × 10 [illegible]
Iridium (77)	Ir 190	S	1 × 10 [illegible]	6 × 10 [illegible]	4 × 10 [illegible]	2 × 10 [illegible]
		I	4 × 10 [illegible]	5 × 10 [illegible]	1 × 10 [illegible]	2 × 10 [illegible]
	Ir 192	S	1 × 10 [illegible]	1 × 10 [illegible]	4 × 10 [illegible]	4 × 10 [illegible]
		I	3 × 10 [illegible]	[illegible]	9 × 10 [illegible]	4 × 10 [illegible]
	Ir 194	S	2 × 10 [illegible]	1 × 10 [illegible]	8 × 10 [illegible]	3 × 10 [illegible]
		I	2 × 10 [illegible]	9 × 10 [illegible]	5 × 10 [illegible]	3 × 10 [illegible]
Iron (26)	Fe 55	S	9 × 10 [illegible]	2 × 10 [illegible]	3 × 10 [illegible]	8 × 10 [illegible]
		I	1 × 10 [illegible]	7 × 10 [illegible]	3 × 10 [illegible]	2 × 10 [illegible]
	Fe 59	S	1 × 10 [illegible]	2 × 10 [illegible]	5 × 10 [illegible]	6 × 10 [illegible]
		I	5 × 10 [illegible]	6 × 10 [illegible]	2 × 10 [illegible]	5 × 10 [illegible]
Krypton (36)	Kr 85m	Sub	6 × 10 [illegible]		1 × 10 [illegible]	
	Kr 85	Sub	1 × 10 [illegible]		3 × 10 [illegible]	
	Kr 87	Sub	1 × 10 [illegible]		2 × 10 [illegible]	
	Kr 88	Sub	1 × 10 [illegible]		2 × 10 [illegible]	
Lanthanum (57)	La 140	S	3 × 10 [illegible]	7 × 10 [illegible]	5 × 10 [illegible]	2 × 10 [illegible]
		I	1 × 10 [illegible]	7 × 10 [illegible]	4 × 10 [illegible]	2 × 10 [illegible]
Lead (82)	Pb 203	S	3 × 10 [illegible]	1 × 10 [illegible]	9 × 10 [illegible]	4 × 10 [illegible]
		I	2 × 10 [illegible]	1 × 10 [illegible]	6 × 10 [illegible]	4 × 10 [illegible]
	Pb 210	S	1 × 10 [illegible]	4 × 10 [illegible]	4 × 10 [illegible]	1 × 10 [illegible]
		I	2 × 10 [illegible]	5 × 10 [illegible]	8 × 10 [illegible]	2 × 10 [illegible]
	Pb 212	S	2 × 10 [illegible]	6 × 10 [illegible]	4 × 10 [illegible]	1 × 10 [illegible]
		I	1 × [illegible]	6 × [illegible]	[illegible]	[illegible]
Lutetium (71)	Lu 177	S	6 × 10 [illegible]	3 × 10 [illegible]	2 × 10 [illegible]	1 × 10 [illegible]
		I	5 × 10 [illegible]	3 × 10 [illegible]	2 × 10 [illegible]	1 × 10 [illegible]
Manganese (25)	Mn 52	S	[illegible]	1 × 10 [illegible]	7 × 10 [illegible]	3 × 10 [illegible]
		I	1 × 10 [illegible]	9 × 10 [illegible]	5 × 10 [illegible]	3 × 10 [illegible]
	Mn 54	S	4 × 10 [illegible]	4 × 10 [illegible]	1 × 10 [illegible]	1 × 10 [illegible]
		I	4 × 10 [illegible]	3 × 10 [illegible]	1 × 10 [illegible]	1 × 10 [illegible]
	Mn 56	S	8 × 10 [illegible]	4 × 10 [illegible]	[illegible]	1 × 10 [illegible]
		I	5 × 10 [illegible]	3 × 10 [illegible]	2 × 10 [illegible]	1 × 10 [illegible]
Mercury (80)	Hg 197m	S	7 × 10 [illegible]	6 × 10 [illegible]	1 × 10 [illegible]	2 × 10 [illegible]
		I	8 × 10 [illegible]	5 × 10 [illegible]	3 × 10 [illegible]	2 × 10 [illegible]
	Hg 197	S	1 × 10 [illegible]	9 × 10 [illegible]	4 × 10 [illegible]	3 × 10 [illegible]
		I	3 × 10 [illegible]	1 × 10 [illegible]	9 × 10 [illegible]	5 × 10 [illegible]
	Hg 203	S	7 × [illegible]	[illegible]	2 × 10 [illegible]	2 × 10 [illegible]
		I	1 × [illegible]	[illegible]	4 × 10 [illegible]	1 × 10 [illegible]
Molybdenum (42)	Mo 99	S	7 × 10 [illegible]	[illegible]	1 × 10 [illegible]	2 × 10 [illegible]
		I	3 × [illegible]	1 × 10 [illegible]	7 × 10 [illegible]	4 × 10 [illegible]
Neodymium (60)	Nd 144	S	8 × [illegible]	[illegible]	3 × 10 [illegible]	7 × 10 [illegible]
		I	3 × 10 [illegible]	2 × 10 [illegible]	1 × 10 [illegible]	8 × 10 [illegible]
	Nd 147	S	4 × 10 [illegible]	7 × 10 [illegible]	1 × 10 [illegible]	6 × 10 [illegible]
		I	3 × 10 [illegible]	[illegible]	8 × 10 [illegible]	6 × 10 [illegible]
	Nd 149	S	2 × 10 [illegible]	8 × 10 [illegible]	6 × 10 [illegible]	3 × 10 [illegible]
		I	1 × 10 [illegible]	8 × 10 [illegible]	5 × 10 [illegible]	3 × 10 [illegible]

APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See notes at end of appendix)

Element (atomic number)	Isotope[1]		Table I Column 1 Air (μc/ml)	Table I Column 2 Water (μc/ml)	Table II Column 1 Air (μc/ml)	Table II Column 2 Water (μc/ml)
Neptunium (93)	Np 237	S	4 × 10 [illegible]	9 × 10 [illegible]	1 × 10 [illegible]	3 × 10 [illegible]
		I	1 × 10 [illegible]	9 × 10 [illegible]	4 × 10 [illegible]	3 × 10 [illegible]
	Np 239	S	8 × 10 [illegible]	4 × 10 [illegible]	3 × 10 [illegible]	1 × 10 [illegible]
		I	7 × 10 [illegible]	4 × 10 [illegible]	7 × 10 [illegible]	1 × 10 [illegible]
Nickel (28)	Ni 59	S	5 × 10 [illegible]	6 × 10 [illegible]	2 × 10 [illegible]	2 × 10 [illegible]
		I	8 × 10 [illegible]	4 × 10 [illegible]	3 × 10 [illegible]	7 × 10 [illegible]
	Ni 63	S	6 × 10 [illegible]	8 × 10 [illegible]	2 × 10 [illegible]	3 × 10 [illegible]
		I	3 × 10 [illegible]	2 × 10 [illegible]	1 × 10 [illegible]	7 × 10 [illegible]
	Ni 65	S	9 × 10 [illegible]	4 × 10 [illegible]	3 × 10 [illegible]	1 × 10 [illegible]
		I	5 × 10 [illegible]	3 × 10 [illegible]	2 × 10 [illegible]	1 × 10 [illegible]
Niobium (Columbium) (41)	Nb 93m	S	1 × 10 [illegible]	1 × 10 [illegible]	4 × 10 [illegible]	4 × 10 [illegible]
		I	2 × 10 [illegible]	1 × 10 [illegible]	3 × 10 [illegible]	4 × 10 [illegible]
	Nb 95	S	3 × 10 [illegible]	3 × 10 [illegible]	2 × 10 [illegible]	1 × 10 [illegible]
		I	1 × 10 [illegible]	3 × 10 [illegible]	3 × 10 [illegible]	1 × 10 [illegible]
	Nb 97	S	6 × 10 [illegible]	3 × 10 [illegible]	3 × 10 [illegible]	9 × 10 [illegible]
		I	5 × 10 [illegible]	3 × 10 [illegible]	2 × 10 [illegible]	9 × 10 [illegible]
Osmium (76)	Os 185	S	5 × 10 [illegible]	2 × 10 [illegible]	2 × 10 [illegible]	7 × 10 [illegible]
		I	5 × 10 [illegible]	2 × 10 [illegible]	2 × 10 [illegible]	7 × 10 [illegible]
	Os 191m	S	2 × [illegible]	7 × 10 [illegible]	6 × 10 [illegible]	3 × 10 [illegible]
		I	9 × 10 [illegible]	7 × 10 [illegible]	2 × 10 [illegible]	2 × 10 [illegible]
	Os 191	S	1 × 10 [illegible]	5 × 10 [illegible]	4 × 10 [illegible]	2 × 10 [illegible]
		I	4 × 10 [illegible]	5 × 10 [illegible]	1 × 10 [illegible]	2 × 10 [illegible]
	Os 193	S	4 × 10 [illegible]	2 × [illegible]	5 × [illegible]	[illegible]
		I	3 × [illegible]	9 × 10 [illegible]	9 × [illegible]	3 × 10 [illegible]
Palladium (46)	Pd 103	S	1 × 10 [illegible]	1 × 10 [illegible]	3 × 10 [illegible]	3 × 10 [illegible]
		I	7 × 10 [illegible]	8 × 10 [illegible]	3 × 10 [illegible]	3 × 10 [illegible]
	Pd 109	S	6 × 10 [illegible]	3 × 10 [illegible]	2 × 10 [illegible]	9 × 10 [illegible]
		I	4 × 10 [illegible]	5 × 10 [illegible]	1 × 10 [illegible]	7 × 10 [illegible]
Phosphorus (15)	P 32	S	7 × 10 [illegible]	5 × 10 [illegible]	2 × 10 [illegible]	2 × 10 [illegible]
		I	8 × 10 [illegible]	7 × 10 [illegible]	3 × 10 [illegible]	2 × 10 [illegible]
Platinum (78)	Pt 191	S	8 × 10 [illegible]	4 × 10 [illegible]	3 × 10 [illegible]	1 × 10 [illegible]
		I	6 × 10 [illegible]	3 × 10 [illegible]	2 × 10 [illegible]	1 × 10 [illegible]
	Pt 193m	S	7 × 10 [illegible]	3 × 10 [illegible]	2 × 10 [illegible]	1 × 10 [illegible]
		I	5 × 10 [illegible]	3 × 10 [illegible]	2 × 10 [illegible]	1 × 10 [illegible]
	Pt 197m	S	6 × 10 [illegible]	3 × 10 [illegible]	2 × 10 [illegible]	1 × 10 [illegible]
		I	5 × 10 [illegible]	3 × 10 [illegible]	2 × 10 [illegible]	9 × 10 [illegible]
	Pt 197	S	8 × 10 [illegible]	4 × 10 [illegible]	3 × 10 [illegible]	1 × 10 [illegible]
		I	6 × 10 [illegible]	3 × 10 [illegible]	2 × 10 [illegible]	1 × 10 [illegible]
Plutonium (94)	Pu 238	S	2 × 10 [illegible]	1 × 10 [illegible]	7 × 10 [illegible]	5 × 10 [illegible]
		I	3 × 10 [illegible]	8 × 10 [illegible]	1 × 10 [illegible]	3 × 10 [illegible]
	Pu 239	S	[illegible]	1 × 10 [illegible]	4 × 10 [illegible]	5 × 10 [illegible]
		I	4 × 10 [illegible]	8 × 10 [illegible]	1 × 10 [illegible]	3 × 10 [illegible]
	Pu 240	S	2 × 10 [illegible]	1 × 10 [illegible]	6 × 10 [illegible]	5 × 10 [illegible]
		I	4 × 10 [illegible]	8 × 10 [illegible]	1 × 10 [illegible]	3 × 10 [illegible]
	Pu 241	S	9 × 10 [illegible]	7 × 10 [illegible]	3 × 10 [illegible]	7 × 10 [illegible]
		I	4 × 10 [illegible]	4 × 10 [illegible]	1 × 10 [illegible]	[illegible]
	Pu 242	S	2 × 10 [illegible]	1 × 10 [illegible]	6 × 10 [illegible]	3 × 10 [illegible]
		I	4 × 10 [illegible]	9 × 10 [illegible]	1 × 10 [illegible]	3 × 10 [illegible]
	Pu 243	S	2 × 10 [illegible]	1 × 10 [illegible]	6 × 10 [illegible]	3 × 10 [illegible]
		I	2 × 10 [illegible]	1 × 10 [illegible]	8 × 10 [illegible]	3 × 10 [illegible]

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PART 20 - STANDARDS FOR PROTECTION AGAINST RADIATION

APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See notes at end of appendix)

Element (atomic number)	Isotope		Table I Column 1	Table I Column 2	Table II Column 1	Table II Column 2
			Air (μc/ml)	Water (μc/ml)	Air (μc/ml)	Water (μc/ml)
Plutonium (94)	Pu 244	S	2×10^{-12}	1×10^{-4}	6×10^{-14}	4×10^{-6}
		I	3×10^{-11}	3×10^{-4}	1×10^{-12}	1×10^{-5}
Polonium (84)	Po 210	S	5×10^{-10}	2×10^{-5}	2×10^{-11}	7×10^{-7}
		I	2×10^{-10}	8×10^{-4}	7×10^{-12}	3×10^{-5}
Potassium (19)	K 42	S	2×10^{-6}	9×10^{-4}	7×10^{-8}	3×10^{-5}
		I	1×10^{-7}	6×10^{-4}	4×10^{-9}	2×10^{-5}
Praseodymium (59)	Pr 142	S	2×10^{-7}	9×10^{-4}	7×10^{-9}	3×10^{-5}
		I	2×10^{-7}	9×10^{-4}	5×10^{-9}	3×10^{-5}
	Pr 143	S	3×10^{-7}	1×10^{-3}	1×10^{-8}	5×10^{-5}
		I	2×10^{-7}	1×10^{-3}	6×10^{-9}	5×10^{-5}
Promethium (61)	Pm 147	S	6×10^{-8}	6×10^{-3}	2×10^{-9}	2×10^{-4}
		I	1×10^{-7}	6×10^{-3}	3×10^{-9}	2×10^{-4}
	Pm 149	S	3×10^{-7}	1×10^{-3}	1×10^{-8}	4×10^{-5}
		I	2×10^{-7}	1×10^{-3}	8×10^{-9}	4×10^{-5}
Protoactinium (91)	Pa 230	S	2×10^{-9}	7×10^{-3}	6×10^{-11}	2×10^{-4}
		I	8×10^{-10}	7×10^{-3}	3×10^{-11}	2×10^{-4}
	Pa 231	S	1×10^{-12}	3×10^{-5}	4×10^{-14}	9×10^{-7}
		I	1×10^{-10}	8×10^{-4}	4×10^{-12}	2×10^{-5}
	Pa 233	S	6×10^{-7}	4×10^{-3}	2×10^{-8}	1×10^{-4}
		I	2×10^{-7}	3×10^{-3}	6×10^{-9}	1×10^{-4}
Radium (88)	Ra 223	S	2×10^{-9}	2×10^{-5}	6×10^{-11}	7×10^{-7}
		I	2×10^{-10}	1×10^{-4}	8×10^{-12}	4×10^{-6}
	Ra 224	S	5×10^{-9}	7×10^{-5}	2×10^{-10}	2×10^{-6}
		I	7×10^{-10}	2×10^{-4}	2×10^{-11}	5×10^{-6}
	Ra 226	S	3×10^{-11}	4×10^{-7}	3×10^{-12}	3×10^{-8}
		I	5×10^{-11}	9×10^{-4}	2×10^{-12}	3×10^{-5}
	Ra 228	S	7×10^{-11}	8×10^{-7}	2×10^{-12}	3×10^{-8}
		I	4×10^{-11}	7×10^{-4}	1×10^{-12}	3×10^{-5}
Radon (86)	Rn 220	S	3×10^{-7}		1×10^{-8}	
	Rn 222	S	1×10^{-7}		3×10^{-9}	
Rhenium (75)	Re 183	S	3×10^{-6}	2×10^{-2}	9×10^{-8}	6×10^{-4}
		I	2×10^{-7}	8×10^{-3}	5×10^{-9}	3×10^{-4}
	Re 186	S	6×10^{-7}	3×10^{-3}	2×10^{-8}	9×10^{-5}
		I	2×10^{-7}	1×10^{-3}	8×10^{-9}	5×10^{-5}
	Re 187	S	9×10^{-6}	7×10^{-2}	3×10^{-7}	3×10^{-3}
		I	5×10^{-7}	4×10^{-2}	2×10^{-8}	2×10^{-3}
	Re 188	S	4×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
		I	2×10^{-7}	9×10^{-4}	6×10^{-9}	3×10^{-5}
Rhodium (45)	Rh 103m	S	8×10^{-5}	4×10^{-1}	3×10^{-6}	1×10^{-2}
		I	6×10^{-5}	3×10^{-1}	2×10^{-6}	1×10^{-2}
	Rh 105	S	8×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
		I	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
Rubidium (37)	Rb 86	S	3×10^{-7}	2×10^{-3}	1×10^{-8}	7×10^{-5}
		I	7×10^{-8}	7×10^{-4}	2×10^{-9}	2×10^{-5}
	Rb 87	S	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	7×10^{-8}	5×10^{-3}	2×10^{-9}	2×10^{-4}

APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See notes at end of appendix)

Element (atomic number)	Isotope¹		Table I Column 1	Table I Column 2	Table II Column 1	Table II Column 2
			Air (μc/ml)	Water (μc/ml)	Air (μc/ml)	Water (μc/ml)
Ruthenium (44)	Ru 97	S	2×10^{-6}	1×10^{-2}	8×10^{-8}	4×10^{-4}
		I	2×10^{-7}	1×10^{-2}	6×10^{-9}	3×10^{-4}
	Ru 103	S	5×10^{-7}	2×10^{-3}	2×10^{-8}	8×10^{-5}
		I	8×10^{-8}	2×10^{-3}	3×10^{-9}	8×10^{-5}
	Ru 105	S	7×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
	Ru 106	S	8×10^{-8}	4×10^{-4}	3×10^{-9}	1×10^{-5}
		I	6×10^{-9}	3×10^{-4}	2×10^{-10}	1×10^{-5}
Samarium (62)	Sm 147	S	7×10^{-11}	2×10^{-3}	2×10^{-12}	6×10^{-5}
		I	3×10^{-10}	2×10^{-3}	9×10^{-12}	7×10^{-5}
	Sm 151	S	6×10^{-8}	1×10^{-2}	2×10^{-9}	4×10^{-4}
		I	1×10^{-7}	1×10^{-2}	5×10^{-9}	4×10^{-4}
	Sm 153	S	5×10^{-7}	2×10^{-3}	2×10^{-8}	8×10^{-5}
		I	4×10^{-7}	2×10^{-3}	1×10^{-8}	8×10^{-5}
Scandium (21)	Sc 46	S	2×10^{-7}	1×10^{-3}	8×10^{-9}	4×10^{-5}
		I	2×10^{-8}	1×10^{-3}	8×10^{-10}	4×10^{-5}
	Sc 47	S	6×10^{-7}	3×10^{-3}	2×10^{-8}	9×10^{-5}
		I	5×10^{-7}	3×10^{-3}	2×10^{-8}	9×10^{-5}
	Sc 48	S	2×10^{-7}	8×10^{-4}	6×10^{-9}	3×10^{-5}
		I	1×10^{-7}	8×10^{-4}	5×10^{-9}	3×10^{-5}
Selenium (34)	Se 75	S	1×10^{-6}	9×10^{-3}	4×10^{-8}	3×10^{-4}
		I	1×10^{-7}	8×10^{-3}	4×10^{-9}	3×10^{-4}
Silicon (14)	Si 31	S	6×10^{-6}	3×10^{-2}	2×10^{-7}	9×10^{-4}
		I	1×10^{-6}	6×10^{-3}	3×10^{-8}	2×10^{-4}
Silver (47)	Ag 105	S	6×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	8×10^{-8}	3×10^{-3}	3×10^{-9}	1×10^{-4}
	Ag 110m	S	2×10^{-7}	9×10^{-4}	7×10^{-9}	3×10^{-5}
		I	1×10^{-8}	9×10^{-4}	3×10^{-10}	3×10^{-5}
	Ag 111	S	3×10^{-7}	1×10^{-3}	1×10^{-8}	4×10^{-5}
		I	2×10^{-7}	1×10^{-3}	8×10^{-9}	4×10^{-5}
Sodium (11)	Na 22	S	2×10^{-7}	1×10^{-3}	6×10^{-9}	4×10^{-5}
		I	9×10^{-9}	9×10^{-4}	3×10^{-10}	3×10^{-5}
	Na 24	S	1×10^{-6}	6×10^{-3}	4×10^{-8}	2×10^{-4}
		I	1×10^{-7}	8×10^{-4}	5×10^{-9}	3×10^{-5}
Strontium (38)	Sr 85m	S	4×10^{-5}	2×10^{-1}	1×10^{-6}	7×10^{-3}
		I	3×10^{-5}	2×10^{-1}	1×10^{-6}	7×10^{-3}
	Sr 85	S	2×10^{-7}	3×10^{-3}	8×10^{-9}	1×10^{-4}
		I	1×10^{-7}	5×10^{-3}	4×10^{-9}	2×10^{-4}
	Sr 89	S	3×10^{-8}	3×10^{-4}	3×10^{-10}	3×10^{-6}
		I	4×10^{-8}	8×10^{-4}	1×10^{-9}	3×10^{-5}
	Sr 90	S	1×10^{-9}	1×10^{-5}	3×10^{-11}	3×10^{-7}
		I	5×10^{-9}	1×10^{-3}	2×10^{-10}	4×10^{-5}
	Sr 91	S	4×10^{-7}	2×10^{-3}	2×10^{-8}	7×10^{-5}
		I	3×10^{-7}	1×10^{-3}	9×10^{-9}	5×10^{-5}
	Sr 92	S	4×10^{-7}	2×10^{-3}	2×10^{-8}	7×10^{-5}
		I	3×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
Sulfur (16)	S 35	S	3×10^{-7}	2×10^{-3}	9×10^{-9}	6×10^{-5}
		I	3×10^{-7}	8×10^{-3}	9×10^{-9}	3×10^{-4}
Tantalum (73)	Ta 182	S	4×10^{-8}	1×10^{-3}	1×10^{-9}	4×10^{-5}
		I	2×10^{-8}	1×10^{-3}	7×10^{-10}	4×10^{-5}

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APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See notes at end of appendix)

Element (atomic number)	Isotope[1]		Table I Column 1 Air (μc/ml)	Table I Column 2 Water (μc/ml)	Table II Column 1 Air (μc/ml)	Table II Column 2 Water (μc/ml)
Technetium (43)	Tc 96m	S	8 × 10[illegible]	4 × 10[illegible]	3 × 10[illegible]	1 × 10[illegible]
		I	3 × 10[illegible]	3 × 10[illegible]	1 × 10[illegible]	1 × 10[illegible]
	Tc 96	S	6 × 10[illegible]	3 × 10[illegible]	2 × 10[illegible]	1 × 10[illegible]
		I	2 × 10[illegible]	1 × 10[illegible]	8 × 10[illegible]	5 × 10[illegible]
	Tc 97m	S	2 × 10[illegible]	1 × 10[illegible]	8 × 10[illegible]	4 × 10[illegible]
		I	2 × 10[illegible]	5 × 10[illegible]	5 × 10[illegible]	2 × 10[illegible]
	Tc 97	S	1 × 10[illegible]	5 × 10[illegible]	4 × 10[illegible]	2 × 10[illegible]
		I	3 × 10[illegible]	2 × 10[illegible]	1 × 10[illegible]	8 × 10[illegible]
	Tc 99m	S	4 × 10[illegible]	2 × 10[illegible]	1 × 10[illegible]	6 × 10[illegible]
		I	1 × 10[illegible]	8 × 10[illegible]	5 × 10[illegible]	3 × 10[illegible]
	Tc 99	S	2 × 10[illegible]	1 × 10[illegible]	7 × 10[illegible]	3 × 10[illegible]
		I	6 × 10[illegible]	5 × 10[illegible]	2 × 10[illegible]	2 × 10[illegible]
Tellurium (52)	Te 125m	S	4 × 10[illegible]	5 × 10[illegible]	1 × 10[illegible]	2 × 10[illegible]
		I	1 × 10[illegible]	1 × 10[illegible]	4 × 10[illegible]	1 × 10[illegible]
	Te 127m	S	1 × 10[illegible]	2 × 10[illegible]	5 × 10[illegible]	6 × 10[illegible]
		I	4 × 10[illegible]	2 × 10[illegible]	1 × 10[illegible]	5 × 10[illegible]
	Te 127	S	2 × 10[illegible]	8 × 10[illegible]	6 × 10[illegible]	3 × 10[illegible]
		I	9 × 10[illegible]	2 × 10[illegible]	3 × 10[illegible]	3 × 10[illegible]
	Te 129m	S	8 × 10[illegible]	1 × 10[illegible]	3 × 10[illegible]	3 × 10[illegible]
		I	3 × 10[illegible]	6 × 10[illegible]	1 × 10[illegible]	2 × 10[illegible]
	Te 129	S	5 × 10[illegible]	2 × 10[illegible]	2 × 10[illegible]	8 × 10[illegible]
		I	4 × 10[illegible]	2 × 10[illegible]	1 × 10[illegible]	8 × 10[illegible]
	Te 131m	S	4 × 10[illegible]	2 × 10[illegible]	1 × 10[illegible]	6 × 10[illegible]
		I	2 × 10[illegible]	1 × 10[illegible]	6 × 10[illegible]	4 × 10[illegible]
	Te 132	S	2 × 10[illegible]	9 × 10[illegible]	7 × 10[illegible]	3 × 10[illegible]
		I	1 × 10[illegible]	6 × 10[illegible]	4 × 10[illegible]	2 × 10[illegible]
Terbium (65)	Tb 160	S	1 × 10[illegible]	1 × 10[illegible]	3 × 10[illegible]	4 × 10[illegible]
		I	3 × 10[illegible]	1 × 10[illegible]	1 × 10[illegible]	4 × 10[illegible]
Thallium (81)	Tl 200	S	3 × 10[illegible]	1 × 10[illegible]	9 × 10[illegible]	4 × 10[illegible]
		I	1 × 10[illegible]	7 × 10[illegible]	4 × 10[illegible]	2 × 10[illegible]
	Tl 201	S	2 × 10[illegible]	9 × 10[illegible]	7 × 10[illegible]	3 × 10[illegible]
		I	9 × 10[illegible]	5 × 10[illegible]	3 × 10[illegible]	2 × 10[illegible]
	Tl 202	S	8 × 10[illegible]	4 × 10[illegible]	3 × 10[illegible]	1 × 10[illegible]
		I	2 × 10[illegible]	2 × 10[illegible]	8 × 10[illegible]	7 × 10[illegible]
	Tl 204	S	6 × 10[illegible]	3 × 10[illegible]	2 × 10[illegible]	1 × 10[illegible]
		I	3 × 10[illegible]	2 × 10[illegible]	9 × 10[illegible]	6 × 10[illegible]
Thorium (90)	Th 228	S	9 × 10[illegible]	2 × 10[illegible]	3 × 10[illegible]	7 × 10[illegible]
		I	6 × 10[illegible]	4 × 10[illegible]	2 × 10[illegible]	1 × 10[illegible]
	Th 230	S	2 × 10[illegible]	5 × 10[illegible]	8 × 10[illegible]	2 × 10[illegible]
		I	1 × 10[illegible]	9 × 10[illegible]	3 × 10[illegible]	3 × 10[illegible]
	Th 232	S	3 × 10[illegible]	5 × 10[illegible]	1 × 10[illegible]	2 × 10[illegible]
		I	3 × 10[illegible]	1 × 10[illegible]	1 × 10[illegible]	4 × 10[illegible]
	Th natural	S	3 × 10[illegible]	3 × 10[illegible]	1 × 10[illegible]	1 × 10[illegible]
		I	3 × 10[illegible]	3 × 10[illegible]	1 × 10[illegible]	1 × 10[illegible]
	Th 234	S	6 × 10[illegible]	5 × 10[illegible]	2 × 10[illegible]	2 × 10[illegible]
		I	[illegible]	5 × 10[illegible]	1 × 10[illegible]	2 × 10[illegible]
Thulium (69)	Tm 170	S	4 × 10[illegible]	1 × 10[illegible]	1 × 10[illegible]	5 × 10[illegible]
		I	3 × 10[illegible]	1 × 10[illegible]	1 × 10[illegible]	5 × 10[illegible]
	Tm 171	S	1 × 10[illegible]	1 × 10[illegible]	4 × 10[illegible]	5 × 10[illegible]
		I	2 × 10[illegible]	1 × 10[illegible]	8 × 10[illegible]	5 × 10[illegible]

APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See notes at end of appendix)

Element (atomic number)	Isotope[1]		Table I Column 1 Air (μc/ml)	Table I Column 2 Water (μc/ml)	Table II Column 1 Air (μc/ml)	Table II Column 2 Water (μc/ml)
Tin (50)	Sn 113	S	4 × 10[illegible]	2 × 10[illegible]	1 × 10[illegible]	9 × 10[illegible]
		I	5 × 10[illegible]	2 × 10[illegible]	2 × 10[illegible]	8 × 10[illegible]
	Sn 125	S	1 × 10[illegible]	5 × 10[illegible]	4 × 10[illegible]	2 × 10[illegible]
		I	8 × 10[illegible]	5 × 10[illegible]	3 × 10[illegible]	2 × 10[illegible]
Tungsten (Wolfram) (74)	W 181	S	2 × 10[illegible]	1 × 10[illegible]	8 × 10[illegible]	4 × 10[illegible]
		I	1 × 10[illegible]	1 × 10[illegible]	4 × 10[illegible]	3 × 10[illegible]
	W 185	S	8 × 10[illegible]	4 × 10[illegible]	3 × 10[illegible]	1 × 10[illegible]
		I	1 × 10[illegible]	3 × 10[illegible]	4 × 10[illegible]	1 × 10[illegible]
	W 187	S	4 × 10[illegible]	2 × 10[illegible]	2 × 10[illegible]	7 × 10[illegible]
		I	3 × 10[illegible]	2 × 10[illegible]	1 × 10[illegible]	6 × 10[illegible]
Uranium (92)	U 230	S	3 × 10[illegible]	1 × 10[illegible]	1 × 10[illegible]	5 × 10[illegible]
		I	1 × 10[illegible]	1 × 10[illegible]	4 × 10[illegible]	5 × 10[illegible]
	U 232	S	1 × 10[illegible]	8 × 10[illegible]	3 × 10[illegible]	3 × 10[illegible]
		I	3 × 10[illegible]	8 × 10[illegible]	9 × 10[illegible]	3 × 10[illegible]
	U 233	S	5 × 10[illegible]	9 × 10[illegible]	2 × 10[illegible]	3 × 10[illegible]
		I	1 × 10[illegible]	9 × 10[illegible]	4 × 10[illegible]	3 × 10[illegible]
	U 234	S	6 × 10[illegible]	9 × 10[illegible]	2 × 10[illegible]	3 × 10[illegible]
		I	1 × 10[illegible]	9 × 10[illegible]	4 × 10[illegible]	3 × 10[illegible]
	U 235	S	5 × 10[illegible]	8 × 10[illegible]	2 × 10[illegible]	3 × 10[illegible]
		I	1 × 10[illegible]	8 × 10[illegible]	4 × 10[illegible]	3 × 10[illegible]
	U 236	S	6 × 10[illegible]	1 × 10[illegible]	2 × 10[illegible]	3 × 10[illegible]
		I	1 × 10[illegible]	1 × 10[illegible]	4 × 10[illegible]	3 × 10[illegible]
	U 238	S	7 × 10[illegible]	1 × 10[illegible]	3 × 10[illegible]	4 × 10[illegible]
		I	1 × 10[illegible]	1 × 10[illegible]	5 × 10[illegible]	4 × 10[illegible]
	U 240	S	2 × 10[illegible]	1 × 10[illegible]	8 × 10[illegible]	3 × 10[illegible]
		I	2 × 10[illegible]	1 × 10[illegible]	6 × 10[illegible]	3 × 10[illegible]
	U natural	S	7 × 10[illegible]	5 × 10[illegible]	3 × 10[illegible]	2 × 10[illegible]
		I	6 × 10[illegible]	5 × 10[illegible]	2 × 10[illegible]	2 × 10[illegible]
Vanadium (23)	V 48	S	2 × 10[illegible]	9 × 10[illegible]	6 × 10[illegible]	3 × 10[illegible]
		I	6 × 10[illegible]	8 × 10[illegible]	2 × 10[illegible]	3 × 10[illegible]
Xenon (54)	Xe 131m	Sub	2 × 10[illegible]		4 × 10[illegible]	
	Xe 133	Sub	1 × 10[illegible]		3 × 10[illegible]	
	Xe 133m	Sub	1 × 10[illegible]		3 × 10[illegible]	
	Xe 135	Sub	4 × 10[illegible]		1 × 10[illegible]	
Ytterbium (70)	Yb 175	S	7 × 10[illegible]	3 × 10[illegible]	2 × 10[illegible]	1 × 10[illegible]
		I	6 × 10[illegible]	3 × 10[illegible]	2 × 10[illegible]	1 × 10[illegible]
Yttrium (39)	Y 90	S	1 × 10[illegible]	6 × 10[illegible]	4 × 10[illegible]	2 × 10[illegible]
		I	1 × 10[illegible]	6 × 10[illegible]	3 × 10[illegible]	2 × 10[illegible]
	Y 91m	S	2 × 10[illegible]	1 × 10[illegible]	8 × 10[illegible]	3 × 10[illegible]
		I	2 × 10[illegible]	1 × 10[illegible]	6 × 10[illegible]	3 × 10[illegible]
	Y 91	S	4 × 10[illegible]	8 × 10[illegible]	1 × 10[illegible]	3 × 10[illegible]
		I	3 × 10[illegible]	8 × 10[illegible]	1 × 10[illegible]	3 × 10[illegible]
	Y 92	S	4 × 10[illegible]	2 × 10[illegible]	1 × 10[illegible]	6 × 10[illegible]
		I	3 × 10[illegible]	2 × 10[illegible]	1 × 10[illegible]	6 × 10[illegible]
	Y 93	S	2 × 10[illegible]	8 × 10[illegible]	6 × 10[illegible]	3 × 10[illegible]
		I	1 × 10[illegible]	8 × 10[illegible]	5 × 10[illegible]	3 × 10[illegible]
Zinc (30)	Zn 65	S	1 × 10[illegible]	3 × 10[illegible]	4 × 10[illegible]	1 × 10[illegible]
		I	6 × 10[illegible]	5 × 10[illegible]	2 × 10[illegible]	2 × 10[illegible]
	Zn 69m	S	4 × 10[illegible]	2 × 10[illegible]	1 × 10[illegible]	7 × 10[illegible]
		I	3 × 10[illegible]	2 × 10[illegible]	1 × 10[illegible]	6 × 10[illegible]

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APPENDIX B

Concentrations in Air and Water Above Natural Background—Continued

(See notes at end of appendix)

Element (atomic number)	Isotope [1]		Table I Column 1 Air (μc/ml)	Table I Column 2 Water (μc/ml)	Table II Column 1 Air (μc/ml)	Table II Column 2 Water (μc/ml)
Zinc (30)	Zn 69	S	7×10^{-6}	5×10^{-2}	2×10^{-7}	2×10^{-3}
		I	9×10^{-6}	5×10^{-2}	3×10^{-7}	2×10^{-3}
Zirconium (40)	Zr 93	S	1×10^{-7}	2×10^{-2}	4×10^{-9}	8×10^{-4}
		I	3×10^{-7}	2×10^{-2}	1×10^{-8}	8×10^{-4}
	Zr 95	S	1×10^{-8}	2×10^{-3}	4×10^{-10}	6×10^{-5}
		I	3×10^{-8}	2×10^{-3}	1×10^{-9}	6×10^{-5}
	Zr 97	S	1×10^{-7}	5×10^{-4}	4×10^{-9}	2×10^{-5}
		I	9×10^{-8}	5×10^{-4}	3×10^{-9}	2×10^{-5}
Any single radionuclide not listed above with decay mode other than alpha emission or spontaneous fission and with radioactive half life less than 2 hours		Sub	1×10^{-6}		3×10^{-8}	
Any single radionuclide not listed above with decay mode other than alpha emission or spontaneous fission and with radioactive half life greater than 2 hours			3×10^{-9}	9×10^{-5}	1×10^{-10}	3×10^{-6}
Any single radionuclide not listed above, which decays by alpha emission or spontaneous fission			6×10^{-13}	4×10^{-7}	2×10^{-14}	3×10^{-8}

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a Soluble (S); Insoluble (I).

b "Sub" means that values given are for submersion in a semispherical infinite cloud of airborne material.

NOTE: In any case where there is a mixture in air or water of more than one radionuclide, the limiting values for purposes of this Appendix should be determined as follows:

1. If the identity and concentration of each radionuclide in the mixture are known, the limiting values should be derived as follows: Determine, for each radionuclide in the mixture, the ratio between the quantity present in the mixture and the limit otherwise established in Appendix B for the specific radionuclide when not in a mixture. The sum of such ratios for all the radionuclides in the mixture may not exceed "1" (i.e., "unity").

EXAMPLE: If radionuclides A, B, and C are present in concentrations C_A, C_B, and C_C, and if the applicable MPC's are MPC_A, and MPC_B, and MPC_C respectively, then the concentrations shall be limited so that the following relationship exists:

$$\frac{C_A}{MPC_A}+\frac{C_B}{MPC_B}+\frac{C_C}{MPC_C}\le 1$$

2. If either the identity or the concentration of any radionuclide in the mixture is not known, the limiting values for purposes of Appendix B shall be:

a. For purposes of Table I, Col. 1—6×10^{-13}
b. For purposes of Table I, Col. 2—4×10^{-7}
c. For purposes of Table II, Col. 1—2×10^{-14}
d. For purposes of Table II, Col. 2—3×10^{-8}

3. If any of the conditions specified below are met, the corresponding values specified below may be used in lieu of those specified in paragraph 2 above.

a. If the identity of each radionuclide in the mixture is known but the concentration of one or more of the radionuclides in the mixture is not known, the concentration limit for the mixture is the limit specified in Appendix "B" for the radionuclide in the mixture having the lowest concentration limit; or

b. If the identity of each radionuclide in the mixture is not known, but it is known that certain radionuclides specified in Appendix "B" are not present in the mixture, the concentration limit for the mixture is the lowest concentration limit specified in Appendix "B" for any radionuclide which is not known to be absent from the mixture; or

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c. Element (atomic number) and isotope	Table I Column 1 Air (μc/ml)	Table I Column 2 Water (μc/ml)	Table II Column 1 Air (μc/ml)	Table II Column 2 Water (μc/ml)
If it is known that Sr 90, I 125, I 126, I 129, I 131, (I 133, Table II only), Pb 210, Po 210, At 211, Ra 223, Ra 224, Ra 226, Ac 227, Ra 228, Th 230, Pa 231, Th 232, Th-nat, Cm 248, Cf 254, and Fm 256 are not present	....	9×10^{-5}	....	3×10^{-6}
If it is known that Sr 90, I 125, I 126, I 129, (I 131, I 133, Table II only), Pb 210, Po 210, Ra 223, Ra 226, Ra 228, Pa 231, Th-nat, Cm 248, Cf 254, and Fm 256 are not present	....	6×10^{-5}	....	2×10^{-6}
If it is known that Sr 90, I 129, (I 125, I 126, I 131, Table II only), Pb 210, Ra 226, Ra 228, Cm 248, and Cf 254 are not present	....	2×10^{-5}	....	6×10^{-7}
If it is known that (I 129, Table II only), Ra 226, and Ra 228 are not present	....	3×10^{-5}	....	1×10^{-7}
If it is known that alpha-emitters and Sr 90, I 129, Pb 210, Ac 227, Ra 228, Pa 230, Pu 241, and Bk 249 are not present	3×10^{-9}	....	1×10^{-10}	....
If it is known that alpha-emitters and Pb 210, Ac 227, Ra 228, and Pu 241 are not present	3×10^{-10}	....	1×10^{-11}	....
If it is known that alpha-emitters and Ac 227 are not present	3×10^{-11}	....	1×10^{-12}	....
If it is known that Ac 227, Th 230, Pa 231, Pu 238, Pu 239, Pu 240, Pu 242, Cm 248 and Cf 249 are not present	3×10^{-12}	....	1×10^{-13}	....

4. If a mixture of radionuclides consists of uranium and its daughters in ore dust prior to chemical processing of the uranium ore, the values specified below may be used in lieu of those determined in accordance with paragraph 1 above or those specified in paragraphs 2 and 3 above.

a. For purposes of Table I, Col. 1, 1×10^{-10} μc/ml gross alpha activity; or 2.5×10^{-11} μc/ml natural uranium; or 75 micrograms per cubic meter of air natural uranium.

b. For purposes of Table II, Col. 1, 3×10^{-12} μc/ml gross alpha activity; or 8×10^{-13} μc/ml natural uranium; or 3 micrograms per cubic meter of air natural uranium.

5. For purposes of this note, a radionuclide may be considered as not present in a mixture if (a) the ratio of the concentration of that radionuclide in the mixture (C_A) to the concentration limit for that radionuclide specified in Table II of Appendix "B" (MPC_A) does not exceed $\frac{1}{10}$,

$$\left(\text{i.e. } \frac{C_A}{MPC_A}\le\frac{1}{10}\right)$$

and (b) the sum of such ratios for all the radionuclides considered as not present in the mixture does not exceed $\frac{1}{4}$, i.e.

$$\left(\frac{C_A}{MPC_A}+\frac{C_B}{MPC_B}+\cdots\le\frac{1}{4}\right).$$

*Revised 30 FR 15801

**ERRATUM: This line should read: "210, Ac 227, Ra 228, Pa 230, Pu 241, and Bk 249 are not"

Appendix B from 25 FR 10914 except as otherwise noted.

APPENDIX C

Material	Micro-curies
[illegible]	[illegible]

[illegible] or any of [illegible] listed [illegible]

NOTE: For purposes of §§ 20.303 and 20.304 where there is involved a combination of isotopes in known amounts, the limit for the combination should be derived as follows: Determine, for each isotope in the combination, the ratio between the quantity present in the combination and the limit otherwise established for the specific isotope when not in combination. The sum of such ratios for all the isotopes in the combination may not exceed "1" (i.e., "unity").

EXAMPLE: For purposes of § 20.304, if a particular batch contains 20,000 µc of Au^{198} and 25,000 µc of C^{14}, it may also include not more than 3,000 µc of I^{131}. This limit was determined as follows:

$$\frac{20{,}000\ \mu c\ Au^{198}}{100{,}000\ \mu c}+\frac{25{,}000\ \mu c\ C^{14}}{50{,}000\ \mu c}+\frac{3{,}000\ \mu c\ I^{131}}{10{,}000\ \mu c}=1$$

The denominator in each of the above ratios was obtained by multiplying the figure in the table by 1,000 as provided in § 20.304.

APPENDIX D

UNITED STATES ATOMIC ENERGY COMMISSION COMPLIANCE OFFICES

Region	Address	Telephone: Daytime	Telephone: Nights and holidays
I Connecticut, Delaware, District of Columbia, Maine, Maryland, Massachusetts, New Hampshire, New Jersey, New York, Pennsylvania, Rhode Island, and Vermont.	Region I, Division of Compliance, USAEC, [illegible] Newark, N.J. [illegible]	201-645-3940	201-645-3940
II Alabama, Arkansas, Florida, Georgia, Kentucky, Louisiana, [illegible] Carolina, Puerto Rico, [illegible] Tennessee, Virginia, Virgin Islands, and West Virginia.	Region II, Division of Compliance, USAEC, [illegible] St., NE., [illegible]	404-526-4557	404-526-4557
III Illinois, Indiana, Iowa, Michigan, Minnesota, Missouri, Ohio, and Wisconsin.	Region III, Division of Compliance, USAEC, [illegible] Road, [illegible]	312-858-2660	312-858-2660
IV Colorado, Idaho, Kansas, Montana, Nebraska, New Mexico, North Dakota, Oklahoma, South Dakota, Texas, Utah, and Wyoming.	Region IV, Division of Compliance, USAEC, [illegible]	303-297-4211	303-297-4211
V Alaska, Arizona, California, Hawaii, Nevada, Oregon, Washington, and U.S. territories and possessions in the Pacific.	Region V, Division of Compliance, USAEC, 2111 Bancroft Way, Berkeley, Calif. 94704.	415-841-5121, Ext. [illegible]	415-841-4204

[illegible]

NOTE: The reporting and recordkeeping requirements contained in this part have been approved by the Bureau of the Budget in accordance with the Federal Reports Act of 1942.

Appendix C from 25 FR 10914.

Appendix D from 27 FR 10826.

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BASIC PHYSICS

2.0 Introduction

In order to understand and appreciate nuclear physics and reactor kinetics, a good foundation in the principles of basic physics must be established. The easiest approach to this undertaking is to first analyze the natural phenomena which surround us in our everyday life, and then explain these phenomena by means of the language of mathematics. Having established the basic principles of physics, one can then feel more confident to advance into more complex physics.

2.1 Units of Measurement

Three fundamental quantities are required to set up a system of units. The kind of units set up depends upon the choice. Most commonly, the fundamental quantities are length, mass, and time. Another system of fundamental quantities utilizes length, force and time.

Mass is independent of the place at which observation is made, and hence a system of units based on length, mass, and time is called an "absolute" system. In the alternative choice, length, force and time, the force commonly chosen is a gravitational force, or weight, and hence the system of units is called a "gravitational" system.

In the Cgs system, the centimeter, gram, and second are the fundamental units. In the British gravitational system the fundamental unit, the pound (lb.), is 1/2.2046 the force with which the earth pulls on a standard kilogram. In this system we define a unit of mass called the slug. The slug is the mass to which a force of one pound will give an acceleration of one foot per second per second. A mass of 1 slug weighs approximately 32 lb. (at the earths surface).

It is unfortunate that in the different sets of units the same word is used to designate a unit of mass in one set, but a unit of force in another. Therefore in Basic Physics section of this book we shall employ only the British gravitational system and the Cgs absolute system. Whenever the term gram is used, it will refer to mass. Whenever the term pound is used, it will refer to force.

2.2 Mass (Units of Grams, Kilograms)

Mass is defined as that property of a body which resists a change in motion of that body. This simply means that a body which has mass will resist being moved by a force. Mass is independent of the place at which the observation is made.

Table I

System of Units

Name of System	Unit of Mass	Unit of Force	Unit of Acceleration
MKS	Kilogram	Newton	m/sec^2
Cgs Absolute	Gram	Dyne	cm/sec^2
Cgs Gravitational	No name $m = w/g$	Gram	cm/sec^2
British Absolute	Pound	Poundal	$Feet/sec^2$
British Gravitational	Slug	Pound	$Feet/sec^2$

2.3 Length (Meter, Yard)

To specify a distance we must use some unit of length. The unit commonly employed for scientific use and accepted as an international standard is the meter. The meter is defined as the distance between two lines on a certain bar of platinum-iridium when the temperature of the bar is that of melting ice (0°C). One hundredth of the meter is called the centimeter (1 cm = 0.01 m). Other decimal fractions of the meter are the decimeter (1 dm = 0.1 m) and the millimeter (1 mm = 0.001 m). For large distances the kilometer (1 km = 1,000 m) is employed.

The British unit of length is the yard and has its legal definition as the distance between two lines on a bronze bar at a temperature of 62°F. Other common British units of length are the mile (1 mi. = 1760 yards), the foot (1 ft. = 1/3 yard), and the inch (1 in. = 1/36 yard).

In the United States the yard is legally defined in terms of the meter: 1 yd. = 3600/3937 m. This leads to the simple approximate relation: 1 in. = 2.54 cm.

2.4 Time (Second)

The fundamental unit of time is the mean solar second. The mean solar second has been defined as 1/86,400 (86,400 = 24 x 60 x 60) of the mean solar day, which is the average, throughout the year, of the time between successive transits of the sun across the meridian at any place. Thus, the time it takes the earth to turn once on its axis, with respect to the sun, serves as the basis for the unit of time.

2.5 Weight (Unit of Pounds)

As a result of universal gravitation everything on or near the surface of the earth is attracted toward the earth with a force that we call weight. The force with which the earth pulls on a mass of one pound under standard conditions is called the weight of one pound or the "pound of force." Therefore, the weight of a body is a force.

$w = mg$

w = weight of a body in pounds

where;

m = mass of a body in slugs

g = acceleration of a freely falling body equal to 32 ft/sec^2

2.6 Newton's Laws of Motion

1. A body will maintain its state of rest or of uniform motion along a straight line unless compelled by some unbalanced force to change that state.

2. An unbalanced force F acting on a body produces in it an acceleration which is in the direction of the force and directly proportional to the force, and inversely proportional to the mass of the body.

3. To every action, or force, there is an equal and opposite reaction, or force.

2.7 Force (Dyne, Pound)

Whenever an unbalanced force acts on a body, it produces an acceleration in the direction of the force, an acceleration that is directly proportional to the force and inversely proportional to the mass of the body. This is a law known as "Newton's Second Law of Motion."

If successively greater forces are applied to a given body, correspondingly greater accelerations are produced. A given force applied in turn to bodies of successively greater mass will produce successively smaller accelerations. The accelerations are inversely proportional to the masses of the bodies.

According to the second law, the following proportions may be written :

$$a \propto F \quad \text{and} \quad a \propto 1/m$$

These proportions may be combined as:

$$a \propto F/m$$

and written as an equation

$$F = kma \quad .$$

Any unit of force, mass, and acceleration can be used, provided the proper value is assigned to the proportionality constant k. In general, a different value of k would have to be assigned for each combination chosen. For simplicity, it is customary to use a system of units for which k has a value of 1.

2.8 Velocity (cm/sec, ft/sec)

The simplest kind of motion that an object can have is uniform motion in a straight line. In every second the body moves the same distance in the same direction as it does in each other second. An object moving in this manner is moving with constant velocity. Note that constant velocity implies not only constant speed but unchanging direction as well.

The speed of a moving body is the distance it moves per unit time. If the speed is uniform, the object moves the same distances in each second of time. Whether or not the speed is constant, the average speed is the distance the body moves divided by the time required for the motion.

$$\text{Average speed} = \frac{\text{distance}}{\text{time}}$$

$$\text{Avg. speed} = s/t$$

where; s = distance
t = time

Example: If, for example, an automobile travels 200 mi. in 4 hr. Its average speed is 50 mi/hr. In 6 hr. at the same average speed it would travel 300 mi.

The concept of speed does not involve the idea of direction. An object moving with constant speed may move in a straight line, in a circle, or in any one of an infinite variety of paths so long as the distance moved in any unit of time is the same as that moved in any other equal unit of time.

Constant velocity is a particular case of constant speed. Not only does the distance traveled in unit time remain the same, but the direction as well does not change. An automobile that travels for 1 hr. at a constant velocity of 20 mi/hr north reaches a place 20 mi. north of its first position. If, on the other hand, the automobile travels around a race track for 1 hr. at a constant speed of 20 mi/hr, it may traverse the same distance without getting anywhere. At one instant its velocity may be 20 mi/hr east, at another 20 mi/hr south, etc.

The statement, "An automobile is moving with a velocity of 20 mi/hr" is incorrect by virtue of incompleteness, since the direction of motion must be stated in order to specify a velocity. For this reason one should always use the word speed when one does not wish to state the direction of motion, or when the direction is changing.

2.9 Acceleration (cm/sec^2, ft/sec^2)

Objects seldom move with constant velocity. In almost all cases the velocity of an object is continually changing in magnitude or in direction or in both. Motion in which the velocity is changing is called accelerated motion. The rate at which the velocity changes is called the acceleration.

$$\text{Average acceleration} = \frac{\text{change in velocity}}{\text{Time}} = \frac{v_2 - v_1}{t}$$

Example: An automobile accelerates at a constant rate from 15 mi/hr to 45 mi/hr in 10 sec. while traveling in a straight line. What is the acceleration?

$$a = \frac{45\ \text{mi/hr} - 15\ \text{mi/hr}}{10\ \text{sec}} = \frac{30\ \text{mi/hr}}{10\ \text{sec.}} = 3\ \text{mi/hr/sec}$$

This indicates that the speed increases 3.0 mi/hr during each second. Since

$$30\ \text{mi/hr} = \frac{(30\ \text{mi/hr})(5280\ \text{ft/mile})}{3600\ \text{sec/hr}} = 44\ \text{ft/sec}$$

the acceleration can be written also as:

$$a = \frac{44\ \text{ft/sec}}{10\ \text{sec.}} = 4.4\ \text{ft/sec}^2$$

The velocity of a body may be changed by changing the speed or by changing the direction or by changing both speed and direction. If the direction of the acceleration is parallel to the direction of motion, only the speed changes, while if the acceleration is at right angles to the direction of motion, only the direction changes. Acceleration in any other direction produces changes in both speed and direction.

For the present we shall confine our attention to the simplest type of accelerated motion, called uniformly accelerated motion, in which the speed changes at a constant rate.

In uniformly accelerated motion, the acceleration is always the same and is parallel to the direction of the original motion. The acceleration in this case is equal to the rate of change of speed, since there is no change in direction. The acceleration is called positive if the speed is increased, negative if the speed is decreased.

The distance traveled during any time is given as s = $\overline{v}t$, but the average speed v must be obtained from the initial and final speeds v_1 and v_2. Since the speed changes at a uniform rate, the average speed v is equal to the average of the initial and final speeds or:

$$\overline{v} = \frac{v_1 + v_2}{2}$$

Example: How far does an automobile move in 10 sec. while it is increasing its speed uniformly from 15 to 45 mi/hr?

$\overline{v}$ = 1/2 (15 + 45) mi/hr = 30 mi/hr = 44 ft/sec

and s = 44 ft/sec x 10 sec = 440 ft.

2.10 Momentum ($\frac{\text{gm-cm}}{\text{sec}}$, $\frac{\text{slug-foot}}{\text{sec}}$)

Momentum is the quantity of motion in a body and is expressed as the product of the mass and velocity of a body.

Momentum = mass x velocity

or P = mv

Example: What is the momentum of a 100 lb. shell as it leaves a gun with a speed of 1200 ft/sec?

$m = w/g = \frac{100 \text{ lb.}}{32 \text{ ft/sec}^2} = 3.1$ slugs

P = mv = (3.1 slugs)(1200 ft/sec)

= 3700 slug ft/sec

Momentum is a vector quantity. Its direction is that of the velocity. To find the momentum of a system of two or more bodies, we must add their momenta vectorially. Consider two 4 lb. balls moving toward each other with equal speeds of 4 ft sec as shown below:

The mass of each is 4/32 slug. The momentum of A is:

$$P_a = (4/32 \text{ slug})(4 \text{ ft/sec}) = 0.5 \text{ slug-ft/sec}$$

to the right. The momentum of B is similarly 0.5 slug-ft/sec to the left. The vector sum of the two is zero, and hence the momentum of the system is zero.

2.11 Conservation of Momentum

According to Newton's first Law of motion, the velocity of a body does not change unless it is acted upon by a net force. Since the mass of the body is constant, we find that the momentum does not change unless an external force acts upon the body. The statement that the momentum of a body, or system of bodies, does not change except when a net external force is applied is known as the "law of conservation of momentum."

If an external force does act upon a system of bodies, the momentum of the system is changed. But, in the process, some other set of bodies must gain (or lose) an amount of momentum equal to that lost (or gained) by the system. In every process where velocity is changed the momentum lost by one body or set of bodies is equal to that gained by another body or set of bodies.

Momentum lost by A = momentum gained by B

Consider the two balls shown earlier. If they continue to move toward each other, they will collide and in the collision each will exert a force on the other. The momentum of the system of two balls is zero before the impact. If the balls are "elastic," they will rebound and the conservation law requires that the speeds of recoil shall be equal to each other (but not necessarily equal to the original speed) so that the momentum of the system remains zero. If the balls are perfectly inelastic, there will be no rebound, and the total momentum of the system will still remain zero.

Example 1) A 2.0 oz. bullet is fired from a 10-lb. gun with a speed of 2000 ft/sec. What is the speed of recoil of the gun?
NOTE: The momentum of the system of gun and bullet is zero before the gun is fired and therefore must be zero after the firing. Hence the momentum of the gun is equal in magnitude but opposite in direction to that of the bullet.

$$M_1 v_1 = M_2 v_2$$

Where: $m_1 = \dfrac{10\text{ lb. gun}}{32\text{ ft/sec}^2}$

$m_2 = \dfrac{2\text{ oz. bullet}}{32\text{ ft/sec}^2}$

v_1 = speed of recoil

v_2 = speed of bullet

$$v_1 = \frac{M_2 v_2}{m_1} = \frac{w_2/g}{w_1/g} v_2 = \frac{w_2}{w_1} v_2$$

$$v_1 = \left(\frac{2/16\text{ lb.}}{10\text{ lb.}}\right)(2000\text{ ft/sec}) = \underline{\underline{25\text{ ft/sec}}})$$

Example 2) Two inelastic masses of 16 and 4 grams move in opposite directions with velocities of 30 and 50 cm/sec respectively. Determine the resultant velocity v on collision if they are stuck together.

momentum before impact = momentum after

$$16\text{ gm} \times 30\text{ cm/sec} - 4\text{ gm} \times 50\text{ cm/sec} = (16 + 4)\text{ gm} \times v$$

$$480\ \frac{\text{gm cm}}{\text{sec}} - 200\text{ gm cm/sec} = (20\text{ gm})(v)$$

$$v = \frac{280\text{ gm cm/sec}}{20\text{ gm}} = \underline{\underline{14\text{ cm/sec}}}$$

2.12 Centripetal Force (Dynes, Pounds)

When an object is moving in a circular path at constant speed, its velocity is changing because of the continual change in direction. According to Newton's laws of motion, therefore, an unbalanced force is acting upon the object. This force, called the "centripetal force," is directed toward the center of the circular path. The magnitude of the centripetal force, therefore, is given by the relation:

centripetal force = mass x centripetal acceleration

$$F_c = mv^2/r$$

where:

m = mass of moving object

v = its linear velocity

r = radius of the circular path

Example: A 10-lb. stone, held in a horizontal circular path by a string 4.0 ft. long, moves at a constant speed of 8.0 ft/sec.

Find the centripetal force.

$$\underline{m = w/g} = \frac{10\ \text{lb.}}{32\ \text{ft/sec}^2}$$

$$F_c = \frac{mv^2}{r} = \frac{(10\ \text{lb.})(8.0\ \text{ft/sec})^2}{(32\ \text{ft/sec}^2)(4.0\ \text{ft.})} = \underline{\underline{5.0\ \text{lb.}}}$$

2.13 Centrifugal Reaction

A string that constrains an object to a circular path exerts on the object the centripetal force that changes the velocity. In reaction against this change of motion, the object pulls outward on the string with a force called the centrifugal reaction. This force, which is exerted by the object in its tendency to continue along a straight path, is just equal in magnitude to the inward (centripetal) force.

2.14 Work (erg, foot-pound)

The term work used in physics is restricted to cases in which there is a force and a displacement along the line of force.

$$\underline{\text{Work} = Fs}$$

Example: A box is pushed 15 ft. along a horizontal floor against a frictional force of 50 lb. How much work is done?

$$\text{Work} = Fs = (50\ \text{lb.})(15\ \text{ft.}) = \underline{\underline{750\ \text{ft-lb.}}}$$

2.15 Energy

The ability to do work is called "energy." Though energy can be neither created nor destroyed, it can exist in many forms and can be transformed from one form to another. The energy possessed by an object by virtue of its motion is called "kinetic energy," or energy of motion. Energy of position or configuration is called "potential energy." Since the change in the kinetic energy or potential energy is equal to the work done on a body or group of bodies, it follows that the units in which energy is expressed are the same units used for work.

2.16 Potential Energy

The most common form of potential energy is gravitational potential energy. Since the earth attracts every body, work is required to lift the body to a higher level. When a brick is carried to the top of a building, the work done on the brick (weight of the brick times vertical distance) represents energy that can be recovered. By virtue of its position at the top of the building, the brick possesses a capacity for doing work. It has potential energy.

$$\text{P.E.} = wh \text{ (foot-pounds)}$$

$$\text{P.E.} = mgh \text{ (ergs)}$$

Example: A 20-lb. stone is carried to the top of a building 100 ft. high. How much does its potential energy increase?

$$\text{P.E.} = Fs$$

$$\text{P.E.} = (20 \text{ lb.})(100 \text{ ft.}) = 2000 \text{ ft. lb.}$$

2.17 Kinetic Energy

In addition to energy of position or state; objects may possess energy due to their motions. A car or bullet in motion, a stream of water, or a revolving flywheel possesses kinetic energy K.E. The kinetic energy of a moving object can be measured by the amount of work it will do if brought to rest, or by the amount of work originally needed to impart the kinetic energy to it.

$$\text{K.E.} = (1/2)\left(\frac{w}{g}\right)(v^2)$$

$$\text{K.E.} = (1/2)(m)(v^2)$$

Example: What is the kinetic energy of a 3000 lb. automobile which is moving at 30 mi/hr (44 ft/sec)?

$$\text{K.E.} = (1/2)\left(\frac{w}{g}\right) v^2 = (1/2)(3000 \text{ lb.})\left(\frac{44}{32}\,\frac{\text{ft/sec}^2}{\text{ft/sec}^2}\right)^2$$

$$\text{K.E.} = 91{,}000 \text{ ft-lb.}$$

Note: If a moving body is stopped by a uniform force F, the work done in stopping must be equal to the kinetic energy.

$Fs = (1/2)(\frac{W}{g})(v^2)$ where s is the distance required to stop the object

Example: What average force is necessary to stop a bullet of mass 20 gm and speed 250 m/sec as it penetrates wood to a distance of 12 cm?

$Fs = 1/2\ mv^2$

$(F)(12\ cm) = (1/2)(20\ gm)(25,000\ cm/sec)^2$

$F = 5.2 \times 10^8$ dynes

2.18 Heat and Temperature

Heat and temperature are probably the most misunderstood parameters. Most laymen will tend to equate the meaning of heat and temperature. However, they are not the same. Heat is a measure of energy and temperature is, qualitatively speaking, a property which governs the direction of flow of heat. Temperature usually is measured by comparison with a standard body at a known temperature. All material is made up of molecules. These molecules are all in a state of random motion within the body of material. If heat were added to the body of material, the random motion of the molecules would increase much like a swarm of excited bees. Since each molecule has mass and is in motion, its energy then is its kinetic energy of motion. The total energy associated with the random motion of these molecules is what is meant by heat. Since the molecules are in random motion, not all the molecules will have the same kinetic energy, but the average kinetic energy per molecule can be considered the temperature of the body.

Temperature, as mentioned earlier, is usually a comparison with another standard body. Two such bodies are the temperature of frozen water and boiling water. On the Fahrenheit scale the freezing and boiling temperatures of water are 32° and 212° respectively, as compared to 0° and 100° on the centigrade scale. The temperature conversion between Fahrenheit and Centigrade is as follows:

Temperature in °C = 5/9 (°F-32)

Example: If the temperature of a room is 70°F, what is it in units of Centigrade?

°C = 5/9 (°F - 32)

°C = 5/9 (70 - 32)

°C = 21.1°

2.19 Avogadro's number

An atom is a very small particle (approximate diameter of 10^{-8} cm). Therefore, it is conceivable that any material substance is made up of billions and billions of atoms. The quantity of any substance which has a mass in grams numerically equal to its atomic or molecular weight is called one gram atom or one gram mole.

The number of atoms or molecules in one gram atom or one gram mole is called "Avogadro's number" and is equal to 6.02×10^{23} atoms per gram atom or molecule per gram mole.

Example #1. Find the number of atoms in 5 grams of calcium (Ca).

atomic mass of Ca = 40.08

therefore, number of atoms =

$$\left(\frac{5 \text{ gm}}{40.08}\right)\left(\frac{6.02 \times 10^{23} \text{ atoms}}{\text{gm atom}}\right)$$

$$= 7.51 \times 10^{22} \text{ atoms}$$

Example #2. Find the number of molecules in 5 grams of NaCl (salt)

atomic mass of Na = 23.0

atomic mass of Cl = 35.46

TOTAL 58.46

therefore, the number of molecules =

$$\left(\frac{5 \text{ gm}}{58.46}\right)\left(\frac{6.02 \times 10^{23} \text{ molecules}}{\text{gm mole}}\right)$$

$$= \underline{5.16 \times 10^{22} \text{ molecules}}$$

2.20 Coulomb's Law

Coulomb's Law is an experimental law which states that if two electrical point charges are placed a distance r apart the force on either charge is proportional to the product of the two charges and inversely proportional to the square of the distance between them.

$$F = \frac{q_1 q_2}{r^2}$$

where; q_1 & q_2 = magnitudes of two charges

r = distance between two charges

F = force on either charge

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ATOMIC PHYSICS

3.0 Introduction

With the ever increasing number of reactor power plants being constructed throughout the world, the demand for reactor operators and supervisors is beginning to increase. The responsibilities of these persons are great; and, therefore, adequate training in areas of nuclear physics, health physics, reactor kinetics, reactor instrumentation, and reactor operation is required. The more knowledge one has in these areas the better equipped he is to handle his job in a most efficient and safe manner, and the more valuable he becomes to the industry or utility he serves. With this in mind, let us start at the beginning.

3.1 Building Blocks of Matter

The material world we live in is made up of one or more of the basic 103 elements (See Table 1). By combining one or more of these elements, we form compounds such as the compound salt (NaCl). If we had a gram of salt and then subdivided the gram of salt down to a point where we had a single atom of sodium bound to a single atom of chlorine, then we can say that we now have a single molecule of salt. Any material, when subdivided into the smallest particle, having all the properties of that element is called an atom (See Figure 1).

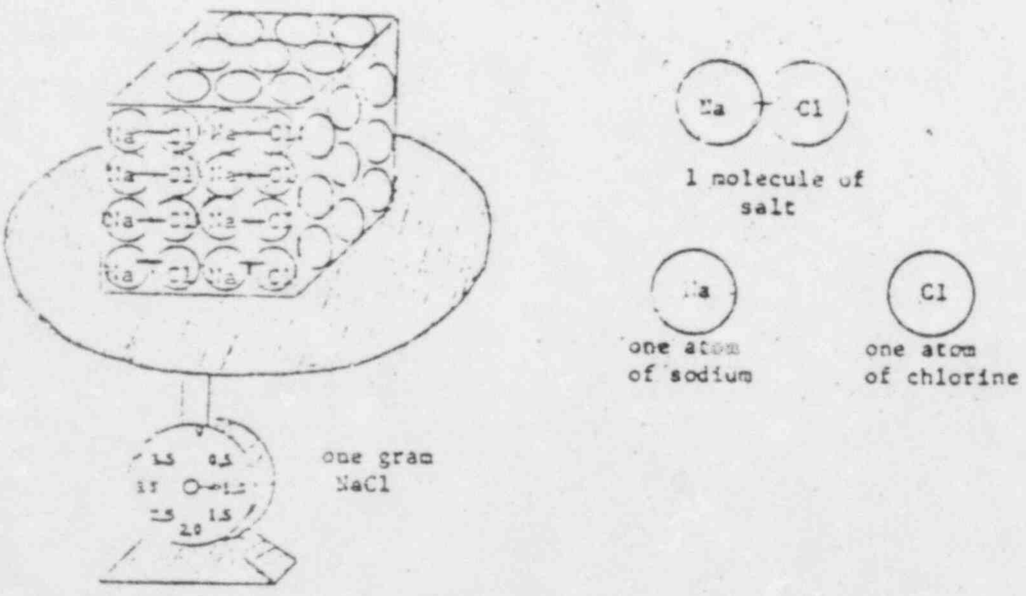
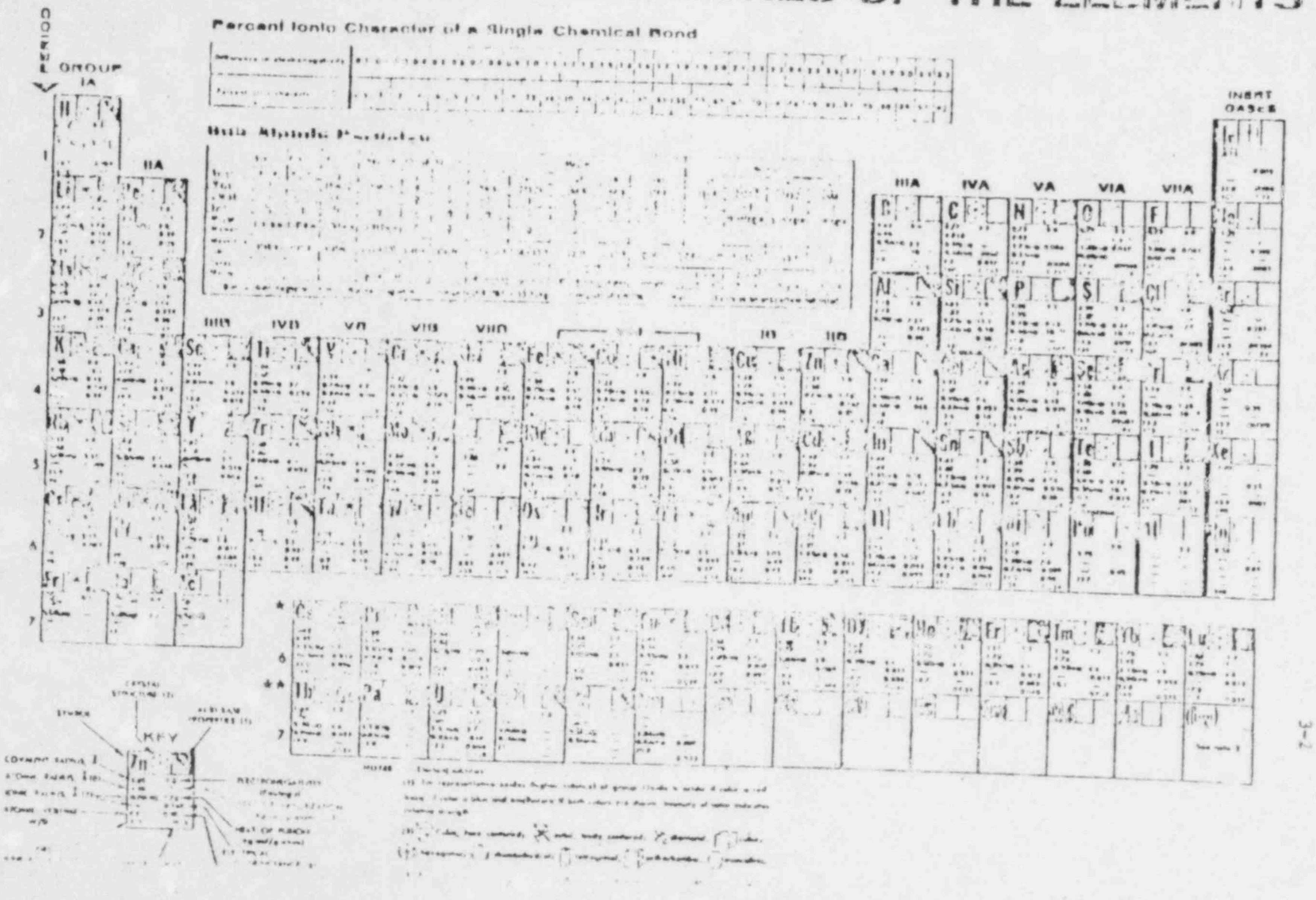


Figure 1

TABLE OF PERIODIC PROPERTIES OF THE ELEMENTS

Percent Ionic Character of a Single Chemical Bond



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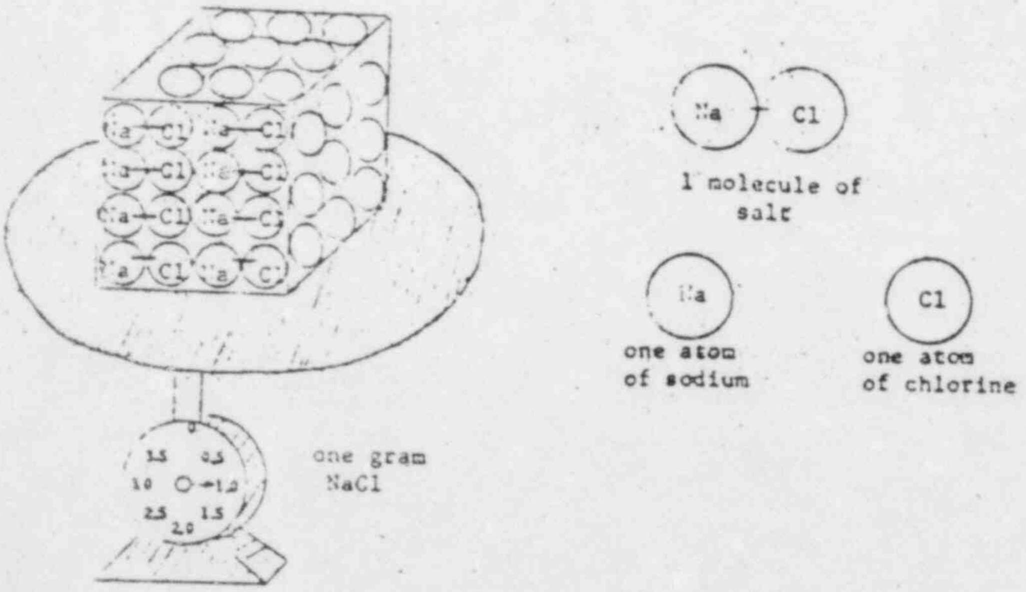


Figure 1

3.2 The Atom

The atom (See Figure 2) consists of a densly populated core called the nucleus (approximate dia. 10^{-13} cm) which contains protons and/or neutrons which are tightly bound together. The nucleus is surrounded by electrons, much like the earth orbits the sun.

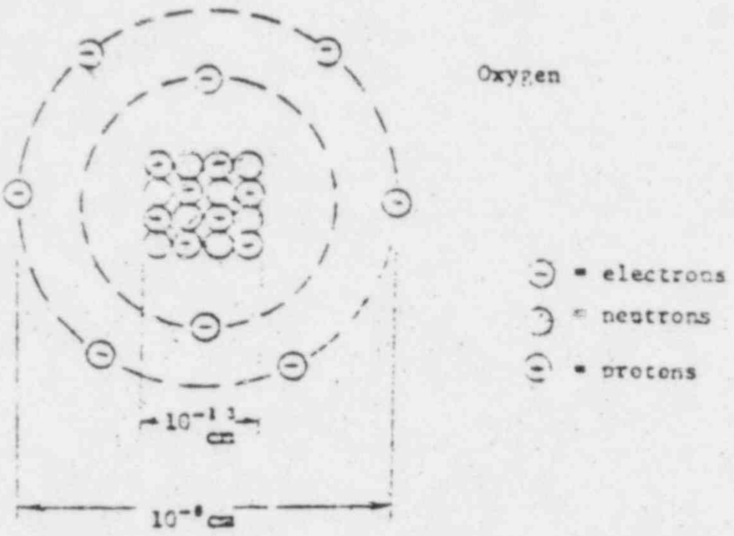
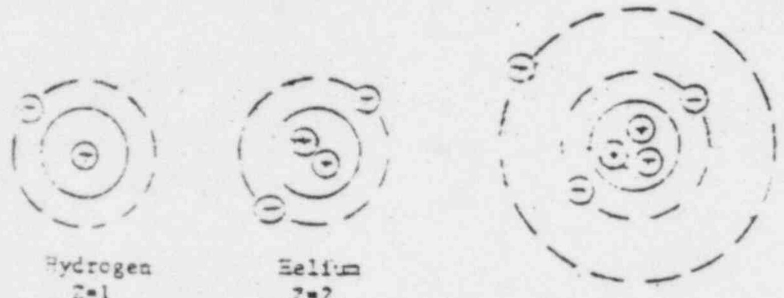


Figure 2

The proton is a positively charged particle (+1) having a charge of $+1.9 \times 10^{-19}$ coulombs and a mass of 1.6×10^{-24} grams. Electrically balancing each positively charged proton in the nucleus is a negative charged electron (-1) in orbit about the nucleus. The electron has a negative charge of -1.6×10^{-19} coulombs and a mass of 9.1×10^{-28} grams. The neutron is a neutral particle (no charge) and has a mass of 1.67×10^{-24} grams which is nearly equal to that of the proton. Note that since the mass of the proton and neutron are nearly equal, it is obvious that practically all of the mass of an atom is contained in the nucleus.

3.3 Element Identification

Each element (See Table 1) is identified, one from another, by the number of protons which are contained in the nucleus. For example, hydrogen has one proton in its nucleus, helium has two, and lithium has three (See Figure 3). For further classification, each element is assigned an atomic number (Z) which represents the number of protons in the nucleus. For example, the element hydrogen is assigned the atomic number Z=1, for helium Z=2, and for lithium Z=3 (See Figure 3).



Hydrogen
Z=1

Helium
Z=2

Lithium
Z=3

Figure 3

The number of sub-atomic particles in the nucleus (protons plus neutrons) is identified by its assigned atomic mass (A). For example, hydrogen, having only one proton in its nucleus, would have A=1 and sodium having eleven protons and twelve neutrons in its nucleus would have A=23 (See Figure 4). Therefore, each atom can be represented by the following notation:

$${}_{Z}X^{A}$$

where X = element symbol

A
= atomic mass number
= total number of nucleons
= sum of proton and neutrons in nucleus
= atomic weight to the nearest whole number

Z
= atomic number of an element
= number of protons in nucleus
= number of orbital electrons

From the above notation we can note that A-Z equals the number of neutrons in the nucleus.

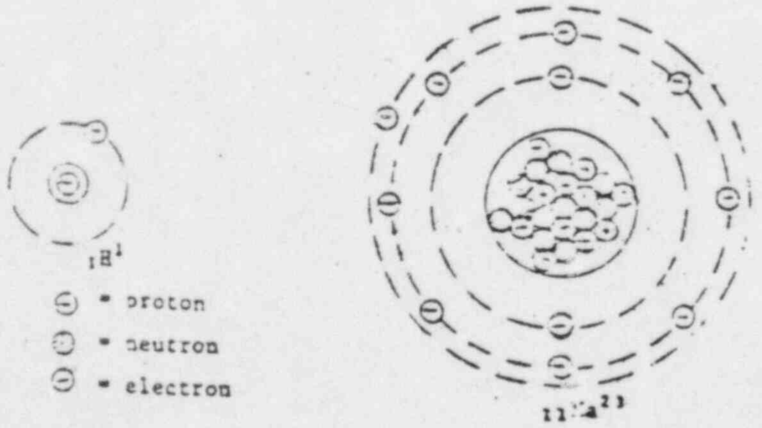


Figure 4

3.4 Laws of Chemistry

In 1869 a Russian chemist, Dmitri Mendelijeff, using research information from previous scientists, devised the system of grouping all the elements by order of their atomic weight. With Mr. Mendelijeff's information and present day knowledge, the acceptance of a classification system known as the "Periodic Law" is now used. The law simply states that the properties of the elements are periodic functions of their atomic numbers.

Upon investigation of the elements, certain periodic properties among the elements were found. By arranging the elements into what is known as a "Periodic Table" (See Table 1) these properties can be compared and easily discussed.

3.5 The Periodic Table

The periodic table (See Table 1) has the elements arranged in order of their atomic number with the elements of similar properties placed under each other in vertical columns called groups and horizontal rows called periods.

3.5.1 Groups

There are eight groups numbered from I to VIII. Group IA is called the alkali group; group IIA the alkaline earth group; groups IIIA, IVA, VA, VIA are called the non-metals; group VIIA the halogen family; group VIIIA the inert gases, and groups IB through IIIB are the transition elements which include the metals.

3.5.2 The Periods (horizontal rows)

The periods represent the number of shells of electrons which surround the atoms of each element. For example, hydrogen has one shell, lithium has two shells, and sodium has three shells (See Figure 4A).

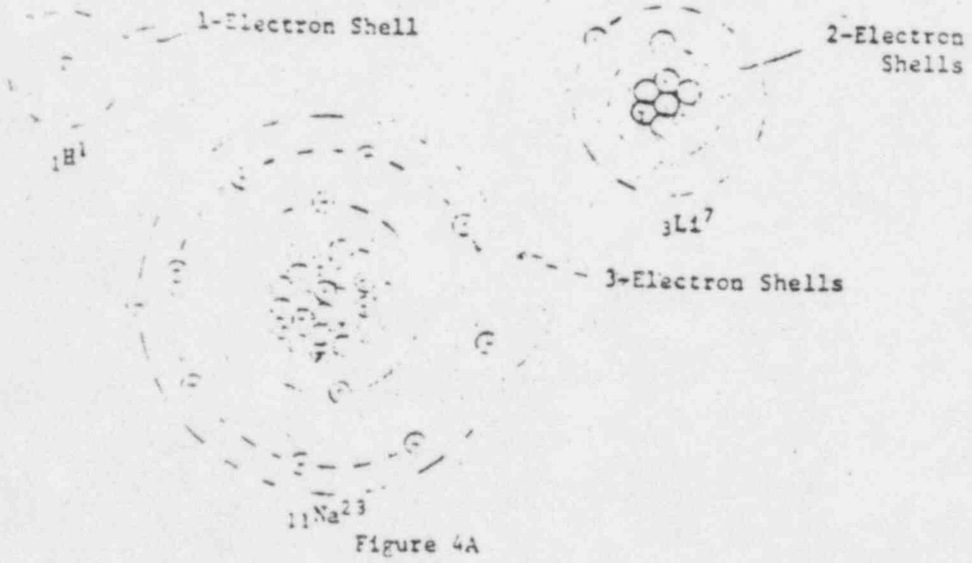


Figure 4A

3.6 Electron Shells

Both Lord Rutherford and Niels Bohr were considered the forerunners of the modern day physics. Around the year 1900, Lord Rutherford described the atom as consisting of a hard dense core, called the nucleus.. and surrounded by orbital electrons. Then Neils Bohr, following along this line, discovered an interesting relationship with respect to the number of electrons in each shell. He discovered that the number of electrons which could be in any one shell followed the value of $2(n)^2$, where "n" is the number of electron shells starting with the shell nearest the nucleus. He called each shell level a quantum level. For example, the first electron shell could contain a maximum of $2(1)^2 = 2$ electrons; the second shell could have $2(2)^2 = 8$ electrons, and the third shell could have $2(3)^2 = 18$ electrons. It must be noted, however, that this does not mean that each shell must contain the maximum number of electrons. It could contain less, but not more, than the maximum $2(n)^2$.

For the purpose of identifying each shell, each shell is lettered starting with the closest shell to the nucleus (k) and continuing outward l, m, n, o, p, ... etc. (See Figure 4B).

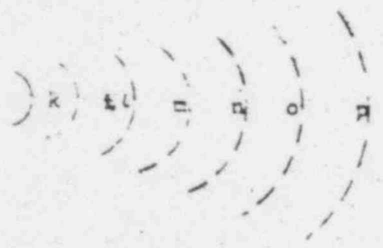
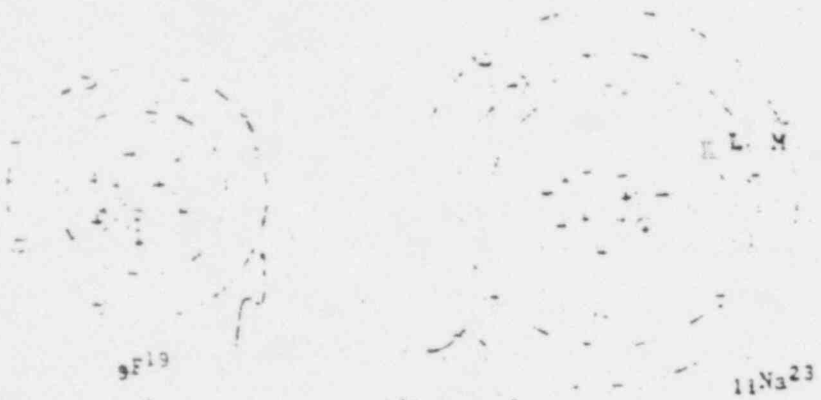


Figure 4B

3.7 Atom Symmetry

Each atom prefers to have a balance or symmetrical array of electrons within each orbit $2(n)^2$ and will go to almost any extreme to achieve this array. The atom may give up electrons or even borrow them in order to obtain this symmetry. For example, if we look at an atom of fluorine, we see that it contains 2 electrons in the "k" shell and 7 electrons in the "l" shell. In order to obtain symmetry, it could attract another electron or give up seven electrons. However, the chances are that it will take the easiest approach and attract one electron rather than give away seven (See Figure 4C).



needs one electron for symmetry

sodium needs seven electrons for symmetry

Figure 4C

From the above example, it can be seen that the fluorine atom would like very much to have the electron which is located in the "m" shell of the sodium atom, and the sodium atom would like very much to give up its electron to the fluorine atom. Therefore, as a result, they can combine very easily to form the compound, sodium fluoride (See Figure 5).

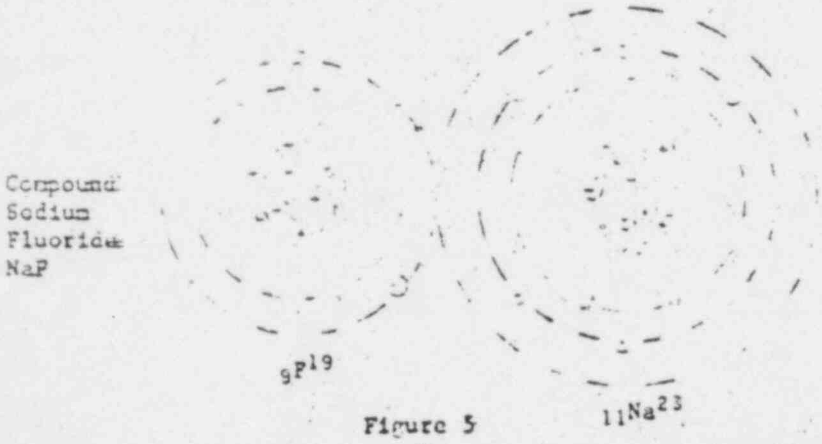


Figure 5

This theory of symmetry also explains why the inert gases Ne, Ar, He, Kr, Xe, do not combine with other elements, since each atom contains its full allotment of electrons in its outer shell.

The important concept to keep in mind is the constant adjustments and devious means to which an atom will go to achieve symmetry. This concept becomes very important later on in this course when talking about radioactive isotopes, for the same type of adjustments tend to occur in the nucleus of unstable isotopes.

3.8 The Bohr-Rutherford Atom

A few years after Lord Rutherford first described the atom as consisting of a hard dense core, called the nucleus, and surrounded by orbital electrons. Niels Bohr devised a theory about the conclusion of Rutherford's atom which was later to be called "Quantum Mechanics". Quantum mechanics describes the radiation from an atom in terms of wave length.

Bohr assumed that the electrons were kept in the orbits of a nucleus by ordinary coulomb forces. He also assumed that only certain orbits were possible. This means that the radius of any orbital electron would depend upon the energy of the electron. Since only specific orbits are possible, then only certain electron energies are possible. The electrons closest to the atom will have the smallest energy and those further out will have the most energy (See Figure 6).

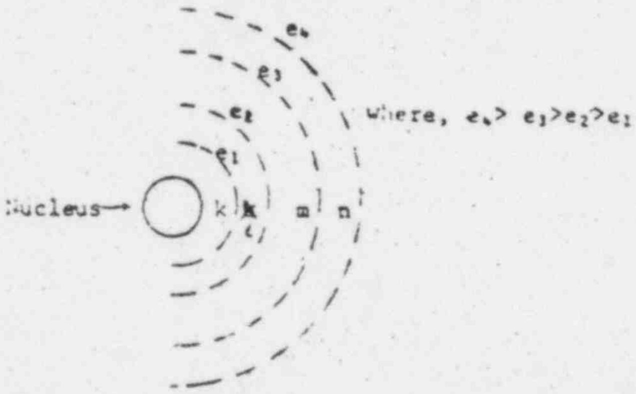


Figure 6

From the theory of orbital electron energies, Bohr postulated that now he can explain how light and magnetic radiation from an atom are formed. For example, if an electron in "n" shell were to fall into "m" shell, the energy required of "n" shell is now less; and the energy difference, it is postulated, is given off as a photon or packet of energy (See Figure 7).

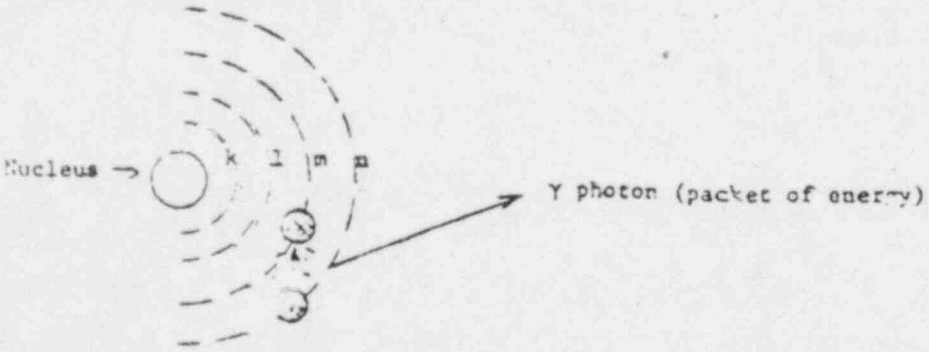


Figure 7

Note that for each transition between orbits there will be definite energy photons produced depending upon the type of transition.

1.9 Photons

The photons (or light energy) have energies equal to hV (h = Plancks constant, and v = frequency). Therefore, an electron transition from one quantum level to another will produce an energy ΔE = hV. It is this very principle which accounts for the production of x-rays. Any material of high "Z" (large number of electrons) when bombarded with charged particles will have some of its electrons stripped off; whereupon an electron from an outer shell will fall back into the vacancy, and the energy difference appears as a photon of light energy which is electromagnetic in nature.

1.10 Electromagnetic Radiation

Radiation, whether it be the transmission of radio-signals, television pictures, x-rays, gamma rays, infrared waves, etc., may all be considered a form of energy transmission. The radiation emitted by the above mentioned may be absorbed, reflected, and even stored by certain materials. To know which will occur we must better understand what is meant by the radiation of energy and such terms as radio waves, cycles, frequency and wavelength.

When a local radio station is broadcasting, it transmits energy through the air by electromagnetic radiation. This electromagnetic radiation may be thought of as waves and can be likened to the effect you might expect to see if you were to throw a stone into the center of a still lake. A depression would be developed where the stone hit the water, thus producing a wave which would move out equally in all directions until it reaches shore (See Figure 8).

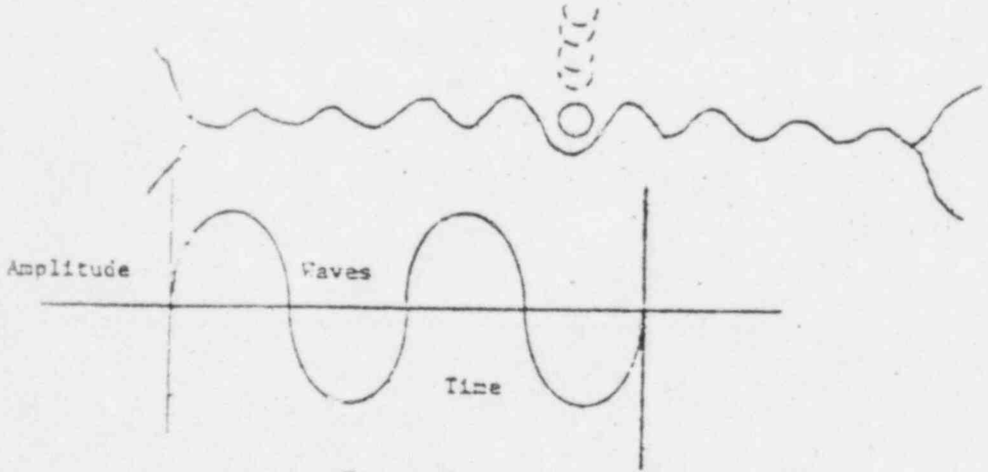


Figure 8

3.11 Waves

Waves or wave motion is the propagation of a periodic disturbance in a medium. The transverse wave is one which is most important in nuclear physics. For example, electromagnetic waves, like waves in water, are transverse waves and move up and down and move outwardly in a horizontal direction (See Figure 9).

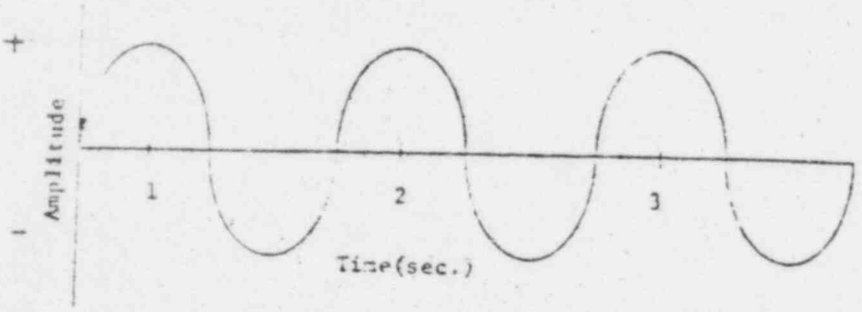


Figure 9

It can be seen by the figure above that the wave first rises in a positive direction, then reverses direction and goes negative. Also note that this occurs in a regular or periodic fashion. That is, if we measure the distance between two successive corresponding points on the wave, we find that this distance is the same between any other two successive points. This distance is called the wave length (λ) (See Figure 10).

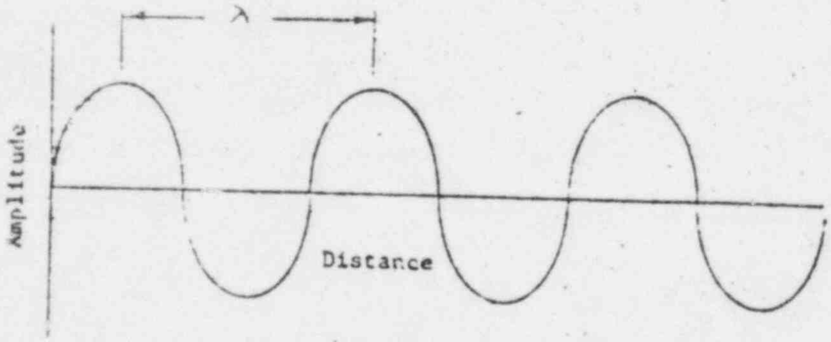


Figure 10

3.12 Frequency

The number of wavelengths or complete waves (one complete wave means one full cycle) which pass a given point per second is called the frequency (f) of the wave (See Figure 11).

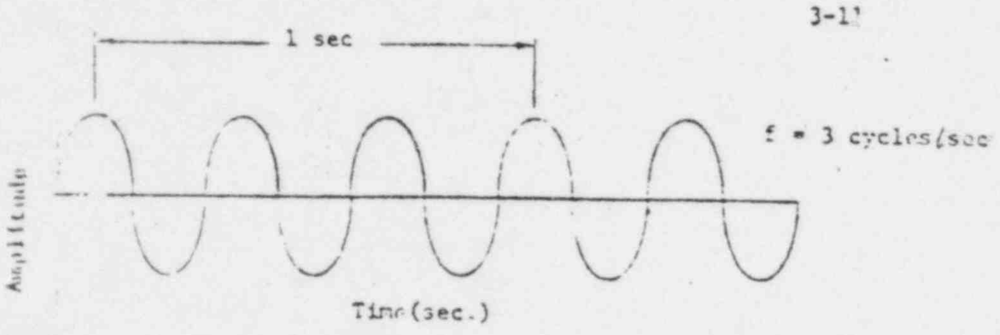


Figure 11

3.13 Frequency - Wavelength Relationship

Electromagnetic waves travel at the speed of light (c) (3×10^{10} cm/sec); therefore, a wave of wavelength λ traveling at speed c will produce a frequency (cycles/sec) of

$$f = \frac{c}{\lambda}$$

Note also that the period (T) of a wave is the time required for a complete wave to pass a given point and is the reciprocal of the frequency.

$$T = \frac{1}{f}$$

Example: 1. What is the wavelength of a radio station broadcasting at 1500 KC/sec (KC = kilocycles, which equals 1,000 cycles/sec)?

$$\lambda = \frac{c}{f}$$

$$\lambda = \frac{3 \times 10^{10} \text{ cm/sec}}{1.5 \times 10^{6} \text{ cycles/sec}}$$

$$\lambda = 2 \times 10^{4} \text{ cm/cycle}$$ (usually expressed as cm only)

or

$$\underline{\underline{\lambda = 200 \text{ meters}}}$$

2. What is the frequency of an unidentified signal which completes a cycle in 3 milliseconds (1 millisecond = 0.001 seconds)?

$$T = \frac{1}{f}$$

$$f = \frac{1}{T} = \frac{1}{3 \times 10^{-3} \text{ sec}}$$

$$f = \underline{\underline{3.33 \times 10^{2} \text{ cycles/sec}}}$$

3.14 Quantum Mechanics

In Section 3.10 (Electromagnetic Radiation), it was pointed out that radiation (such as radar, radio waves, x-rays, gamma rays) was considered a form of energy transmission, having some known frequency and wavelength. Quantum theory assumes that the energy of electromagnetic radiation of any frequency cannot be varied or emitted continuously but must be some exact integral multiple of a small basic unit of energy. This packet of energy is called the photon. Described in many ways, a photon could be described as a package of light or bundle of energy, etc. In any case, the energy of the photon is dependent upon the frequency of the radiation. It follows the relationship:

$$E = h\nu$$

where

E = energy of photon in Joules

h = Planck's constant (Joule-sec)

ν = frequency (cycles per second)

$h = 6.6 \times 10^{-34}$ Joule-sec

Returning to Neils Bohr theory on electron shells, you will remember that he assumed that the electrons were restricted to certain particular orbits. He further discovered that there was no radiation emitted from an atom as long as the electrons remained in orbit, but that energy is absorbed by an atom when the electron is moved from an orbit of lower energy to one of higher energy and that energy is emitted when the electron moves in the reverse direction. Since there are only discrete orbits, this restriction requires that energy be absorbed or emitted in quanta (energy packets) described by the above equation. This process is one source of x-ray production.

3.15 Mass-Energy Relationship

The idea that it was possible for a bundle of energy called a photon to exist without mass was hard for early scientists to conceive until one of the most well-known scientists, Albert Einstein, developed his theory on relativity. Out of this theory came the now famous equation:

$$E = MC^2$$

where

E = energy (Joules)

M = mass (kilogram)

C = velocity of light (meters/sec)

Now, for the first time, it is realized that energy and mass are convertable. That is, energy can be converted to mass and mass can be converted to energy.

A term more useful than ergs for describing energy in nuclear applications is the electron volt (ev) which is described as that amount of energy acquired by a particle having one electron charge when it passes in a vacuum through a potential difference of one volt.

Since the mass of an atom is described in terms of its atomic mass unit (1 AMU = 1.66×10^{-24} grams), its equivalent energy would be:

$$1 \text{ AMU} = \frac{(1.66 \times 10^{-24} \text{ grams})(3 \times 10^{10} \text{ cm/sec})^2}{(1.602 \times 10^{-6} \text{ ergs/Mev})}$$

where 1-Mev = one million electron volts

Therefore, 1-AMU = 931 Mev.

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BASIC NUCLEAR PHYSICS

4.0 Introduction

In the previous sections we have discussed the characteristics of an atom. Now we will turn our attention to the nuclear forces which interact between the nucleons within the atom. We will see how certain internal transitions within the nucleus of an atom can cause an atom to become either stable or unstable and how instability produces energy changes which result in the emission of radiation.

4.1 Isotopes

The chemical properties of an element are determined by the number of protons within the nucleus (Z). However, it is possible for atoms of tha same element to have the same number of electrons but differ in its atomic mass. Therefore, atoms which have the same "Z" but different "A" are called isotopes. For example, hydrogen has three isotopes (See Figure 1), hydrogen 1, commonly called protium; hydrogen 2, commonly called deuterium; and hydrogen 3, commonly called tritium.

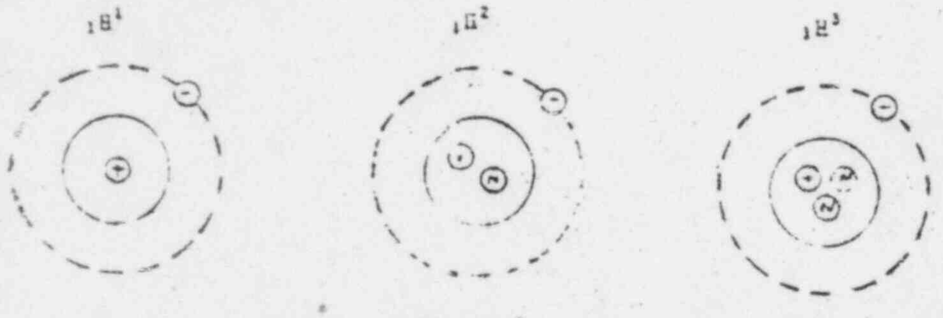


Figure 1

All the other elements, as shown on a chart of nuclides, have a various number of isotopes.

Generally speaking, isotopes can be found in either a stable state or an unstable state. A stable isotope is one in which there is a low probability of disintegration to some other combination. Unstable nuclei can disintegrate and transform themselves into other elements by the emission of radiation in the form of alpha, beta, or gamma radiation. To find out why some isotopes are stable and others unstable, we must further investigate the nuclear forces within the nucleus for the answer.

4.2 Nuclear Stability

Earlier in the course, we talked about the long-range coulomb forces which were forces of attraction or repulsion between two charged bodies. The question most often asked about the atom is, "Why isn't the negative charged electron attracted to the positive charged nucleus"? For the answer, we must realize that the electron is in constant motion around the nucleus and thus has centrifugal force (mv^2/r). Therefore, the centrifugal force must just balance out the coulomb force of attraction (q_1q_2/r^2), thus allowing the electron to remain in constant motion about the nucleus (See Figure 2).

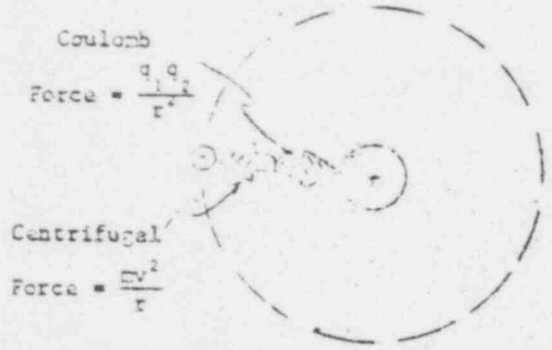


Figure 2.

As long as these forces are equal and opposite, stability external to the nucleus is maintained. Note, however, if the electron should gain energy (e.g., collision with a charged particle or gamma photon), the centrifugal forces could exceed the coulomb forces giving the electron enough energy to break away from its orbit to become a free electron. (See Figure 3)

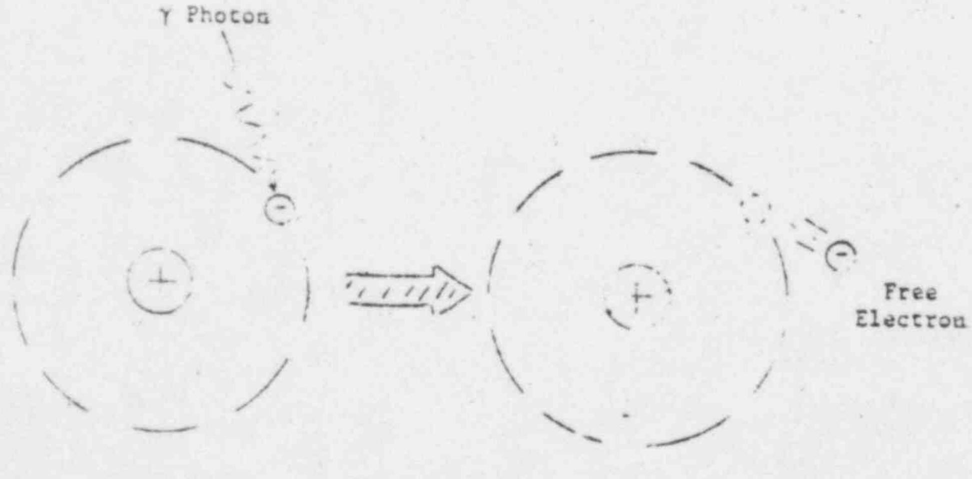


Figure 3

In the nucleus there exists a similar situation in which protons are bound tightly together. Since protons carry a plus charge and plus charges repel each other, one would expect that no more than one proton could exist within the nucleus. However, since there are so many elements which are stable and contain more than one proton in their nucleus, it is obvious that somehow the coulomb forces are overcome by some type of attractive force.

4.3 Nuclear Forces

The attractive forces which exist between proton-proton, proton-neutron, and neutron-neutron are called short range nuclear forces. The exact nature of these forces is unknown; however, it is known that they act only through very short distances (approximately 10^{-13} cm separation) and are strong enough to overcome the coulomb forces of repulsion. (See Figure 4).

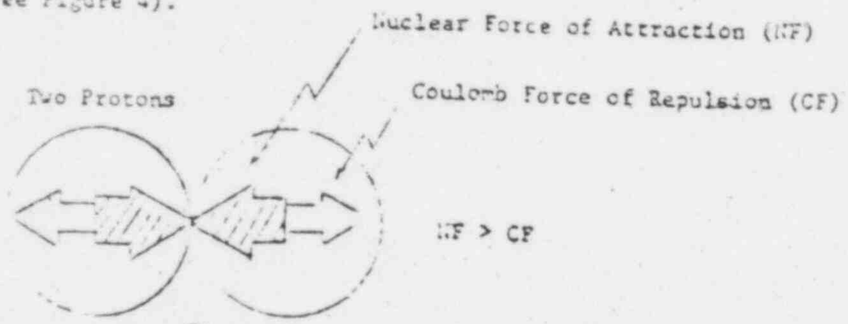


Figure 4

The nuclear forces can be considered a sort of nuclear glue which holds the nucleons together very tightly. For example, to separate a bond between a neutron and a proton in the hydrogen-2 nucleus would require about 2.2 Mev of energy. This energy increases with the mass of the element to as high as 8 Mev.

4.4 Binding Energy

The nucleons being tightly bound together would require a certain amount of energy to break them apart. This energy is calleo the binding energy of the nucleon.

If one were to take an atom of any element and sum up the masses of all the constituent particles, one might erroneously conclude that this would equal the mass of an atom.

$$\text{Mass(atom)} = M_p(Z) + M_n(A-Z) + M_e(Z)$$

where

M_p = mass of the proton

M_n = mass of the neutron

M_e = mass of the electron

For example, the mass of a Beryllium-8 ($_4Be^8$) atom should be:

$$\text{Mass }(Be^8) = M_p(Z) + M_n(A-Z) + M_e(Z)$$

$$= (1.007593)(4) + (1.008982)(4) + (5.4876 \times 10^{-4})(4)$$

$$= 8.068495 \text{ AMU}$$

However, by actual weight, the Beryllium-8 atom weighs 8.005310 AMU. The mass difference of:

$$\text{Mass difference} = (8.068495 - 8.005310)$$

$$MD = 0.063185 \text{ AMU}$$

or an energy difference of

$$\left(\frac{931 \text{ Mev}}{\text{AMU}}\right)\left(\frac{6.3185 \times 10^{-2} \text{ AMU}}{1}\right) = \underline{\underline{58.9 \text{ Mev}}}$$

which is equivalent to the binding energy or the energy which is required to break the nucleons apart. Therefore, the equation for calculating binding energy (BE) is:

$$BE = M_p(Z) + M_n(A-Z) + M_e(Z) - M_x$$

where

M_p = mass of the proton

M_n = mass of the neutron

M_e = mass of the electron

M_x = mass of the element

If we find the average binding energy per nucleon for all the nuclides by dividing the total binding energy by the atomic mass number and then plot this against the atomic mass number, we would produce a curve as shown in Figure 5.

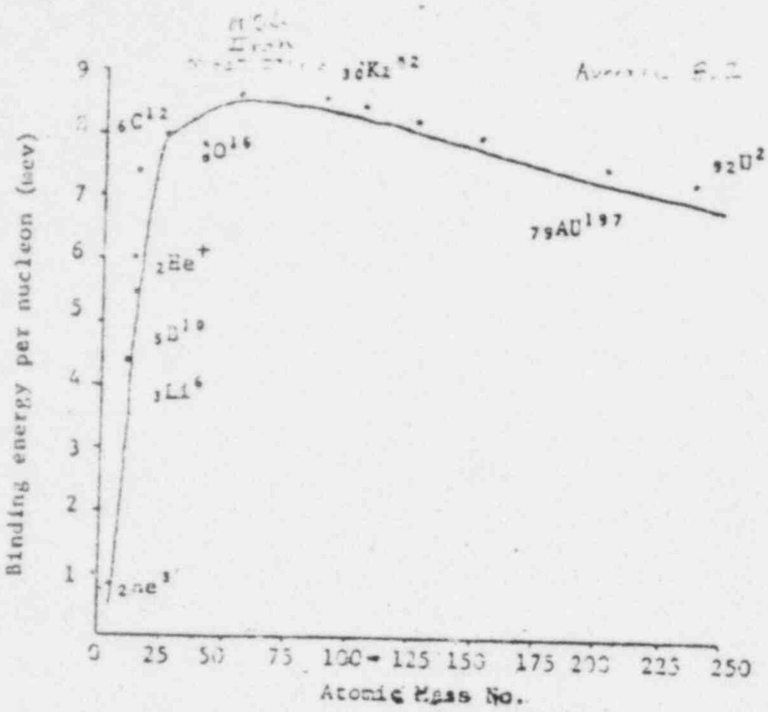


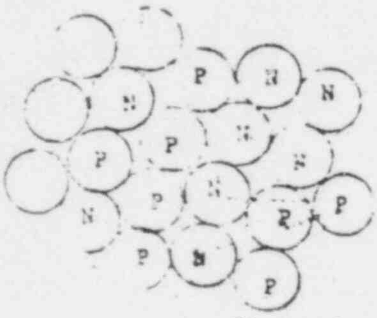
Figure 5

Note that the binding energy per nucleon rises very sharply to a maximum of about 8.7 Mev and then drops gradually to about 7.4 Mev for uranium. The binding energy for light nuclei is small, and high (approximately 8 Mev per nucleon) for heavy nuclei. This is the reason a great amount of energy is released when heavy nuclei such as Uranium-235 breaks up into two parts.

4.5 Nuclear Instability

Investigation of the nuclear forces shows us that they are saturable; that is, given a particular nucleon, it may strongly attract perhaps up to three nucleons, but the fourth one will be repelled easily. Later studies clarify that the total energy within the nucleus is quantized much like the electron shells around the nucleus. However, the biggest contributor to nuclear instability is the neutron to proton ratio.

As mentioned earlier, there is a competition of forces interacting between the nucleons within the nucleus. With this in mind, we can see that the protons will tend to separate from one another if possible. However, in most cases, the nuclear space between the protons is occupied by the neutrons (in the case of heavy nuclides); and, since there is no charge of repulsion or attraction between the neutrons and the protons, only nuclear forces occur. But, remember, the protons located at various points within the nucleus do experience the coulomb forces of repulsion between one another (See Figure 6).



$_9F^{19}$ nuclei

Example of the coulomb force acting on a single proton in the nucleus

Figure 6

If the coulomb forces become large enough, the nuclear binding energy will be overcome, causing the atom to become very unstable and rid itself of this excess energy by the emission of radiation in the form of alpha particles, beta particles, or gamma photons.

The contributing factor to nuclear stability, then, is the neutron to proton ratio:

$$\text{Ratio} = \frac{A-Z}{Z}$$

In the case where there are too many protons (A-Z/Z small ratio), the nucleus will normally remain bound. But in a few cases, a proton will change its identity to a neutron with the emission of a positron ($\beta+$). When the nucleus contains too many neutrons (A-Z/Z large ratio), a neutron may change its identity to a proton with the emission of a beta particle ($\beta-$). The neutron to proton ratio for all the nuclides may vary from 1, for light nuclei, to 1.6 for heavy nuclei (See Figure 7).

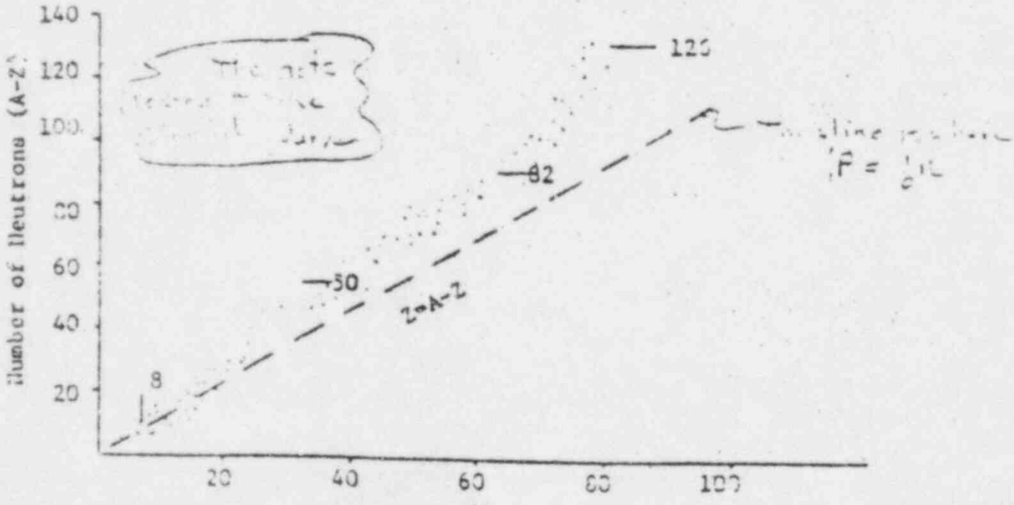


Figure 7

4.6 Radioactivity

Since unstable isotopes disintegrate spontaneously to a stable configuration by the emission of radiation, radioactivity can then be defined as "the spontaneous nuclear disintegration of unstable nuclei with the emission of corpuscular or electromagnetic radiations".

Because unstable nuclei constantly seek stability, it isn't surprising to find that most of the isotopes found in nature are stable. Only a few of the natural (high Z) isotopes such as K^{40}, Rb^{87}, In^{115}, La^{138}, Nd^{144}, Sm^{147}, Lu^{176}, W^{180}, Re^{187} and Pt^{190} are found to be unstable. All elements above Z=83 exist in an unstable radioactive form.

Naturally occurring isotopes or those produced by man-made devices such as cyclotrons, Van de Graff generators, or linear accelerators produce radiation in one or more of the following forms:

a) alpha particle emission

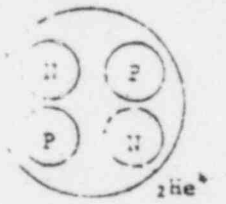
b) beta particle emission

c) gamma rays or photons

4.7 Alpha Particles (α)

Alpha particle emission usually occurs with the spontaneous breakdown of very heavy nuclei. The alpha particle is a unit package consisting of two neutrons and two protons having a net mass of approximately 4 x 10^{-24} grams and a net charge of (+2). The unit package looks identically like a helium nuclei; that is, a helium atom stripped of its electrons ($_2He^4$) and is emitted from the nucleus with considerable kinetic energy (See Figure 8).

Alpha Particle (Helium nuclei)



Net charge +2

Atomic Mass 4 x 10^{-24} gms

Figure 8

As mentioned earlier, there are two types of forces acting on nucleons within the nucleus - the long-range coulomb forces and the short-range nuclear forces. As a result of these forces, a potential wall is formed around the alpha particle (See Figure 9).

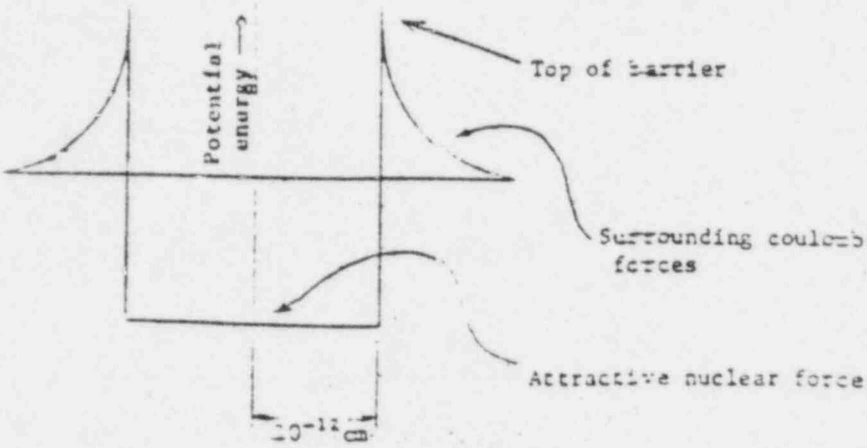


Figure 9

The alpha particle is located at the bottom of the well due to the very strong nuclear forces. For the alpha particle to escape, it must acquire enough energy to surmount the top of the barrier. This would require approximately 25 Mev in order to escape. In addition, quantum mechanics predicts that there is a small probability for an alpha particle to travel to any other location, including the area outside the nucleus even if it did not have enough energy to surmount the barrier. In other words, a small chance exists that the alpha particle could be anywhere in space including somewhere outside the potential well; and, the more energy it acquires, the better its chance of escape.

The classical example of an unstable isotope disintegrating by the emission of an alpha particle is as follows:

$$_{92}U^{238} \longrightarrow {}_{2}He^{4}(\alpha) + {}_{90}Th^{234}$$

The uranium-238 isotope decays by the emission of an alpha particle to form thorium-234. In many cases when an alpha particle is emitted, the parent isotope is left in an excited state (unstable state) and rids itself of this excess energy by the emission of a gamma photon. Note that when an alpha particle is emitted, the parent isotope decreases its atomic mass by four and its atomic number by two.

4.8 Beta Particles (β+ or β-)

Beta particles are electrons emitted from the nucleus of unstable isotopes. They have essentially no mass (10^{-27} grams) and have a charge of plus one (sometimes called a positron) or minus one (sometimes called a negative beta). Experimental evidence has shown that electrons do not exist within the nucleus. If this is the case, where do they come from?

From experimentation with beta emitting isotopes, it has been shown that when the nucleus of the parent atom emits a beta particle, the atomic number (Z) changes with no change in the atomic mass (A). For example, the decay of Au-203 is as follows:

$$^{203}_{79}Au \longrightarrow {}^{203}_{80}Hg + {}^{0}_{-1}\beta$$

Probably the first question which arises at this time is, "If gold emits a negative beta particle, why does the atomic number (Z) of the parent isotope increase by one?" The reason this occurs is because a neutron within the nucleus transforms itself into a proton and an electron and a neutrino as shown:

$${}_0n^1 \longrightarrow {}_1H^1 + {}_{-1}e^0 + \text{Neutrino}$$

Therefore, since the neutron changed identity with a proton, the atomic number (Z) of the parent isotope is increased by one without a change in the atomic mass (A).

In the case of an isotope which emits a positive beta ($\beta+$), the atomic number of the parent isotope decreases by one. For example, the isotope Sb-120 decays as follows:

$$^{120}_{51}Sb \longrightarrow {}_{+1}e^0 + {}^{120}_{50}Sn$$

In this example, a proton in the nucleus transformed into a positron and a neutron and a neutrino.

$${}_1H^1 \longrightarrow {}_{+1}e^0 + {}_0n^1 + \text{Neutrino}$$

Again, the atomic mass remains the same but the atomic number of the parent isotope decreases by one.

4.9 Gamma Photons (γ)

Gamma photons, like x-rays are electromagnetic radiations with the exception that gamma photons originate from the nucleus whereas x-rays originate from the electron shells. When most radioactive isotopes decay by the emission of an alpha particle or beta particle, the nucleus of the parent isotope is left in a very excited state and usually returns to a ground state (stable state) by the emission of a gamma photon with a discrete energy equal to (hν). Since there is only a change in energy, the atomic number (Z) and the atomic mass (A) of the parent isotope remain the same. For example:

$${}_{27}Co^{60*} \longrightarrow {}_0\gamma^0 + {}_{27}Co^{60}$$

4.10 Radioactive Transformation

Unstable nuclei, like the electrons in the orbital shells of an atom, want to obtain stability. In the process, they may decay by going through several readjustments. That is, a radioactive parent isotope may decay by beta emission and then the daughter isotope might decay by alpha emission and a continuation of this process until the isotope reaches its ground state or stable state.

For elements of atomic number 81 to 92 we find that they are naturally radioactive and are continually decaying into other radioactive elements until some stable, non-radioactive nuclide is formed. We call this formation of nuclides a "radioactive series."

There are three such series for the heavy nuclides - the thorium series, the uranium series, and the actinium series. All three series have one thing in common, that is, they all have the element lead as a final product. (See Figures 10, 11, and 12).

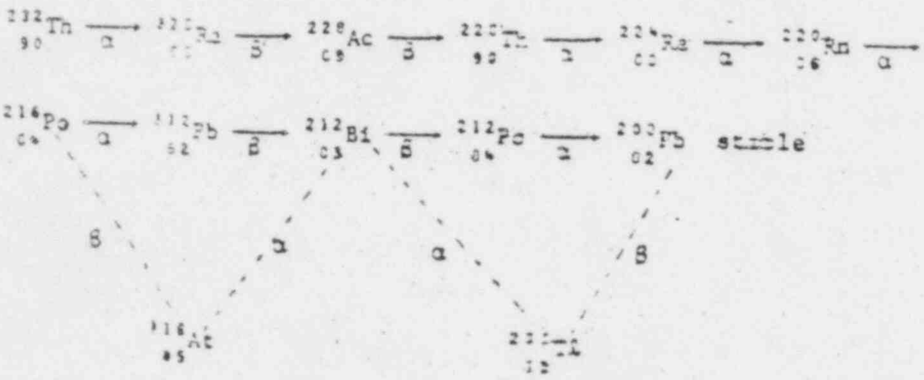


Figure 10

Thorium Series

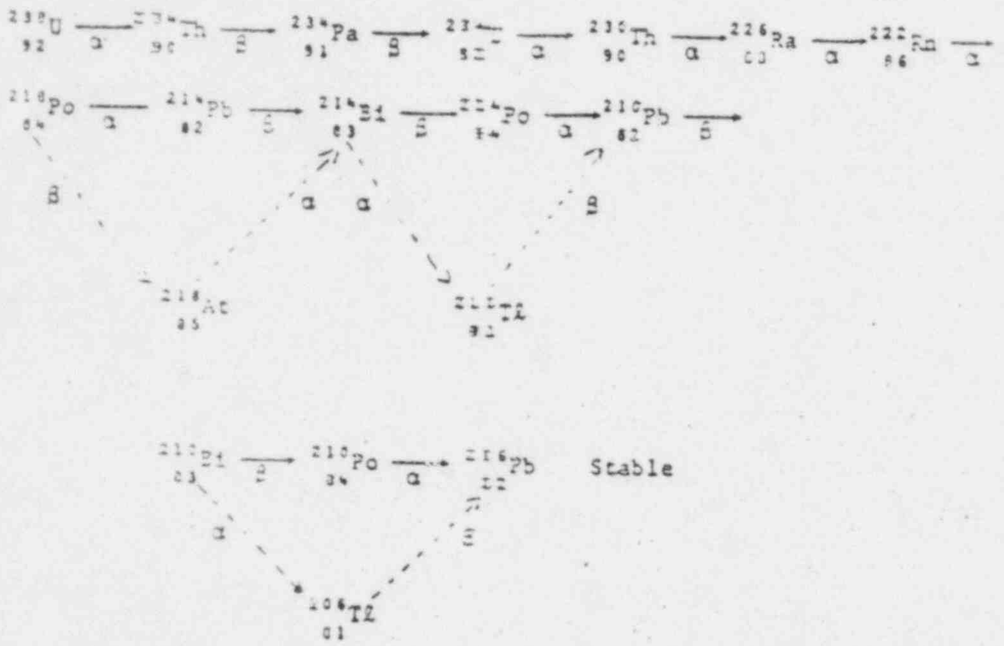


Figure I1

Uranium Series

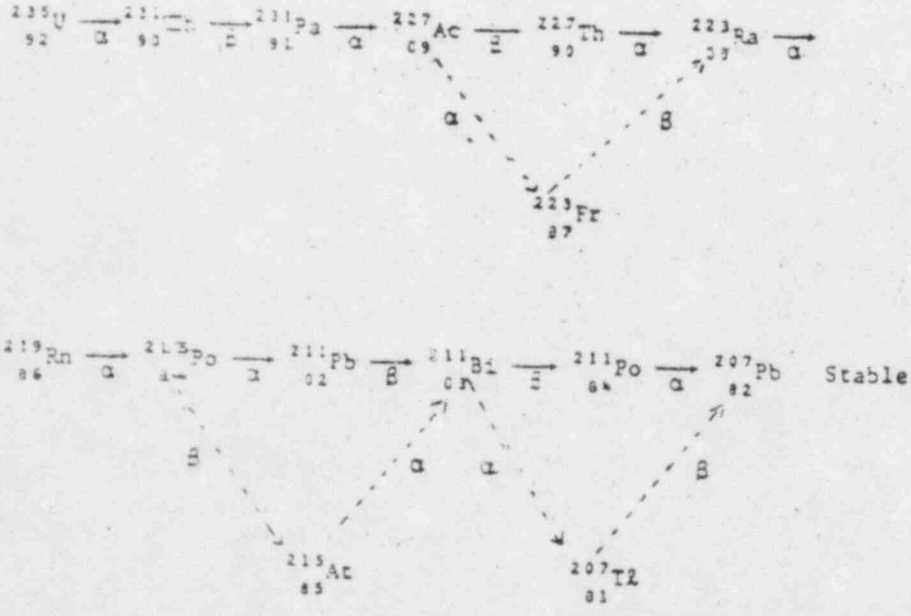


Figure I2

Actinium Series

4.11 Radioactive Decay

Each radioactive isotope decays at a constant rate and no physical, chemical or other means can change this rate. A sample containing billions of atoms will decay by the emission of these atoms at a particular rate, but it is impossible to determine which atom will decay when. Therefore, this process is a random process.

Assuming that the number of atoms of a particular radioactive isotope is very very large, then we can assume the average time it will take for any isotope to decay.

The number of atoms which are decaying with time is proportional to the number of atoms present and we may write

$$\frac{\Delta N}{\Delta T} = \lambda N$$

where ΔN is the change in the number of atoms originally present per change in time (ΔT). Since the rate of decay is proportional to the number of atoms originally present, λ (pronounced lambda) represents the proportionality constant or decay constant and "N" represents the original number of atoms present.

Since the number of atoms of the radioactive isotope decreases with time, a negative sign is needed as follows:

$$\frac{\Delta N}{\Delta T} = -\lambda N$$

This formula now represents the rate of decay (the number of atoms disintegrating with time) which is equal to a decay constant times the number of atoms originally present.

By the use of calculus, we can integrate the above equation and arrive at another equation which will allow us to calculate the number of atoms remaining at any time "t" if we know how many atoms we had originally (N) and the decay constant fot this isotope. This relationship is as follows:

$$N = N_o e^{-\lambda t}$$

where, N = number of atoms present at time "t"

N_o = number of atoms originally present

λ = decay constant

t = time of decay

The decay constant (λ) is defined in terms of another relationship called "half-life". Where the half-life ($T_{1/2}$) is the time it takes for one half of the radioactive nuclei to decay. In terms of the above equation, $t = T_{1/2}$ when N equals one half of N_o. Then the following equation evolves:

$$\frac{N_o}{2} = N_o e^{-\lambda T_{1/2}}$$

or $$\frac{1}{2} = e^{-\lambda T_{1/2}}$$

or $$2 = e^{\lambda T_{1/2}}$$

then, $$\ln 2 = \ln (e^{\lambda T_{1/2}})$$

$$0.693 = \lambda T_{1/2}$$

or $$\lambda = \frac{0.693}{T_{1/2}}$$

We now find that the decay constant is found by dividing the half-life of the radioactive isotope into 0.693. The half-life for all known isotopes has been experimentally determined and can be found in a chart of the nuclides.

Taking a sample of radioactive material and plotting its acitvity with time would yield a curve which looks like the following (See Figure 13).

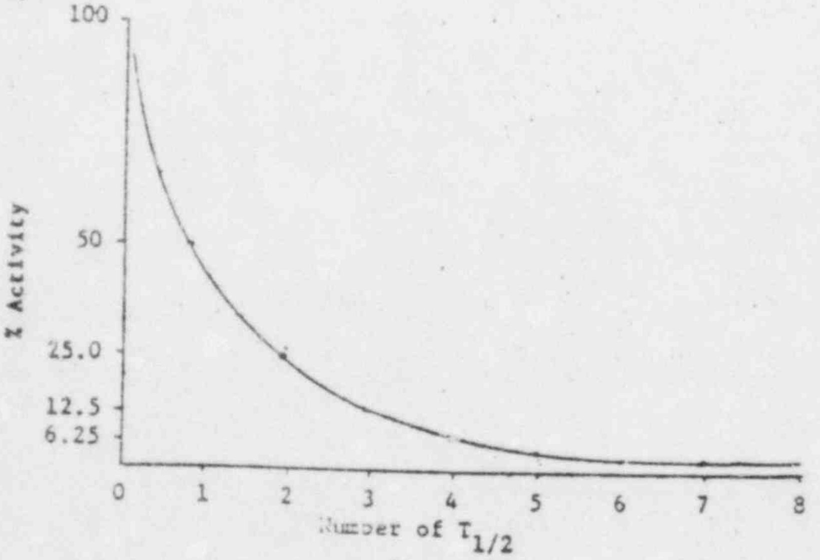


Figure 13

Notice that the activity decreases by one half for each half-life but never reaches zero, only approaches it. Also, observe that after about six half-lives the percent activity remaining is less than one percent.

Examples:

1. Determine the number of disintegrations per second from 2 grams of Pb-209 which has a half-life of 3.3 hours.

$$\frac{\Delta N}{\Delta T} = -\lambda N$$

$$\frac{\Delta N}{\Delta T} = \left(\frac{0.693}{3.3\ \text{hr}}\right)\left(\frac{2\ \text{gm}}{2.09 \times 10^{2}}\right)\left(\frac{6.02 \times 10^{23}\ \text{atoms}}{\text{gm-atom}}\right)\left(\frac{\text{hr}}{3600\ \text{sec}}\right)$$

$$\frac{\Delta N}{\Delta T} = 3.3 \times 10^{17}\ \text{disintegrations/sec}$$

2. If a reactor operator were accidentally exposed to a high beam of neutrons which activated his gold wrist band producing Au-198 ($T_{1/2}$ = 64.8 hours), how many atoms of Au-198 would be remaining at the end of 5 days assuming there were 5 gm of Au-198 at the time of exposure?

$$N = N_o e^{-\lambda t}$$

$$N_o = \left(\frac{5\ \text{gm}}{198}\right)\left(\frac{6.02 \times 10^{23}\ \text{atoms}}{\text{gm-atom}}\right) = 1.52 \times 10^{22}\ \text{atoms}$$

$$T_{1/2} = 64.8\ \text{hr} = \left(\frac{64.8\ \text{hr}}{1}\right)\left(\frac{\text{day}}{24\ \text{hr}}\right) = 2.7\ \text{days}$$

$$\lambda = \frac{0.693}{T_{1/2}} = \frac{0.693}{2.7\ \text{days}} = 2.56 \times 10^{-1}\ \text{days}^{-1}$$

$$N = (1.52 \times 10^{22}\ \text{atoms})\, e^{-(2.56 \times 10^{-1})(5)}$$

$$N = \underline{\underline{4.22 \times 10^{21}\ \text{atoms remaining after 5 days}}}$$

4.12 Ionization

In atomic physics, we learned that an atom of any element is normally electrically neutral; that is, for each positive charged proton within the nucleus, there is a negative charged electron in orbit around the nucleus. Ionization, then, is simply the removal of one of the orbital electrons from the atom creating a positive charged atom and a free electron (See Figure 14).

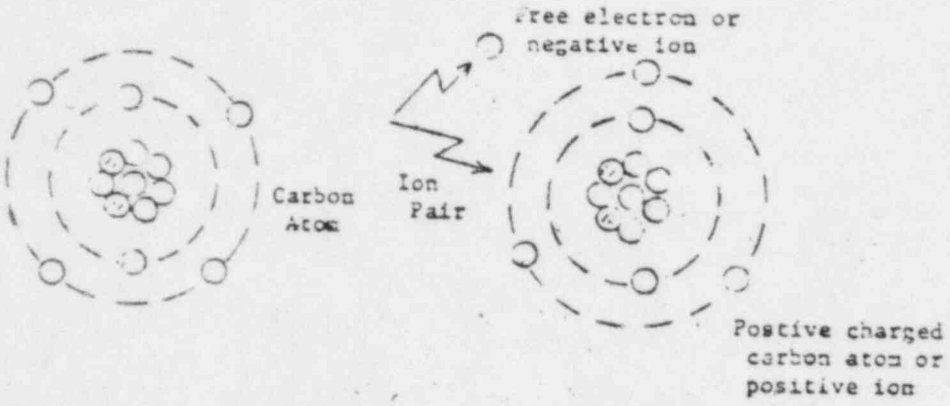


Figure 14

Ionization of an atom can occur when energy from some external source such as radiation is transferred to the electrons in the target medium. The electrons may then acquire enough energy to escape from their atom creating an ion pair.

Since radiation is classified into two general groups, charged and uncharged particles, we might expect, and do find, that the interaction of these particles with matter falls into two categories. For example, we find that charged particles (e.g., beta and alpha particles) interact with materials producing direct ionization, while uncharged particles (gamma photons) cause ionization by secondary means (to be discussed later).

The number of ion pairs produced per unit length of travel is called the specific ionization. The higher the charge on the interacting particle, the higher the specific ionization. For two particles having the same charge, the one with the highest mass creates the highest specific ionization. This is true because the heavier the particle, the slower its speed through the target material and the longer it spends in a given area. For example, an alpha particle with its two protons and two neutrons is a very heavy particle. For a given energy, the alpha particle would produce approximately 2,000 more ion pairs per unit length of travel than a beta particle of the same energy.

In air it takes approximately 32.5 ev of energy to produce a single ion pair. Therefore, for an alpha particle of 5 Mev, the total number of ion pairs formed would be:

$$\text{I.P.} = \frac{5 \times 10^6 \text{ ev}}{32.5 \text{ ev/I.P.}} = \underline{1.54 \times 10^5 \text{ ion pairs}}$$

and since a 5 Mev alpha particle can be stopped in approximately 3 cm of air, the specific ionization of an alpha particle in air would be:

$$\text{S.I.} = \frac{1.54 \times 10^5 \text{ I.P.}}{3 \text{ cm air}} = \underline{5.13 \times 10^4 \text{ ion pairs/cm}}$$

A beta particle of the same energy (5 Mev) would be stopped in approximately 5000 cm of air. Therefore, its specific ionization would be:

$$S.I. = \frac{1.54 \times 10^5 \text{ ion pairs}}{5 \times 10^3 \text{ cm}} = \underline{\underline{30.8 \text{ ion pairs/cm}}}$$

4.13 Interaction with matter

4.13.1 Alpha radiation

The alpha particle, as mentioned earlier, when traveling through matter produces ionization. If we would measure the specific ionization of an alpha particle as a function of distance, through air, a curve with the following shape would be formed (See Figure 15).

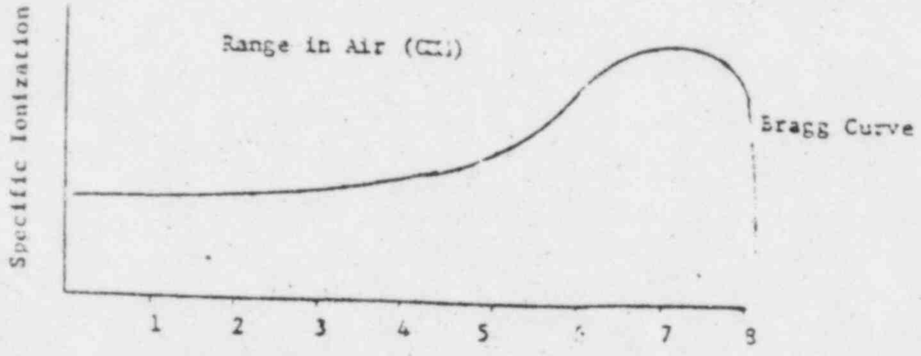


Figure 15

This type curve is called a Bragg curve. You will notice that the specific ionization increases slowly as it travels through the air, then increases very rapidly to a peak, then sharply returns to zero. This indicates that the alpha particle has a definite range in a given material which is easily defined. As the alpha particle travels through matter, it loses some of its energy for each ion pair formed and finally, at the end of its path, combines with two electrons to form a helium atom. The fact that the specific ionization falls off very rapidly indicates that from a single alpha emitting source all the alphas have the same or discrete energy. Therefore, the range of an alpha particle depends upon the source of particles (each alpha emitting isotope having a specific energy alpha) and the medium through which the alpha particle is traveling.

Experimentation has shown that the energy, and thus the range of an alpha particle, is inversely proportional to the half-life of alpha emitting isotopes. This means that the shorter the half-life, the longer the range.

The range of an alpha particle in a solid decreases as the density of the solid increases. The range of an alpha particle can be determined by the following relationship:

$$R(\text{solid}) = \frac{(3.2 \times 10^{-4})(X)(A^{1/2})}{\rho}$$

where, λ = range in air (approx. 3.45 cm for a 5 Mev alpha)

A = atomic mass of solid

ρ = density of the solid

The approximate range of a 5 Mev alpha particle in air is 1.35 inches and in human tissue about 0.001 inches; which means that since a sheet of paper is approximately twice the thickness of human skin, it will effectively stop all alpha radiation.

4.13.2 Beta Radiation

Beta particles like alpha particles cause ionization in the material through which they travel. However, the specific ionization is much less than that of alphas due to the smaller mass and speed of travel. Because of the small electrical field which surrounds the beta particle, it must come much closer to the atom of material to cause ionization than the alpha particle.

Various absorption experiments with beta emitting isotopes have shown that beta particles are <u>not</u> emitted in discrete energies but in a continuous range of energies up to a definite value (See Figure 16).

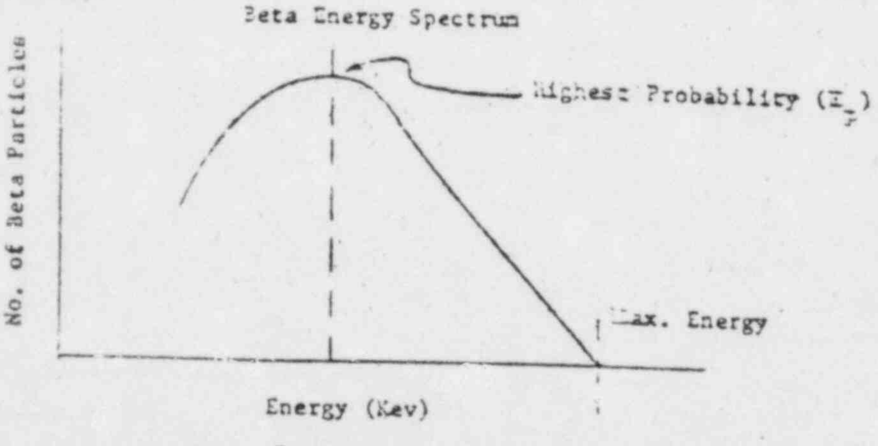


Figure 16

The continuous energy spectrum is due to the sharing of energy between the beta particle and the neutrino, with the beta particle carrying approximately one-third the energy and the neutrino the remaining two-thirds. Also note from the curve that there is some maximum energy but, the probability of a beta particle having this energy is very small as compared to having an energy equal to E_p.

Beta particles, because of their small mass, are easily scattered as they pass through matter. This is due not only to head-on collisions with particles but also to the strong coulomb forces of repulsion. Because of this scattering and the continuous energy spectrum, the beta particles do not have a definite range.

Even though beta particles do not have a definite range, it is possible to obtain a penetration thickness of some material which will attenuate virtually all of the radiation (See Figure 17).

Penetration of Beta Particles

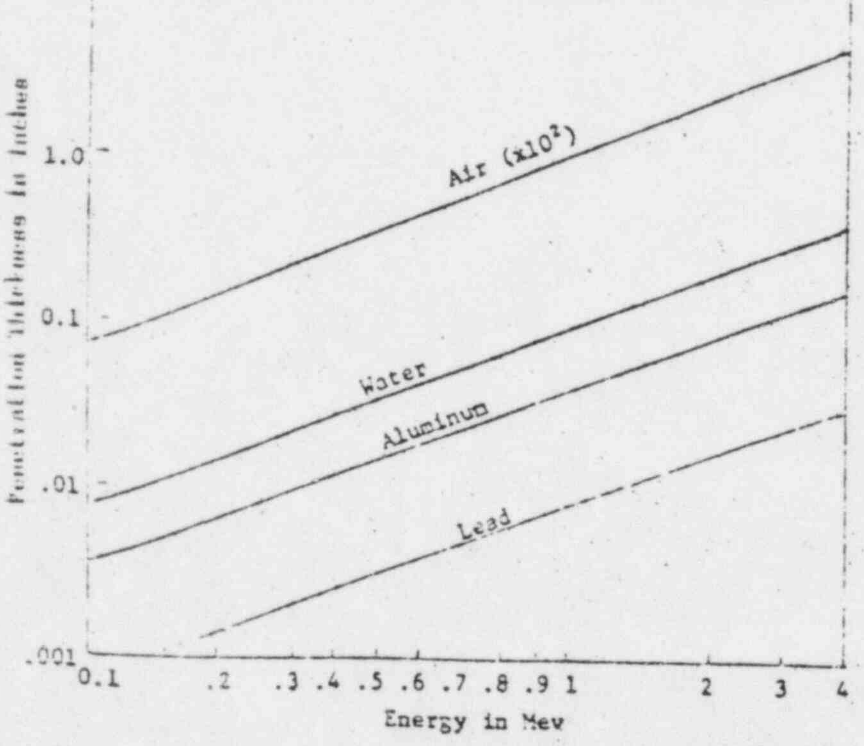


Figure 17

For betas between 0.8-3 Mev, approximate range in

$$gm/cm^2 = 0.546\,E(Mev) - 0.16$$

Example 1: Find the range of 1 Mev betas in aluminum ($\rho = 2.82$ gm/cm^3) and convert to penetration in inches (1 in. = 2.54 cm).

Range = 0.546(1)-0.16 = 0.386 gm/cm^2

$$\text{Penetration} = \frac{\text{range}(gm/cm^2)}{\rho(gm/cm^3)\ 2.54(in/cm)}$$

Penetration = 0.386/2.82(2.54) = 0.054 in.

4.13.3 Gamma Radiation

Since gamma radiation is electromagnetic in nature, it cannot cause ionization directly like charged particles, but does so by three indirect processes - photoelectric effect, Compton effect, and pair production.

4.13.3.1 Photoelectric Effect

In this process, a gamma photon collides with one of the electrons of an atom and disappears transferring <u>all</u> of its energy to the electron. The energy of the electron is now greater than its energy in its bound state, thus allowing it to break away from its parent atom to become a free electron (See Figure 13).

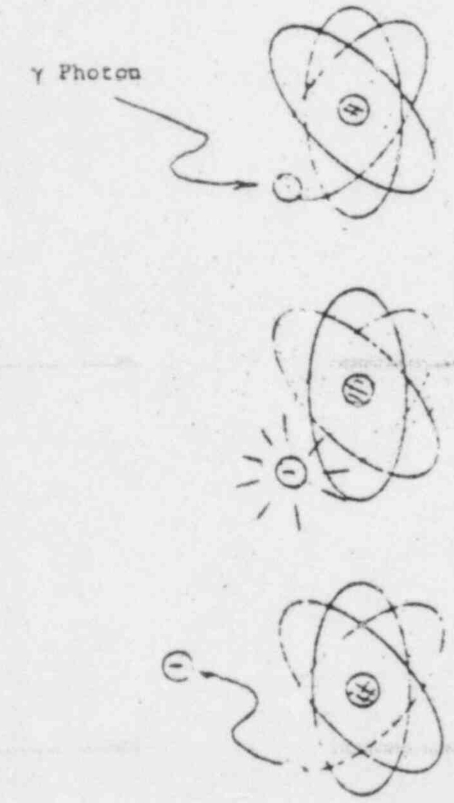


Figure 13

Photoelectric Effect

This effect occurs for gamma energies ≤ 1.0 Mev.

4.13.3.2 Compton Effect

For gamma photons of energy less than or equal to 5 Mev, it is possible for the gamma photon to be only partially absorbed by an orbital electron. The electron obtaining enough energy from the photon to break its bond will leave its parent atom to become a free electron, while the reduced energy gamma photon continues on its way (See Figure 19).

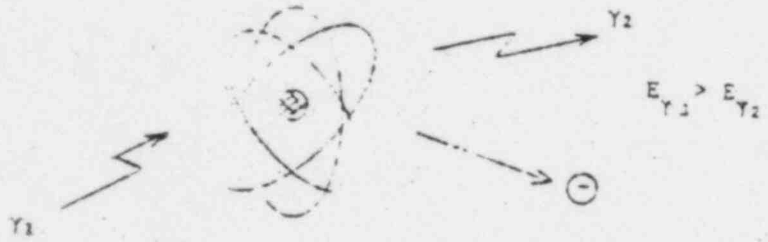


Figure 19

Compton Effect

4.13.3.3 Pair Production

Gamma photons with energies greater than or equal to 1.02 Mev have, in most cases, enough energy to penetrate the electron shells of an atom and enter the nucleus. The gamma photon then is completely absorbed by the nucleus and in its stead is produced a positron and an electron which are ejected from the nucleus (See Figure 20).

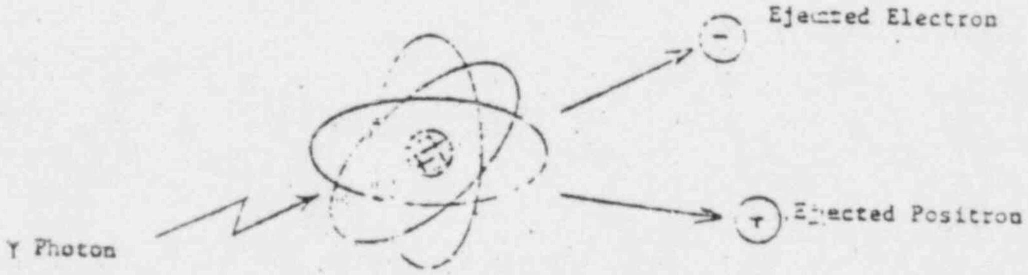


Figure 20

Pair Production

The summarization of the three processes and their probability of reaction for each energy is shown in Figure 21.

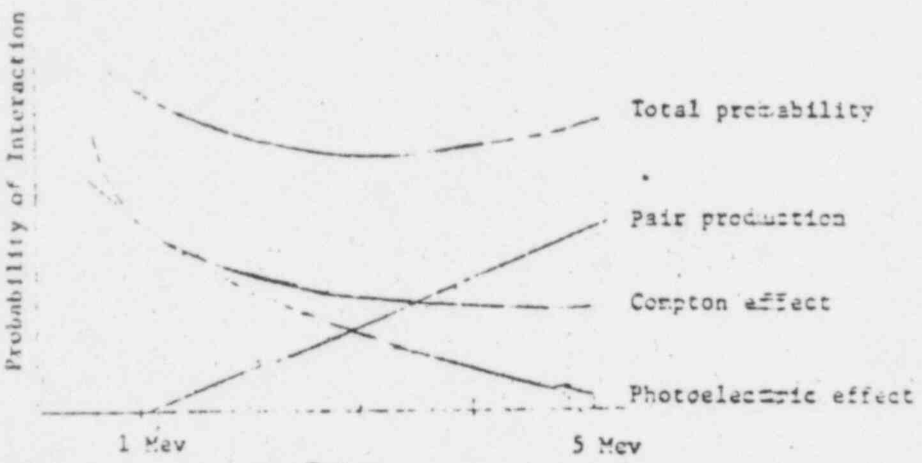


Figure 21

4.14 Nuclear Interactions

Most of us at one time or another had an opportunity to mix various chemicals together and observe the results. The chemical reactions which took place were due to the bonding or separation of atoms to form compounds, gases, etc. The results depended upon two atoms to rearrange their structure to form a more stable configuration. All of these rearrangements involved the orbital electrons or molecules of the atoms. In nuclear reactions, all of the rearrangements and adjustments occur within the nucleus.

Nuclear reactions occur when a particle, either charged or uncharged enters the nucleus of an atom. As a result, the nucleus struck is usually converted into another element or isotope. The particle entering the nucleus is usually called the bombarding or incident particle, and the nucleus being struck is called the target nucleus. If a new element or isotope is formed, it is called the compound nucleus which, in some cases, is unstable and further emits radiation called the ejected particle or photon. This process is usually expressed as follows:

Incident particle + Target nucleus —> compound nucleus

ejected particle.

The incident particles usually consist of nuetrons, protons, or helium nuclei. Particles of this type are usually accelarated at great speeds by machines such as the nuclear accelerator or cyclotron, then left to strike a target nucleus. These interactions then result in the formation of new elements or isotopes and, in some occasions, the release of a great deal of energy. For example, consider the following nuclear reaction:

$${}_7N^{14} + {}_2He^4 \longrightarrow {}_9F^{18} \dashrightarrow {}_8O^{17} + {}_1H^1$$

In this reaction, ${}_7N^{14}$ is the target nucleus, ${}_2He^4$ (alpha particle) is the incident particle, ${}_9F^{18}$ is the compound nucleus, ${}_8O^{17}$ is the recoil nucleus, and ${}_1H^1$ (a proton) is the ejected particle.

Interaction, like the one just described, can occur in one of five different ways: elastic scattering, inelastic scattering, capture reactions, capture with particle ejection, and nuclear fission.

4.14.1 Elastic Scattering

In this type of reaction, the incident particle may collide with a target nuclei and then bounce off much like collision between billiard balls. The only reaction that occurs is the transfer of kinetic energy. This means that although kinetic energy is transferred, no energy is transferred to cause nuclear instability. An example of this type of collision is the one between a neutron and a hydrogen atom. Since each has about the same mass, the only effect of the collision is that the neutron loses some of its energy and is slowed down (See Figure 22).

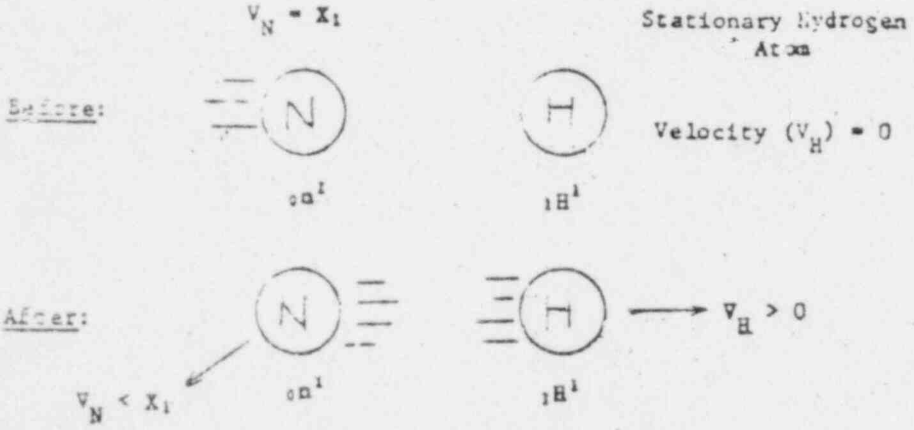


Figure 22

Elastic Scattering

4.14.2 Inelastic Scattering

Inelastic scattering is a process in which an incident particle collides with or glances off the nucleus of the target atom. In the process, it transfers energy to the target nuclei raising its energy above the ground state. The compound nucleus then returns to its ground state by emitting a gamma photon (See Figure 23).

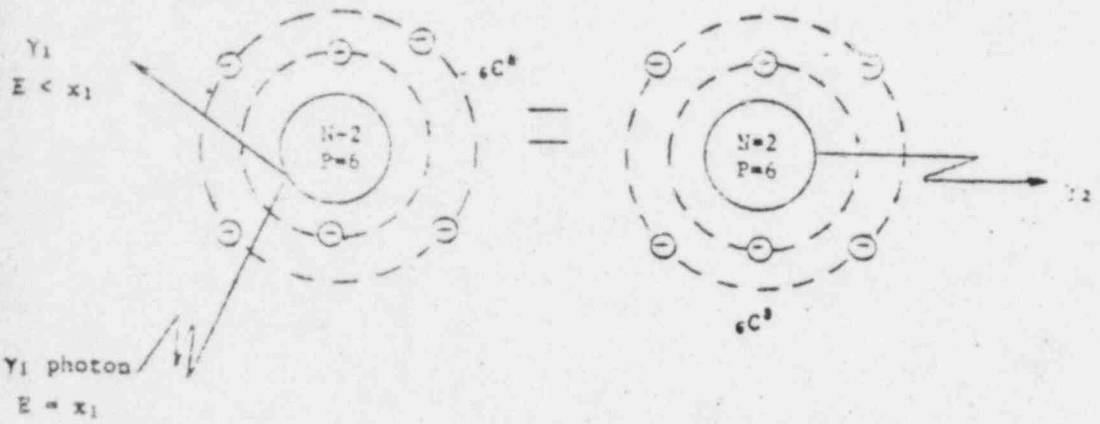


Figure 23

Inelastic Scattering

4.14.3 Capture Reactions

Unlike the inelastic scattering process, the incident particle is completely absorbed by the target nuclei. The energy state of the target nuclei is now above its ground state. It returns to ground state by the emission of a gamma photon. Note that in capture reactions, the target atom absorbs the incident particle; therefore, its nucleus is no longer the same. For example, consider the interaction between a neutron and the isotope Cd^{113}.

$$ {}_{48}Cd^{113} + {}_0n^1 \longrightarrow {}_{48}Cd^{114} + \gamma $$

Notice that the incident particle (${}_0n^1$) has been captured by the target nuclei (Cd^{113}) producing the compound nucleus (Cd^{114}) which recoiled with the ejection of a gamma photon.

4.14.4 Capture with Particle Ejection

This process is identical to the capture process except that the target nucleus is raised to a much higher energy state. The compound nucleus then recoils ejecting another particle.

For example, the interaction between the isotope O-16 and a deutron ${}_1H^2$ is as follows:

$${}_8O^{16} + {}_1H^2 \longrightarrow {}_9F^{18} \longrightarrow {}_7N^{14} + {}_2He^4$$

In this reaction, the compound nucleus (F^{18}) decays to an alpha particle (${}_2He^4$) and a recoil nucleus (${}_7N^{14}$).

4.14.5 Nuclear Fission

Fission reactions occur when a heavy nucleus is struck by some particle, the result of which is a splitting of the heavy nuclei into two lighter nuclei. In the case of the fission reaction between U-235 and a neutron, the U-235 nuclei split into two lighter nuclei called fission products with the production of two or more neutrons. (This process will be discussed in greater detail in a later section.)

4.15 Typical Reactions

The following listed reactions are those which are most commonly utilized in the nuclear field.

1) ${}_1H^2 + \gamma \longrightarrow {}_1H^1 + {}_0n^1$
   This is an example of a photo neutron source used extensively in heavy water reactors.

2) ${}_6C^{12} + {}_0n^1 \longrightarrow {}_0n^1 + {}_6C^{12}$
   This is an example of a neutron moderator (Graphite) used to slow down neutrons.

3) ${}_5B^{10} + {}_0n^1 \longrightarrow {}_5B^{11} \longrightarrow {}_3Li^7 + {}_2He^4$
   Process used in the detection of neutrons for reactor instrumentation.

4) ${}_{48}Cd^{112} + {}_0n^1 \longrightarrow {}_{48}Cd^{113}$
   Process used as absorber material in reactor control rods.

5) ${}_{92}U^{238} + {}_0n^1 \longrightarrow {}_{-1}B^0 + {}_{93}Np^{239} \longrightarrow {}_{-1}B^0 + {}_{94}Pu^{239}$
   Process used by breeder reactors for the production of fissile fuel.

4.16 Nuclear Cross Section

Nuclear cross section is defined as the probability of a nuclear reaction to occur. It is measured in terms of the effective area which a material presents to an incident particle.

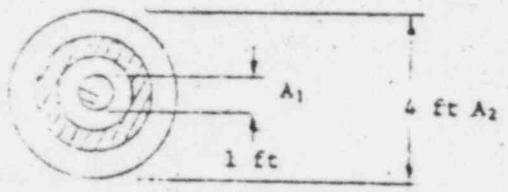
To better understand the basic concept of cross section, an analogy can be made to an analogy can be made to an archer shooting at a "bulls eye" target.

if the following assumptions are made:

1. All the arrows shot hit the target in a completely <u>random</u> manner.
2. The target consists of a central "bulls eye" with concentric rings surrounding it.

Then for each group of arrows shot, there is some probability that some of the arrows will hit the "bulls eye". Since the arrows were shot in a random manner, then the probability of hitting the target would be equal to the ratio of the area of the "bulls eye" to the area of the whole target. If we normalize the area of the target to unity (1), then the probability of hitting the "bulls eye" is just the area of the "bulls eye".

Example: Suppose an archer used the following target. What is the probability of an arrow hitting the "bulls eye"?



$$\text{Probability} = \frac{\text{area } A_1}{\text{area } A_2} = \frac{(3.14)(0.5)^2}{(3.14)(2)^2} = \frac{0.25}{4} = \underline{\underline{0.0625}}$$

Then if the archer shot 200 arrows, the probable number of arrows which would hit the "bulls eye" is

$$\text{Probable number of hits} = (0.062)(200) = \underline{\underline{12}}$$

In reactor physics we will be most interested in the probability of scattering or absorption occurring when a neutron passes through certain matter. In this case, we will be comparing the effective area of the target matter to the area of the neutron. For example, if a beam of neutrons (I) per cm^2/sec were to strike a wall of matter one atom thick and containing N_o number of atoms per cm^2 (See Figure 24), the nuclear cross section would be derived from the following equation:

$$\text{cross section } (\sigma) = \frac{\text{number of reactions (C)}}{(N_o)(I)}$$

<u>Probability of Interaction</u>

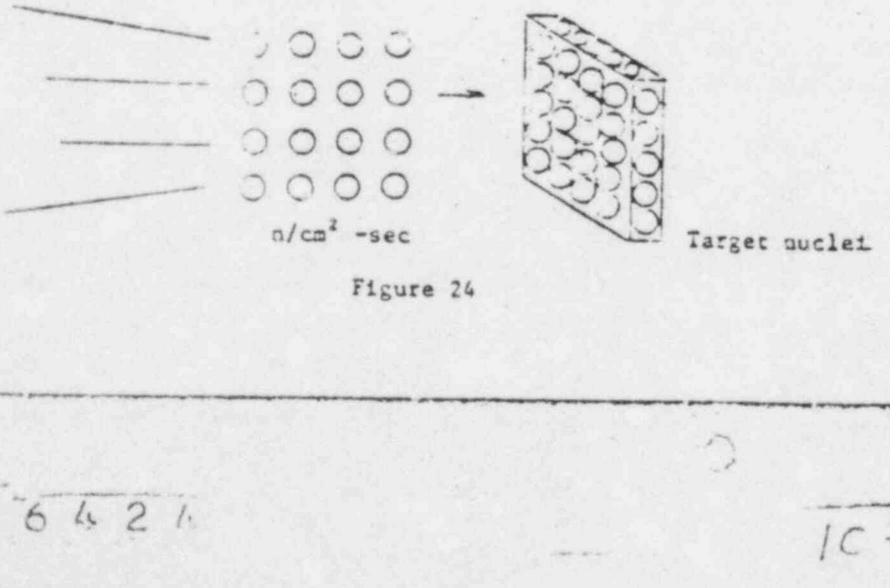


Figure 24

N 6 4 2 7

IC P3 10 8 6

From this relationship we can see that the fraction C/I is a measure of the fraction of neutrons which strike the wall and react with its nuclei. Dividing this by the number of nuclei per square cm will give us the square cm per nucleus that takes part in the reaction

Since the diameter of atom is approximately 10^{-12} cm, which is very small, the unit of cross section (σ), called a <u>barn</u>, has been established. One <u>barn</u> equals 10^{-24} cm^2. In the previous example, we defined the wall of material as one atom thick and the cross section as the square cm/nucleus that took part in the reaction. This type of cross section is called the <u>microscopic cross section</u> having units of (cm^2). If we would add depth to our wall (that is, <u>add volume</u>), then the cross section would be equal to the following equation:

<u>cross section</u> = (number of nuclei/cm^3)(σ) and is called the <u>macroscopic cross section</u> (Σ) having units of (cm^{-1}).

In the case of neutrons interacting with material in a reactor, we would like to know what effects can be expected from such interactions. Therefore, we define materials as having a scattering cross section (σ_s) or an absorption cross section (σ_a). For example, if we wanted to surround a reactor core with some material which would effectively reflect neutrons back into the core, we would want a material which has a high scattering cross section. On the other hand, if we wanted to find some material for use as reactor control rods to absorb neutrons, we would look for a material with a high absorption cross section.

In conclusion, we can say that nuclear cross section is a probability of different interactions; and, therefore, the total probability or total cross section of any material is the sum of these events as shown by this equation:

$$\sigma_{total} = \sigma_s + \sigma_a$$

Examples of cross sections for various materials are shown in Table 1.

4.17 <u>Moderation</u>

In a nuclear reactor we are concerned with the fissioning of uranium fuel. Neutrons born as a result of fission have a very high kinetic energy and are called fast neutrons. However, slow neutrons or low kinetic energy neutrons have a better probability of causing fission. Therefore, if some material is placed around the fuel which will attenuate the fast neutrons to thermal energy ranges with a small probability of absorption, then we say this material is a good moderator. The quality of a material as a moderator can be determined from the moderator ratio which equals:

$$\text{moderator ratio} = \frac{\Sigma_s}{\Sigma_a}$$

where, Σ_s = scattering cross section

Σ_a = absorption cross section

THERMAL NEUTRON CROSS-SECTIONS

Material	σ_s (barns)	σ_a (barns)	$M = \frac{\sigma_s}{\sigma_a}$
Aluminum	1.4	0.24	5.83
Beryllium	7.0	0.10	70.0
Bismuth	9.0	0.034	264.0
Boron	4.0	755.0	0.00529
Cadmium	7.0	2450.0	0.00285
Carbon	4.8	0.0034	1411.0
Deuterium	7.0	0.0005	14000.0
Helium	0.8	0.007	114.0
Hydrogen	38-100	0.33	115.0-303.0
Iron	11.0	2.62	4.198
Lead	11.0	0.17	64.7
Nitrogen	10.0	1.88	5.319
Oxygen	4.2	0.0002	21000.0
Sodium	4.0	0.53	7.54
Uranium (natural)	8.3	7.68	1.08
U-235	8.2	650.0	0.0126
U-238	8.2	2.80	2.92
Zirconium	8.0	0.185	43.2
*Stainless steel	—	—	—

*Contains a small amt. of nickel, which contains Co 60, which has a σ_a of 30 barns.

<u>good moderators</u>	<u>good structural materials</u>
deuterium	aluminum
beryllium	zirconium
carbon	iron or steel

Table 1

Examples of moderator ratios for various materials can be found in Table 1.

4.18 Conclusion of Nuclear Physics

We have now covered a great deal of information in a very short time. It is obvious that almost any of the individual sections which we have covered could be a major topic in itself, and to learn every detail would consume a great deal of time and effort. However, it is not important that every detail be learned. What is important is that the fundamental principles are understood. Having learned these principles, it is now possible to deal more effectively with the problems associated with reactor kinetics and reactor operations.

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REACTOR PHYSICS

5.0 Introduction

In the previous chapters on Basic and Atomic Physics some insight was developed into the elemental properties of matter. For instance, it was learned that all matter is composed of the same kind of building blocks or particles, that there is a mysterious thing called binding energy which is a measure of the amount of energy needed to take these particles apart. It was also learned that elements have a characteristic property called a nuclear cross section and that there are different kinds of cross sections displayed by any one element.

It shall be the purpose of this section to show that with the proper consideration given to the characteristic properties of the various materials one can put together a critical mass capable of sustaining a chain reaction. It is intended that some of the seemingly unrelated facts and phenomena presented in the preceding chapters might be brought into focus, that the subject of basic physics might be blended with a bit of reactor physics and engineering and the end result will be a nuclear reactor.

5.1 Fission and the Fission Process

To outline the chapter at hand one can define an area of interest by stating that when conditions are right fission will occur and as a result one obtains fission fragments, the release of neutrons, and most important the release of energy.

The first topic to be explored is the fission process. Fission will be defined. The type of elements one would seek out and call fissionable material will be investigated as will some of the consequences of fission.

Next, the chain reaction will be considered along with what kind of properties material must exhibit if they are to be useful in sustaining and controlling a chain reaction.

Continuing into the chapter, the concept of criticality will be considered in terms of an infinite reactor followed by a description of the finite reactor.

Finally, this chapter will deal with the approach to criticality and then will become very practical and an actual critical experiment will be described. Having devoted enough space to preliminaries, the chapter begins with attention being directed to the basic material of any reactor system, the fuel.

5.2 Fissionable Materials

When a neutron is captured in the nucleus of certain heavy elements enough energy is imparted to the nucleus to render the new isotope unstable. One result is that the nucleus releases the excess energy by actually breaking apart and forming two different radioactive isotopes called fission products. The heavy elements so affected by encountering a neutron are called fissionable nuclei. Some examples of fissionable material are Thorium-232, Uranium-233, Uranium-235, Uranium-238, and Plutonium-239

A further subdivision of this group of elements can be made by considering the energy of the neutron which encounters the nuclei. If neutrons with energy of 0.025 ev (thermal neutrons) will cause fission to occur, then the material is called fissile material. Such is the case with U-233, U-235 and Pu-239. Of this group of fissile materials, only U-235 is found in nature. The two remaining elements, Th-232 and U-238 are said to be fertile materials because they can be made fissile by the following reactions:

$$\begin{tikzcd}
{}_{90}\mathrm{Th}^{232} + {}_{0}n^{1} \arrow[r] & {}_{90}\mathrm{Th}^{233} + \gamma \arrow[d, "22.1\text{ min.}"] \\
 & {}_{91}\mathrm{Pa}^{233} + {}_{-1}\beta^{0} \arrow[d, "27.4\text{ days}"] \\
 & {}_{92}\mathrm{U}^{233} + {}_{-1}\beta^{0}
\end{tikzcd}$$

$$\begin{tikzcd}
{}_{92}\mathrm{U}^{238} + {}_{0}n^{1} \arrow[r] & {}_{92}\mathrm{U}^{239} + \gamma \arrow[d, "23.5\text{ min}"] \\
 & {}_{93}\mathrm{Np}^{239} + {}_{-1}\beta^{0} \arrow[d, "2.35\text{ days}"] \\
 & {}_{94}\mathrm{Pu}^{239} + {}_{-1}\beta^{0}
\end{tikzcd}$$

Figure I

Fertile Material Made Fissile

An upper limit has been set on the neutron energy involved with fissionable nuclei of about 10 Mev. Although there are heavy nuclei such as Pb-208 which will fission with extremely fast neutrons, (20 Mev in this specific instance) for practical purposes they are considered non-fissionable.

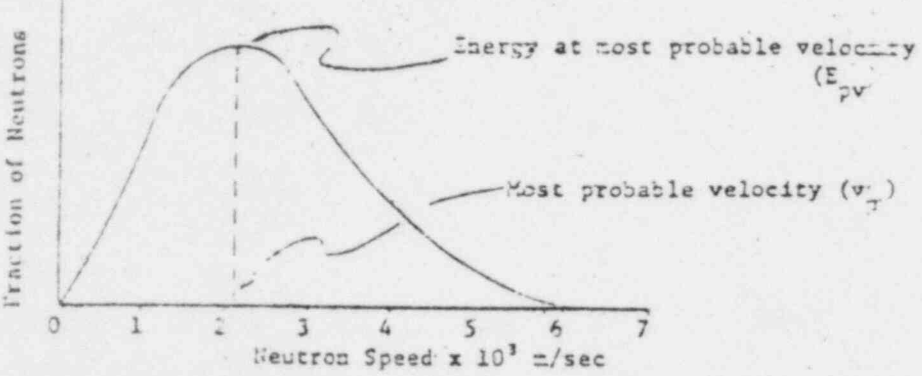

Figure 2

Maxwellian Distribution of Thermal Neutrons

At this point the term thermal neutron will be considered in more detail. If one considers that a closed cylinder containing neutrons would be analogous to a cylinder of gas in that the molecules of gas lose and gain velocity as they collide with each other and the walls of the cylinder, then it can be appreciated that the neutrons so contained at a constant temperature would not have one single velocity, but would have a distribution of velocities from zero up to thousands of meters per second. However, the energy of the neutrons at the most probably velocity would be

$$E_{pv} = \frac{1}{2} mv_p^2$$

and relating this energy to a constant 20°C temperature by the use of Boltzmann's constant $k = 1.38 \times 10^{-16}$ erg/°K one can write

$$\frac{1}{2} mv_p^2 = kT$$

$$v_p^2 = \frac{2kT}{m}$$

$$v_p = (\frac{2kT}{m})^{\frac{1}{2}}$$

$$= (\frac{2 \times 1.38 \times 10^{-16} \times 293}{1.67 \times 10^{-24}})^{\frac{1}{2}}$$

$$= 2.2 \times 10^5 \text{ cm/sec}$$

$$= 2.2 \times 10^3 \text{ meters/second}$$

Now using this most probable velocity in the expression for the energy at the most probable velocity

$$E_{pv} = \frac{1}{2} mv_p^2$$

one finds that

$$E_{pv} = \frac{1.67 \times 10^{-24}(2.2 \times 10^{3})^2}{2} \frac{g\text{-}cm^2}{sec^2}$$

Knowing that 1 erg = 1 dyne-cm = $\frac{1 \text{ gm-cm}}{sec^2}$ cm and 1 m = 10^2 cm, then

$$E_{pv} = 4 \times 10^{-14} \text{ ergs}$$

and recalling that 1 ev is 1.6×10^{-12} ergs then

$$E_{pv} = \frac{4 \times 10^{-14}}{1.6 \times 10^{-12}} = 0.025 \text{ev}$$

thus, it can be said that thermal neutrons have velocities and consequently energies no greater than the molecules of their environment. By extension then, a fast neutron has greater velocity and consequently higher energy. Neutrons in the Mev range are considered fast. It might be noted here that a 1 Mev neutron has energy associated with approximately 10^{10} °K.

Having thus defined the thermal neutron one can write an expression which describes what happens when a thermal neutron encounters the nucleus of a fissile material and for the purposes of this example U-235 will be used.

$${}_0n^1 + {}_{92}U^{235} \longrightarrow {}_{92}U^{236*} \longrightarrow {}_{Z_1}FP^{A_1} + {}_{Z_2}FP^{A_2} + E + 2.5\,{}_0n^1$$

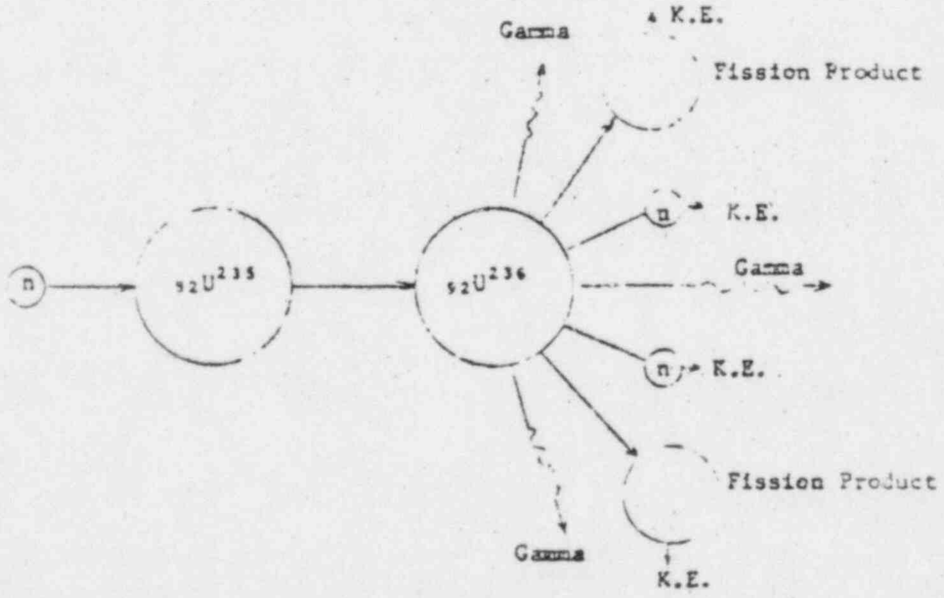

Figure 3

The Fission Process

Notice that it is really U-236 (a very unstable isotope) which fissions; however, this occurs so rapidly that one can consider the absorption of the thermal neutron and the fission event practically simultaneous. The fission products (FP) which are formed and the average number of neutrons produced by the fissioning of U-235 are subjects of future sections. The next topic to be discussed will be the energy considerations in the fission process.

5.3 Energy of Fission

From the power plant operations point of view, the most important consequence of the fission process must be the release of energy which accompanies each fission event. To get some idea of the amount of energy released during a fission event let attention be focused upon the binding energy per nucleon versus number of nucleons curve (Basic Nuclear Physics, Page 4-5). If one were to estimate the binding energy per nucleon (BE/A) for U-235 and In-117, one should get numbers on the order of 7.6 and 8.5 Mev per nucleon respectively (see points A and B in Figure 4).

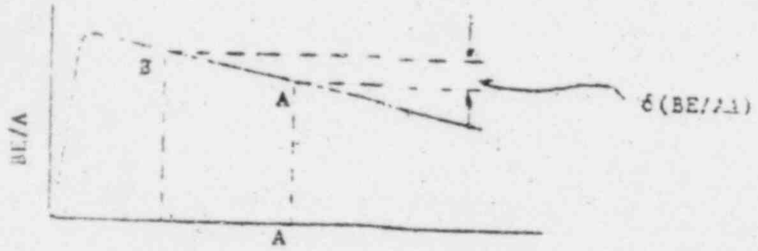

Figure 4

Binding Energy Curve

Let it then be assumed that as the result of the fission of U-235 the two fission products formed were Indium-117 and Indium-118. Then the binding energy released as a result of the fission event would be

$$B.E. = \left(\frac{8.5\ \text{Mev}}{\text{nucleon}}\right)(117 + 118\ \text{nucleons}) - \left(\frac{7.6\ \text{Mev}}{\text{nucleon}}\right)(2235\ \text{nucleons})$$

$$\approx 211\ \text{Mev}$$

which is very close to the tabulated amount of energy per fission event in Table 1. The majority of the fission energy is carried off as kinetic energy of the fission products which will be passed on to the reactor components in the form of heat energy. The neutrons which are released during fission carry away about 5 Mev as kinetic energy. Prompt gammas and fission product decay beta and gamma account for about 22 Mev of the total. The following table gives an account of the fission energy.

Table 1

Distribution of Fission Energy

Particle	Energy (Mev)
Fission Products	168
Fission Product Decay	
β	8
γ	7
Neutrinos	12
Released Neutrons	5
Prompt Gammas	7
Total	207

If so much energy, one might ask, is available to be released when a zero energy neutron enters a U-235 nucleus, why don't all heavy nuclei fission with thermal neutrons? The answer can be found in the way in which the encountered neutron changes the binding energy per nucleon of the nucleus with respect to a threshold energy for fission. A list of important nuclei and their thresholds for fission is shown in Table 2. Also shown in the same table is the added binding energy of the last neutron in the target nucleus.

Table 2

Fission Threshold of Some Fissionable Materials

Target Nucleus	Threshold Energy (Mev)	Binding Energy (Mev)
Th-233	7.5	5.4
U-239	7.0	5.5
U 236	6.5	6.8
U-234	6.0	7.0
Pu-240	5.0	6.6

It can be seen then that a zero energy or thermal neutron cannot impart enough energy to reach the threshold of U-239 fission falling short by 7.0 - 5.5 Mev. Thus, the additional energy must be supplied to the target nucleus by the kinetic energy of the incoming neutron. A neutron with this much energy must be removed from the thermal neutron category and placed in the fast neutron group. This is in agreement with the contention in Section 5.2 that U-238 is not a fissile material but is fissionable. By the same analysis it is seen that, as previously stated, U-235, U-233, and Pu-239 are indeed fissile materials or they do fission with zero energy neutrons.

Before leaving the topic of fission energy, it would be interesting to make a calculation which would give more meaning to the fission energy release number. Assuming that a fission product production-decay equilibrium had been reached, then all the energy per fission event is recoverable except that which is carried off by the neutrinos or

$$E = (207-12)\ \text{Mev per fission event}$$

$$= \left(\frac{195\ \text{Mev}}{\text{fission}}\right)\left(\frac{1.6 \times 10^{-6}\ \text{erg}}{\text{Mev}}\right)\left(\frac{10^{-7}\ \text{watt seconds}}{\text{erg}}\right)$$

$$= 3.2 \times 10^{-11}\ \text{watt seconds/fission}$$

The student should show that it would require 1 gram of U-235 per day to yield a power of approximately 1 Megawatt.

5.4 Fission Product Distribution

Having dealt with the most important resultant of the fission event, namely the release of about 200 Mev per event, attention is now directed to the vehicle which transfers most of the 200 Mev to the reactor surroundings - the fission product.

Because of the highly radioactive nature of fission products, great effort is expended to contain the fission products within the fuel elements by providing the elements with a cladding material which stays in tact. Although the fission products have much kinetic energy, it must be remembered that they are quite massive; consequently, their energy is deposited in the fuel element material within 10^{-3} centimeters of travel. In addition to being highly radioactive, the fission products are highly ionized (See Page 4-14 - Basic Nuclear Physics). It would not be too difficult to imagine an atom with the kinetic energy which the fission products possess moving away from some of its orbital electrons, virtually leaving them behind in the rush, and this is precisely what happens to the fission products. Incidentally, the specific ionization of these products is much greater than specific ionization of alpha particles; thus making fission products very useful as neutron detectors as in a fission chamber.

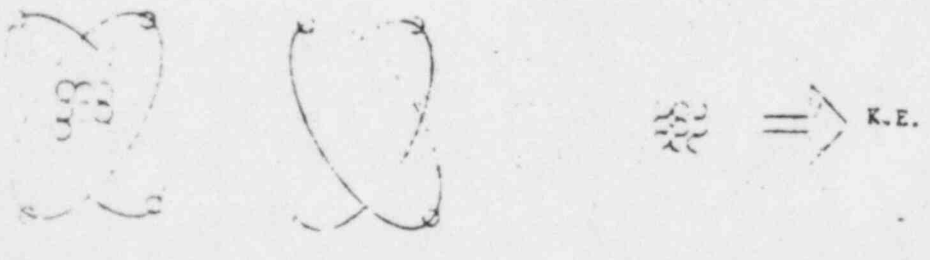

Figure 5
Fast Moving Fission Product is Stripped of its Electrons

Surprisingly enough when a fission event occurs there is very little tendency for the fissioning nucleus to split into equal or near equal parts. Assuming that U-235 is the fissionable material, then upon absorbing a neutron the number of nucleons becomes 236 or A=236. If the fission process were symmetric, the most probable yield of fission products would be two new elements each with A = 118. As observed from Figure 6, which shows a semilogarithmic plot of fission yield as a function of mass number A, the abundance of elements with A = 118 is less than 0.02% of the total yield, or this type of splitting occurs only once in 10,000 events. It can also be seen by the pairs which occur at mass numbers 95 and 140 that the most probable split is in a 3 to 2 ratio. The other two fissile materials have fission product distribution curves very similar to Figure 6.

At this point one could rewrite the fission expression shown in general terms on Page 5-4 in the following manner:

$$^{1}_{0}n + {}^{235}_{92}U \longrightarrow {}^{236*}_{92}U \longrightarrow {}^{140}_{54}Xe + {}^{94}_{38}Sr + (2)\ {}^{1}_{0}n + 207 \text{ Mev}$$

When the fission fragments are formed they are neutron rich, that is, the neutron to proton ratio is too high to render them stable; thus, as pointed out in Basic Nuclear Physics, Page 4-6, the unstable isotope will approach the stability curve by emitting beta particles usually in a series of decays. One example would be

$$\begin{tikzcd}
{}^{140}_{54}Xe \arrow[r, "16 \text{ sec.}"'] & {}^{140}_{55}Cs + {}^{0}_{-1}\beta \arrow[d, "66 \text{ sec.}"] \\
& {}^{140}_{56}Ba + {}^{0}_{-1}\beta \arrow[d, "12.8 \text{ days}"] \\
& {}^{140}_{57}La + {}^{0}_{-1}\beta \arrow[d, "40 \text{ hrs.}"] \\
& {}^{140}_{58}Ce + {}^{0}_{-1}\beta
\end{tikzcd}$$

Thus, by the emission of four beta particles (or the conversion of four neutrons to four protons) the radioactive fission product Xe-140 reaches the stable isotope Ce-140.

Up to this point the consequences of the fission process which have been considered will be condensed as follows:

a. Energy - each fission event yields about 200 Mev.

b. Fission Products - each fission event yields two fission products with a distinct distribution depending upon the fuel material. Also, the kinetic energy of these products represents the greatest portion of the energy release.

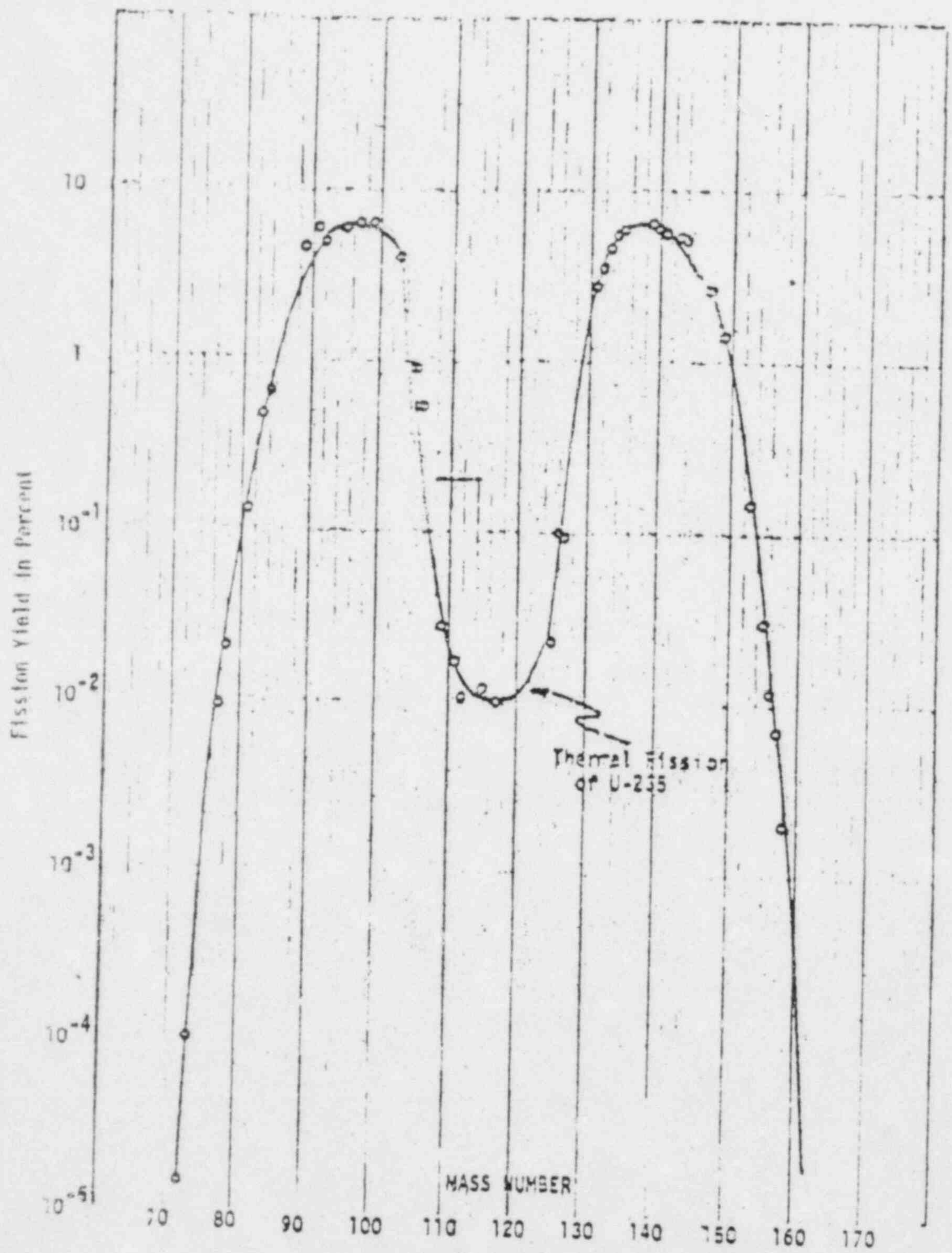

Figure 6

Fission Product Distribution for Thermal Fission of U-235

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c. Release of particles - the fission products release beta particlas because they are unstable isotopes and each fission event releases an average 2.5 neutrons.

Still to be considered is the fact that prompt gammas are released and this is done in the chapter on Shielding. Another consideration is the poison effect certain of the fission products have upon the reactor and this will be covered in the chapter on Reactor Operations. Finally, the neutrons which are released merit much more consideration than they have been given in this section. In subsequent sections the fate of these neutrons will be investigated. Also, time will be spent in showing that most neutrons are released simultaneously with the fission event; but there is a small fraction of neutrons which become available after the fission event occurs. These delayed neutrons come from the fission products and play a most important role in reactor control. The next topic to be considered will be the fate of the prompt neutrons which are released during fission.

5.5 The Controlled Chain Reaction

In perhaps an over simplified statement, one can say that the neutrons which have been released via fission events are going to be used to perpetuate fission events. With a few short sections then it is hoped that a feel for the physical environment necessary for this perpetuation can be developed. The necessary environment can be broken down into sections as follows:

a. Fuel or critical mass
b. Moderation
c. Reflection
d. Neutron absorbing material
    1. Structural
    2. Control

This list comprises the minimum essentials for a controlled chain reaction and by definition then, the essentials of a reactor.

5.6 Critical Mass

If again, one were to inquire why, with the supply of fissile material in the bowels of the earth, does not fission occur spontaneously? The answer is that some spontaneous fission does occur; however, not to any appreciable amount because the competition for the neutrons is very keen. Since U-235 occurs only one part in 140 in natural uranium, the rest consisting mainly of U-238, there is fierce competition for neutrons right in the uranium orewithout even considering the other elements in the surrounding rocks and minerals with their "hostile" cross sections for interactions other than the fission event. Quite clearly, then, the limiting factor is one of concentration.

The subject of cross section has come up again in this section and there is need for a little expansion on the material that was presented in Basic Nuclear Physics, Page 4-24, Nuclear Cross Sections. The total cross section σ_{total} was defined as the sum of the scattering cross section and the absorption cross section or $\sigma_{total} = \sigma_s + \sigma_a$. To show where the probability of a fission event occurring fits into this cross section scheme, it can be said that $\sigma_a = \sigma_c + \sigma_f$ where σ_f is the fission cross section and σ_c is the radiative capture cross section so that now

$$\sigma_{total} = \sigma_s + \sigma_c + \sigma_f$$

A list of various cross sections for the fissionable fuels is shown in Table 3.

Table 3

Nuclear Fuel Cross Sections

Nucleus	σ_a	σ_c	σ_f	σ_s
U-233	581	54	527	
U-235	683	106	577	15
Pu-239	1029	287	742	9.6
U-238	2.71	2.71	—	8.3
U (natural)	7.7	3.5	4.2	8.3

All data is given for 0.025 ev neutrons. Observe that only about 85% of the absorptions in U-235 lead to fission (577/683).

Assuming now that an unlimited amount of enriched U-235 (high concentration, low contaminant) is available, what would be the fate of neutrons produced via fission in a mass of U-235 which would be assembled? Two things could happen in such an assembly (it is further assumed that the reaction has not reached uncontrolled proportions). First, the neutrons could be absorbed within the material of the assembly leading to the production of more neutrons. This is quite obviously a function of the volume of the assembly. Secondly, the neutrons could escape from the assembly and be lost to the system and this is a function of the surface area through which their escape must be made.

Two variables have now been identified in the problem of putting together a critical mass, namely, concentration of fuel and the geometry or shape which the assembly will take. The first factor has been touched upon by postulating enriched fuel but carried a step further by letting the fuel be a constant enrichment. This leaves a decision to be made as to what shape the assembly shall be given. Thinking about the situation a bit leads one to conclude that if the production to loss ratio of neutrons could be maximized, one would be taking a step in the right direction. Well from prior discussion, it is clear that production/loss α volume/surface area and now it looks like the decision might be guided by a small investigation into the volume/surface area, (V/SA), of some common geometric figures. Given a cube as shown in Figure 7, with a=r then the volume is $V = r \times r \times r = r^3$ and the surface area is $SA = 6\ (r \times r) = 6r^2$ and finally $V/SA = r^3/6r^2 = r/6 = 0.167r$.

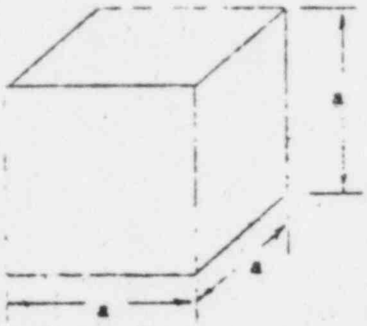

Figure 7

A Cubical Assembly

Next, consider a right cylinder with $r = h$, then $V = \pi r^2 h$, $V = \pi r^3$ and the surface area is $SA = 2(\pi r^2) + \pi dh = 2\pi r^2 + \pi(2r)\ r = 2\pi r^2 + 2\pi r^2 = 4\pi r^2$ and it follows then that $V/SA = \pi r^3/4\pi r^2 = r/4 = 0.250r$. (See Figure 8)

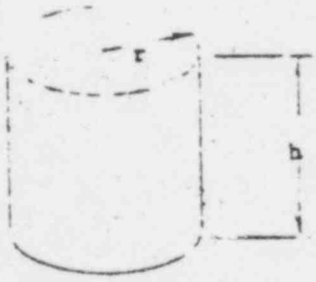

Figure 8

A Cylindrical Assembly

A distinct improvement over the V/SA of the cube but one more case will be looked at before making a decision. Given now a sphere of radius r, then the volume of Figure 9 is $V = 4\pi r^3/3$ and the surface area of the sphere is $SA = 4\pi r^2$ making the $V/SA = \frac{4}{3}\pi r^3/4\pi r^2 = r/3 = 0.333r$ or clearly indicating that of the three geometric forms considered the best V/SA, consequently, the best production/loss factor is to be obtained from a spherical assembly. It is left to the good faith of the student to accept the sphere as having the best V/SA of all practical geometric forms without a rigorous calculus proof.

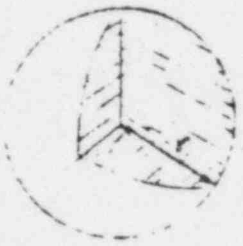

Figure 9

Spherical Assembly

Upon accepting this as fact then, the assembly of the critical mass takes on a spherical form with a constant enrichment of fuel. If it is now assumed that one starts with a very small sphere then it is conceivable that the neutrons produced in the enriched U-235 mass can escape from the system before they cause another fission. As the radius of the sphere of U-235 is increased the losses increase as r^2 and the production increases as r^3; thus, the overall effect of increasing r is to decrease losses by 1/r. Clearly then a point is reached where for some radius r_c the production of neutrons will equal the losses and for each neutron that is released by fission, it goes on to release another neutron which will follow suit, thus, one has a critical mass which can support a sustained chain reaction. One might approach the question of critical mass from another point of view and select a given size sphere and proceed to change the concentration of a water and U-235 slurry within the sphere and get the same results as above with this change of variable.

As the average student who has been subjected to a good deal of molding, shaping, brainwashing, and pedantic browbeating might suspect, the above is an over-simplification. Section 5.3, Energy of Fission, hints at the next consideration. The fact that fission neutrons are "born" fast or with high energies and that U-235, like all fissile materials, has a fission cross section for thermal neutrons, pretty clearly brackets the next section's discussion, that of moderation.

5.7 Moderation

The subject of moderation has been approached in Basic Nuclear Physics, Page 4-26, and one recalls from this section that moderation is the function a material performs in a reactor when it takes fast neutrons and slows them down to thermal energies thereby increasing the probability of a fission interaction occurring between the neutron and the fissile fuel nuclei.

Let a quantity ξ be defined as follows:

$$\xi = \overline{\Delta(\ln E)}$$

$$= \ln E_1 - \ln E_2$$

$$= \ln E_1/E_2$$

where E_1 is the neutron energy before collision and E_2 is the energy of the neutron after collision with the moderating material. Then ξ can be described as the average logarithmic energy decrement per collision. Notice also that

$$e^{\xi} = \frac{E_1}{E_2}$$

A mathematical treatment beyond the scope of this course would show the following approximation:

$$\xi = \frac{2}{A + \frac{2}{3}}$$

which is accurate to within 1% for $A > 10$ and also shows that ξ is independent of the energies involved but is an inverse function of the mass number A. To help bring this notion into focus, consider a collision between a high velocity marble (fast neutron by analogy) and a bowling ball (nucleus with large A by analogy). Clearly there will be little velocity imparted to the bowling ball by the marble. The marble would leave the collision with about the same velocity i.e., energy as it brought into the collision or ξ of the bowling ball is very low. On the other hand, when a collision is envisioned between two marbles of equal size, then one witnesses a greater transfer of energy between the two equal mass particles and as a result ξ is larger. The conclusion, then, to be drawn is that one of the factors to be considered when selecting a moderator material is how closely the mass of the nucleus of the moderator approximates the mass of the neutron. An interesting number can be calculated by first calculating ξ for a number of materials (See Table 4, Page 5-15); then assume that one has a neutron population of 2 Mev neutrons and wishes to thermalize these neutrons (0.025 ev). The average number of collisions to thermal ($\overline{C}_{th}$) is:

$$\overline{C}_{th} = \frac{\ln E_1/E_2}{\xi}$$

$$= \frac{\ln \frac{2 \times 10^6}{0.025}}{\xi} = \frac{18.2}{\xi}$$

A list of $\overline{C}_{th}$ values for various materials is also included in Table 4.

Second and third factors to be considered in defining a good moderator are the scattering and absorption cross sections of the material. A material with a high ξ but a low Σ_s would be of very little value as would a material such as boron which has a very high absorption cross section. Thus, it can be said that a material's worth as a moderator is directly related to ξ and Σ_s and inversly related to Σ_a.

Referring back to Table I, Page 4-27, Basic Nuclear Physics, one obtains a value of moderating ratio for bismuth of 264. This value was obtained by considering only the scattering and absorption cross sections for bismuth.

Table 4

Slowing Down Data

Element	A	ξ	$\bar{C}_{th}$
Hydrogen	1	1.000	18
Water	18	0.920	19
Deuterium	2	0.725	25
Heavy Water	20	0.590	35
Helium	4	0.425	43
Lithium	7	0.268	67
Beryllium	9	0.209	86
Carbon	12	0.158	114
Oxygen	16	0.120	150
Sodium	23	0.0825	221
Uranium	238	0.00838	2172

When the moderating ratio is redefined to include the average logarithmic energy decrement, moderating ratio becomes $M' = \xi\Sigma_s/\Sigma_a = \xi\sigma_s/\sigma_a$ and for Bi-209

$$M' = \left(\frac{2}{209 + \frac{2}{3}}\right)(264) = 0.0095\ (264) = 2.51$$

which would be a rather discouraging figure to choose for a moderating material. Table 5 shows some common moderator materials with their respective $\xi\Sigma_s/\Sigma_a$ moderating ratios.

Table 5

Slowing Down Property

Moderator	Moderating Ratio
Water (H_2O)	72
Heavy Water (D_2O)	12,000
Helium (atmospheric T and P)	83
Berryllium	159
Carbon	170

Scattering cross sections are assumed constant from 1 to 10^5 ev.

Although heavy water has an excellent moderating ratio, the price per pound of D_2O was $28.50 as of April 1968.

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5.8 Reflection

The reflector area of a reactor would in general cover the outside area of the core; thus, it is sometimes referred to as a blanket. Refer back to the critical mass which was assembled as a "bare sphere". This type of reactor is affectionately known as a Godiva Reactor (for obvious reasons). If the bare sphere were to be blanketed with a layer of material which had properties similar to moderator materials (high Σ_s), then after scattering around in the reflector medium there would be a 50-50 chance that a neutron, which had just been lost to the bare system, would scatter back into the critical mass. With a region like this surrounding the critical mass, it is obvious that the loss figure goes down and the production figure goes up, and consequently, one can decrease the concentration of fuel or the critical size of the sphere. Any time one can effect a change like this in a reactor one has just saved many dollars in fuel costs. So primarily, a reflector is a savings device and does so in the following ways:

1. The refelctor reduces core leakage.
2. The reflector allows a smaller core, thus less fuel, to be constructed.
3. The reflector flattens the thermal flux, thus a more even flux density.

5.9 Neutron Absorbing Material

As more and more material is added to the original bare sphere making it more practical in nature, one finds that foreign materials, that is, materials which have very little to contribute to the continuity of the chain reaction, have to be added to hold the various parts of the assembly together. These structural materials have the effect of robbing neutrons from the chain reaction because of their absorption cross sections which do not lead to fission. Since they are a necessary evil in performing functions like removing heat, cladding fuel, shielding from radiation, containment, etc., materials are usually chosen with low absorption cross sections. However, to obtain certain desirable characteristics like structural strength and heat capacity one will overlook a relatively high Σ_a occasionally. The net result is that this type of neutron absorber increases the size of the critical mass if a neutron economy is to be maintained.

A second class of absorber material is the control material. As was postulated in Section 5.6, the assembly was taken critical by one of two means. Either the radius of the sphere was changed until r_c was reached, or the concentration of the fuel was changed via a fuel liquid moderator slurry. From an engineering standpoint, neither of these methods won wide acceptance. The more acceptable method is to introduce, purposely, this time, material with a high absorption cross section Σ_a such as boron or cadmium to upset the neutron economy allowing one to attain and depart from critical at will. This deliberate introduction of "poison" material is done in two general ways as follows:

a. Lumped poisons contained in control rods which are inserted and removed by electromechanical means.

b. Distributed poisons which are in solution in the coolant material and perform their control functions by changes of concentration.

More will be said about control in subsequent chapters.

In summary, then it is seen that from the time neutrons are born with fast energies until the time they are absorbed or "die" at thermal energies, there are a number of phenomena which occur. A careful treatment of the neutron economy as the bare sphere became more sophisticated allowed a reactor to emerge. The next section goes back to the fissioning fuel nuclei again and with some simplifying assumptions sets up conditions for criticality.

5.10 Critical Infinite Reactor

To start simplifying matters, let it be assumed that the reactor system under discussion is an infinite system. By making this assumption, one can say that the only losses which occur are absorption losses within the system as obviously no neutrons can escape from an infinite system.

The object of the subsequent sections will then be to devise a mathematical scheme to describe the physical condition which is referred to as criticality. As previously indicated, a critical reactor is one in which neutrons perpetuate themselves. Putting it another way, one can look at the numbers of neutrons in successive generations, i.e., the n^{th} generation, the n + 1 generation, etc., and if the ratio of two successive generations of neutrons is equal to unity then the reactor is critical. Let the ratio between generations be referred to as k and since the system under consideration is an infinite system let it be k_∞. Then

$$k_\infty = \frac{\text{number of neutrons in the n + 1 generation}}{\text{number of neutrons in the } n^{th} \text{ generation}}$$

and k_∞ shall be called the infinite multiplication factor. Another way of expressing k_∞ is

$$k_\infty = \frac{\text{production}}{\text{absorption}}$$

A series of four terms will be developed which will account for all the production and absorption of neutrons which occurs in the infinite system under consideration. Two further assumptions made in the following development are that all fission neutrons are born fast and that the fuel is a mixture of U-235 and U-238. The product of these four terms, since they account for all of the neutrons, will indicate whether or not the system is critical. This expression is the four factor equation and is written

$k_\infty = \eta \epsilon \rho f$ Should be $k_\infty = \eta \epsilon p f$

and its development follows.

5.11 The Reproduction Factor η

It has been mentioned many times in previous sections that about 2.5 neutrons are released in each fission event. This figure will be assigned the Greek letter ν (nu) and will be defined as follows:

$$\nu \equiv \frac{\text{number of fast neutrons produced}}{\text{thermal neutron fission}}$$

A second term, η(eta) will be defined as follows:

$$\eta \equiv \frac{\text{number of fast neutrons produced by thermal fission}}{\text{number of thermal neutrons absorbed in fuel}}$$

This term is called the reproduction factor and attention is called to the subtle difference between ν and η which shows up in the denominator of the previously defined expressions. The larger of the two terms ν speaks of the production of fast neutrons in terms of FISSION, whereas η speaks of the production of the same commodity in terms of ABSORPTION. Since not all thermal neutrons which are absorbed lead to a fission event, the relationship between ν and η is governed by the non-fission captures in the fuel or one may write

$$\eta = \nu \frac{\Sigma_f^{235} + \Sigma_f^{238}}{\Sigma_f^{235} + \Sigma_c^{235} + \Sigma_f^{238} + \Sigma_c^{238}}$$

Notice, however, that U-238 has no cross section for thermal neutron fission (Table 3, Page 5-11); thus, a simplification results

$$\eta = \nu \frac{\Sigma_f^{235}}{\Sigma_f^{235} + \Sigma_c^{235} + \Sigma_c^{238}}$$

where Σ_c^{235} and Σ_c^{238} are the respective radiative capture cross sections for U-235 and U-238 and Σ_f^{235} is the thermal fission cross section for U-235. The reproduction factor is ≥ 1 and a typical calculation of η for natural uranium which contains 1 part U-235 for every 140 parts follows:

$$\eta = \nu \frac{N^{235}\,\sigma_f^{235}}{N^{235}\,\sigma_f^{235} + N^{235}\,\sigma_c^{235} + N^{238}\,\sigma_c^{238}}$$

where N^{235} and N^{238} represent the number of atoms of the respective isotopes per cubic centimeter of natural uranium. Dividing by N^{235}, one gets

$$\eta = \nu \frac{\sigma_f^{235}}{\sigma_f^{235} + \sigma_c^{235} + \sigma_c^{238}/R}$$

where $R = N^{235}/N^{238} = 0.00715$ and $\nu = 2.5$

using values from Table 3, Page 5-11,

$$\eta = \frac{2.5(577)}{577 + 125 + 2.71/0.00715} = 1.33$$

A table of typical values of η and ν for fissionable materials is shown in Table 6 below.

Table 6

Some Values of η and ν

U-233	U-235	Pu-239	Natural U
η - 2.27	2.06	2.10	1.33
ν - 2.51	2.43	2.89	2.50

Values are for 2200 m/sec neutrons.

5.12 The Fast Fission Factor ε

In reactors which contain large amounts of fissionable but non-fissile fuel nuclei, one must account for an increase in the number of neutrons available as a result of the fast neutron fissions that take place in the non-fissile fuel. Attention is called to Figure 10 below and Figure 11 on Page 5-20, which clearly show that in natural uranium, because of the high abundance of the U-238 isotope, most of the fissions occurring at 2 Mev (most probable fission neutron energy) will be the result of fissioning of U-238. So then to account for this increase in neutron population, the fast fission factor ε is introduced and defined as follows:

$$\varepsilon \equiv \frac{\text{number of fast neutrons produced by all fission}}{\text{number of fast neutrons produced by thermal fission}}$$

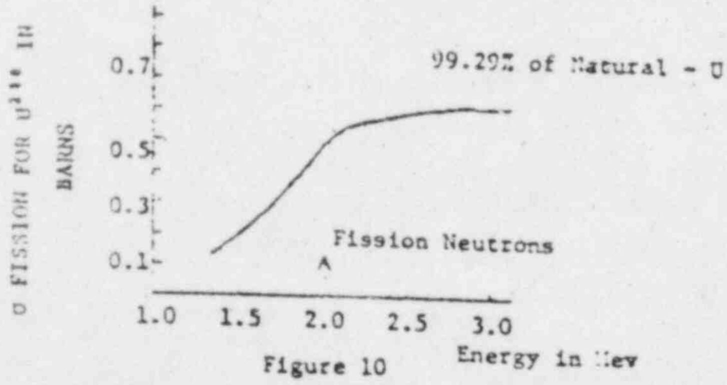

Figure 10

Fission Cross Section of U-238

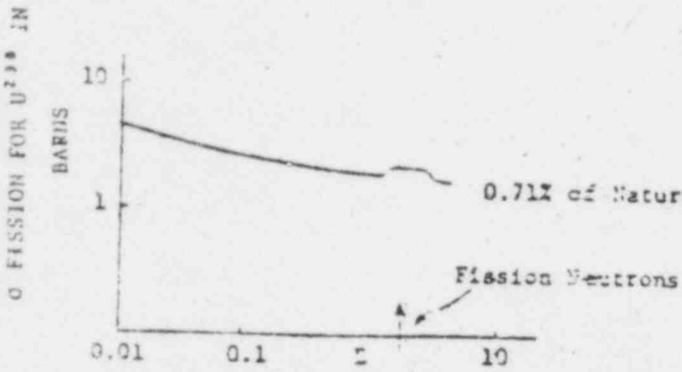

Figure 11

Fission Cross Section of U-235

Even though there is an increase in neutron population because of fast fission, it is slight because there is a great probability of inelastic scattering of high energy neutrons with U-238. A typical value of ε for natural uranium would be ε = 1.03. The actual calculation is based on probability theory and requires a knowledge of fission and non-fission captures, inelastic and elastic scattering cross sections.

Observe that the first two terms considered, i.e., η and ε, have been production terms and are both > 1. One might feel that the parameters chosen here should be the ones to yield the highest ηε product; but there are some considerations involved that have inverse relationships with subsequent factors in k_∞ so that a happy medium must be reached.

5.13 Resonance Escape Probability p

Now that the fast effects have been accounted for, i.e., the neutrons from U-235 fission are born fast and some of these fast neutrons cause additional fission while they are still fast, the topic of slowing down these fast neutrons to thermal energies needs to be considered.

As indicated previously, the most probable energy for fission neutrons is 2 Mev and the most probable energy for thermal neutrons is 0.025 ev. This leaves a rather large gap covering about a 10^8 ev energy span which, for the most part, is called the epi - thermal region. The neutrons born of fission must pass through this energy region and there is a good possibility that they will be captured in the U-238 resonance region. These peaks of high capture probability can be seen in Figure 12, page 5-21. Although U-235 has similar resonances, in natural uranium one is only concerned about the highly abundant U-238.

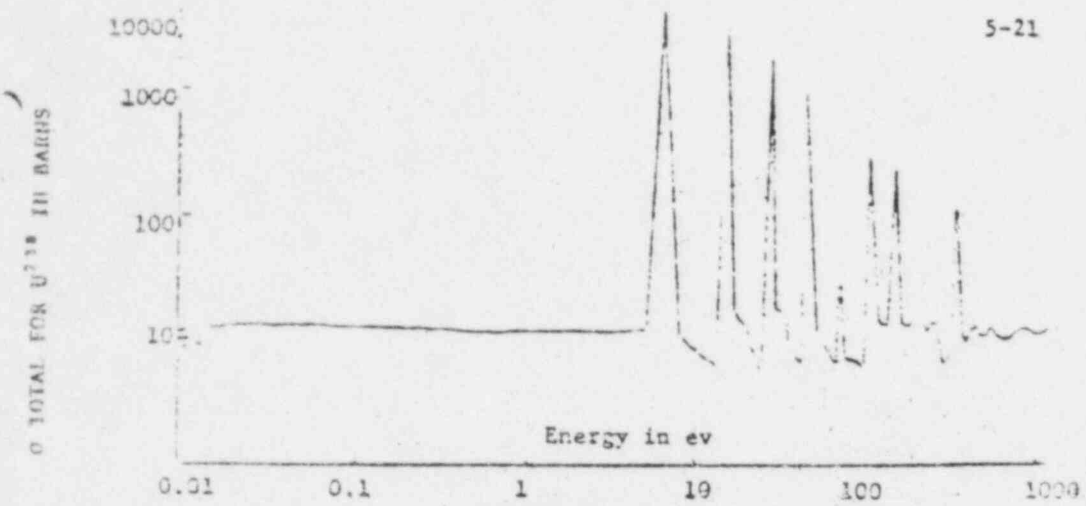

Figure 12

Resonance Peaks in U-238

The resonance escape probability is defined by

$$p \equiv \frac{\text{number of neutrons reaching thermal}}{\text{number of neutrons not absorbed as fast}}$$

A point worth mentioning here but which will be discussed more fully later on is Doppler broadening. As the temperature of the fuel in a reactor system increases, the width of the resonance peaks increases which is one of the contributing factors to a negative temperature coefficient for a given reactor system.

It was thought by some of the scientists working on the Manhattan Project that a possible way of circumventing the resonance problem was to separate the fuel and the moderator material. This was done in the CP-1 reactor with natural uranium rods and graphite in between the fuel rods. The underlying idea was that when the fission event occurred in the fuel rod the fast neutrons would escape from the fuel rod into the graphite moderator; thus, slowing down of thermalization would occur away from the menacing influence of the resonance cross sections. After becoming thermal, the neutrons would diffuse back into the fuel rods and fission. The success of the CP-1 in maintaining the first chain reaction prompts one to agree that this was a darn good idea. However, the complexity brought about by the interplay between concepts can be illustrated here. From the above discussion it is quite obvious that if one makes the radius of the fuel rod quite small so that there is little chance for a neutron to encounter a U-238 nucleus before escaping into the graphite, one has gone a long way toward keeping the value of p high.

On the other hand, go back to ε, the fast fission factor. It was the intent to keep ε high also thereby keeping the neutron population high. Since it would require a large radius fuel rod allowing a greater probability of a fission interaction between the fast neutron and the U-238 nucleus to keep ε high, and a small radius to keep p high, therein lies the dilema. The solution comes from experimentation with fuel rod dimensions and lattice spacing resulting in optimizing by compromising.

Resonance escape probability also presents a challenge through direct calculation of a value; however, Figure 14 indicates some values of p as a function of rod radius for various lattice spacings. Notice, also, that over the range of radii investigated in Figure 14, there was an increase from 1.02 to 1.035 in ε as shown in Figure 13 below.

One final comment on p would be that p < 1 because it is a probability. If every neutron that wasn't absorbed while fast got to thermal energy that would account for 100% of the neutrons and the probability would then be one. Since they don't all make thermal energies, p must always be less than 100%.

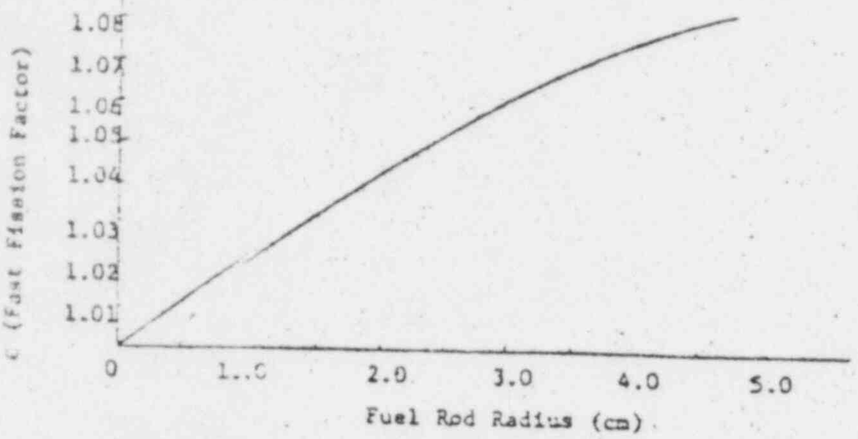

Figure 13

Fast Fission Factor Versus Fuel Rod Radius

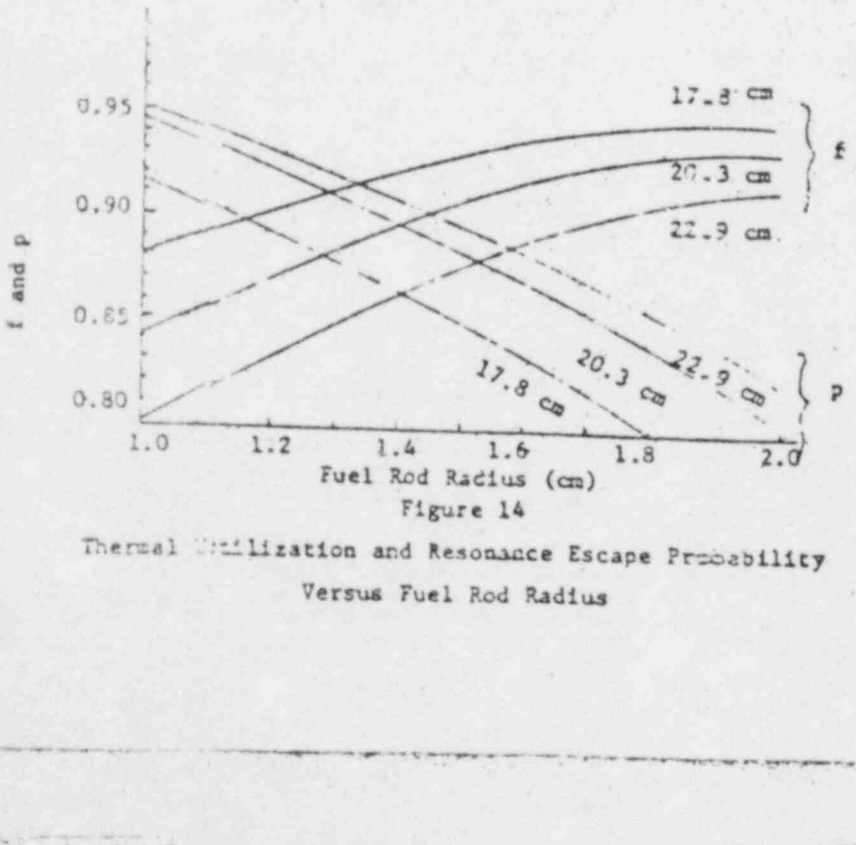

Figure 14

Thermal Utilization and Resonance Escape Probability Versus Fuel Rod Radius

5.14 Thermal Utilization f

The thermal utilization factor like the resonance escape is a probability and, therefore, can never be greater than unity. In simple form, the thermal utilization factor is a measure of how efficiently the fissile fuel competes with other reactor materials for thermal neutrons. The definition of thermal utilization is:

$$f \equiv \frac{\text{number of thermal neutrons absorbed in fuel}}{\text{number of thermal neutrons absorbed}}$$

Examination of numerator and denominator indicates that all the neutrons are thermal. The neutron population, having reached thermal energies, will diffuse for some time then be absorbed. Depending upon how many neutrons are absorbed by materials other than fuel, one can write the following

$$f = \frac{\phi V\,\Sigma_a^{235}}{\phi V\,\Sigma a^{235} + \phi V\,\Sigma a^{mod} + \phi V\,\Sigma a^{all}}$$

where Σa^{235}, Σa^{mod}, and Σa^{all} are the macroscopic absorption cross sections for U-235, moderator material, and all other materials such as core structure, shielding material and control rods. Note that the thermal flux ϕ is not, except in the simplest examples, the same in the fuel as in the moderator or other structural materials. The ratio of ϕ_{fuel}/ϕ_{mod} is known as the disadvantage factor F and accounts for a decrease in neutron flux in the fuel rods of a reactor where there is absorption of thermal neutrons but no replacement because the fission neutrons born in the fuel rods are fast. The F factor also corrects for the depression of thermal flux in the moderator adjacent to the fuel rod because of the rod's acting as a thermal neutron sink.

The counter effects of changing dimensions are again manifest in the thermal utilization. A look at Figure 14, Page 5-22, shows an increase of f with rod radius; also consistently higher values for shorter lattice spacings. This is explained when one considers that in both situations the neutrons are spending less time in the moderator, thus there is less chance of being absorbed there - consequently, f is higher.

At this point, some numbers will be collected from the above sections to determine whether or not criticality has been obtained. If it is assumed that $\eta = 1.33$ as calculated in Section 5.11, then for the same type of fuel, i.e., natural uranium, let $\epsilon = 1.03$. Now refer to Figure 14 again and select a radius for fuel rods of 1.4 cm and rod spacings of 20.3 cm yielding f and p values of 0.905 and 0.890, respectively. Using these values in $k_\infty = \eta\epsilon pf$, one finds that $k_\infty = 1.104$ so that not only is the reactor critical (because $k_\infty > 1$), but the number of neutrons in the (n + 1) generation is 1.104 times greater than the number of neutrons in the n^th^ generation. This is not an undesirable situation and is actually necessary if one is to get power from a reactor. More will be said later about the amount k_∞ is in excess of unity.

The situation must be controlled, however, and this is accomplished by the addition of absorber material as suggested in Section 5.9. Neutron

Absorbing Materials. The way in which k_∞ will be controlled by the absorbing material is implicit in the thermal utilization factor f. Simplifying f so that one can write

$$f = \frac{\Sigma_a^{235}}{\Sigma_a^{235} + \Sigma_a^{m} + \Sigma_a^{all}}$$

it becomes clear that when more absorbing material is added to the reactor system Σ_a^{all} will increase with an accompanying decrease in f and, consequently, k_∞.. Therein lies the secret of neutron population adjustment with control rods or distributed poisons.

To summarize the treatment of the infinite natural uranium reactor system, it is found convenient to describe the following capture processes:

a. Absorption of fast neutrons above the 1 Mev fission threshold of U-238, the fast fission factor.

b. Radiative capture in the resonance region (10-1000ev) of U-238 during slowing down, leading to the resonance escape probability.

c. Parasitic capture of slow neutrons in moderator, structural, and control materials resulting in the thermal utilization factor.

d. Fission captures in the thermal region by U-235 which is described by the reproduction factor.

Let it here by pointed out that all of these effects are microscopic in nature or intensive properties of the materials involved. The next section will deal with problems of size. In other words, the finite reactor and the modifications of some equations and ideas already presented to make the finite reactor workable.

5.15 The Finite Reactor

The difference between the finite and the infinite system is quite obviously one of size. Granting that an infinite reactor is an unimaginably large reactor, the size will be cut to some conceivable, measurable dimensions, but more importantly, neutrons will now be allowed to escape from the system. Two more terms will be added to the multiplication factor and it will be indicated as Keff or just k as opposed to the already familiar k_∞. The new k is still the ratio of the (n + 1) and the nth generations of neutrons; however, because it is a finite system now being described and neutrons can escape one must now account for the escaped neutrons or, of greater concern, those that remain behind. This will be done by defining

$$k = \frac{\text{Production}}{\text{Absorption} + \text{Leakage}}$$

and introducing two non-leakage terms P_t and P_f, the thermal and fast probabilities of non-leakage. The relationship between k and k_∞ is then

$$k = k_\infty P_t P_f$$

$$= \eta \varepsilon p f P_t P_f$$

The above modification will be known as the six factor equation. Its development will be the subject of the next two sections.

5.16 Probability of Fast Non-Leakage P_f

This factor accounts for leakage of neutrons from the finite system, which occurs between the time that the fission neutrons are born (fast) and the time that they escape resonance capture (epithermal), and is defined as follows:

$$P_f = \frac{\text{number of neutrons not leaking while fast}}{\text{number of fast neutrons produced}}$$

Thus, it is again evident that P_f must be less than one and that the fraction of neutrons which actually leak is $1 - P_f$.

$P_f < 1$

From "two group theory", as opposed to "one group" where all phenomena are assumed to occur at a single energy, the following expression for fast non-leakage probability evolves:

$$P_f = e^{-B^2 \tau}$$

Where B^2 is the buckling of the finite system and τ is the Fermi age and both terms will be explained more fully.

In explaining Fermi age, it is helpful to look first at a factor called the slowing down length or $\sqrt{\tau}$ where $\sqrt{\tau}$ is proportional to the average "crow-flight" distance traveled by a neutron in its journey from birth to thermalization. A pictorial representation of slowing down length is given in Figure 15.

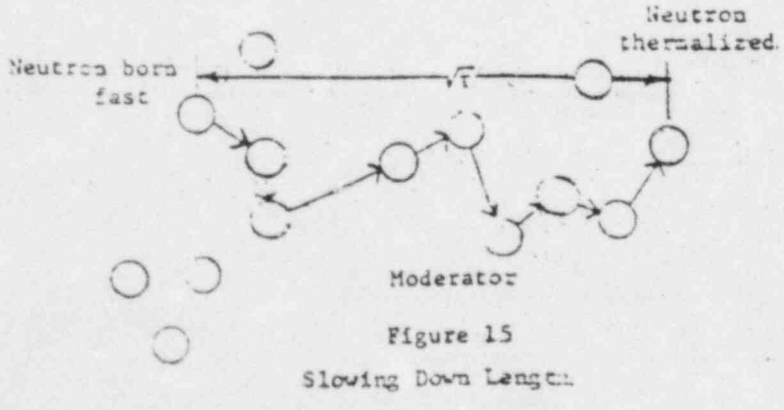

Figure 15

Slowing Down Length

For reasons which will become apparent in a future section, the Fermi age is taken to be the square of the slowing down length.

As indicated in Section 5.7, the neutrons lose a certain increment of energy per collision. If it takes many collisions to degrade the neutron energy from fast to thermal then the "crow-flight" distance that the neutron travels becomes larger and τ increases accordingly. As τ becomes larger, the expression $e^{-B^2\tau}$ becomes smaller which is rationalized in the following manner: If the distance the neutron has to travel to become thermal is greater, the chance of leaking is greater. Clearly then, τ is a function of the material selected to moderate the neutrons in the system and is concerned with fast neutrons.

The remaining term B^2 in the exponent of the expression for P_f, known as the buckling of the system, is the term which demonstrates the dependence of the finite system upon size, shape, or geometry effects. The B^2 term is actually proportional to the curvature of the flux in the reactor system. If the system is infinite, then the flux is everywhere equal and has no curvature and

$$P_f = e^{-B^2\tau} = e^0 = 1$$

Consequently, as postulated, there is no leakage from an infinite system because the flux does not "buckle". However, in a finite system where the flux does buckle, it can be shown that B^2 increases with decreasing size. For a very small reactor, it was previously suggested intuitively that many neutrons would leak. Now one can see that as the size of the system decreases, causing B^2 to increase, P_f must decrease.

Perhaps this conceptual point of view may help to understand the term geometrical buckling. Neutrons born near the center of a critical mass have little probability of escaping the system. On the other hand, those born near the outer regions of the core have an excellent chance of escaping. The change in the escape rate of neutrons is a measure of the system's buckling.

Finally, it can be concluded that the fast non-leakage probability is a term dependent upon the material and size of the reactor system. Much the same argument is used in the next section to illustrate the remaining non-leakage term.

5.17 Probability of Thermal Non-Leakage P_t

The sixth factor of the six factor equation speaks of the neutrons available to the reactor system after the neutrons have reached thermal energies and those that are going to leak out of the system have departed. The probability of thermal non-leakage is defined in the following way:

$$P_t = \frac{\text{number of thermal neutrons absorbed}}{\text{number of neutrons reaching thermal}}$$

as with all probability terms, $P_t < 1$ and the fraction of neutrons which leak are given by $1-P_t$.

Again, from two group theory, the term which describes the thermal non-leakage probability is:

$$P_t = \frac{1}{1 + L^2 B^2}$$

where B^2 is buckling as before, and L^2 is the square of the thermal diffusion length.

The diffusion length L can be described as being proportional to the "crow-flight" distance from the point where a neutron becomes thermal to the point where it is absorbed. Note - that if the moderator is not pure then

$$L^2 = L_o^2 (1 - f)$$

or the square of the diffusion length is dependent upon that of the pure moderator times an inefficiency factor 1 - f where f is thermal utilization.

To get some physical feel for the dependence of P_t upon the material and geometrical situations, assume that the reactor is large so that B^2, as before, is small. Then one sees that the reciprocal of a small number, plus 1, is being taken and P_t remains large (but < 1). By the same token, if a relatively few number of collisions are necessary before capture occurs, P_t remains large. Some typical values for the parameters associated with non-leakage probabilities are tabulated in Table 7.

Table 7

Some Material Properties

Material	Density grms/cm³	L_o cm	τ cm²	Σ_a cm⁻¹
H_2O	1.0	2.88	33	0.017
D_2O	1.1	100	120	0.00008
Be	1.84	23.6	93	0.0013
C	1.62	50.2	350	0.00036

Some numerical examples will be offered to illustrate the non-leakage terms. As a working model (to assure successful calculation) use the X-10 Reactor which is natural uranium graphite moderated and has a buckling figure of 9.5×10^{-5} cm^{-2} and f=0.89, then

$$P_f = e^{-B^2\tau}$$

$$= e^{-(9.5 \times 10^{-5} \times 3.5 \times 10^{2})}$$

$$= e^{-0.0333}$$

$$= 0.967$$

and

$$L^2 = L_o^{\,2}(1 - f)$$

$$= 50.2^2\ (1 - 0.890)$$

$$= 2520\ (0.11)$$

$$= 277\ \text{cm}^2$$

From the thermal non-leakage expression

$$P_t = \frac{1}{1 + L^2 B^2}$$

$$= \frac{1}{1 + (2.77 \times 10^{2})(9.5 \times 10^{-5})}$$

$$= \frac{1}{1.026}$$

$$= 0.975$$

Since the relationship between k and k_∞ is

$$k = k_\infty\, P_t\, P_f$$

k = keff

then it is seen that to have a critical finite reactor one must have a minimum k_∞ as follows:

$$k_\infty = \frac{k}{P_t\, P_f}$$

$$= \frac{1}{(0.975)(0.967)}$$

$$= \frac{1}{0.942} = 1.062$$

Now, referring back to Section 5.14, for an infinite natural uranium graphite moderated system where

$$k_\infty = \eta\, \varepsilon\, p\, f$$

it was shown that the four factor product would be 1.104 which is greater than the minimum k_∞ figure required from above.

$$k = \frac{\eta \varepsilon p f e^{-B^2\tau}}{1 + L^2 B^2}$$

$$= \frac{k_\infty e^{-B^2\tau}}{1 + L^2B^2}$$

and from this equation calculations can be made for various geometries and given sets of conditions to determine the feasibility of a particular reactor system. Before much more can be done with calculations, however, additional information will be presented pertaining to buckling.

5.19 Geometry Factors

If one were to write a neutron balance equation for a reactor system it might look like the following:

Change in neutrons = Production - leakage - absorption

rewriting

$$\frac{\partial n}{\partial t} = S + D\nabla^2\phi - \Sigma_a\phi \qquad (1)$$

where S is the rate at which neutrons are produced per unit volume.

If a beam of collimated neutrons moves with a velocity v in a single direction, then the number of neutrons which pass through 1 square centimeter of area perpendicular to the path of motion of the neutron beam is the neutron flux, or flux

$$\phi\ (\text{neutrons/cm}^2\text{-sec}) = n\ (\text{neutrons/cm}^3)\ v\ (\text{cm/sec})$$

or flux is the number of neutrons incident upon 1 cm² of target material each second.

Recall that Σ is the macroscopic cross section or

$$\Sigma\ (\text{cm}^{-1}) = N\ (\text{nuclei/cm}^3)\ \sigma(\text{cm}^2)$$

thus, the probability of interaction in a unit volume. From this, one can then express the reaction rate R as follows:

$$R = \Sigma(\text{cm}^{-1})\ \phi(\text{neutron/cm}^2\text{-sec})$$

$$= \Sigma\ \phi(\text{neutrons/cm}^3\text{-sec})$$

Assuming steady state conditions where there is no change in the number of neutrons and thus $\partial n/\partial t = 0$

$$S = \phi\ \Sigma_a\ k_\infty \qquad (2)$$

Material Buckling must = Geometric Buckling (Criticality)

or the neutrons produced are equal to the reaction rate times the multiplication factor, then one can write

$$D\nabla^2\phi - \Sigma_a\phi + k_\infty\Sigma_a\phi = 0 \qquad (3)$$

where ∇^2 is the Laplacian operator and is shorthand notation for

$$\nabla^2 = \frac{\partial^2}{\partial x^2} + \frac{\partial^2}{\partial y^2} - \frac{\partial^2}{\partial z^2} \qquad (4)$$

in Cartesian coordinate systems. Factoring the second and third terms in Equation (3) and dividing by D yields

$$\nabla^2\phi - \frac{\Sigma_a}{D}(k_\infty - 1)\phi = 0 \qquad (5)$$

Now if one defines a term

$$B^2 = \frac{\Sigma_a}{D}(k_\infty - 1) \qquad (6)$$

then the following can be said. By substituting Equation (6) into Equation (5)

$$\nabla^2\phi - B^2\phi = 0 \qquad (7)$$

If it is given that

$$L^2 = \frac{D}{\Sigma_a} \qquad (8)$$

then by substitution of Equation (8) into Equation (6)

$$B^2 = \frac{k_\infty - 1}{L^2} \qquad (9)$$

As indicated previously, B^2 is called buckling and here buckling is represented in terms of reactor materials. This is the material buckling and must be equal to the geometric buckling at criticality.

Assume, for a moment, that one has an infinite slab reactor. For the purposes of illustration, then,

$$\nabla^2\phi - B^2\phi = 0 \qquad (10)$$

would reduce to

$$\frac{d^2\phi}{dx^2} - B^2\phi = 0, \qquad (11)$$

or the flux would vary in the only finite direction, namely, the x direction. The above equation has a solution

$$\phi = A_1 \cos Bx + A_2 \sin Bx \qquad (12)$$

and differentiating Equation (12) gives

$$\frac{d\phi}{dx} = - A_1 B \sin Bx + A_2 B \cos Bx \qquad (13)$$

from symmetry ϕ is a maximum at $x = 0$, thus, $d\phi/dx = 0$ at $x = 0$. Substituting these conditions into Equation (13) yields

$$0 = 0 + A_2 B, \quad A_2 = 0 \qquad (14)$$

thus Equation (12) becomes

$$\phi = A_1 \cos Bx \qquad (15)$$

Now applying the boundary condition $\phi = 0$ at $x = \pm a/2$, where a is the actual width of the reactor, to Equation (15), one gets

$$\phi = A_1 \cos \frac{Ba}{2} = 0 \qquad (16)$$

Recalling that the cosine function is zero at 90° or $\pi/2$ radians, then

$$\frac{Ba}{2} = \frac{\pi}{2} \qquad (17)$$

$$B = \frac{\pi}{a} \qquad (18)$$

substituting back into Equation (15)

$$\phi = A_1 \cos \frac{\pi}{a} x \qquad (19)$$

or the geometric buckling term, by comparison with Equation (15), is

$$B^2 = \left(\frac{\pi}{a}\right)^2 \qquad (20)$$

which is an expression that relates physical values, namely, dimensions with the buckling; thus, the term is called geometric buckling. Recall now that the material buckling and the geometric buckling are equal in a critical reactor. If one looks back to the critical equation or

$$1 = \frac{k_\infty e^{-B^2 \tau}}{1 + L^2 B^2}$$

a means is now afforded the student to couple material values inherent in k_∞, L^2, and τ with dimensional values in B^2, the geometric buckling and come up with finite sizes for critical reactor calculations.

A similar mathematical analysis could be done for a sphere, cylinder, and rectangular parallelepiped and the results for geometric buckling would be as indicated in the following table:

Table 8

Geometric Buckling Factors

Shape	B^2	
Sphere	$(\frac{\pi}{R})^2$	R = Radius
Cylinder	$(\frac{2.405}{R})^2 + (\frac{\pi}{H})^2$	R = Radius H = Height
Rectangular Parallelepiped	$(\frac{\pi}{a})^2 + (\frac{\pi}{b})^2 + (\frac{\pi}{c})^2$	a, b, c are lengths of sides
Cube	$3(\frac{\pi}{a})^2$	Special case of above where a = b = c

The student is now able to perform the following types of calculations:

a. Determining critical composition when the reactor size is given.

b. Determination of the critical size when concentrations are given.

5.20 Numerical Examples

Example: If the geometric buckling for a critical cylindrical reactor, whose height is equal to its diameter is 2×10^{-4} cm^{-2}, what is the radius of this reactor?

$$B^2 = 2 \times 10^{-4}\ cm^{-2}$$

$$H = D = 2R$$

$$B^2 = (\frac{2.405}{R})^2 + (\frac{\pi}{H})^2$$

$$= \frac{5.76}{R^2} + \frac{10}{4R^2}$$

$$4B^2 = \frac{23}{R^2} + \frac{10}{R^2}$$

$$4B^2 = \frac{33}{R^2}$$

$$R = \left(\frac{33}{8 \times 10^{-4}}\right)^{\frac{1}{2}} = \underline{\underline{202 \text{ cm}}}$$

Example: Find k of the system if one has a cubical, natural uranium, graphite moderated reactor which is 450 cm on a side, f = 0.90, p = 0.910 and ε = 1.03.

$$B^2 = 3\left(\frac{\pi}{450}\right)^2 = 1.47 \times 10^{-4} \text{ cm}^{-2}$$

$$P_f = e^{-B^2\tau} = e^{-(1.47 \times 10^{-4})(350)}$$

$$= e^{-0.0515}$$

$$= 0.950$$

$$P_t = \frac{1}{1 + L^2 B^2} = \frac{1}{1 + L_o^2 (1-f)B^2}$$

$$= \frac{1}{1 + 50^2 (1-0.9)(1.47 \times 10^{-4})}$$

$$= 0.965$$

$$k = k_\infty P_t P_f$$

$$= \eta \varepsilon p f P_t P_f$$

$$= 1.33 (1.03)(0.91)(0.9)(0.965)(0.95)$$

$$= \underline{\underline{1.03}}$$

Find the critical mass for the reactor in the above example if 3 cm diameter fuel rods are spaced 20 cm on centers in a square lattice array. The density of natural uranium is 18.9 gm/cm³.

Area of a cell = $20^2 = 400$ cm²

Area of a side = $450^2 = 2.025 \times 10^5$ cm²

$$\text{Number of cells} = \frac{2.025 \times 10^5}{4 \times 10^2}$$

$$= 506 \text{ cells or rods}$$

16661

1095 072

$$\text{Volume of fuel} = \pi r^2 \ L(\text{rods})$$

$$= (3.14)(1.5)^2(450)(506)$$

$$= 1.61 \times 10^6 \ \text{cm}^3$$

$$\text{Critical Mass} = \rho_u V_u$$

$$= (18.9)(1.61 \times 10^6)$$

$$= \underline{\underline{30.4 \times 10^6 \ \text{gm U}}}$$

Example: Solution of the critical equation with a simplification for a large reactor.

Given a heterogeneous array of natural uranium in heavy water moderator with $k_\infty = 1.20$, $L^2 = 175 \ \text{cm}^2$ and $\tau = 120 \ \text{cm}^2$. Determine the critical buckling.

If one were to use the form of the critical equation

$$k = \frac{k_\infty e^{-B^2\tau}}{1 + L^2 B^2}$$

it is quite obvious that one runs into a problem when a choice of B^2 must be made. The problem can be simplified considerably by the following analysis:

Recall from Section 5.16, that $\sqrt{\tau}$ is the slowing down length of fast neutrons approaching thermalization. Recall, also from Section 5.17 that L was defined as the thermal diffusion length or the distance traveled from thermal to absorption by a neutron. Observe that when a neutron scatters there is an equal probability that it will scatter forward or backward. One might conclude then that on the average the scattering angle will be 90° or $\pi/2$ radians. Having defined $\sqrt{\tau}$ as the "crow-flight" distance from birth to thermalization of a neutron and L as the "crow-flight" distance from thermal to death, and the most probable scatter angle as 90°, then, by Pythagorian theorem

$$c^2 = a^2 + b^2$$

Now if $\sqrt{\tau} \equiv a$ and $L \equiv b$ (See Figure 17), then the sum of the squares of these terms would be the square of the "crow-flight" distance from birth to death of a neutron. This new quantity will be called the migration area and expressed as follows:

$$M^2 = L^2 + \tau$$

As stated previously, this quantity applies for large reactors and can be justified in the following manner: If the system is large then B^2 is very small and in the equation

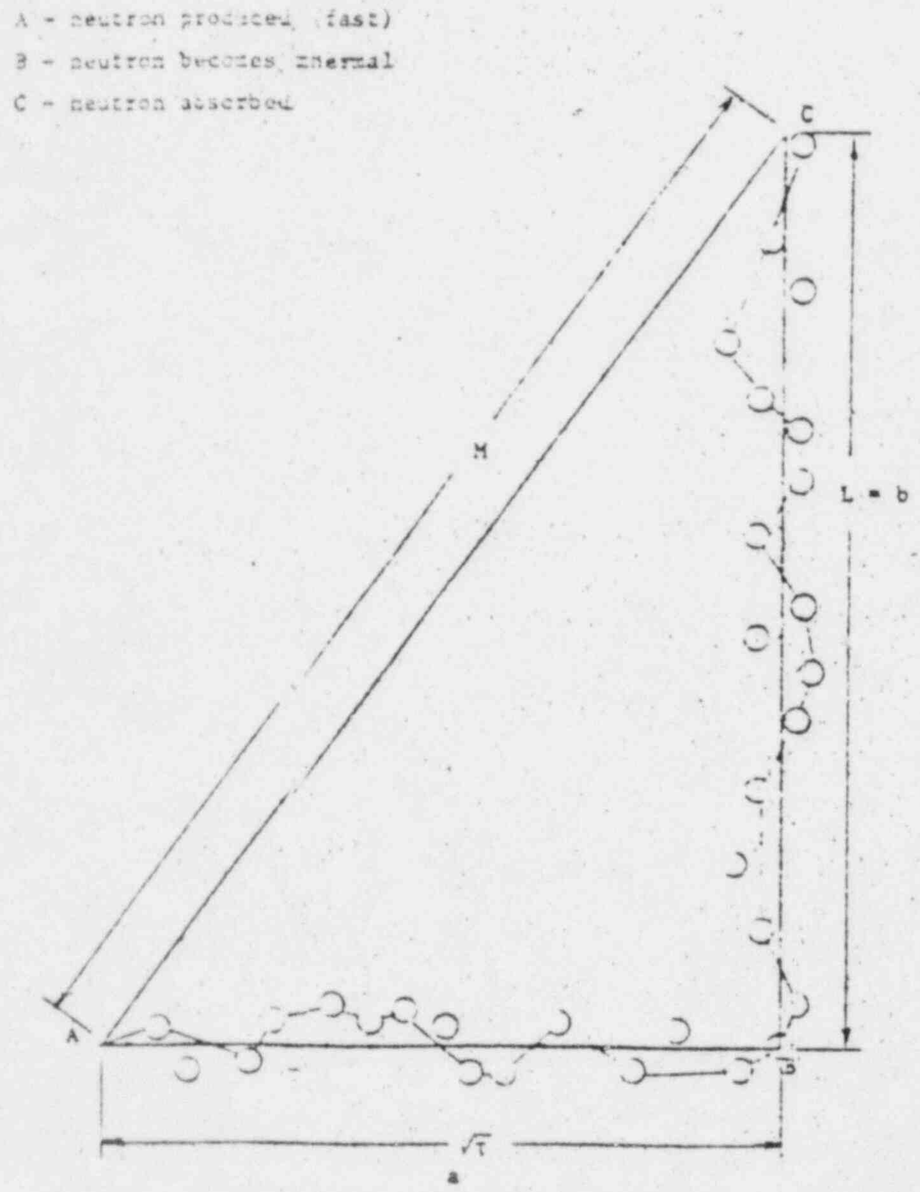

Figure 17
Migration Area

$$k = \frac{k_\infty e^{-B^2\tau}}{1 + L^2B^2} = 1$$

$e^{-B^2\tau}$ can be written as e^{-x} where x is very small and can be approximated by (1-x) or

$$1 = \frac{k_\infty (1-B^2\tau)}{1 + L^2 B^2}$$

Again, when x is small (1-x) can be approximated by 1/1+x which makes the expression above

$$1 = \frac{k_\infty}{(1 + L^2B^2)(1 + B^2\tau)}$$

thus,

$$1 = \frac{k_\infty}{1 + L^2B^2 + B^2\tau + L^2B^4\tau}$$

and if B^2 is small, B^4 is very small and can be neglected. With some factoring in the denominator, the expression becomes

$$1 = \frac{k_\infty}{1 + (L^2 + \tau) B^2}$$

and if, as stated, $M^2 = L^2 + \tau$

then,

$$1 = \frac{k_\infty}{1 + M^2 B^2}$$

Now returning to the data of the example with another expression to work with, an expression which is simplified by the occurrence of only one B^2 term, B^2 can be obtained as follows:

$$\begin{aligned} B^2 &= \frac{k_\infty - 1}{M^2} \\ &= \frac{k_\infty - 1}{L^2 + \tau} \\ &= \frac{1.28 - 1}{175 + 120} \\ &= \underline{\underline{9.5 \times 10^{-4} \text{ cm}^{-2}}} \end{aligned}$$

With an expedient means of estimating B^2, one can use the more exact form of the critical equation to solve for the buckling as illustrated in the next example. Example of the iterative process of solving the critical equation.

Taking the value for B^2 which was determined from the simplification of the critical equation in the previous example and now using a more exact form of the critical equation obtained from the age diffusion theory, one obtains

$$k = \frac{k_\infty \, e^{-B^2\tau}}{1 + L^2B^2} = 1$$

rewriting to get B^2 on both sides

$$e^{-B^2\tau} = \frac{1 + L^2 \cdot B^2}{k_\infty}$$

now inserting $9.5 \times 10^{-4} = B^2$ on the right

$$e^{-B^2\tau} = \frac{1 + 175\ (9.5 \times 10^{-4})}{1.28}$$

$$= 0.910$$

$$B^2\tau = 0.094$$

$$B^2 = \frac{0.094}{120} = 7.83 \times 10^{-4}\ \text{cm}^{-2}$$

which is not in good agreement with the estimated value of B^2 from the previous example. Obviously, the large reactor approximation gave too large a value for B^2. Now the iteration process, try $B^2 = 8.5 \times 10^{-4}\ \text{cm}^2$

$$e^{-B^2\tau} = \frac{1 + 175 \cdot (8.5 \times 10^{-4})}{1.28}$$

$$= 0.897$$

$$B^2\tau = 0.109$$

$$B^2 = \frac{0.109}{120}$$

$$= 9.07 \times 10^{-4}$$

thus, $8.5 \times 10^{-4}\ \text{cm}^{-2}$ is too small. A third attempt is made with $B^2 = 8.75 \times 10^{-4}$

$$e^{-B^2\tau} = \frac{1 + 175\ (8.75 \times 10^{-4})}{1.28}$$

$$= 0.901$$

$$B^2\tau = 0.1049$$

$$B^2 = \frac{0.1049}{120}$$

$$= 8.74 \times 10^{-4}\ \text{cm}^{-2}$$

which is reasonably close to the 8.75×10^{-4} cm^{-2} value chosen for the third iteration.

The Fermi age τ and diffusion length are very basic properties to the solution of the critical equation. These properties can be determined for a particular lattice arrangement and material composition, by well established experiments performed with the reactor in question, or can be calculated from cross section information.

One final example before moving on to the next section will concern itself with loss calculations. If a homogeneous mixture in an atom ratio of 10^5 to 1 of graphite and U-235 is used for a critical mass where σ_a for graphite = 0.003 barn, σ_a for U-235 = 698 barn, L_o = 54 cm, τ = 364 cm^2, η = 2.08, p ε = 1 (because U-235 is not present), then find the probability of leakage for a critical core.

$$f = \frac{1}{1 + \frac{\Sigma_a^m}{\Sigma_a^f}} = \frac{1}{1 + \frac{N^m \sigma_a^m}{N^f \sigma_a^f}}$$

$$= \frac{1}{1 + \frac{10^5(.003)}{1\ (698)}} = 0.699$$

$$k_\infty = \eta\, \varepsilon\, p\, f$$

$$= 2.08\ (1)(0.699)$$

$$= 1.45$$

$$L^2 = L_o^2\ (1-f) = 54^2\ (.301)$$

$$= 878\ \text{cm}^2$$

The total non-leakage is

$$\boxed{P_f P_t} = \frac{k}{k_\infty} = \frac{1}{1.45}$$

$$= 0.690$$

therefore, total leakage is

$$L_f\, L_t = 1 - 0.690 = \underline{\underline{31\%}}$$

total non leakage [illegible] fast & thermal [illegible] n.l.

1 - Non leakage = [illegible]

By methods described previously, i.e., big reactor approximation and iteration

$$B^2 = 3.25 \times 10^{-4}$$

thus,

$$e^{-B^2\tau} = P_f = e^{-(3.25 \times 10^{-4})(364)}$$

$$P_f = 0.888$$

and the fast leakage is

$$L_f = 1 - 0.888 = 11.2\%$$

$$P_t = \frac{1}{1 + L^2 B^2}$$

$$= \frac{1}{1 + 878\ (3.25 \times 10^{-4})}$$

$$= 0.778$$

and the thermal leakage is

$$L_t = 1 - 0.778 = 22.2\%$$

Since only 88.8% of the original neutrons ever slow down, then 22% of this number leak as thermal neutrons.

$$L_t = 0.222\ (0.888) = 19.7\%$$

thus, the total leakage is

$$L = L_f + L_t = 11.2\% + 19.7\%$$

$$= \underline{\underline{30.9\%}}$$

which compares favorably with the 31% found above.

5.21 Approach to Critical

Up to this point the reactor system in question was always a "paper system", i.e., the system components were only numbers in a critical equation. Now consider a practical situation where one has fuel elements, moderator, control rods, and a small neutron source available to put together into a critical system. The first problem which should become apparent is that of knowing when enough fuel has been put together to sustain a chain reaction. For this purpose a neutron sensitive detector will be used to determine a count rate. From this information the nearness to criticality or how close k_{eff}

is to unity will be obtained. More about the mechanics of the method in the next section. First, let the relationship between count rate and k_{eff} be established in this manner: Suppose a neutron source which is available emits n_o neutrons in the first generation. Then by

$$k = \frac{\text{number of neutrons in the } (n + 1) \text{ generation}}{\text{number of neutrons in the } n^{th} \text{ generation}}$$

it is seen that in the second generation there are present $k\ n_o$ neutrons. The following tabulation is then possible

Generation	Number of neutrons
1	n_o
2	$n_o\ (k + 1)$
3	$n_o\ (k + k^2 + 1)$
4	$n_o\ (k + k^2 + k^3 + 1)$
...	....
n	$n_o\ (k + k^2 + k^3 + \ldots + k^{n-1} + 1)$

then one would write

$$n = n_o\ (1 + k + k^2 + k^3 + \ldots + k^{n-1})$$

It follows, then, that

$$\frac{n}{n_o} = \frac{1 - k^n}{1 - k}$$

Hint: If one were to actually perform the division of $1 - k^n/1 - k$, the quotient would be $(1 + k + k^2 + \ldots + k^n)$. Let it now be assumed that the critical mass has not yet been obtained so that $k < 1$. In this case, as n approaches ∞, k^n approaches o and

$$\frac{n}{n_o} = \frac{1}{1 - k}$$

at this point, let $M \equiv$ multiplication of the count rate, thus

$$M = \frac{C\ R_1}{C\ R_o}$$

where $C\ R_o$ might be some count rate with just the neutron source and a detector in place and $C\ R_1$ is the count rate with an initial amount of fuel added to the system. Clearly, then, the count rate is equal to the numbers of neutrons detected so that

$$\frac{n}{n_o} = \frac{C\ R_1}{C\ R_o}$$

and then one can write

$$M = \frac{1}{1 - k}$$

or from the above equation it is seen that as k approaches 1, M goes to ∞. Perhaps by solving the above expression for k as follows

$$k = 1 - \frac{1}{M}$$

and noting that as M goes to ∞, k approaches one, one can better appreciate the way that the count rate information will be used to predict criticality. As fissionable material is added to the assembly, the count rate will continue to increase and the reciprocal of the count rate becomes smaller. If the reciprocal count rate curves are plotted as fuel is added, one can extrapolate to the point where 1/M will be zero, and hence, when criticality will be reached. The next section will describe the procedure for conducting such an experiment.

5.22 Critical Mass Experiment

The purpose of a critical mass experiment is to determine the minimum critical fuel loading for a particular reactor system; consequently, it is performed when the reactor is to go critical for the first time. A procedure that would be followed in performing a critical mass experiment might be of the following form:

a. Position source and detectors

b. Measure initial count rate with rods in and with rods out. Plot reciprocal of count rate.

c. Position the first charge of fuel.

d. Measure new count rate with rods in and out. Plot new value and make the first estimate of the fuel needed to go critical.

e. If results of estimate allow, add another charge of fuel.

f. Steps d & e are repeated until criticality is reached. The amount of fuel added is made smaller as one gets closer to criticality.

A more detailed discussion of the critical mass experiment is now offered. For reasons of safety, two or more neutron detection channels, independent of each other, are used to determine the count rate of the growing assembly. The details of selection of detector type are discussed in the chapter on Instrumentation and Control.

The ideal arrangement of the fuel, detectors, and neutron source will be that geometry which will make the detector most sensitive to increases in count rate brought about by the addition of fuel to the assembly. Speaking in general terms, this situation exists whenever the maximum amount of fuel in the system is located between the source and the detector thereby offering the greatest amount of multiplying medium for source neutrons to travel through prior to detection. Figure 18 shows the ideal location of the components as opposed to the geometry in Figure 19 where, because of the close proximity of the source to the detector, the measured count rate would be insensitive to fuel which would be added.

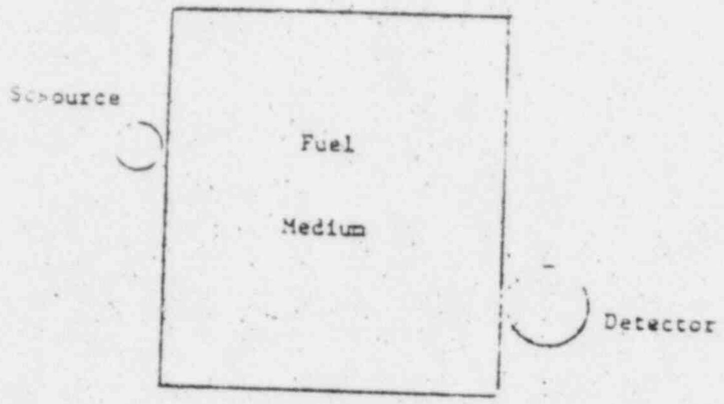

Figure 18

Core Geometry for Critical Experiment (ideal)

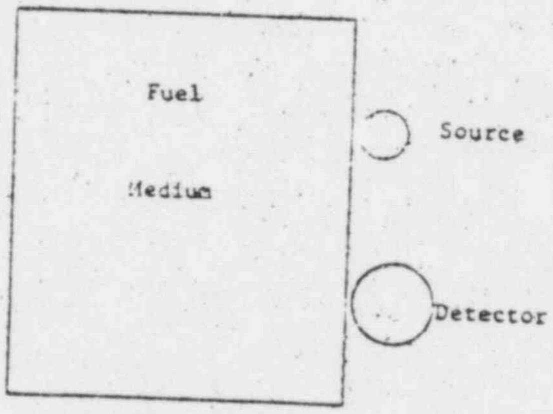

Figure 19

Core Geometry for Critical Experiment (non-ideal)

A typical neutron source used in this type of experiment might be Antimony - Beryllium source with the following reactions occurring:

$$Sb^{123} + {}_{0}n^{1} \longrightarrow Sb^{124}$$

$$\Big\downarrow \text{ 60 days}$$

$$Te^{124} + {}_{-1}\beta^{0} + \gamma\,(2.1 \text{ Mev}) + \gamma\,(0.6 \text{ Mev})$$

$$Be^{9} + \gamma(>1.67 \text{ Mev}) \longrightarrow Be^{8} + {}_{0}n^{1}\ (\sim 30 \text{ Kev})$$

The antimony is regenerated by keeping the source in-core while the reactor is operating at high power levels.

Recall from the previous section that

$$M = \frac{1}{1 - k}$$

where M is the multiplication of the count rate and k is the multiplication factor of the core, then the count rate goes to infinity as k approaches unity. To get any usable information from a plot of count rate versus fuel aboard one would need a rather large sheet of graph paper. For this reason then, the reciprocal of the count rate is plotted against fuel aboard and this quantity goes to zero as k approaches one. A reciprocal count rate versus fuel mass is shown in Figure 20.

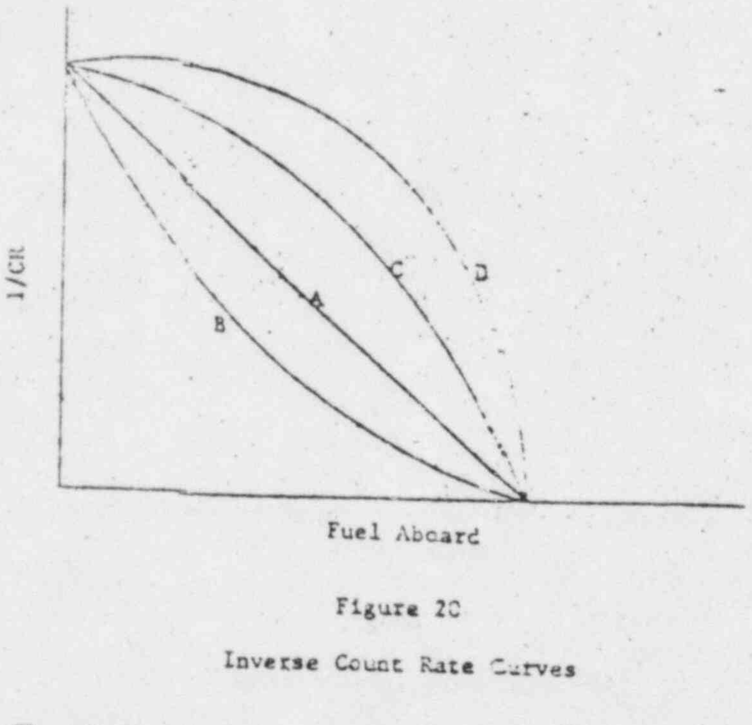

Figure 20

Inverse Count Rate Curves

The line marked "A" in Figure 20 is the ideal curve one might hope for in performing the critical mass experiment. Notice that the first two points plotted predict the exact critical mass. This of course, is not the case and in actual practice one experiences curves similar to "B" and "C" in Figure 20. The more desirable situation is that shown in "B" where the curve always predicts conservatively until the point of criticality is reached. A curve similar to "C" would reflect an insensitivity to the initial fuel loaded and carrying this notion to an extreme one could get a curve similar to "D" where criticality is over predicted until the critical mass is practically assembled then the unpleasant surprise of super criticality may occur.

The step by step procedure of adding fuel, counting, and predicting is carried out until the count rate appears to increase slowly without limit. At this point the neutron source is removed from the core and if the count rate holds its own the reactor is indeed critical. If the count rate decreases after removal of the source then the reactor was not critical and an additional small amount of fuel is necessary to obtain the critical mass. By limiting the mass of fuel to small quantities as one gets very close to critical, and by checking count rates with rods in and out, one safeguards against the possibilities of getting so much fuel into the core that criticality is possible with rods in the core.

The above description is just one way of performing this exercise. Other methods can and must be used under circumstances where fuel is not readily accessible. For instance, in power reactors that are loaded and sealed off, one could approach criticality by adjusting the level of the moderator. Another possibility is to change fuel concentration in a liquid uranium fuel mixture until criticality is reached. Obviously, the particular design will dictate the procedure to be used but the basic idea is the same.

This concludes the discussions on Reactor Physics which took the student from considerations of the intensive properties of reactor materials to the point where these materials, and their nuclear properties, were used to create a critical assembly capable of supporting a steady state chain reaction. The subject of the next chapter will be a natural extension of the investigation thus far. In particular, the behavior of the reactor when the steady state is disturbed will be the point of interest.

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REACTOR KINETICS

6.0 Introduction

The past chapter dealt with the steady state reactor system of a just critical reactor in which the multiplication factor k was unity. Any change from the condition k = 1 will result in either a super-critical or sub-critical reactor depending upon whether k > 1 or k < 1. Figure 1 shows the behavior of the flux with respect to time for the three conditions described above.

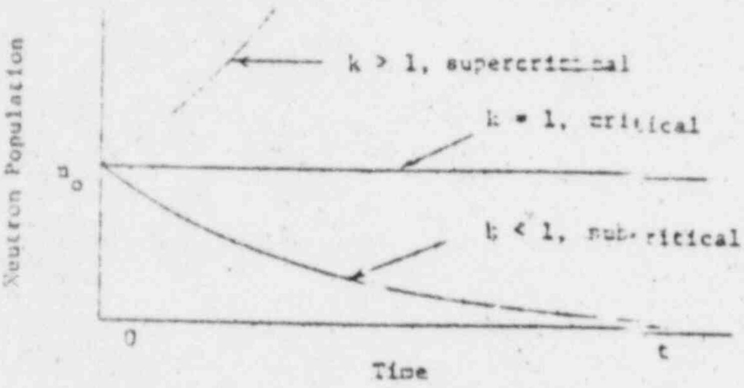

Figure 1

Time Variation of Neutron Flux for Various Reactor Conditions

The study of this dynamic behavior is what is encompassed in Reactor Kinetics.

For the most part, this chapter will deal with the dynamic or transient effects brought about by the movement of control rods. The control rods, of necessity, initiate immediate or short term effects in reactor behavior. Other time dependent effects are caused by fuel burn-up, fission product inventory, and temperature changes; however, these changes are long term and will be discussed in more detail in subsequent sections.

6.1 Reactivity

As previously mentioned, the subject of reactor kinetics deals with a system changing from k = 1. Reactivity (ρ) is defined then as a measure of the deviation of a reactor system from the critical condition k = 1.

$$\rho = \frac{k_{ex}}{k_{eff}} \qquad (1)$$

where k_{ex} is the excess reactivity. Perhaps a more accurate mathematical definition of reactivity would be

$$\rho = \frac{k_{eff} - 1}{k_{eff}} \qquad (2)$$

where it can now be seen that ρ can be either positive or negative. If one limits ρ so that k_{eff} is never far from unity, then the following is true.

$$\rho = \frac{k_{eff} - 1}{k_{eff}} \cong \frac{k_{eff} - 1}{1} = k_{eff} - 1 \qquad (3)$$

and frequently for small reactivity changes one writes

$$\rho = k_{eff} - 1 = \delta k \qquad (4)$$

Suppose one had a critical reactor and at a given instant of time a small amount of fissionable material whose reactivity worth $\delta k = 0.0001$ was inserted into the reactor. What would k be equal to after the insertion? Notice that the subscript on k is dropped and k is intended to mean k_{eff}.

$$\begin{aligned} k &= \delta k + 1 \\ &= 0.0001 + 1 = \underline{1.0001} \end{aligned}$$

The same reactor is made critical again, i.e., k = 1, and a small amount of material is inserted once more into the core and the resulting k is 0.9998. What was the reactivity worth of the second material? Again

$$\begin{aligned} \delta k &= k - 1 \\ &= 0.9998 - 1 \\ &= - 0.0002 \end{aligned}$$

Notice though that δk can be either negative as in this example or positive as in the previous example. The sign of δk will determine whether one has a super-critical or a sub-critical reactor assuming the system is just critical when the addition is made. In Figure 1, the situation described by $k > 1$ was obviously the result of adding a positive δk and, conversely, the decreasing exponential associated with $k < 1$ was negative δk added to the critical system.

Reactivity is quite frequently referred to in percent. For instance, the result of the second example would be referred to as −0.02% δk/k. Notice that something new has been added. Reactivity is a unitless quantity and to keep it so, yet not confuse a reactivity value with just an ordinary decimal or percentage value, the "unitless units" of δk/k are carried along with the number. So then a reactivity of

$$\delta k = 0.001 \ \delta k/k$$

is the same as 0.1% δk/k but has the opposite effect when inserted into a critical reactor has a reactivity of

$$\delta k = -0.1\%\ \delta k/k$$

In a later section another unit of reactivity, the "dollar", will be defined.

Another numerical example will again assume that initially k = 1 for a given reactor and δk = 0.0025. How many neutrons will there be in the next generation?

$$k = \delta k + 1$$

$$= 0.0025 + 1 = 1.0025$$

also,

$$k = \frac{\text{neutrons in n + 1 generation}}{\text{neutrons in n generation}}$$

Substituting the value for k into the above equation, one gets

$$1.0025 = \frac{n + 1}{n}$$

therefore,

$$n + 1 = 1.0025\ n$$

or the population has multiplied 1.0025 times in the next generation.

6.2 Time Dependence of Flux

The fact that neutron population, flux, or power (since they are all proportional) increase with time for a positive reactivity insertion will be shown mathematically as follows:

Assuming once more that ρ is small, one can write as before

$$\delta k = k - 1 \qquad (5)$$

from which

$$k = 1 + \delta k \qquad (6)$$

and

$$k = \frac{n + 1}{n} \qquad (7)$$

where n + 1 and n refer to numbers of neutrons in the first and second generations.

Therefore,

$$\frac{n + 1}{n} = 1 + \delta k \qquad (8)$$

Equation (8) can be rewritten

$$n + 1 = n\ (1 + \delta k) \qquad (9)$$

and then the ratio between second and third generations is

$$\frac{n+2}{n+1} = 1 + \delta k \qquad (10)$$

substituting Equation (9) into Equation (10)

$$\frac{n+2}{n(1+\delta k)} = 1 + \delta k \qquad (11)$$

or

$$\frac{n+2}{n} = (1 + \delta k)^2 \qquad (12)$$

which is the ratio between the first and third generations of neutrons. It follows then that the ratio between the first and the m^{th} generation is

$$\frac{n+m}{n} = (1 + \delta k)^m \qquad (13)$$

Since flux is equal to the number of neutrons in a generation, one can write

$$\frac{\phi}{\phi_o} = (1 + \delta k)^m \qquad (14)$$

where ϕ_o is flux at time zero and ϕ is flux some time later. Then

$$\ln\frac{\phi}{\phi_o} = m\ln(1 + \delta k) \qquad (15)$$

and by a series expansion of ln (1 + δk)

$$\ln\frac{\phi}{\phi_o} = m\left(\delta k - \frac{\delta k^2}{2} + \frac{\delta k^3}{3} - \ldots\right) \qquad (16)$$

Because δk is purposely kept small all the terms in the expansion, except the first term, are negligible so that

$$\ln\frac{\phi}{\phi_o} = m\delta k \qquad (17)$$

Taking antilogs

$$\phi = \phi_o\, e^{m\delta k} \qquad (18)$$

Now let t be the time in seconds which elapses after δk is inserted into the reactor. If ℓ is the neutron generation lifetime, then

$$m = \frac{t}{\ell} \qquad (19)$$

or the number of generations is determined by the time available for the generations to come into existence divided by time to go through one generation. Substituting Equation (19) into Equation (18) yields

$$\phi = \phi_o\, e^{t\delta k/\ell} \qquad (20)$$

Now let the reactor period T be defined as the amount of time required for the flux to change a factor of 2.718 (e). Period is sometimes referred to as "e" folding time.

Thus in Equation (20), when $t = T$, the following is true by definition

$$\frac{\phi}{\phi_o} = e^1 \tag{21}$$

Comparing exponents in Equations (20) and (21), one sees that

$$1 = \frac{T\delta k}{\ell} \tag{22}$$

or

$$T = \frac{\ell}{\delta k} \tag{23}$$

which, when substituted into Equation (20), allows one to write

$$\phi = \phi_o \, e^{t/T} \tag{24}$$

Equation (24) now describes the time behavior of reactor flux, consequently power, for a small positive reactivity insertion.

Before showing some numerical examples, a closer look at neutron lifetime, or generation time is perhaps in order. Generation time is the quantity ℓ introduced in Equation (19) above. For an infinite medium of pure moderator the mean (or average) neutron lifetime is described by

$$\ell_o = \frac{\lambda_a}{v} \tag{25}$$

where λ_a is the mean free path a neutron travels between absorptions, and v is the velocity. To account for the decrease in lifetime when one considers a finite system, and a system which is not pure moderator, some new expressions emerge. The latter effect is shown first by re-writting Equation (25) as

$$\ell_o = \frac{1}{\Sigma_a v} \tag{26}$$

Now one clearly sees that as other materials are added besides moderator, Σ_a becomes larger which decreases the lifetime. Then when one multiplies Equation (26) by the thermal non-leakage probability, lifetime becomes

$$\ell = \ell_o P_t = \frac{\ell_o}{1 + L^2B^2} = \frac{1}{v\Sigma_a\,(1 + L^2B^2)} \tag{27}$$

Assuming that a good moderator is used, then the high energy, or fast neutrons will spend little time slowing down to thermal; thus, ℓ precludes slowing down time from fast to thermal and is primarily a measure of diffusion time at thermal energies. The validity of this assumption is demonstrated by the following approximation for graphite:

$$\ell_{total} = \ell_{fast} + \ell_{thermal}$$

$$= \frac{1}{\xi\Sigma_s\, c}\left(\frac{2mc^2}{E_{th}}\right)^{\frac{1}{2}} + \frac{1}{\frac{N_a\,\rho}{A}\,\sigma_a\, v}$$

$$= \frac{(2 \times 9331 \times 10^6/0.0225)^{\frac{1}{2}}}{0.158(0.385)(3 \times 10^{10})}$$

$$+ \frac{12}{6.022 \times 10^{23}(1.622)(3.4 \times 10^{-27})(2.2 \times 10^5)}$$

$$= 1.5 \times 10^{-4} + 1.64 \times 10^{-2} \text{ seconds}$$

$$= 1.655 \times 10^{-2} \text{ seconds}$$

which for all practical purposes is the thermal diffusion time as indicated for graphite in Table 1 below.

Table 1

Slowing Down and Diffusion Times for Neutrons $E_o = 2$ Mev to $E_{th} = 0.0225$ev

Moderator	Slowing Down Time (Seconds)	Diffusion Time (Seconds)
Water (H_2O)	1×10^{-5}	2×10^{-4}
Water (D_2O)	4.6×10^{-5}	1.5×10^{-1}
Beryllium	6.7×10^{-5}	4.3×10^{-3}
Graphite	1.5×10^{-4}	1.2×10^{-2}

For a large reactor like the X-10, a generation time on the order of $\ell = 10^{-3}$ seconds is a valid number. Recall that X-10 is a heterogeneous mixture of natural uranium and graphite.

Let an example be offered here to examine the change of flux for a given reactivity insertion. Starting with a critical reactor in which neutron lifetime $\ell = 10^{-3}$ seconds and for a positive reactivity insertion of $\delta k = 1\%$ $\delta k/k$, let the flux increase for 1 second. What will be the flux ratio at the end of 1 second?

From Equation (23)

$$T = \frac{\ell}{\delta k}$$

$$= \frac{10^{-3}}{10^{-2}} = 0.1 \text{ second}$$

and from Equation (24)

$$\phi = \phi_o \, e^{t/T}$$

$$= \phi_o \, e^{1/0.1}$$

$$= \phi_o \, e^{10}$$

$$\frac{\phi}{\phi_o} = 21000$$

or in the one second time during which power was allowed to increase, the level would go up 21000 times.

Looking at the same example, let the lifetime of the neutrons be increased so that now $\ell = 10^{-2}$ seconds, then for the same δk and the same time

$$T = \frac{\ell}{\delta k}$$

$$= \frac{10^{-2}}{10^{-2}} = 1 \text{ second}$$

and

$$\frac{\phi}{\phi_o} = e^{t/T}$$

$$\frac{\phi}{\phi_o} = e^{1/1} = 2.718$$

Quite a reduction in the power increase for the same reactivity insertion. Notice that the big difference is in the lifetime ℓ of the neutron generations. "Is this only of academic interest?", one may ask. Indeed not. for the interplay between period, lifetime, and reactivity will be established in a few sections hence. Before that development, however, a subject which must now be considered is that of the neutrons in the various generations. As indicated in the last chapter, not all of the neutrons which result from the fission process appear immediately as prompt neutrons. A small portion of the neutrons are delayed and these two categories of neutrons, i.e., prompt and delayed, will be discussed in the next sections.

6.3 Prompt Neutrons

As mentioned in one of the early sections of the last chapter, one of the most important products of the fission process is neutrons. Greater than 99% of the fission neutrons are liberated instantaneously (from 10[illegible] seconds to 10^{-13} seconds after the fission event occurs).

These neutrons are called prompt neutrons and it is to these neutrons with their relatively short lifetime that the ℓ in the expression

$$\phi = \phi_o e^{t\delta k/\ell} \qquad (28)$$

in the last section refers.

Since prompt neutron lifetime has been so specifically defined, now let the symbol used for prompt neutron lifetime be designated as ℓ^* to differentiate it from ℓ which will be used to mean something differant in subsequent presentations.

The numerical examples of the last section show that when one considers only prompt neutrons in the reactor time behavior, a rather large control problem develops because of the brief generation time unless one keeps reactivity insertions very small. The saving feature as witnessed again from numerical examples is the lengthening of generation time. This is accomplished by the less than 1% of the fission neutrons not accounted for as prompt (not to be confused with fast which refers to neutron energy). This small fraction of neutrons is discussed in the next section.

6.4 Delayed Neutrons

Unlike their prompt brothers, delayed neutrons do not come directly from the fission process but originate from the unstable nuclides or fission products. These nuclides are known as precursors and since they are radioactive they have associated with them specific decay times. The decay times of the precursors actually determine the amount of delay or lag that the neutrons experience in entering the chain reaction.

A typical example of a delayed neutron precursor would be the 54.4 second half-life isotope of Bromine-87. Ultimately Br-87 goes to stable Krypton-86 or to radioactive Kr-87 which decays by the following scheme:

$${}_{36}Kr^{87} - {}_{-1}\beta^{0} \longrightarrow {}_{37}Rb^{87} - {}_{-1}\beta^{0} \longrightarrow {}_{38}Sr^{87} \text{ (stable)}$$

The interesting feature though is the way in which Br-87 gets to Kr-87 and this scheme is shown below:

$${}_{35}Br^{87} - {}_{-1}\beta^{0} \longrightarrow {}_{36}Kr^{87} \qquad \text{30\% of the time}$$

$${}_{35}Br^{87} - {}_{-1}\beta^{0} \longrightarrow {}_{36}Kr^{87} - 2\gamma \qquad \text{60\% of the time}$$

$${}_{35}Br^{87} - {}_{-1}\beta^{0} \longrightarrow {}_{36}Kr^{87} - {}_{0}n^{1} \longrightarrow {}_{36}Kr^{86} \qquad \text{2\% of the time}$$

and it is the last scheme with its emission of one neutron 2% of the time which places Br-87 in the delayed neutron precursor category.

Physicists have identified at least a half dozen and possibly 20 nuclei such as Br-87 which may eventually lead to a delayed neutron emission. These precursors can be grouped into six distinct groups depending upon the half-lives of their respective decay schemes. This gives rise to the six delayed neutron groups which are tabulated for U-235, Pu-239, and U-233 as seen in Table 2 on Page 6-10.

As mentioned previously, the delayed neutrons can be identified as coming from six distinct groups of precursors. The term λ_i then refers to the decay constant of the ith group of delayed neutrons (actually the decay constants of the precursors comprising the ith group). The relationship between half-life ($t_{1/2}$) and λ_i then is

$$\lambda_i = \frac{0.693}{t_{1/2i}} \tag{29}$$

where 0.693 is the natural log of two. From the reciprocal value of λ_i one finds t_i or the mean lifetime of the ith group of delayed neutrons.

$$t_i = \frac{1}{\lambda_i} \tag{30}$$

Of course, β_i is the fraction of delayed neutrons in the ith group and as indicated in Table 2

$$\sum_{i=1}^{6} \beta_i = \beta \tag{31}$$

and is the total fraction of delayed neutrons referred to as the beta fraction.

A few words of explanation about some conflicting values of β that might be obtained from various texts. As a result of some work done back in 1947, a value of 0.00755 was assigned to β. Now, by virtue of more contemporary techniques, β is found to be 0.0064 for U-235. This is a very valid number and is characteristic of U-235 in any reactor. However, when β is measured for a particular core configuration, by first determining ℓ then oscillating the core to determine β/ℓ, a value of β may emerge which is closer to the 0.00755 number.

One resolves this situation by resorting to an effectiveness factor which shall be termed γ. Then the expression for the fraction of delayed neutrons becomes

$$\beta_{eff} = \gamma \beta \tag{32}$$

where β_{eff} is the actual measured value from an operating core. The γ accounts for the fact that delayed neutrons are given off at lower energies than prompt neutrons; therefore, their effectiveness is enhanced because they do not leak out of the system as readily as prompt neutrons. Also, if one is talking about a large reactor, there is less chance for leakage of the lower energy delayed neutrons because of the increased size. Then depending upon the features of a particular system β can look like a larger number by a factor γ. For instance, in the PSTR, β is

Table 2

Delayed Neutron Data for Thermal Fission of U-235, Pu-239, and U-233

Neutron Group i	Fraction in ith Group β_i	Decay Constant λ_i sec^{-1}	Mean Life t_i sec	Product $\beta_i\ t_i$ sec x 10^{-2}
		U-235		
1	0.00027	3.01	0.33	0.009
2	0.00074	1.14	0.88	0.065
3	0.00252	0.301	3.31	0.834
4	0.00125	0.111	8.97	1.125
5	0.00140	0.0305	32.8	4.592
6	0.00021	0.0124	80.4	1.688
	$\sum_{i=1}^{6} \beta_i = .0064$			$\sum_{i=1}^{6} \beta_i t_i = .08315$
		Pu-239		
1	0.000092	2.70	0.371	0.00345
2	0.000178	1.12	0.892	0.016
3	0.000684	0.326	3.07	0.206
4	0.000443	0.124	8.08	0.358
5	0.000625	0.0301	33.2	2.073
6	0.000072	0.0128	78.3	0.564
	$\sum_{i=1}^{6} \beta_i = .0021$			$\sum_{i=1}^{6} \beta_i t_i = .03222$
		U-233		
1	0.000086	2.50	0.4	0.00344
2	0.000132	1.13	0.887	0.0117
3	0.000722	0.326	3.07	0.222
4	0.000651	0.139	7.22	0.470
5	0.000773	0.034	29.7	2.295
6	0.000224	0.0126	79.4	1.780
	$\sum_{i=1}^{6} \beta_i = .0026$			$\sum_{i=1}^{6} \beta_i t_i = .04782$

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0.0064 for U-235 but the reactor behaves (via measurement) as though β were 0.007 so one can determine γ by

$$\gamma = \frac{\beta_{eff}}{\beta}$$

$$\gamma = \frac{0.0070}{0.0064}$$

$$= 1.09$$

or for the PSTR

$$\beta_{eff} = 1.09\ \beta$$

Unfortunately, the subscript β_{eff}, just like the subscript in k_{eff}, is dropped most of the time and when one says β for an operating reactor he generally means β_{eff}.

To illustrate, further, how β might be affected by a particular reactor system, an interesting example can be cited. If beryllium or H_2 is used to any great extent in the reactor core, the (γ,n) reaction which results will actually look like the seventh group of delayed neutrons with an obvious change in β resulting.

Now to show quite simply the effect of delayed neutrons on the lifetime of neutron generations, recall that

$$\beta = \sum_{i=1}^{6} \beta_i \qquad (33)$$

where all terms are as previously defined. Then the fraction of neutrons remaining are prompt or

$$\text{Prompt Fraction} = 1 - \beta \qquad (34)$$

then the mean lifetime of all fission neutrons, prompt and delayed, shall be ℓ and shown mathematically as follows:

$$\ell = (1 - \beta)\ \ell^* + \sum_{i=1}^{6} \beta_i\ t_i \qquad (35)$$

and simplifying still further by realizing that $\beta << 1$, one can write

$$\ell = \ell^* + \sum_{i=1}^{6} \beta_i\ t_i \qquad (36)$$

If some relative values are now inserted into Equation (36), the role of the delayed neutrons in stretching the lifetime is apparent.

For a large reactor $\ell^* = 10^{-3}$ seconds is feasible and the summation term is approximately 10^{-1} so that

$$\ell = 0.001 + 0.1 = 0.1$$

or ℓ is essentially the same as the mean weighted lifetime of the delayed neutron groups. Now the examples shown previously on Page 6-7, perhaps take on more meaning.

6.5 Time Dependence of Flux - Stable Period

As a result of the discussion in Section 6.2, Time Dependence of Flux, the following equation emerged

$$\phi = \phi_o e^{t/T} \qquad (37)$$

This expression is of course an over simplification of the dependence of flux upon time because, as pointed out previously, only the prompt neutrons are considered. To illustrate the contrast between the behavior of the flux from the above equation and an expression which will be investigated a little later, the flux expression will be developed in this manner:

$$\frac{d\phi}{dt} = \frac{\delta k}{\ell} \phi \qquad (38)$$

where $d\phi/dt$ is the change in flux and ℓ is the prompt neutron lifetime. One sees then that the change in flux is proportional to the flux present, the change in reactivity, and the inverse of the neutron generation lifetime. Integrating Equation (38) yields

$$\ln \phi = \frac{\delta k}{\ell} t + C \qquad (39)$$

where C is a constant of integration and $\ell/\delta k = T$ as previously defined; or

$$\ln \phi = \frac{1}{T} t + C \qquad (40)$$

To evaluate C let it be assumed that at $t = 0$ $\phi = \phi_o$ then substituting into Equation (40) one obtains

$$\ln \phi_o = \frac{1}{T} (0) + C \qquad (41)$$

or

$$C = \ln \phi_o \qquad (42)$$

Now Equation (40) can be written as

$$\ln \phi = \frac{1}{T} t + \ln \phi_o \qquad (43)$$

which corresponds to the equation of a straight line as shown below.

$$y = m x + b \qquad (44)$$

where m is the slope of the straight line and is represented as 1/T in Equation (43). Now if one were to plot Equation (43), the results would be as shown in Figure 2 on Page 6-13.

Observe that when the natural log of the flux is plotted against time, straight line power increases result. Also note that as the

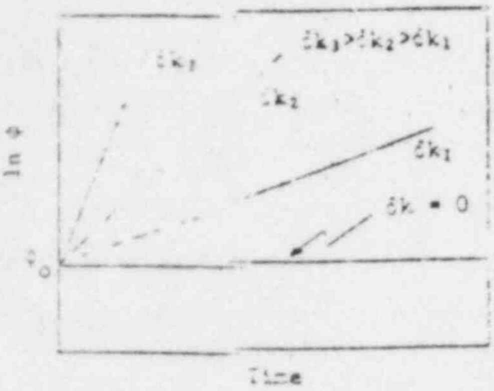


Figure 2

Flux Variation with Time for Prompt Neutrons Only

reactivity which causes the power change is increased, the reciprocal of the period 1/T increases (this is the slope of the line). Thus, the inverse relationship between period and reactivity is simply demonstrated.

A more exact expression for relating period and reactivity is the expression commonly called the inhour equation and is written as follows:

$$\rho = \frac{\omega \ell^*}{1 + \omega \ell^*} + \frac{1}{1 + \omega \ell^*} \sum_{i=1}^{6} \frac{\omega \beta_i}{\omega + \lambda_i} \longrightarrow 0 \text{ as } \omega \to 0 \qquad (45)$$

where ω = 1/T and all other terms are as previously defined. In a future section, the actual inhour equation will be given in addition to defining an inhour of reactivity.

For the present, however, observe that Equation (45) goes to zero as ω approaches zero (implying that T approaches ∞) which is consistent with the inverse relationship. Notice, also, as ω approaches ∞, ρ approaches unity which is consistent with the definition of ρ (Equation (2)). Finally, it is apparent that Equation (45) is a seventh degree polynomial which has seven roots which correspond to seven groups of neutrons (assuming prompt neutrons to be one group). Clearly as each $-\omega$ approaches a λ_i, the term goes to ∞. These seven roots lead to the following expression for the variation of flux with time:

$$\phi = A_o e^{\omega_o t} - A_1 e^{\omega_1 t} + \ldots\ldots. + A_6 e^{\omega_6 t} \qquad (46)$$

An evaluation of the constants shows that all A except A_o are negative as are all ω except ω_o, thus Equation (46) can be rewritten:

$$\phi = A_o e^{\omega_o t} - A_1 e^{-\omega_1 t} - \ldots\ldots. - A_6 e^{-\omega_6 t} \qquad (47)$$

It is now obvious that for anytime after t = 0, or the time when one inserts positive reactivity into the operating reactor, the terms, except the first term, will rapidly go to zero and after a time comparable to $1/\lambda_1$ Equation (47) reduces to

$$\phi = A_o e^{\omega_o t} \qquad (48)$$

where $T \equiv 1/\omega_0$ or the stable period. This is essentially the same as Equation (37) of this section except that after a short time interval when the stable value of T is reached, the generation time becomes ℓ and not ℓ^*.

The short time interval after $t = 0$, known as the transient time, is of interest because this is when a rapid increase in flux occurs. This rapid increase in flux is referred to as the prompt jump. One can think of this prompt jump as happening before the delayed neutrons have a chance to enter into the chain reaction and lengthen neutron lifetime.

The prompt jump effect can be shown by an evaluation of Equation (47) with the appropriate constants but this is quite cumbersome and the same situation can be demonstrated by making some simplifications to Equation (45).

If one clears fractions in Equation (45) and regroups, one gets the following:

$$\rho = \omega\ell^* (1 - \rho) + \sum_{i=1}^{6} \frac{\omega\beta_i}{\omega + \lambda_i} \qquad (49)$$

Now from the definition of ρ, one finds that

$$k_{eff} = \frac{1}{1 - \rho} \qquad (50)$$

then after substituting Equation (50) into Equation (49), one finds that

$$\rho = \frac{\omega\ell^*}{k_{eff}} + \sum_{i=1}^{6} \frac{\omega\beta_i}{\omega + \lambda_i} \qquad (51)$$

Performing the summation yields

$$\rho = \omega\ell^* + \frac{\omega\beta}{\omega + \lambda} \qquad (52)$$

where β is the sum of all the fractions of precursors, λ is the average decay constant, and $k_{eff} \simeq 1$ because ρ is small.

By clearing fractions in Equation (52) and regrouping, one can write

$$\ell\omega^2 + (\beta - \rho + \ell\lambda)\,\omega - \rho\lambda = 0 \qquad (53)$$

which is quadratic in ω. Now the solutions to Equation (53) are, after simplification:

$$\omega_o = \frac{\lambda\rho}{\beta - \rho} \quad \omega_1 = -\left(\frac{\beta - \rho}{\ell}\right) \qquad (54)$$

The whole purpose of the development from Equation (49) to the present is to show that one can illustrate prompt jump by assuming that the six neutron groups can be expressed by an average term corresponding to the right hand side of Equation (52). Then by analogy to Equation (47), one can write for the flux variation with time

$$\phi = A_o e^{\omega_o t} - A_1 e^{\omega_1 t} \qquad (55)$$

where ω_o and ω_1 are given in Equation (54).

Once more it is seen that at $t = 0$ the first term is A_o and the second term is $-A_1$, both terms will behave exponentially with increasing time. Since ω_o is positive, the first term will increase at a rate determined by the size of the reactivity insertion, conversly, the second term will decrease at a rate determined by the same reactivity insertion. The second term is the transient term and will disappear in a short time and the behavior of the reactor will be as in Equation (48). Flux will increase at a steady state or on a stable period.

However, in the short time interval after the reactivity insertion while both terms exist, the behavior of the flux is the sum of the two terms and is shown in Figure 3. Notice that the rapid decay of the transient term allows a rapid rise of the second term to its stable period increase; thus, the prompt jump associated with a sudden increase of reactivity is explained graphically.

If one now compares Figure 3 with Figure 2 where only prompt neutrons were considered, it is obvious that Figure 2 is just the first portion of the curve illustrated in Figure 3.

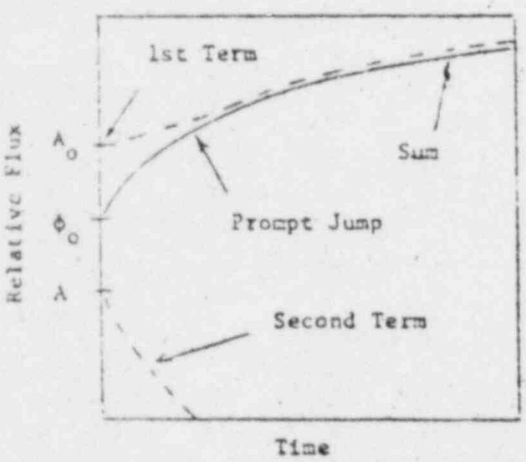


Figure 3

Time Behavior of Flux with Prompt and Delayed Neutrons for Positive ρ

It is interesting to observe that the same equation which was used for the discussion of flux behavior for a positive reactivity insertion also predicts the reactor time response of negative reactivity insertions. The values of A_o and A_1 in Equation (55) are given by

$$A_o = \frac{\beta}{\beta - \rho}\ \phi_o \qquad (56)$$

$$A_1 = -\frac{\rho}{\beta - \rho}\ \phi_o \qquad (57)$$

so that Equation (55) becomes

$$\phi = \phi_o\ \frac{\beta}{\beta - \rho}\ e^{\frac{\lambda\rho}{\beta - \rho}t} - \phi_o\ \frac{\rho}{\beta - \rho}\ e^{-\frac{(\beta - \rho)}{\ell}t} \qquad (58)$$

by the inclusion of Equations (54), (56), and (57).

Notice now that both A_o and A_1 are positive for a negative reactivity insertion and that both exponents become negative. This results in a net flux which as in the +ρ case decreases sharply but then reaches a steady rate of decrease which is governed by some weighted average of the longest half-lived precursor. The stable negative period will actually be the half-life of the longest precursor group or 80.4 seconds. This can be appreciated by again considering the seven polynomial terms in Equation (46). As the transient terms die out the dominant ω becomes the one associated with the longest half-life. Figure 4 shows the affect of a sudden negative reactivity insertion.

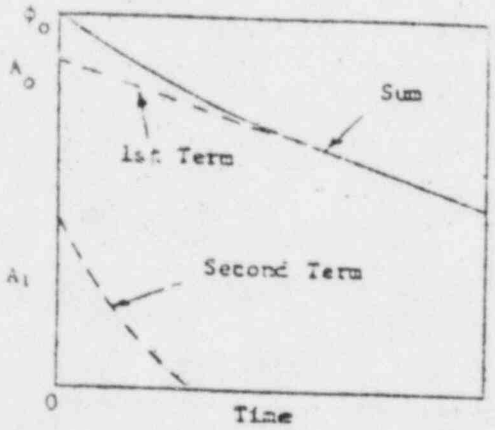


Figure 4

Time Behavior of Flux with Prompt and Delayed Neutrons for Negative ρ

6.6 Prompt Critical

In all discussion up to this point involving a super-critical reactor (k > 1), reactivity insertions were limited to small amounts so that k never deviated much from unity. Under these conditions the reactor is said to be delayed critical and all the equations involving k are valid because k was allowed to be used as unity. If, however, a large amount of reactivity is inserted into a critical reactor it is possible that the reactor would increase power so rapidly that the delayed neutrons would have no affect upon generation time of neutrons and the reactor is then said to be prompt critical. A situation such as prompt-criticality can be quite dangerous if the fuel elements are not capable of handling the large quantities of heat which will be generated in a relatively short period of time. Further, if the reactor does not have a large negative temperature coefficient, temperatures may again reach extreme proportions. Temperature coefficients will be dealt with in subsequent sections. For now all that needs to be said is that some reactors have an inherent ability to add negative reactivity to the system as the core temperature increases. By this device, then, temperatures and powers of prompt critical reactors could be limited to reasonable values.

The amount of reactivity required to make a system prompt-critical is found in the following manner: since only prompt neutrons are involved with the multiplication factor for a prompt-critical reactor, then k becomes $(1 - \beta)\ k$ and now defining the situation for prompt-critical as

$$(1 - \beta)\ k = 1 \tag{59}$$

and recalling again from the definition of k

$$k = \frac{1}{1 - \rho} \tag{60}$$

one can substitute Equation (60) into Equation (59) and find that

$$1 - \beta = 1 - \rho \tag{61}$$

or the minimum condition for prompt criticality is

$$\rho = \beta. \tag{62}$$

That is, the reactor will increase in power under the influence of prompt neutrons only when the positive reactivity insertion is an amount of reactivity equal to the fraction of neutrons which are delayed.

The amount of reactivity which will make a reactor core prompt-critical is the basis for a system of units for reactivity measurements. The "dollar" is defined as a β amount of reactivity. For example, if a given reactor has a $\beta = 0.007$ then the amount of reactivity needed to take the reactor prompt critical is 0.007 $\delta k/k$ or 0.7% $\delta k/k$ or $1.00 of reactivity.

Suppose one were to insert 0.0035 $\delta k/k$ into the same reactor. How much would this be worth in the new reactivity units?

$$\rho_{(\$)} = \frac{\rho\ (\delta k/k)}{0.007\ (\delta k/k/\$)}$$

$$= \frac{0.0035\ (\delta k/k)}{0.007\ (\delta k/k/\$)}$$

$$= \$0.50 \text{ or } 50 \text{ cents}$$

The advantage, then, of a system of units such as this is that the affect of a "dollar" is the same for any reactor. This does not say that a dollar is the same for all reactors. Obviously, if β is different the dollar is different but non-the-less the affect is the same, i.e., prompt criticality.

6.7 The Inhour Equation

Recall that in Section 6.5, Equation (45), was identified as what is commonly referred to as the inhour equation. It was later shown that Equation (45) could be written as

$$\rho = \frac{\omega \ell^*}{k_{eff}} + \sum_{i=1}^{6} \frac{\omega \beta_i}{\omega + \lambda_i} \qquad (63)$$

The actual inhour equation is a ratio of ρ in units of $\delta k/k$ to one inhour or

$$\rho_{In} = \frac{\rho\ \delta k/k}{1\ \text{Inhour}} \qquad (64)$$

where an inhour (short for inverse hour) is that amount of reactivity which, when inserted into a critical reactor, will cause power to increase on a one hour stable period. Since the period involved will be stable or $T = 1/\omega_o$, then Equation (63) can be written as

$$\rho = \frac{\ell^*}{T k_{eff}} + \sum_{i=1}^{6} \frac{\beta_i}{1 + T\lambda_i} \qquad (65)$$

Further, if the stable period is to be 1 hour or 3600 seconds, an inhour of reactivity can be evaluated in the following manner:

$$\rho = \frac{10^{-3}}{3600} + \frac{0.00027}{1 + 3600(3.01)} + \frac{0.00074}{1 + 3600(1.14)}$$

$$+ \frac{0.00253}{1 + 3600(0.301)} + \frac{0.00125}{1 + 3600(0.111)}$$

$$+ \frac{0.00140}{1 + 3600(.0305)} + \frac{0.00021}{1 + 3600(.0124)} \qquad (66)$$

where $\ell^* = 10^{-3}$ seconds, $k_{eff} = 1$, and values for β_i and λ_i are those taken from Table 2 for U-235. It is quite obvious then that an inhour is a very small amount of reactivity on the order of 2.3×10^{-5} $\delta k/k$.

When one desires to express reactivity in inhours then, by Equation (64) it is seen that reactivity in inhours (ρ_{In}) is:

$$\rho_{In} = \frac{\dfrac{\ell^*}{Tk_{eff}} + \displaystyle\sum_{i=1}^{6} \frac{\beta_i}{1 + \lambda_i T}}{\dfrac{\ell^*}{3600} + \displaystyle\sum_{i=1}^{6} \frac{\beta_i}{1 + 3600\ \lambda_i}} \qquad (67)$$

The above equation then is known as the inhour equation.

An example calculation involving the three ways which have been described for expressing reactivity follows: Suppose a given reactor has a $\beta = 0.0046$. How many inhours would \$0.50 be equivalent to in this machine?

$$\$1.00 = 0.0046\ \delta k/k$$

$$\$0.50 = \frac{0.0046}{2}\ \delta k/k = 0.0023\ \delta k/k$$

$$\rho_{In} = \frac{\rho \delta k/k}{1\ \text{Inhour}} = \frac{2.3 \times 10^{-3}\ \delta k/k}{2.3 \times 10^{-5}\ \delta k/k/\text{Inhour}}$$

= 100 inhours is equivalent to 50¢ worth of reactivity.

6.8 Period Approximations

If one were interested in determining what stable period would result from a given ρ insertion or conversly, how much ρ to insert to put the reactor on a certain T, then the need for a solution to the equation

$$\rho = \frac{\omega \ell^*}{k_{eff}} + \sum_{i=1}^{6} \frac{\omega \beta_i}{\omega + \lambda_i} \qquad (68)$$

would arise. Something else which would become apparent is the cumbersome project which arises in making repeated evaluations of Equation (68). Faced with the needs of the above situation, one could start by defining two areas of interest, large and small ρ insertions, then proceed to make assumptions until Equation (68) becomes less cumbersome to manipulate.

First, consider the case where one assumes SMALL reactivity insertions and since one is concerned not with transient effects but with stable periods, then $\omega = \omega_o$ and

$$\rho = \frac{\omega_o \ell^*}{k_{eff}} + \sum_i \frac{\omega_o \beta_i}{\omega_o + \lambda_i} \qquad (69)$$

because ρ is small, $\omega_o \ell^*$ will also be small thereby making the first term on the right hand side of Equation (69) negligible; thus

$$\rho = \sum_i \frac{\omega_o \beta_i}{\omega_o + \lambda_i} \tag{70}$$

Using a weighted average for $\Sigma\lambda_i$ and letting $\omega_o = 1/T$, one writes

$$\rho = \frac{\beta}{1 + \lambda T} \tag{71}$$

Transposing for T

$$T = \frac{\beta - \rho}{\rho\lambda} \tag{72}$$

which allows an approximation of T or ρ to be made for a particular reactor system by the selection of λ. The approximation for T has an obvious limit at $\rho = \beta$ but then again if this limit is approached, ρ is no longer small.

For LARGE reactivities, the assumption that $\omega_o >> \lambda_i$ can be made thus Equation (69) becomes

$$\rho = \frac{\omega_o \ell^*}{k_{eff}} + \sum_i \beta_i \tag{73}$$

and it follows that

$$\rho = \frac{\ell^*}{T k_{eff}} + \beta \tag{74}$$

where as before $\omega_o = 1/T$. Solving for T

$$T = \frac{\ell^*}{k_{eff}} \cdot \frac{1}{\rho - \beta} \tag{75}$$

Now if $\rho >> \beta$, $(\rho - \beta)^{-1} \approx 1/\rho$ and

$$T = \frac{\ell^*}{\rho} = \frac{\ell^*}{\delta k} \tag{76}$$

which is the same expression that was used when T was defined in terms of prompt neutron lifetime back in Section 6.2.

Finally, the results of Equations (72) and (76) can be combined as follows:

$$T = \frac{\ell^* + (\beta - \rho)\, t}{\rho} \tag{77}$$

where $t = 1/\lambda$. The above approximation allows one to pass smoothly from the realm of small reactivity to that of large reactivity. Notice that as ρ approaches β the term $\beta - \rho$ approaches zero and Equation (77) looks like Equation (76). On the other hand, as ρ becomes small $(\beta-\rho)t$ becomes quite significant when compared to the prompt neutron lifetime ℓ^*, and Equation (77) looks like Equation (72) for small reactivities.

In an attempt to develop a feel for what is small and what is large reactivity, some limits will be set on the approximations which have been developed for estimating reactor periods.

Equation	Limits
$T = \frac{\ell^*}{\rho}$	$\rho > \beta$
$T = \frac{(\beta - \rho)}{\rho\lambda}$	$\rho < \frac{\beta}{2}$
$T = \frac{\ell^* + (\beta - \rho)t}{\rho}$	$\frac{\beta}{2} < \rho < \beta$

Where great accuracy is desired, it will be necessary to go to the inhour equation.

A method which can be used with great convenience to estimate reactor period from a reactor which is increasing in power is the doubling time. Since flux is proportional to power then

$$P_{(t)} = P_o e^{t/T} \tag{78}$$

where $P_{(t)}$ is reactor power at any time t and P_o is power at t = o. If one carefully clocks the time it takes the reactor to increase a factor of two in power then

$$\frac{P}{P_o} = 2 = e^{D.T./T} \tag{79}$$

where D.T. is the time it takes for the power to double. Taking the natural log of both sides of Equation (79), one gets

$$\ln 2 = \frac{D.T.}{T} \tag{80}$$

or the relationship between period and doubling time is

$$D.T. = 0.693\ T \tag{81}$$

Thus, when one determines the doubling time from the recorder charts the period is readily found.

A similar situation arises when one speaks of a start-up rate or SUR which is defined as the INTEGRAL, or whole number decades per minute of power increase. From Equation (78) again one can write

$$\frac{P}{P_o} = 10 = e^{60/T} \tag{82}$$

where the start-up rate is 1 decade per minute and the time is one minute or sixty seconds. Taking logs,

$$\ln 10 = \frac{60}{T} \qquad (83)$$

$$2.3 = \frac{60}{T} \qquad (84)$$

or T = 26 seconds when SUR = 1 decade per minute. As indicated previously the expression

$$T = \frac{26}{SUR} \qquad (85)$$

is valid only for integral values of SUR.

6.9 Control Rod Calibration

A direct application of the inhour equation to an operating reactor is a function that must be performed as soon as a critical mass is assembled, namely, control rod calibration. There are a number of ways in which control rods can be calibrated. A partial list might consist of the following methods:

a) Positive Period - Inhour method
b) Rod Drop
c) Distributed poison
d) Rod oscillation

In this section, only the positive period method will be considered in detail while some of the other methods will be discussed in subsequent chapters.

The theory underlying rod calibration by positive period is implicit in the following equations:

$$P = P_o e^{t/T} \qquad (86)$$

$$D.T. = 0.693T \qquad (87)$$

$$\rho = \frac{\ell^*}{Tk_{eff}} + \sum_{i=1}^{6} \frac{\beta_i}{1 + \lambda_i T} \qquad (88)$$

Recognizing from Equation (88) that reactivity and reactor period are related in the indicated manner, then by Equation (76) a given period means a constant increase in reactor power. When the power increase is a factor of two, the stable period and the doubling time are related by Equation (87).

The calibration of a rod is accomplished by first bringing the reactor to critical at some low power and removing the start-up source. Two points to be made here are: a) the low power is necessary to insure that the effect of the negative temperature coefficient of reactivity does not influence the measurements being made and, b) it is important that the reactor be critical (k = 1.000) because any deviation from unity will either add to or subtract from the worth of the rod to be calibrated.

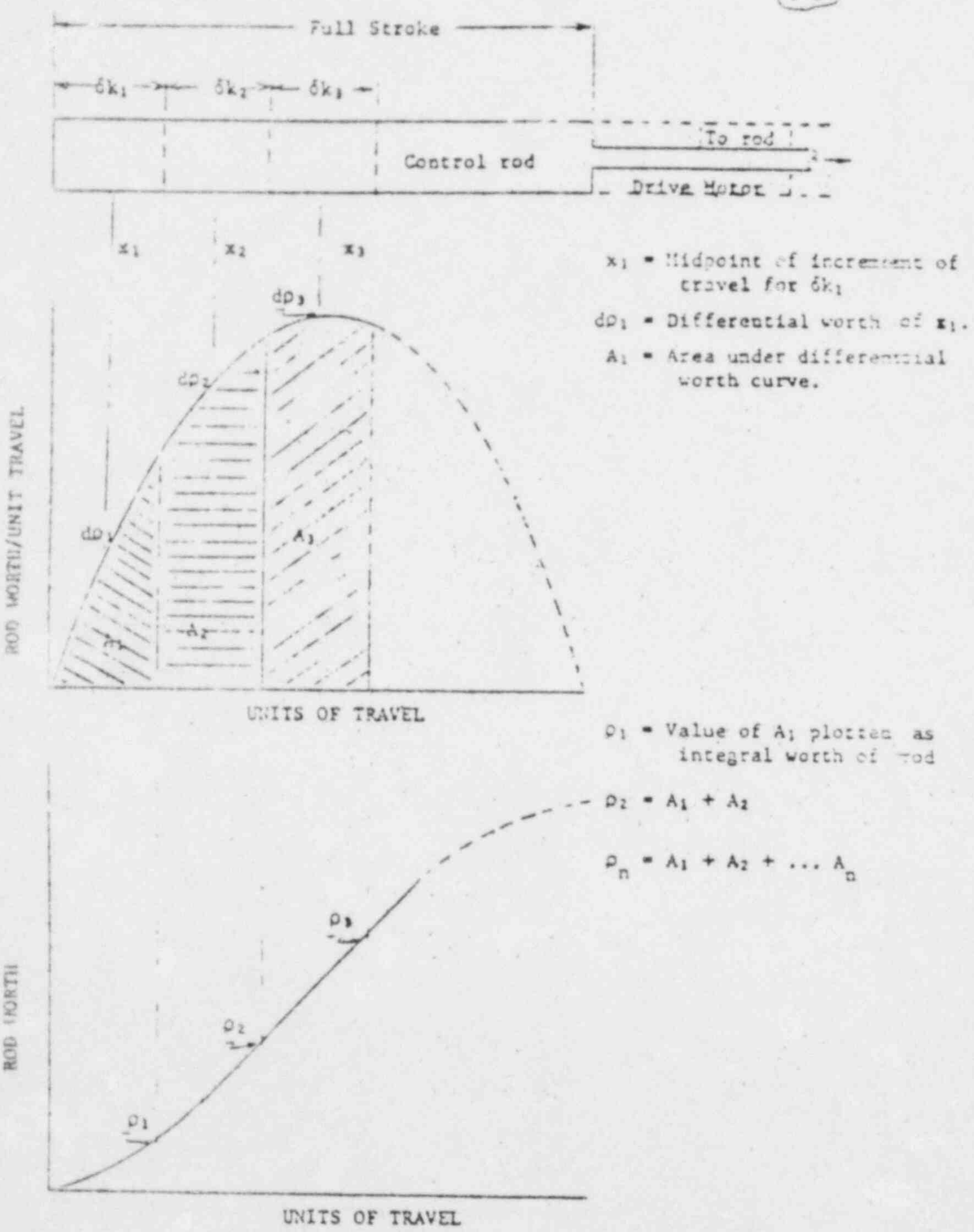
With the rod of interest at the lower limit, withdraw said rod, a small amount (the amount of withdrawal should yield ~ 30 second period). Allow a short period of time for transients to "die out" then determine the doubling time (and by Equation (#7) the period) for the increment of reactivity just inserted. The doubling time should be checked six or eight times before one kilowatt of power is reached. After averaging the best of the six or eight values of period, one would go to a curve which shows reactivity as a function of T or doubling time, as in Equation (#8), and relate the period just determined with a value of reactivity. Then the differential worth of the rod (δk/units of travel) is plotted at the midpoint of the travel of the increment of rod just evaluated.

The resulting curve is a differential rod worth curve and the area under such a curve is the integral rod worth (See Figure 5, Page 6-24). It is the integral rod worth curve which is used in association with a particular rod to relate an increment of rod movement with a reactivity worth.

Repeated reactivity insertions and worth evaluations are performed for the entire stroke of the rod. Each time the doubling time is measured the reactor should be brought back to the original starting power level with some other rod and the operator should be certain that k = 1.000 before making the next reactivity insertion.

In the above manner, each rod or bank of rods is calibrated and a particular calibration curve is assigned to a rod or bank of rods. It must be remembered that rod worths will change as flux changes; consequently, as fuel depletes and fission product poisons build in, rod worths will change. In a research reactor, a heavy experimental loading in the vicinity of a control rod can change the worth of the rod. For these reasons, periodic recalibrations are necessary to keep rod worth curves current.

More remains to be said about the kinetic behavior of the reactor when poisons result from fission products and how temperature changes reactivity; however, these topics will be covered in subsequent discussions on reactor operations.



x_1 = Midpoint of increment of travel for δk_1

$d\rho_1$ = Differential worth of x_1.

A_1 = Area under differential worth curve.

ρ_1 = Value of A_1 plotted as integral worth of rod

$\rho_2 = A_1 + A_2$

$\rho_n = A_1 + A_2 + \ldots A_n$

Figure 5

Rod Travel, Differential and Integral Worth Curves

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REACTOR OPERATIONS

7.0 Introduction

We hope in this section to begin to tie together the various individual lectures which you have already recieved and shall receive during your remaining weeks here. It is hoped that we will be able to draw this material together and relate it to your experience to date in operating our TRIGA reactor and also to relate it to the broader picture of operating a large reactor facility. In this section, we shall discuss some reactor operations philosophy. This section will be mainly intended to stimulate your thinking on developing an attitude towards reactor operations. This area, like most philosophy or philosophical discussions, is open to a great deal of interpretation and consideration depending upon one's past experiences.

7.1 Control Rod Operation

In order for the control rods to serve their intended function, the rate at which each can affect the reactivity balance of the core is important. Safety rods, for example, should have high reactivity worths but do not need to be driven rapidly. On the other hand, they should be capable of being scrammed into the core rapidly. The shim rod, in general, will correct for long term reactivity changes and need not be capable of high speeds. Finally, the regulating rod is required to make, in most cases, small but rapid corrections in power. An exception to this is the situation when the reactor is being taken to power at a time when the xenon concentration is high. Due to the burn-up of xenon present in the core, one generally experiences the most severe reactivity transient. Also to be considered is the fact that provisions for large rates of reacitivity change are built into the control rod system. An accident due to rod drive malfunction becomes more probable because of these large reactivity rates.

Thus, we see some rather complicated and contradicting requirements which must be satisfied. In the PSTR, the problem is relatively easy. To provide a rod with high reactivity worth, we locate it near the core center where the flux is high. (You will recall from your control rod calibration experiment that the rod worth is dependent on flux level). In the case of the regulating rod, a special high speed drive is utilized.

Unfortunately, the situation with a power reactor is not quite that simple. Instead of four control rods, you may have 61 for a PWR or 135 for a BWR type reactor. In a power reactor, several rods are grouped together in a bank and are moved at one time. Also, as operating conditions change, the function of any one control rod cluster or bank of rods may change. Also, the rods will have identical drive systems, thus all the rods will operate at a common speed.

7.2 Control Rod Material

Let us take a look at the material used in a control rod. Since thermal neutrons produce fissioning of Uranium-235, one first considers that any material that will absorb thermal neutrons effectively is a good material to use. However, from a practical point, other considerations are required. The material must be machineable; not suffer from radiation damage; should not change physical properties, (i.e., swell, become brittle, etc.). The list can go on and on. Let us take a look at some materials that have been used in reactor control rods.

Cadmium was used in many research reactors. It has a good thermal neutron cross section (See Table 1), but is expensive and has a low melting point (310°C) which excludes its use in power reactors.

The PSTR uses a boron-carbide rod. The boron-10 isotope has a good absorption cross section. The major disadvantage is the problems caused by the $B^{10}(n,\alpha)Li^7$ reaction. (Gas formation results in internal pressure build-up). It should be noted that as long as low boron-10 concentrations are used in alloys of carbon or steel, satisfactory results are achieved. Such low boron concentrations, however, tend to limit the worth of a control rod.

Table 1

Element of Nuclide	Average thermal cross-section (barns)	Thickness for Blackness
B^{10}	3,470	0.004
Ag	54	.62
Cd	2,210	.0102
In	165	.310
Eu	3,980	.0175
Gd	39,800	.0017
Hf	91	.471

Currently, reactor designers are using materials in their control rods which have effective epithermal cross-sections as well as good thermal cross sections. Thus, if we can effectively capture a neutron in the epithermal region, this in effect supplements the capturing of neutrons after they become thermal. The total worth of the control will be increased in this manner. Many power reactors are using a Cd, In, Ag alloy in their control rods to satisfy not only the neutron absorption requirement but also many of the other physical requirements mentioned earlier.

7.3 Control Rods and Neutron Flux

The shape of the thermal flux in a reactor core at power which is a right circular cylinder with no control rod is shown on Page 7-3.

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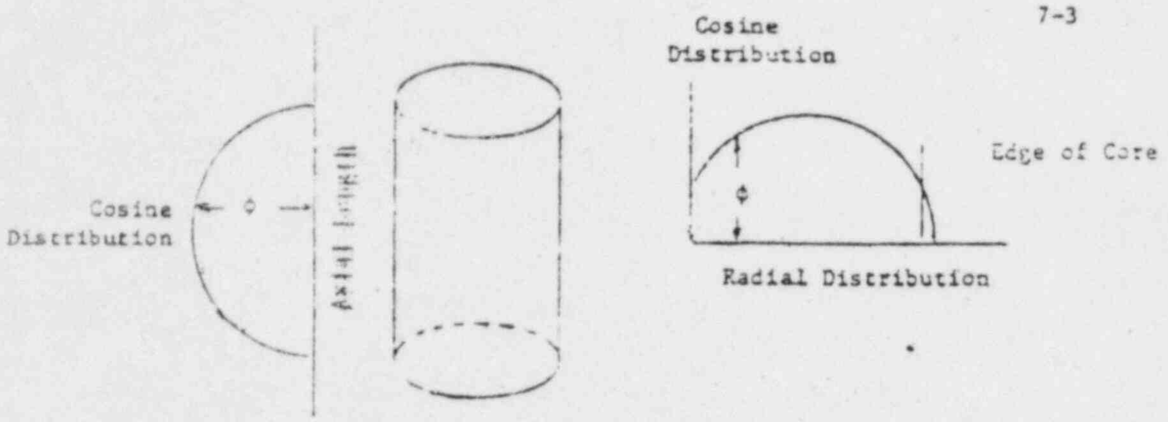

Figure 1

This is the case for a homogeneous core. Now let us see what happens when we place a control rod in the core assuming we maintain the same power level.

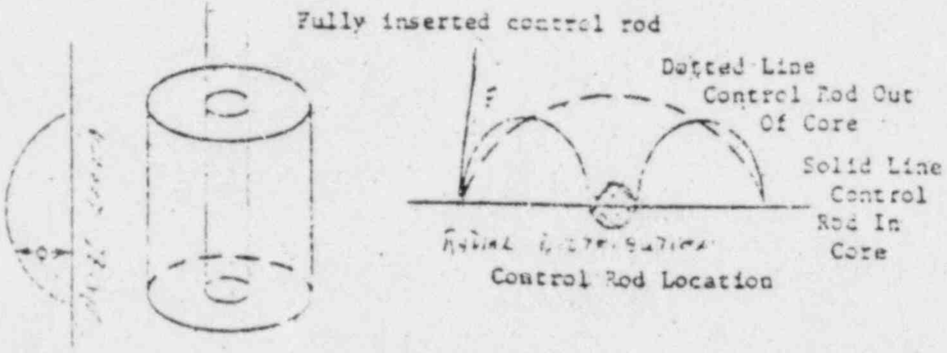

Figure 2

The control rod affects k_{eff} of the core in two ways:

a) by absorption of thermal neutrons

b) by increasing the thermal neutron leakage from the core

It can be shown that the worth of a control rod varies as the square of the flux. From Figure 1, we find the flux distribution to be a cosine, the control rod has its greatest differential worth then when passing through the center of the reactor. You may recall this from your observations during the control rod calibration experiment.

Throughout core life, it will be necessary to have the control rods inserted into the core various amounts for reactivity control purposes. It is of interest then to consider what effect a partially inserted control rod has on flux shapes.

Let us remember that since flux and reactor power are proportional, they may be used interchangeably. The reasoning goes as follows:

$$P = \bar{\phi}\,\bar{\Sigma}_f\,V_f\,C$$

Here we find the reactor power, P, in say thermal megawatts equals the average (indicated by the bar) neutron flux which indicates the number of neutrons present per unit area per unit time. $\bar{\Sigma}_f$ is the average probability per unit neutron path that fissioning will occur. Thus $\phi\,\bar{\Sigma}_f$ is the number of fissions occurring per unit volume per unit time in the core. If this is in turn multiplied by V_f, the total volume of fuel present then we have a measure of the total number of fissions taking place in the core per unit time.

Finally, we must multiply by C a conversion factor (3.2×10^{-17} MW/fission/second) that gives us the reactor power in more meaningful terms.

The importance of this digression again, $P \propto \phi$, i.e., power is proportional to flux. Another way of saying this is when you double the power you double the flux. Now back to control rods and neutron flux.

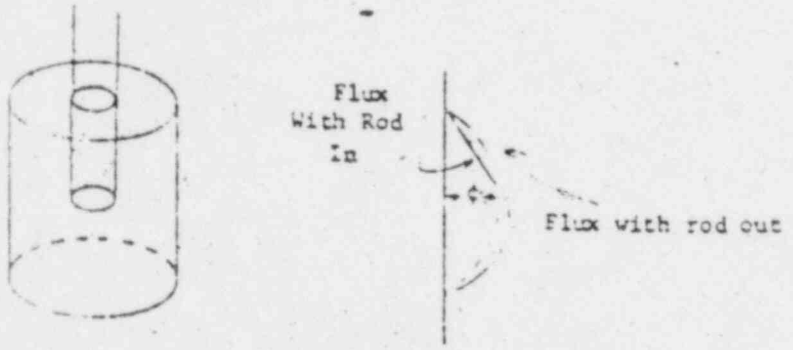

Figure 3

You will find when a control rod is partially inserted, the flux level will tend to bulge out near the bottom of the core. Due to the presence of a strong thermal neutron absorber, the flux will be depressed near the top as in Figure 3. The same situation can arise in the radial direction and is illustrated in Figure 4.

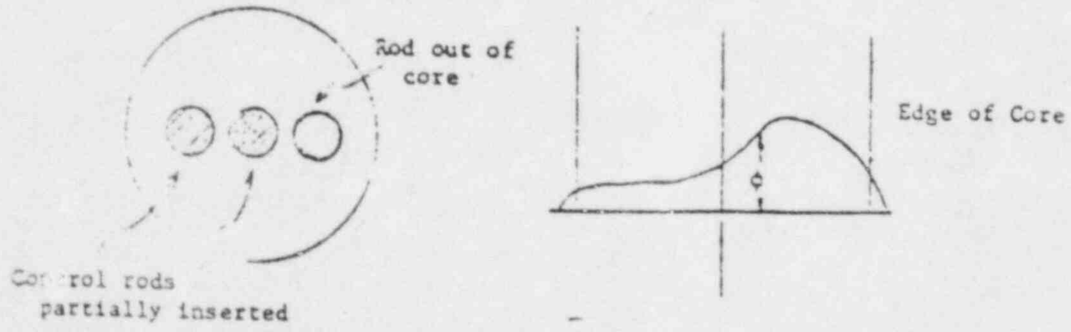

Figure 4

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Now you are ready to ask the question - so what? We will refer back to some previous lectures for the answer.

1. Let us consider what is happening from a thermal standpoint. Recall our digression which demonstrated that power is proportional to flux. Thus, we are really producing a power bulge in the core. Power relates directly to the heat produced in that local region of the core. Thus, overheating of the fuel is possible with fuel melting as a possible result.

2. Let us consider what is happening from the standpoint of nuclear instrumentation. Detectors in close proximity to the flux bulge will read high and those near the flux depression region will read low. Thus, we lose our indication of true power level. This situation is particularly bad when one has to depend on a few detectors to indicate power level.

3. As we will discuss in greater detail, in later chapters, this situation gives rise to xenon oscillations in large reactor cores.

What can be done to minimize this uneven flux distribution? The condition would be alleviated if it were possible to distribute the control rods uniformly over the entire core. The boron which is dissolved in the primary system of a PWR does just that. It provides a good thermal neutron absorber (B^{10}) dispersed over the entire core. This, however, does not do away totally with our need for control rods. Reactivity variations caused by a load change require a more rapid adjustment in reactivity than can be accomplished with a chemical addition or removal system. Thus, we are forced to work with control rods.

Just a few years ago, a reactor used 30 or 40 control rods in a cruciform shape which tended to concentrate the location of the absorber material.

Today's reactors have on the order of 60-90 control rod assemblies, each consisting of 16 or so small individual control rods coupled together in the form of a control rod "spider" (See Figure 5).

In this manner, the designer is able to distribute his control rods over a larger region of the core. This, in turn, makes it possible to obtain a more even power distribution in the core. Ideally we would like to have each section of the core produce the same relative amount of power.

Finally, the movement of control rods by the reactor operator must be carefully considered. One does not want to withdraw rods from one side of the core without removing some from the opposite side. Such action would produce a great imbalance in power distribution. To assist the operator in proper control rod management, incore detectors are used. Such detectors indicate local flux levels and aid the operator in identifying regions of abnormally high or low power

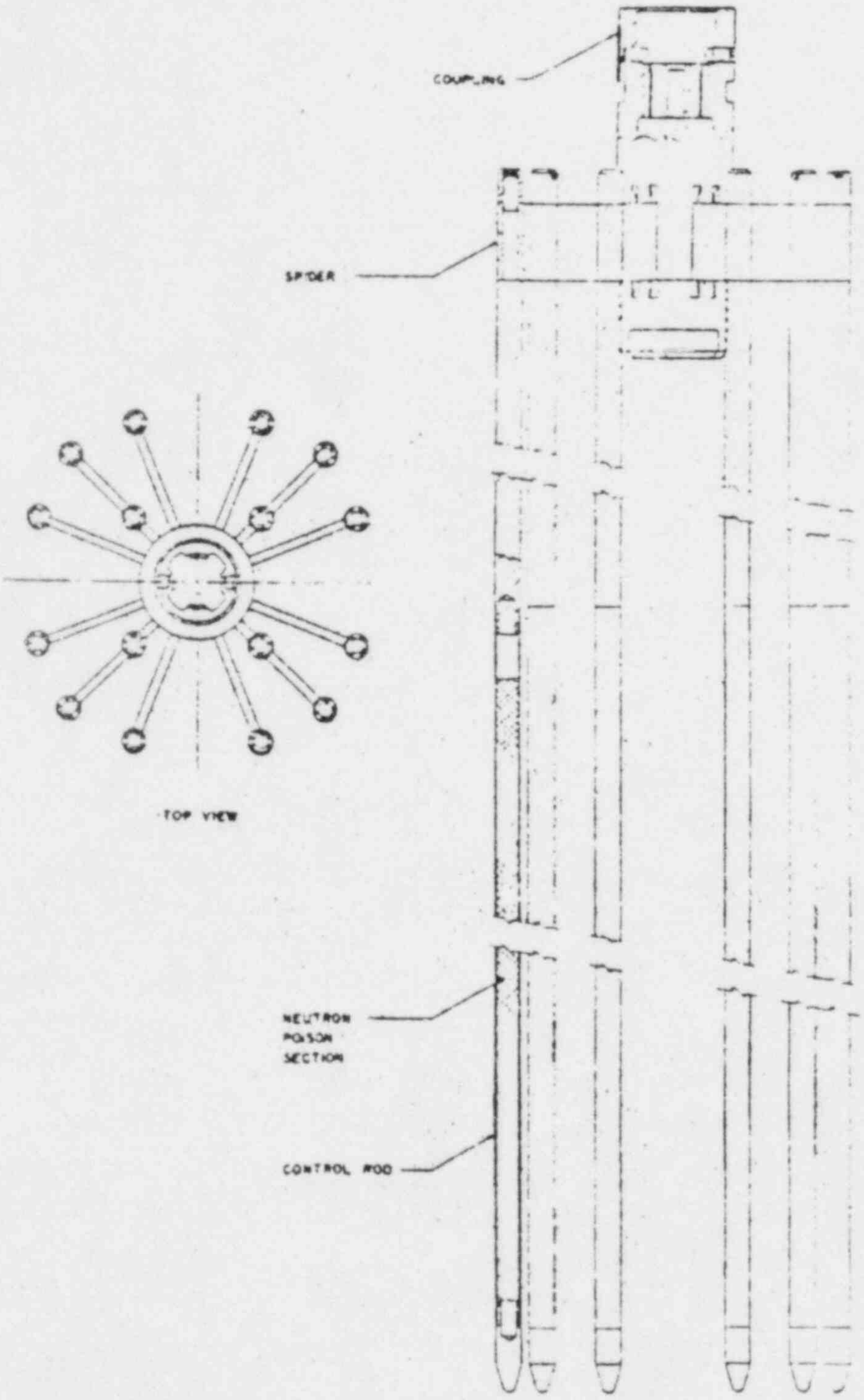


Figure 5

CONTROL ROD ASSEMBLY

FIGURE 3-59

THREE MILE ISLAND NUCLEAR STATION

7.4 Other Methods of Reactivity Control

Although absorbing control rods are the most widely used method for controlling reactivity, they have certain disadvantages. These disadvantages are mechanical complexity, cost, distortions of spatial power distribution, to name a few. A problem also arises frequently in the difficulty in being able to incorporate enough control rods into a core to compensate for the excess reactivity of that core. Often other schemes are combined with control rods to compensate for core excess reactivity.

One such method used commonly in pressurized water reactors today is to dissolve the neutron absorber material in the primary system, as mentioned earlier in this chapter. This is frequently referred to as a soluble poison system. For most power plants of this type, the soluble poison is boron (in the form of boric acid). Boron has the major advantage of being able to uniformly poison down the core and hold down the excess reactivity in the core without creating local flux perturbations. At the same time, one should remember that the soluble poisons generally make the temperature and moderator void coefficients of reactivity less negative.

At high boron concentrations, these coefficients can become positive.

Another means of reactivity control which may be used more and more in future reactor systems is the burnable poison. An example of burnable poison is a material such as gadolinium oxide (Gd_2O_3) which is mixed in with the uranium oxide fuel. In the proper proportion, one is able to utilize burnable poisons not only to hold down excess reactivity in the core at the beginning of core life, but also to provide a poison which will burn out in such a fashion as to give a linear reactivity trend. This allows for compensation of fuel burn-up.

A method sometimes used on the first core is the incorporation of fixed poison control in the core. These could be likened to a control rod without the drive. Generally, during the first fuel loading period these fixed control assemblies can be removed from the core. They are used, therefore, only during the period in which fission product poisons are building into the core. We will, in later sections, take a closer look at a composite picture of control in a power reactor core.

7.5 Reactivity Requirements

One of the basic differences in the mode of operation between a fossil plant and a nuclear plant is that fuel for the fossil plant is continuously added to the system while for the nuclear plant, it is added perhaps once a year. This difference requires that sufficient fuel be loaded into the nuclear plant to allow it to operate until the scheduled refueling shutdown.

The mode of operation requires a certain amount of excess fuel which must be added to the reactor initially to provide for so-called reserve reactivity requirements throughout the lifetime of the core. The reactivity losses which are compensated for by the additional fuel may be broken into the following categories:

a) Fuel depletion
b) Fission product poisoning
c) Temperature effects
d) Control requirements

Let us look at each of these.

7.6 Fuel Depletion

As mentioned earlier, a thermal reactor burns U-235. Therefore, we must add extra fuel to allow for this loss. The amount of extra fuel required for this purpose requires the solution of several differential equations governing fuel burn-up and isotope build-up during core life. Fuel depletion analysis traces a logic as follows:

One fission yields about 200 Mev of energy. This energy yield at the rate of 1 fission per second is equivalent to a power level of 3.2 x 10^{-11} watts (i.e., 3.7 x 10^{10} fissions per second per watt). At this energy yield 1.05 grams of U-235 fissions will result in 1 MW of thermal energy. But we know from previous lectures that not all the neutrons absorbed in U-235 produce fissions, in fact, approximately 18% additional U-235 atoms are destroyed without fissioning. However, fast fission of U 238 in turn reduces the overall consumption of U-235 by 6 to 8% in water reactors. Other non-fission absorptions reduce the total amount of U-235 destroyed even more. The important neutron reactions which occur in uranium fuel are:

$$U^{235} + n \begin{array}{l} \xrightarrow{18\%} U^{236} \\ \xrightarrow{82\%} \text{fission + energy + extra n} \end{array}$$

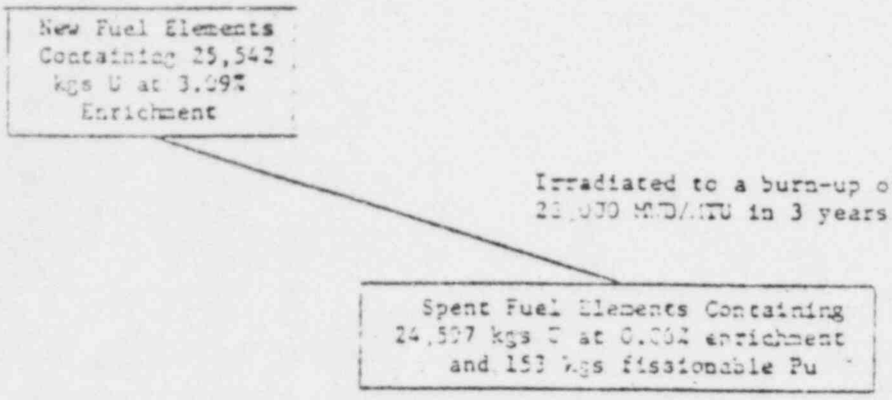
$$U^{238} + n \begin{array}{l} \xrightarrow{7\%} \text{fast fission + energy + extra n} \\ \xrightarrow{93\%} \text{Pu-239} \end{array}$$

$$Pu^{239} + n \begin{array}{l} \xrightarrow{73\%} \text{fission + energy + extra n} \\ \xrightarrow{27\%} \text{Pu-240} \end{array}$$

$$Pu^{240} + n \longrightarrow \text{Pu-241}$$

$$Pu^{241} + n \begin{array}{l} \xrightarrow{73\%} \text{fission + energy + extra n} \\ \xrightarrow{27\%} \text{Pu-242} \end{array}$$

The depletion of fuel in a reactor is frequently referred to as "burn-up". Burn-up is often discussed in units of megawatt-days of heat energy produced per metric ton (i.e., 2000 pounds) of uranium fuel. This is abbreviated MWD/MTU. Let us see what happens to a batch of fuel in a reactor for 3 years with a burn-up of 20,000 MWD/MTU.



7.7 Fission Product Poisoning

You will recall from earlier lectures that every time a uranium atom fissions, two fission fragments are produced; as more and more fuel is burned up, we observe a build-up of fission products within the fuel elements. These fission products have associated with them absorption cross sections which result in an ever increasing loss of neutrons in the core due to parasitic capture. Thus we find a steady loss of reactivity.

Of the fission products produced, two play a most important role in the operation of the reactor. They are xenon-135 and samarium-149. They are important mainly due to their extremely high thermal neutron cross sections. Xenon-135 may be produced as a direct result of the fission process with a relatively low yield of 0.3%. That is to say, 0.003 xenon-135 nuclei are produced on the average per fission. The main source of xenon-135 is the radioactive decay of tellurium-135. The decay chain for tellurium-135 is as follows:

$$\text{Fission of } U^{235} \text{ yields } 6.1\%\ Te^{135} \xrightarrow[1\text{ min}]{} I^{135} \xrightarrow[6.7\text{ hr}]{} Xe^{135} \xrightarrow[9.2\text{ hr}]{}$$

$$Cs^{135} \xrightarrow[2.1\times10^{6}\text{ yrs}]{} Ba^{135} \text{ (stable)}$$

Tellurium-135 has a fission yield of 6.1%, therefore, a relatively large concentration of xenon develops in the reactor core. The thermal neutron absorption cross section of xenon-135 is about 3×10^6 barns.

A convenient way of looking at the situation is to write a balanced equation. Since the amount of xenon-135, present in the core, is greatly dependent upon iodine-135 formation, let us consider what ~~determining~~ its concentration in the core.

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Iodine is produced by the decay of tellurium-135 but we may assume, since its half-life is so short ($T_{1/2} = < 1$ min), that for purposes of computation, iodine-135 is produced directly from fission.

$$y_I \Sigma_f \phi$$

where

y_I = 0.061 (fractional fission yield)

Σ_f(per cm) = macroscopic (thermal neutron) cross section for fission of the reactor fuel like U-235

ϕ(n/cm^2-sec) = thermal flux

Iodine-135 is lost by two different mechanisms. First by its natural radioactive decay

$$\lambda_I I$$

where

λ_I(per sec) = decay constant for iodine-135

I(nuclei/cm^3) = concentration of iodine-135 at any time

Second, iodine is lost by capturing a neutron and being converted to another isotope. This may be represented as

$$\sigma_I \phi I$$

where

I(nuclei/cm^3) = concentration of iodine-135 at any time

σ_I(cm^2/nucleus) = microscopic cross section for thermal neutrons

ϕ(n/cm^2-sec) = thermal flux

Now, assembling these terms into an equation:

$$\frac{\Delta I}{\Delta t} = y_I \Sigma_f - \lambda_I I - \sigma_I \phi I$$

The above equation can be solved to determine the iodine concentration at any time. Eventually the iodine concentration reaches an equilibrium (constant) value after the reactor has operated for approximately two days at a constant power level. At this time:

$$\frac{\Delta I}{\Delta t} = 0 = y_I \Sigma_f \phi - \lambda_I I_o - \sigma_I \phi I_o$$

$$I_o = \frac{y_I \Sigma_f \phi}{\lambda_I + \sigma_I \phi} = \frac{y_I \Sigma_f \phi}{\lambda_I}$$

since $\sigma_I \phi$ is small compared to λ_I.

Now let us write our balance equation for xenon in words.

Rate formation of xenon with time = production - losses

Let us look at the ways by which xenon-135 is produced. First some forms directly as a fission product:

$y_x \Sigma_f \phi$

where

y_x = fractional fission yield = 0.003

Σ_f(per cm) = macroscopic (thermal neutron) cross section for fission of the reactor fuel like U-235

ϕ(n/cm^2-sec) = thermal flux

Secondly, xenon-135 is produced by the decay of iodine-135:

$\lambda_I I$

where

λ_I(per sec) = decay constant for iodine-135

I(nuclei/cm^3) = concentration of iodine-135 at any time

Consider the processes ~~by which xenon~~-135 is lost in the reactor. First some is lost due to natural radioactive decay:

$\lambda_x X$

where

λ_x(per sec) = decay constant for xenon-135

X(nuclei/cm^3) = concentration of xenon-135 at any time

Xenon-135 is also lost by neutron capture.

$$\sigma_x \phi X$$

where

σ_x (cm^2/nucleus) = microscopic cross section for thermal neutron capture

ϕ(neutrons/cm^2-sec) = thermal flux

X(nuclei/cm^3) = concentration of xenon-135 at any time

Therefore:

$$\frac{\Delta X}{\Delta t} = y_x \Sigma_f \phi + \lambda_I I - \lambda_x X - \sigma_x \phi X \qquad (1)$$

The equilibrium xenon concentration may now be found. If we follow the same line of reasoning as we did to determine the equilibrium iodine concentration,

$$\frac{\Delta X}{\Delta t} = 0 = y_x \Sigma_f \phi + \lambda_I I_o - \lambda_x X_o - \sigma_x \phi X_o$$

and

$$X_o = \frac{y_x \Sigma_f \phi + \lambda_I I_o}{\lambda_x + \sigma_x \phi}$$

As mentioned in an earlier section, xenon-135 and samarium-149 are the most important fission product poisons. As we have already seen, xenon build-up and decay leads to certain control considerations.

Samarium is important due to its gradual but continual build-up in the core and its high thermal neutron cross section. Samarium is produced by the decay of the fission product neodymium.

$$Nd^{149} \xrightarrow{2\ hrs} Pm^{149} \xrightarrow{53\ hrs} Sm^{149}\ (stable)$$

The capture cross section for Sm^{149} is 5×10^4 barns for thermal neutrons. Because Sm^{149} is stable, its equilibrium concentration and poisoning (ψ) are not dependent on neutron flux. Also after the reactor is shutdown, the samarium concentration does not pass through a maximum as is the case with xenon, but increases towards some maximum value.

The most obvious effect of xenon and samarium build-up in the core is the loss in reactivity which they cause. Let us look for a moment to see which factor of the 4 factor equation, which makes up the infinite multiplication factor, is being affected by these fission product poisons. We find that the thermal utilization is essentially the only one being affected. You will recall from earlier lectures that the thermal utilization

$$f = \frac{\text{number of thermal neutrons absorbed in fuel}}{\text{number of thermal neutrons absorbed}}$$

Now let us look at the thermal utilization in a reactor with and without poisons.

Without fission product poisons, we may write:

$$f = \frac{\Sigma_a^{U-235}}{\Sigma_a^{U-235} + \Sigma_a^{m} + \Sigma_a^{all}}$$

where Σ_a^{m} is moderator cross section, and Σ_a^{all} is caused by remainder of core material.

If we treat the build-up of the poisons separately, we could write a thermal utilization with poisoning as

$$f^1 = \frac{\Sigma_a^{U-235}}{\Sigma_a^{U-235} + \Sigma_a^{m} + \Sigma_a^{all} + \Sigma_a^{poisons}}$$

Now let us define a new term called poisoning, ψ, of a reactor as the ratio of the number of thermal neutrons absorbed by the poison to those absorbed in the fuel

$$\psi \equiv \frac{\Sigma_a^{poison}}{\Sigma_a^{U-235}}$$

What we are interested in obtaining is relationship between reactivity and poisoning in order to obtain a quantitative measure of its effect on the operation of a reactor. Let us start by looking at the change in effective multiplication due to changes in the thermal utilization, f. Let the effective multiplication factor without poison be k_{eff} and that with poison by k'_{eff}; then

$$\frac{k'_{eff} - k_{eff}}{k'_{eff}} = \frac{f' - f}{f'}$$

If one assumes in the interest of simplicity that:

a) $\Sigma_a^{all} \ll \Sigma_a^{U-235}$ or Σ_a^{m}, then Σ_a^{all} can be neglected

b) $y = \frac{\Sigma_a^{m}}{\Sigma_a^{U-235}}$ by definition for mathematical convenience

c) and that $\psi = \frac{\Sigma_a^{poison}}{\Sigma_a^{U-235}}$

Then it can be shown that

$$\frac{k''_{eff} - k_{eff}}{k'_{eff}} = \frac{\psi}{1 + y} \qquad (2)$$

Remember that for a critical reactor $k_{eff} = 1$ and the left hand side of Equation (2) becomes equal to the reactivity ρ due to poison

$$\rho = -\frac{\psi}{1 + y}$$

7.7.1 Xenon Effects During Operation

Now let us take a look and see what this means in terms of reactor operation. First, refer to Figure 6, Page 7-15. This is a plot of reactivity effect due to xenon vs time. Starting at point A, the reactor has reached some steady state power. At this point, we begin to build up xenon into the reactor core. At time B, we have reached equilibrium value for xenon in the core and as long as we maintain our initial power level, the reactivity loss due to xenon build-up in the core will remain constant. The time duration between points A and B is dependent upon the reactor power level or more particularly the average flux in the core. This is generally on the order of 50-60 hours.

Next, let us assume that at time C, the reactor is either shut down or accidentally scrammed. If we refer back to Equation (1), we note that upon shutting down the reactor, the flux will go to essentially zero. Thus, the formation of xenon by direct fissioning in the core will immediately be stopped, and the burn-up of xenon by absorbing neutrons in an operating reactor will also go to zero. We are therefore left with only two terms: one is the continual decay of iodine to produce xenon and the second being the natural decay of xenon. Since iodine has a shorter half-life than xenon, 6.7 hours vs 9.2 hours for xenon, we find that xenon builds up in the core. This build-up reaches its maximum at time D, which is on the order of 10-12 hours after the reactor was shutdown. After that time the xenon present in the core will continue to decay out reflecting its 9.2 hour half-life. Eventually, after 6 or 7 days, we might reach a time E, in which the reactor core is again free of all xenon poisoning.

Several points should be remembered: first, that the reactivity effect on the xenon in the core is dependent on the average flux in that core. Thus, for a given reactor, the reactivity effect is a function of the power level at which the reactor is operated.

Also, although Figure 6 has arbitrary units on both the X and Y scales, it will be possible for any given reactor to assign values to both these scales. That is, instead of arbitrary reactivity loss due to xenon build-up in the core, one can substitute

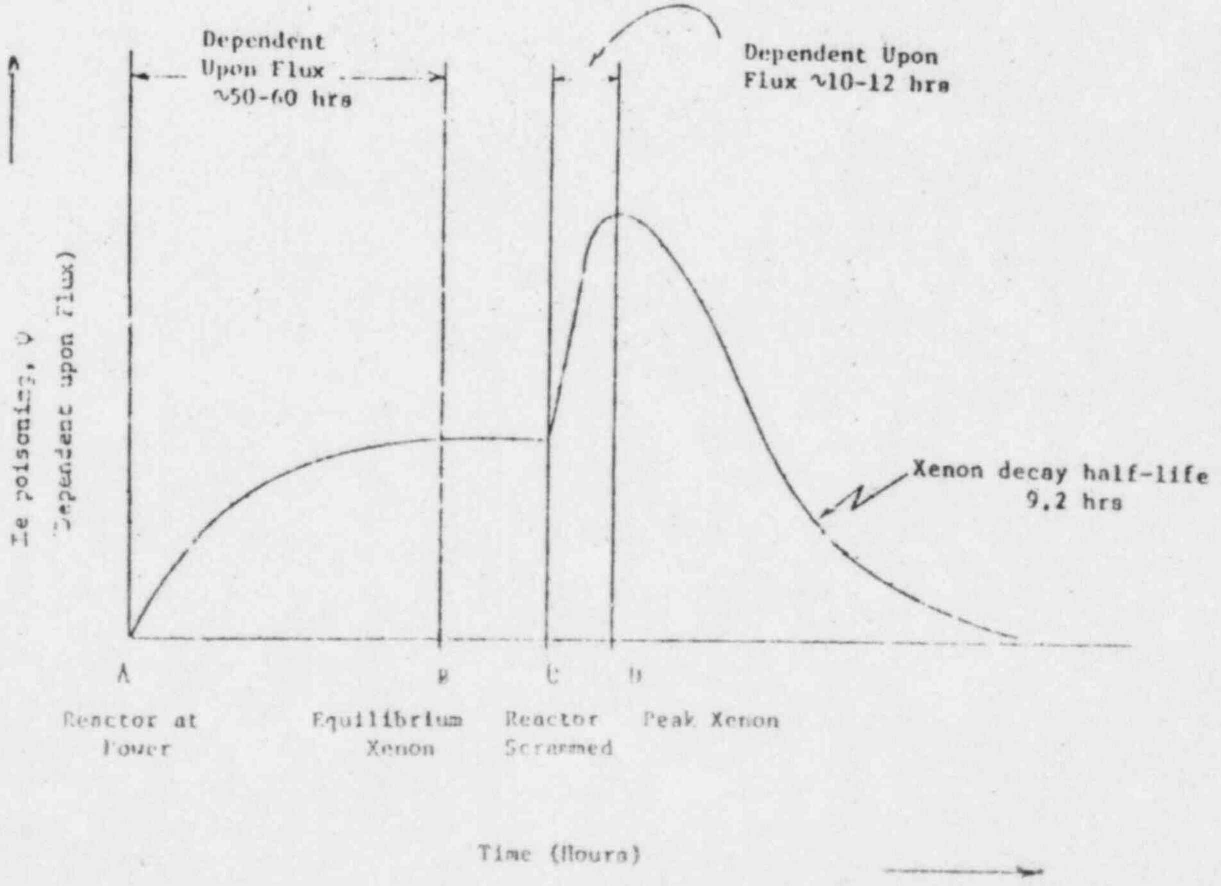

Figure 6

Xenon Poisoning vs Time

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reactivity in dollars and cents, the time scale can be laid out properly in terms of hours reflecting the actual response of the xenon build-up and decay for that machine.

Now let us consider some other conditions that might arise with regards to xenon in the reactor core and their effect on the reactor operation. First, consider a reactor which has been operating for some period of time, sufficient to allow for equilibrium xenon to be reached. Then, due to some inadvertent condition the reactor is accidentally shutdown. One then finds conditions depicted by Figure 7, Page 7-17. You will notice that up to the time the reactor was brought back up to power, the curve is the same as that for Figure 6. However, at the time when one wishes to bring the reactor back to power, we are in a condition of extremely high xenon concentration in the core, in fact, almost twice that of our normal equilibrium concentration for that power. Then what one finds is the following: (again referring to Equation (1)) we find that the production of xenon is beginning but again it's a very small number. We find that there is essentially no iodine present in the core and that we are just beginning to produce iodine in the core. The xenon decay, is of course, continuing to take place, but now the over-riding factor, the factor which is having the greatest influence on the reactivity of the core, is the burnup of xenon due to neutron absorption. We find then that the reactivity in the core is increasing very rapidly due to the burn-up of the xenon and in fact that is one of the most severe transients which the control rods must be able to handle. It will be necessary, as the xenon is burned up in the core, to insert the control rods to counteract this reactivity gain. Eventually, we do reach a turn-around point and pick up our build-up and xenon as normal and the reactor returns to some equilibrium value.

Figure 8, Page 7-18, depicts the variation and poisoning due to a power decrease. Assume that the reactor has been operating at full power for a sufficiently long time to allow equilibrium xenon concentration to be reached. This corresponds to time A. At time B, the reactor power is reduced, let us assume to 50% power. We will notice that the xenon in the core builds up to some maximum value and then gradually decays away until it reaches a new equilibrium value at time C, which corresponds to equilibrium conditions at 50% full power.

The last condition which we shall consider is depicted by Figure 9, Page 7-19. We have essentially reversed the previous condition. Here we have the reactor operating at say 50% of full power and we have reached some equilibrium poison concentration at time A. Then at time B, the reactor power is increased to 100% of full power and we find that for a short time the xenon concentration in the core decreases, then it turns around and builds up to a new equilibrium value corresponding to 100% full power at time C.

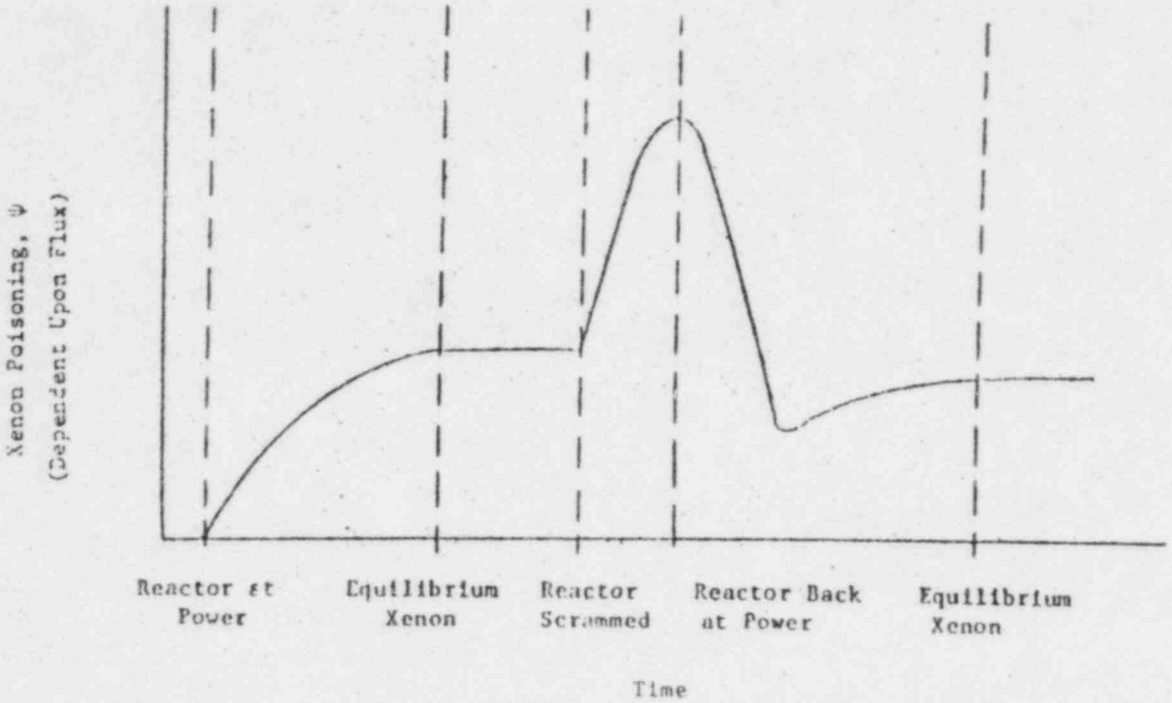

Figure 7

Xenon Poisoning vs Time

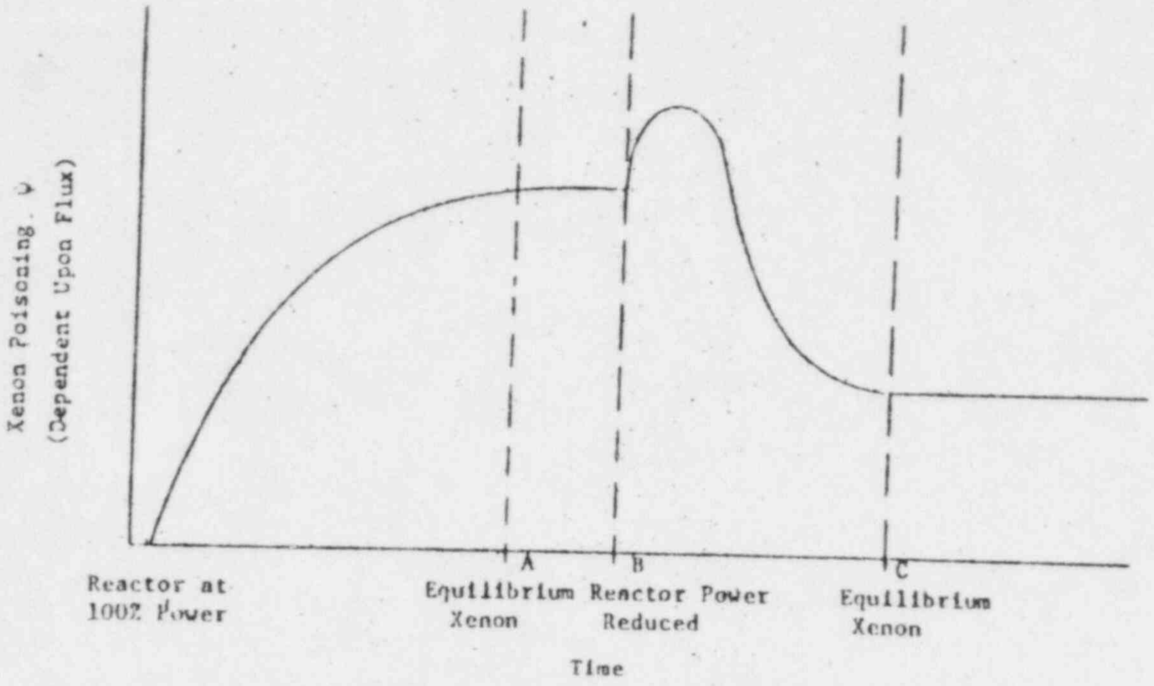

Figure 8

Xenon Poisoning vs Time

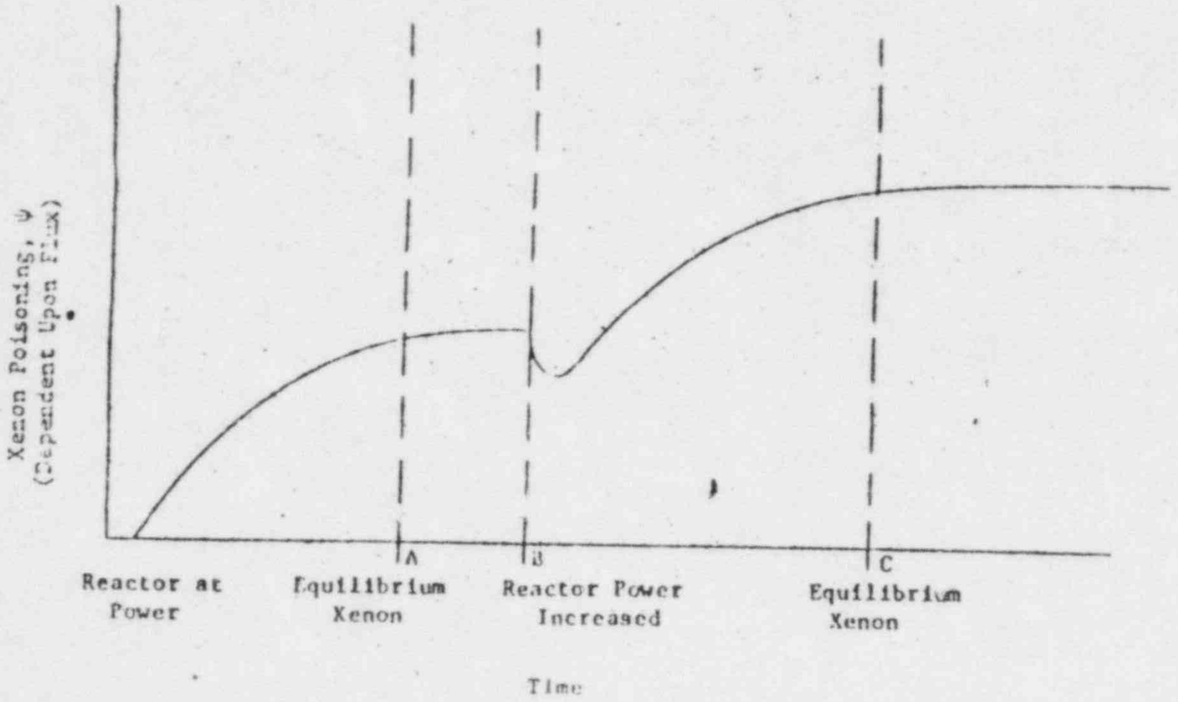

Figure 9

Xenon Poisoning vs Time

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7.7.2 Xenon Oscillations

Large power reactors operating with relatively high thermal neutron flux can experience a problem referred to as xenon oscillation. This oscillation is due to the feedback between the reactor power and xenon production in the core. Variations in the spatial distribution of xenon poisoning in the core can occur even though the reactor may be operating at a constant average power. Such oscillations are associated with a distribution of the xenon poisoning in the core. Let us start by considering a reactor operating at a constant total power and at equilibrium xenon conditions. If a small disturbance in the flux shape is introduced in the axial direction in the core, which causes the xenon to burn out more rapidly at the top and less rapidly at the bottom while the local rate of formation of xenon from iodine decay remains constant for some time, the increased reactivity at the top and the corresponding decrease at the bottom tend to magnify the flux tilt.

Eventually the increase in xenon and iodine decay is forming more rapidly at the top and the rate of decay of xenon at the bottom reverses the reactivity distribution and the power then peaks at the bottom of the reactor. In attempting to correct such oscillations, one can reinforce them by improper control rod movement. Consequently, it is possible to continue these oscillations from the top to the bottom of the core as is the case for our discussion. Such oscillations have a period of about one day. Xenon oscillations are generally most troublesome along the axial direction of the core due to the local disturbance in the neutron flux caused by the control rods. Recall one frequently experiences a bulge in the flux at the bottom of a control rod making it more susceptible to establishing the oscillations. Size is another important factor in determining the possibility of establishing such oscillations. In the PCTR core, since it is an extremely small core, xenon oscillations are not a concern. However, in today's power reactors, the linear dimension of the core in either diameter or height can be several times the migration length for neutrons in that core. It is easy to see where two or more regions of the core can begin to function as independent nuclear units. That is to say, fission that takes place in one region is not caused by neutrons produced in the other regions. In the light water moderated reactor with a migration area on the order of 50 square centimeters, we need to worry about oscillations if one of the dimensions of the core is over 80 inches. Recall that the active length for a typical PWR core is 144 inches.

From a control standpoint, since the oscillations are of such a long period (on the order of about 1/day), one might first think that the oscillation presents no problem since it is easy to make compensating reactivity adjustments. This, however, is not the major problem with xenon oscillations. The concern is one more of variations in the local flux level in individual fuel pins in the core. That is, the local heat generation in the core. Such oscillations could very easily produce local hot spots which possibly could damage several fuel pins.

7.7.3 Reactivity Coefficients

Let us take a look at the five reactivity coefficients which are most important in the operation of a pressurized water reactor. The reason for qualifying the type reactor is that there are other coefficients which influence a boiling water reactor more prominently than they do pressurized water reactors, i.e., void coefficient for one.

7.7.3.1 Doppler Coefficient - a reactivity change caused by changes in the fuel temperature. As the fuel temperature increases, the effective neutron absorption cross section of the fuel in the resonance regions increases. That is to say, as the fuel temperature increases, more neutrons are absorbed in the resonance region with the effect of reducing the neutron production in the core, that is, reducing k_{eff}. This effect is frequently identified as a prompt effect and contributes to the control of reactor power excursions. In the case of our TRIGA reactor, 20% of the prompt temperature coefficient is due to this Doppler broadening effect. The range for the Doppler coefficient in the power reactor under operating conditions is on the order of -0.1×10^{-5} to -1.7×10^{-5} $\delta k/k$ per degree Fahrenheit.

7.7.3.2 Moderator Void Coefficient - causes a change in reactivity due to the presence of voids in the water moderator. These voids can be caused by high heat rate producing local nucleate boiling on the surface of a fuel pin. A reactor controlled by soluble boron in the primary system may initially exhibit a small positive coefficient upon initiation of voids within the core. However, as the percent of voids increases, the coefficient becomes negative. This is due to the initial displacement of the neutron balance within the core than the removal of a small amount of the water moderator. The expected range for the void coefficient in a PWR core might be from $\pm 1 \times 10^{-4}$ to -3×10^{-3} $\delta k/k/\%$ void.

7.7.3.3 Moderator Pressure Coefficient - The moderator pressure coefficient causes changes in reactivity due to changes in moderator density. Changes in moderator density may be caused by several different phenomena. One being the increase in moderator density caused by the increase in primary system pressure. As primary system pressure increases, the density of the moderator increases thus causing a positive coefficient to result. Generally this coefficient is considerably less than the other coefficients considered in this section. A typical value for pressure coefficient might be +3 x 10^{-6} δk/k/psi.

7.7.3.4 Moderator Temperature Coefficient - Indicates the effect upon neutron multiplication due to a change in primary system cooling temperature. Changes in the moderator can affect the reactivity in three ways: 1) there results a change in the density of the materials in the reactor core and thus a change in the atomic density. The density change in the materials, particularly the structural materials of the core are very small over the range of temperatures being considered. It is safe, therefore, to assume that the atomic density of the metals in the core remain relatively constant. This is not the case for the change in atomic density of the water which is significant. In addition, reactors using soluble boron for reactivity control experience, along with the decrease in water density with increasing moderator temperature, a decrease in boron concentrations. These two changes depend considerably upon the concentrations of boron existing in the moderator at the time. This is due to the fact that a reduction in boron concentration results in a positive reactivity change while a reduction in moderator density, i.e., water density, results in a negative reactivity change. Thus we find frequently for higher boron concentrations that the moderator temperature effect may be positive while at low boron concentrations, the temperature coefficient can be negative. 2) the change in moderator temperature also causes a change in the low energy neutron spectrum in the core. This in turn affects the cross section of materials in the core. Recall that cross sections are a function of temperature. 3) a change in the density of the core results in a small change in the apparent size of the core which in turn affects the geometric buckling of the core. This change is brought about by an increase in reflector savings due to the change in density of the water which in turn affects the geometric buckling in the core. This change is quite small compared to the temperature effect mentioned earlier. The operating value ranges from approximately +1 x 10^{-4} to 3 x 10^{-4} δk/k/°F.

7.7.3.5 pH Coefficient - Currently there is no definite correlation for predicting pH reactivity effects. Yankee Rowe, Saxton, and Indian Point have all experienced reactivity changes at the time of pH changes but there is no clear cut evidence that pH is the direct cause. Other items such as clad material, fuel assembly crud deposition, system average temperature, and prior system chemistry are all included. Saxton experiments have indicated a Ph reactivity effect of 1.6 x 10^{-3} δk/k/pH unit change.

7.7.3.6 Measurement of Reactivity Coefficients

- We will now examine the methods one follows to determine the various reactivity coefficients described in the previous section. It is hoped this approach will give the student a better understanding of the many reactivity coefficients. We will use as a reference for this section the NSS Design Summary, Volume 2, prepared by B&W's Nuclear Power Generation Department.

Let us start at the point where the reactor has already gone through the pre-critical test phase. Initial fuel loading has been completed. Initial criticality has also been achieved by control rod withdrawal boron concentration adjustment.

First, a measure of core excess reactivity is obtained. To do this, the reactor is taken critical at "room temperature." This is accomplished by withdrawing all control rods fully from the core. Then the boron concentration is gradually reduced until criticality is reached. Next, the reactor is brought up to normal operating temperature (average temperature of [illegible]°F) in steps. At each step, the boron concentration necessary to achieve criticality with all the control rods removed will be determined.

The boron concentration required to maintain the reactor just critical, with all rods out, is related analytically to the hot core excess reactivity and compared to the cold core excess reactivity to determine the temperature defect.

7.7.3.6.1 Temperature Coefficient Measurements

To evaluate the moderator temperature coefficient and pressure coefficient, the reactor is taken critical at operating temperature and essentially "zero" power. The temperature coefficient is determined by changing the moderator temperature a slight amount at constant boron concentration and rod position and observing the resulting reactivity change. The reactivity change could be determined in the same manner as you followed during the PSTR control rod calibration experiment, i.e., allow the reactor time to reach a stable period, then measure the time it takes for the neutron level to increase by a factor of 2. This so called doubling time can be related to reactivity through the inhour equation. In order to speed-up the data acquisition and reduction process, one frequently determines reactivity by an on-line analog computer. The computer is programmed to solve the inhour equation continuously and indicate core reactivity status from the rate of change of neutron level.

7.7.3.6.2 Pressure Coefficient Measurement

The pressure coefficient is determined by changing the primary system pressure a slight amount, at constant temperature, boron concentration and rod position, and observing the resultant reactivity change. Note: At this time, one will also determine the differential and integral worths for boron and various control rod patterns. Recall that the integral worth can be evaluated by analytically integrating a plot of the differential worth for either rod position or boron concentration.

7.7.3.6.3 Moderator Void Coefficient

The void coefficient is not determined in a straight forward manner at the power plant as is the case for the two previous coefficients. It is necessary to place voids in the core of known volumes and measure their reactivity affect in order to arrive at a determination of $\delta k/k/\%$ void. It would be possible to rely upon calculated void values and observe reactivity changes to arrive at a measure of the void coefficient.

7.7.3.6.4 Power Coefficient Measurement

The power coefficient is generally defined as the change in reactivity resulting from a unit change in reactor power level. Changes in core and moderator temperature and density contribute to this reactivity effect.

One may conclude that two major components of the power coefficient will be the Doppler coefficient and the effects of fuel rod bowing due to thermal expansion. Both of these effects reflect the fuel temperature distribution throughout the core.

The differential power coefficient is measured at a discrete power level by varying the power slightly, at constant control rod position, and allowing the average moderator temperature to compensate for the reactivity change. Control rods are then adjusted to return the moderator average temperature to its normal value.

7.8 Reactivity Control

In the previous sections, we have discussed the effects of fuel depletion, Doppler, moderator temperature, and fission product build-up on core reactivity. Now let us look at the reactivity balance for a typical PWR core.

Excess Reactivity Conditions

Beginning of Core Life	k_{eff}
Cold, Zero Power	1.302
Hot, Zero Power	1.247
Hot, Rated Power	1.229
Hot, Rated Power, equilibrium Xenon	1.192
Hot, Rated Power, equilibrium Xenon and Samarium	1.153

Note: Two cold, fuel assemblies side by side have a keff > 1.0 except when both equilibrium Xe and Sm are present. Three fuel assemblies are super critical regardless of poisons. The total number of fuel assemblies in the core, 177.

Another way of looking at the above effects would be to consider the reactivity balance as a function of core life. We will again consider the first fuel cycle (Refer to Figure 10).

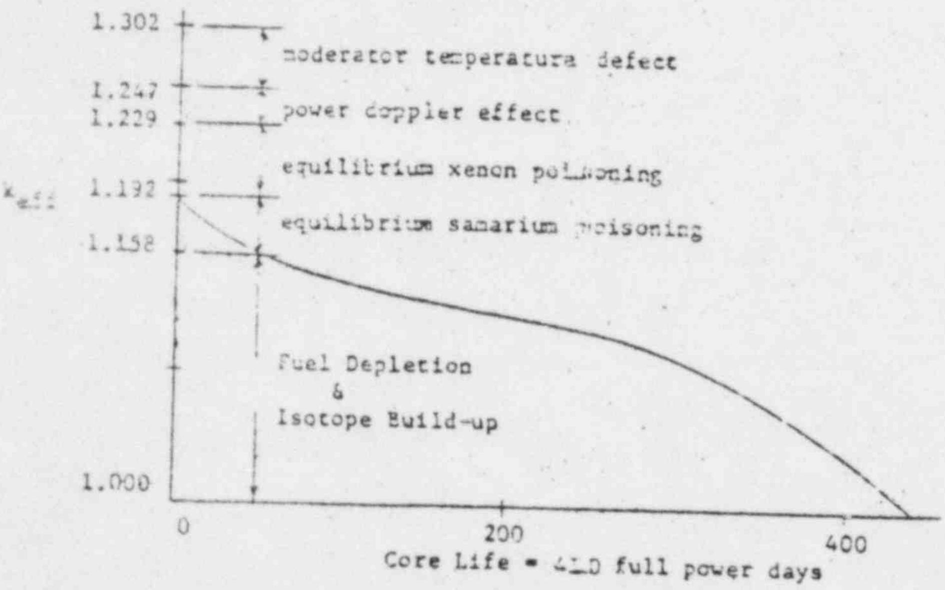

Figure 6

Effective Multiplication vs Core Life

Each reactivity defect has been drawn to scale so that you can compare the relative magnitudes of each defect to the total core excess.

ENERGY REMOVAL

8.0 Introduction

In this series of lectures, we will discuss some of the basic concepts associated with heat transfer and fluid flow as it relates to the removal of energy from a nuclear reactor. Of course, this subject is a very complex discipline in itself, which requires higher mathematics and the use of digital and analog computers when one attempts to solve the problems associated with actual reactors. It will be the purpose of these lectures to attempt to provide an insight into the physical phenomena that take place and to provide a better understanding of the definitions and units utilized in solving the problems associated with energy removal from a reactor.

To accomplish this puspose, the lectures will consist of two parts. The first is a programmed learning section on heat transfer and fluid flow. This section is made available by the Atomic Energy Commission and is provided under separate cover for use by the student outside the classroom. The second part of the lectures will be devoted to classroom discussions of other heat transfer and fluid flow problems associated with energy removal from nuclear reactors using the programmed learning section as a means of introducing concepts, units and elementary examples.

8.1 Energy Removal Systems

One of the unique features of a nuclear reactor is that in theory there is no upper limit to the rate of release of energy as far as the fission process is concerned. Of course, there are practical limits imposed upon the system because one must remove the heat energy from the system or it will eventually destroy itself through meltdown of the fuel elements, core components, etc., or through excessive steam pressures, etc.

An engineer knows that closed thermodynamic power cycles involve the addition of energy at some high temperature (and pressure) and the rejection of a certain portion of this energy to as cold a heat sink as is possible. Thus in a fossil fueled plant, one adds heat energy through the use of hot flames and gases of combustion and one rejects heat in the turbine condenser, in which one utilizes cooling water at as low a temperature as is possible. We know that these two temperatures are important from Carnot, who in 1824 introduced the now famous Carnot Cycle which indicated that the cycle efficiency depends primarily on the absolute temperature of the energy source of the cycle and the absolute temperature of the medium to which the heat is rejected. Thus if we allow T_A to be the absolute temperature in degrees Rankine (degrees F plus 460) at which heat is added and T_R to be the absolute temperature at which the heat is rejected, then Carnot tells us that the maximum efficiency of a power cycle operating between these two limits would be

$$\text{Maximum Efficiency } (\%) = \frac{T_A - T_R}{T_A} \times 100$$

Because the temperature of the cooling water available for use in the condenser is normally fixed by nature one must improve cycle efficficiency by operating with as high a temperature T_A as is possible. Thus, in a power reactor, one wishes to operate with as high a fuel temperature (and steam temperature) as is possible without exceeding the physical limits of the materials present.

8.1.1 PWR and BWR Cycles

To those of you attending these lectures, the concept of a power cycle is well known. You have worked with power plants and know that there is far more to a plant than the problems associated with adding and rejecting heat. I'm sure that you are also aware of the fact that there is a greater difference between a nuclear and a fossil fueled plant than just replacing the firebox, boiler and the steam drum with a reactor and a steam drum, or a reactor and steam generators, etc. However, for our purpose; which is to look briefly at some of the heat transfer and fluid flow considerations in a nuclear reactor core, we will assume that the presence of a reactor core is the primary difference between what you are so familiar with and what you will soon work with.

Serious study of harnessing the energy available from nuclear fission began in the United States before the end of World War II. Because water was available and its use was familiar, it emerged as an early leader as a coolant and moderator for power reactors. Initially, it was believed that water could not be permitted to boil in a reactor vessel. This belief resulted in the early development of pressurized water reactors. The first pressurized water reactor went critical in 1953 at the AEC National Reactor Testing Station in Idaho.

A different type of water cooled and moderated reactor was started in 1953 with the first experiment to permit boiling water in a reactor vessel, thus making steam directly. Successive experiments established the principle that boiling was acceptable and even advantageous for certain purposes.

Thus one is able to sub-divide the general category of water reactors into those where boiling is suppressed and those where a significant amount of boiling is permitted. Boiling is suppressed by maintaining the water at high pressure so that its temperature is below the saturation temperature at that pressure. These reactors are called PWRs for Pressurized Water Reactors. Those reactors which permit a substantial amount of boiling are called BWRs for Boiling Water Reactors.

8.1.2 Pressurized Water Reactors (Non-boiling)

The pressurized water program received its greatest impetus from the extensive U. S. Naval Propulsion Program, so that today the concept is technologically well advanced and the experience gained from the civilian power program has demonstrated that pressurized water reactors are safe, dependable, and reasonably easy to control.

The classes of pressurized water reactors fall into two divisions: "pressure vessel" or "tank" type and "pressure tube" type. In the vessel type, the fuel elements are bundled together in a core that is contained in a tank. The fuel elements are immersed in highly-pressurized water that serves as both moderator and coolant. In the pressure tube type, the fuel is placed in tubes, usually in small bundles, through which water flows at high pressure. The tube separates the fuel and coolant from the moderator, which is usually graphite or heavy water. The present discussion is limited to the pressure vessel type of reactor.

A typical flow diagram for pressurized water reactors is shown in Figure 1, which illustrates the main components of the system. Note that an intermediate heat exchanger or steam generator is required. The water is heated in the reactor and is pumped through the heat exchanger at about 5[illegible]°F to [illegible]°F. In the heat exchanger, the water gives up heat to generate, high pressure (2[illegible] psi) prevents the water from boiling in the reactor and primary system. It is of interest to note that advanced PWR systems (for example Yankee) now permit local boiling in the primary system in order to improve cycle efficiency. To prevent boiling also requires high flow through the primary loop with a resultant high pressure drop which in turn requires substantial structural material to maintain tolerances under the resulting flow forces on the components. The PWR systems require much thicker sections of metal in the pressure vessel design than the boiling water reactors of similar size. Control of the rate of heat generation in most designs is obtained by neutron-absorbing control rods that can enter the reactor core in passages between fuel elements as well as the use of solutions containing boron ("chemical shim").

Operational experience with pressurized water reactors has shown that they are very stable during operation, and can readily accommodate large load changes. The radioactivity of the primary water system at shutdown decays quickly to a tolerable level during the period of operation for several reactors, no long-life radioactivities of uncomfortable magnitude have been observed in the primary coolant loop. Contamination from ruptured fuel elements does pose a decontamination problem. Considerable long-term experience with fuel burnup and plant life is just becoming possible. Pressurized water reactor designers, to compete with steam conditions in existing operating plants, have found it necessary to push the operating temperatures from an average temperature at the reactor outlet of 533°F to 56[illegible]°F (Selni Reactor Italy) and in some of the later designs as high as 5[illegible]3°F (Southern California Edison). This temperature increase has increased fuel surface temperature resulting in some concern that corrosion will limit the life of the fuel. The PWR's have to watch water chemistry closely because of the greater disassociation of water with addition of boric acid. The increase in amount of hydrogen present can tend to undermine fuel reliability.

The U. S. AEC Full Scale Demonstration Plant at Shippingport, Pennsylvania commenced operation in December of 1957. The reactor steam supply system, designed by Westinghouse, operates power

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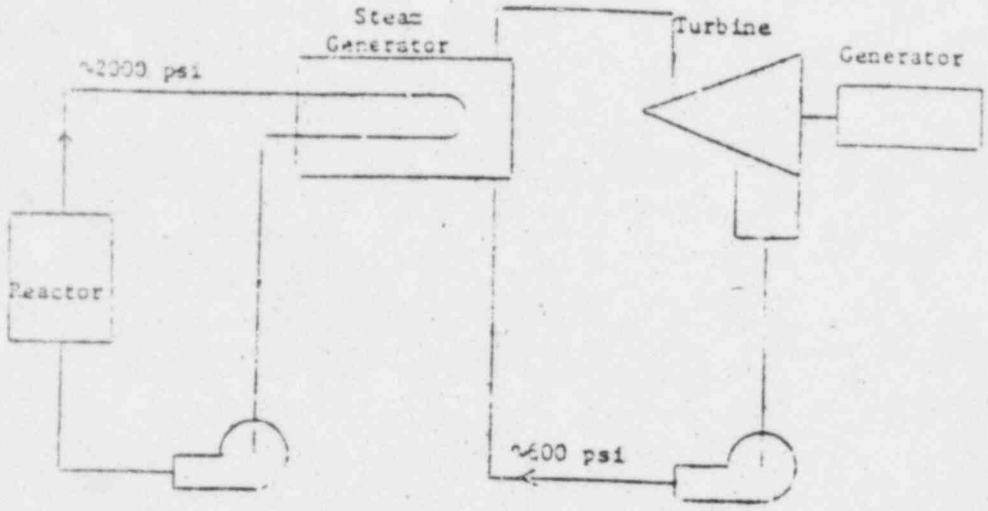


Figure 1

Pressurized Water Reactor

conversion equipment built by the Duquesne Light Company to deliver 60 MWe of electricity. Since first critical, the station has been refueled (seeded) three times; in 1959, 1961, 1963 and was shutdown on February 9, 1964 for a scheduled complete refueling. Left in place, during the first three refuelings, was the natural-uranium major portion of the original core which has generated more than half the 1,790,582 KWh (gross) of electricity produced by the station.

The Yankee Atomic Electric Plant, with a Westinghouse PWR steam supply system, is located near the Massachusetts-Vermont border. It first went critical on August 19, 1960, and full power was achieved in February 1961. The plant was originally licensed for 392 MWt (110 MWe net) and has been steadily and progressively exceeding its design capacity. Yankee reached a record output of 186 MWe gross on its third core on January 1, 1964 — equivalent to 600 MWt and 175 MWe net. The ceiling is the generator's capacity of about 190 MWe gross. Experience during startup showed that mechanical problems can loom large when trying to place a nuclear power station into routine operation. Most of the mechanical problems were of the sort encountered in starting a conventional plant, but the addition of nuclear design factors created minor troubles such as gasketed joints and flanged can sections in the canned motors of the main cooling pump.

The trend toward higher nuclear power plant ratings of 400, 500, and now 1000 MWe makes it urgent that better design techniques and more accurate knowledge of conditions within reactor cores are developed. To this end, the Saxton Nuclear Experimental Reactor, a Westinghouse design, located at Saxton, Pennsylvania, first went critical on April 13, 1962. It began power operation in December 1962. The 5-year experimental program has been aimed at significantly reducing the cost of power from closed-cycle water reactors by improving the core and its conditions of operation.

The Consolidated Edison Indian Point Nuclear Power Plant, with a Babcock and Wilcox pressurized water reactor went critical on August 2, 1962. It uses an oil-fired superheater to increase the steam temperature from 450°F to 1000°F. Fossil-fuel-fired superheat improves thermal efficiency from about 26 percent to 33.5 percent, and provides a lower average plant cost per kilowatt output when compared with the nuclear portion without superheat. The output for the plant is 163 MWe nuclear plus 112 MWe oil superheater, for a gross of 275 MWe and a net of 255 MWe. Another novel feature is the use of thorium fertile material in the fuel rather than U-238 as used in other current projects.

8.1.3 Boiling Water Reactor Systems

In a boiling water reactor, the coolant is water which boils adjacent to the fuel elements thus removing heat from them. The resulting steam-water mixture then proceeds to steam separators, where the liquid water is separated from the steam bubbles. The liquid water then goes back to the reactor where the boiling operation is continued. The steam which is formed passes from the steam drum to a turbine located outside the reactor building. The water which is separated from the steam is retained within the reactor building and does not go outside of the plant.

The major difference in the operating characteristics of a boiling water reactor core as compared to other nuclear systems is affected by the steam voids produced in the assembly. Water serving the dual function of coolant and neutron moderator couples the heat generation and neutron flux characteristics of a nuclear system. If this water is allowed to boil, and thus significantly affect the density of molecules, this coupling becomes more pronounced and thus more significant with regard to performance behavior. The boiling water reactor designs provide a system that produces reactivity changes varying inversely as a function of steam void content in the core. That is, a transient power increase will produce more steam voids, reducing reactivity, which reduces power and thus limits the excursion.

Boiling water reactors, in which significant power operating experience has been acquired, are subdivided into two classes: the single-cycle system and the dual-cycle system. While experience was gained in the operating plants using the boiling water reactors, studies continued on the reactor cycles available to the boiling water concept. As a result of these studies, the cycle types now being offered in the two classes are: the natural circulation single cycle which is best suited for the 50 to 100 MWe range; forced circulation single cycle; and forced circulation, dual cycle, both preferred in the range of 100 to 1000 MWe. The natural circulation plant uses no pumping of the coolant through the core. The forced circulation plant pumps coolant through the core, and the dual cycle plant uses part of the heat of the recirculation water to generate additional steam for the turbine.

8.1.3.1 Single-Cycle Systems

Figure 2 shows the cycle diagram for a single-cycle system. An obvious advantage of this approach is the reduction of plant cost by combining functions in one piece of equipment, thus reducing the number of major system components. Since the steam-water mixture acts as moderator as well as coolant, enough water is recirculated either by natural or forced circulation to keep the fraction of water in the core sufficient to accomplish the moderating function. Technical improvements in the design of the forced circulation single cycle system by the General Electric Company has resulted in a further simplification of the basic concept of the boiling water reactors. This has been made possible by the replacement of the high steam drum and its associated piping with internal separators, which are highly efficient and reliable for providing reactor-produced steam to the turbine.

8.1.3.2 Dual-Cycle System

To gain the attractive features of boiling in a large plant, an early innovation was introduced by the General Electric Company that increased the power density obtainable with the direct boiling cycle. A typical dual-cycle flow diagram is shown in Figure 3. In the dual-cycle reactor, energy is removed from the reactor core directly by a steam-water mixture which flows to the steam drum and then to the primary turbine control valves and indirectly by extracting heat through a steam generator. The fraction of the total energy removed through the secondary or low pressure system controls the temperature of the water returning to the reactor which in turn determines the reactor output. The dual cycle permits a novel and very convenient method of reactor power variation to match load changes.

8.1.4 Reactor Coolants

Most of the energy released from fissioning of the fuel in a reactor appears as heat in the fuel elements and structural materials. Therefore, the reactor must be provided with an adequate cooling system to remove the heat in order that these

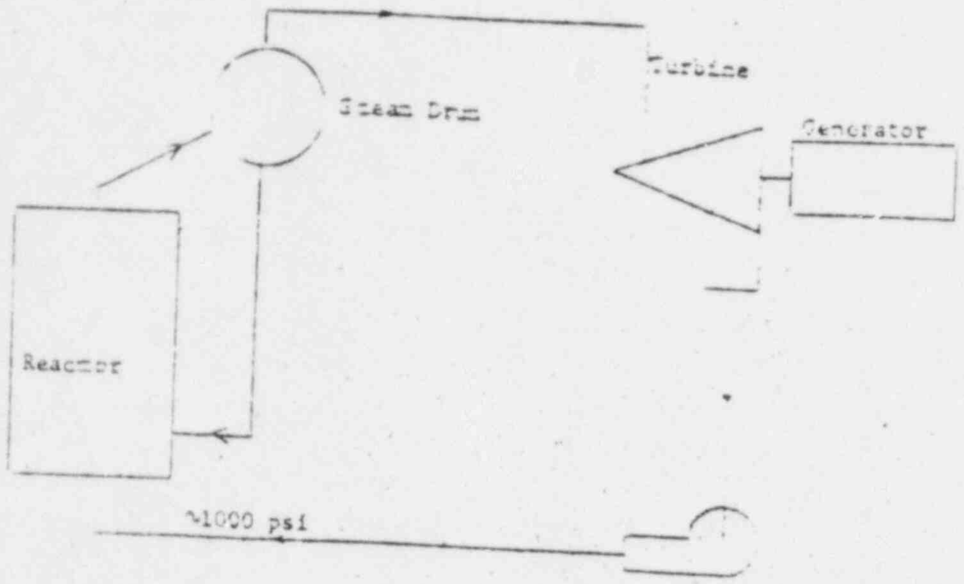


Figure 2
Single Cycle Boiling Water Reactor

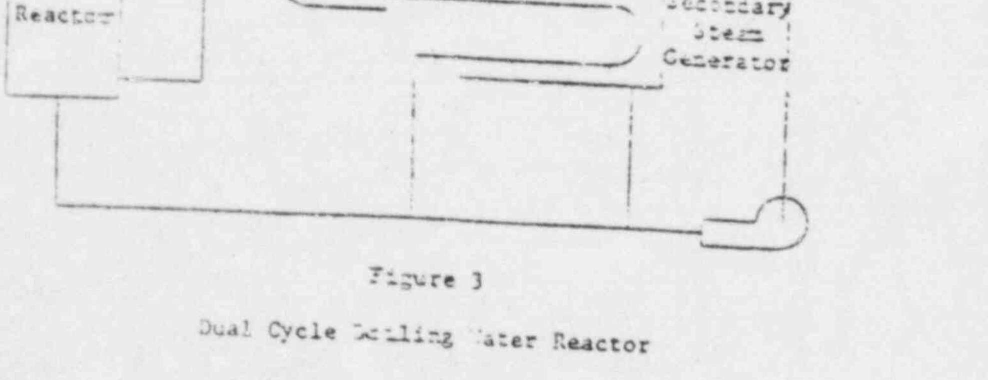


Figure 3
Dual Cycle Boiling Water Reactor

materials do not deform or melt at excessively high temperatures. At the same time, as we have previously briefly discussed, we know that in order to obtain an efficient thermodynamic heat cycle the temperature at which heat is added to the system should be as high as possible. Thus, the limits of the coolant are defined. It must be able to be used at high temperatures and must efficiently remove the heat from the fuel so that the fuel or fuel cladding temperatures do not reach excessive values.

Actually the coolant must satisfy a set of criteria including the general ones we have just described. For example, a coolant should have:

a) good thermal properties (such as high specific heat and high thermal conductivity)
b) high boiling point and low freezing point
c) good stability from breakdown from heat addition and radiation
d) low power requirements for pumping (high density, low viscosity)
e) limited corrosion of components in the system
f) small absorption cross section for neutrons
g) low atomic weight so that it also serves as a good moderator
h) low cost
i) small hazard in handling (non-toxic, not explosive, low stored energy, non-radioactive)

Actually, there is no one coolant which fully satisfies all of these criteria. In the United States, however, water has become the coolant which has been used most extensively in power reactors, as one knows because of the popularity in this country of the PWR and BWR systems.

Perhaps it is not surprising that water became the preferred primary reactor coolant in the U. S. Reliability was a key factor in the early development of reactors, as of course it still is today, and water had many important advantages that did not require extensive experimental and development programs. In other words, water, at least to the extent of its range of pressure and temperature in reactors, was a known quantity. It is cheap and was readily available in the early days of reactor programs. It has relatively good heat transfer characteristics which could be extended beyond its normal narrow temperature range by putting the water under pressure to prevent boiling, and thus overcome the disadvantage of its relatively low boiling point. Furthermore, if the water is kept pure it does not become significantly radioactive. The induced radioactivity of the water itself (not including dissolved mineral content) has a short half-life so that maintenance is not greatly hampered. Water is corrosive but its corrosiveness was reasonably well known. Pressurization intensifies the corrosive action. Of extreme early importance may be the fact that water could serve as a reasonably good moderator as well as a coolant, thus eliminating the complications of the presence of two different materials to serve as moderator and coolant. Water has a relatively high thermal neutron absorption cross section, but this was overcome by enriching the fuel with a greater U-235 content. (A natural uranium-water system will not go critical). The absorption cross section of water becomes a greater problem in breeder reactors where the conservation of neutrons needed to convert fertile materials to fuel is of extreme importance.

Other disadvantages of water include the fact that the pressure necessary to obtain reasonably high temperature are quite high, thus requiring expensive pressurized piping, vessels, fittings, etc. Pure hot water at high pressures is highly corrosive requiring exotic and expensive materials for the system. Because of its high pressure, in the event of a system rupture, the water flashes to steam creating extensive hazards and the necessity of containment systems capable of withstanding pressurization. To maintain this containment pressure within reasonable limits, steam quenching systems (containment spray, vapor suppression using ice or pools of water) are utilized.

Therefore, water is not the perfect coolant that the plant operator would prefer to have, but it has proven quite satisfactory from a number of standpoints. Later on, we will talk briefly about the need to maintain purity of the water, but first let us look at some of the problems associated with transferring the heat from the fuel elements where most of it is produced to the coolant from which one can produce steam to turn the turbine.

8.2 Heat Transmission and Removal

8.2.1 Steady State Heat Transmission Without Internal Generation

One of the primary differences between discussing the transmission and removal of heat in a nuclear reactor system and that in a fossil fueled plant is the fact that in the latter case, heat is transferred from hot flames and gases through tube walls to the coolant, whereas in the case of a reactor the heat is transferred from the fuel (in which heat is internally generated from fission) through a cladding material to the coolant. Thus the primary difference is in the fact that heat is internally generated in the fuel element.

Let us look at the simple case where heat is transferred from a hot region (such as a boiler, a radiator, an iron, the wall of a house, etc.) through an intermediate material into a colder medium. For conceptual and mathematical simplicity, we will look at the case in which heat is being transferred in only one direction, that is we will consider the one dimensional case of a slab or a wall in which one surface is at a uniformly hot temperature, T_o, and the other surface is at uniformly colder temperature, T_1. The thickness of the slab or wall will be considered small compared to the height and width of the wall surface. Thus we would have the following situation:

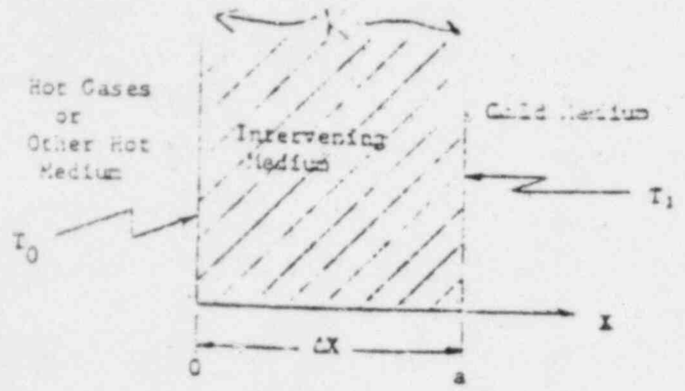


Figure 4

We know that heat will be transferred or conducted from the hot surface where the temperature is T_o to cold surface where the temperature is T_1. Let's define the rate at which heat is conducted through the wall or slab as q, which we will assign the units of BTU per hour (BTU/hr) conducted from the hot to the cold face.

On what things would the amount of heat per hour conducted through the wall depend? Certainly, one would expect that the rate of heat conduction would depend on the difference in temperature between the hot and the cold surfaces, i.e., T_o-T_1. For example, if $T_o=T_1$, then $T_o-T_1=0$ and we would expect that no heat is conducted between the two surfaces. Likewise, if T_o-T_1 is twice as great one time as it was another time, we would expect that the amount of heat conducted would be twice also. Thus, we can say that the rate of heat conduction, q, is proportional to the difference in temperature, T_o-T_1. Mathematically, we would say

$$q \propto T_o - T_1 \qquad (1)$$

Similarly, we would expect the rate of heat conduction to be a function of the thickness of the material. Thus, if the intervening material is thin, we would expect more heat to be transmitted than if the material were thicker. Or in other words the thicker the material, the less the heat conducted. Thus we would say that the rate of heat conduction is inversely proportional to the thickness of material, or mathematically,

$$q \propto \frac{1}{\Delta X} \qquad (2)$$

where ΔX is the thickness of the material under consideration.

Combining the equations of proportionality, (1) and (2), we see that

$$q \propto \frac{\Delta T}{\Delta X}$$

where ΔT is the temperature difference T_o-T_1. Thus we see that the heat rate is proportional to the temperature gradient, $\Delta T/\Delta X$ which has the units of °F/ft.

Further, we would expect that the amount of heat conducted in BTU/hr would depend on the surface area of the wall or slab. Thus, we should expect twice as much heat to be conducted if the wall has 2 ft² of surface area than the case where it would have 1 ft² of area. Thus, we could say that

$$q = A \tag{4}$$

or

$$q = A \frac{\Delta T}{\Delta X} \tag{5}$$

Now, also we know that some materials conduct heat much better than others. For example, we know that a wooden handle on a spoon or fork is better if one is handling hot foods than a metal handle. This is because the metal is a better conductor of heat than is wood. Likewise, we know that a silver spoon conducts heat better than a stainless steel spoon. Thus, the amount of heat conducted is proportional to the thermal conductivity, k, of the material or

$$q = k \left(\frac{BTU/hr}{ft^2\,{}^\circ F/ft}\right) \text{ sometimes abbreviated to } \left(\frac{BTU/hr}{{}^\circ F\text{-}ft}\right) \tag{6}$$

Combining all of these equations, we find that

$$\boxed{q = kA \frac{\Delta T}{\Delta X} \quad BTU/hr} \tag{7}$$

If this equation is rewritten in the form

$$q = \frac{\Delta T}{\Delta X/kA} \tag{8}$$

we can see that this is similar to Ohm's law where

$$I = \frac{E}{R} \tag{9}$$

where the rate of heat conduction q is equivalent to the current I; the temperature difference ΔT is equivalent to the voltage difference or potential E, and the quantity ΔX/kA (called the thermal resistance) is equivalent to the electrical resistance, R.

In order to compare rates of heat transfer on a similar basis (BTU/hr-ft²) one normally divides both sides of Equation (7) by the surface area A. Thus

$$q/A = k \frac{\Delta T}{\Delta X} \quad (BTU/hr\text{-}ft^2) \tag{10}$$

Note that what this does, in essence, is to measure the heat rate q on a basis of 1 ft² of surface area. This quantity of q/A having the units of BTU/hr-ft² is called the heat flux.

Let us now look at an example. Suppose we have a steam radiator in a house which has 4 ft^2 of surface area. The thickness of the steel is 1/4 inch and has a thermal conductivity of 30 BTU/hr/°F-ft. The temperature of the outside of the radiator will be assumed to be the same as the temperature of the room, which is 68°F.

a) What is the rate of heat released from the radiator?

b) What is the heat flux?

To solve the problem, we will assume that the radiator can be converted into a similar slab or wall condition. Further, we will assume that the steam inside the radiator is at atmospheric pressure, thus at 212°F, and we will assume that this is the temperature of the inside surface of the radiator.

Thus, $q = KA \frac{\Delta T}{\Delta X}$

where $K = 30 \frac{BTU/hr}{°F\text{-}ft}$

$A = 4\ ft^2$

$\underline{\Delta T} = 212-68 = \underline{144°F}$

$\Delta X = \frac{1/4}{12} = \frac{1}{48}\ ft$

Then a) $q = 30 \times 4 \times 144 \times 8 = \underline{829,440\ BTU/hr}$

b) $q/A = \frac{829,440}{4} = \underline{207,360\ BTU/hr\text{-}ft^2}$

Actually we would find that the radiator would not transfer this much heat because the air around the outside of the the radiator would tend to insulate the radiator and thus its outside temperature would have to be much higher to drive the heat through this insulating layer. This would reduce the actual ΔT through the metal and thereby reduce q and q/A. We will take that situation up later.

8.2.2 The Effects of Cladding

Let us now look at the case where our slab has for some reason been clad on its outside surface. This <u>cladding</u> might be considered a coat of paint on the outside of the radiator, a layer of chrome plating on a steam iron, a fouling up with crud on the tubes of a condenser or other heat exchanger, etc.

The clad system might be depicted in the following manner:

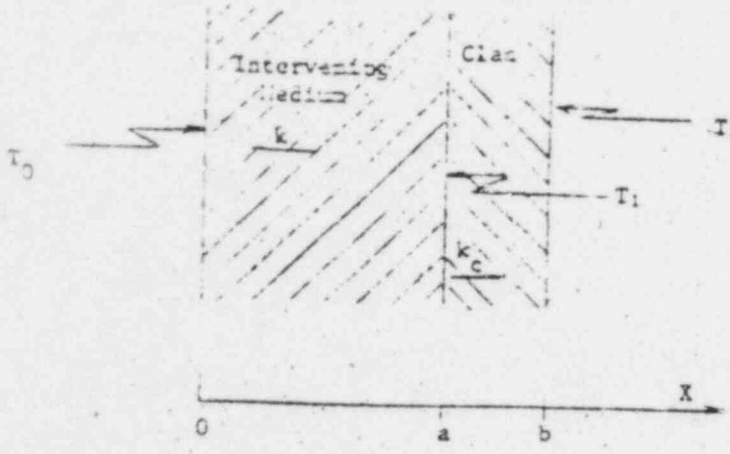


Figure 5

where in addition to what has already been defined, k_c is the thermal conductivity of the clad material and T_1 is the temperature at the interface between the two media.

If we now take Equation (10), $q/A = K \;\; \Delta T/\Delta X$, and rewrite it in the form

$$\Delta T = q/A \; \Delta X/K \qquad (11)$$

we can see that

$$T_o - T_1 = q/A \;\; a/K \qquad (12)$$

and

$$T_1 - T_2 = q/A \; \frac{(b-a)}{k_c} \qquad (13)$$

If we now add these two equations and factor out q on the right side, we find that

$$T_o - T_2 = q \; [\; \frac{a}{KA} + \frac{b-a}{k_c A} \;] \qquad (14)$$

If we divide both sides of the equation by the bracketed term

$$q = \frac{T_o - T_2}{[\frac{a}{KA} + \frac{b-a}{k_c A}]} \qquad 15)$$

where it can be seen that the quantity [a/KA + b-a/k_cA] is the combined thermal resistance of the two media and is equal to the sum of the resistance of the two media. Notice the similarity to the electrical system where resistances in series are added to one another.

When one works with heat transfer situations in which there is more than one type of resistance, one frequently defines another term, the overall coefficient of heat transfer U, which is defined as

$$UA = \frac{1}{\text{Resistance}} \longleftarrow \qquad (16)$$

or for this case

$$UA = \frac{1}{\frac{a}{kA} + \frac{b-a}{k_c A}} \qquad (17)$$

or

$$U = \frac{1}{\frac{a}{k} + \frac{b-a}{k_c}} \ (\text{BTU/hr-ft}^2\text{-°F}) \longleftarrow \qquad (18)$$

Thus Equation (15) can be rewritten as

$$q = UA\,(T_o - T_2)\ (\text{BTU-hr}) \qquad (19)$$

or

$$q/A = U(T_o - T_2)\ (\text{BTU/hr-ft}^2) \qquad (20)$$

where U is defined in Equation (16), or Equation (18) if the surface area remains the same — not true for example in pipes or tubes where the inside and outside area are different.

Let us now look at the effect of adding the cladding to the heat conducting medium. In Figure 6, there are two cases shown. Case number 1 is for a material of thickness a, which is not clad, and which has a hot surface at temperature T_o and a colder surface at temperature T_1. Case number 2 shows the same material of thickness a, but which is clad by a material of thickness b-a. In case 2, the one surface is at the same hot temperature T_o as in case 1, but the outside of the cladding material is at the same temperature, T_2, as T_1 in case 1.

The temperature profile is shown in both case 1 and 2. However, in case 2, there are three conditions considered:

a) for the condition where $k_c = k$. Thus there is no difference between the two materials

b) for the condition where $k_c < k$ (k_c less than k)

c) for the condition where $k_c > k$ (k_c greater than k)

It can be seen that the heat rate, q, or the heat flux, q/A, are both less in the three conditions of case 2 than they are in case 1. This can be seen from looking at the equation

$$q/A = K/a\,(T_o - T_1)$$

where k and a are the same for all conditions of cases 1 and 2 but $T_o - T_1$ in case 1 is the largest for any of the conditions.

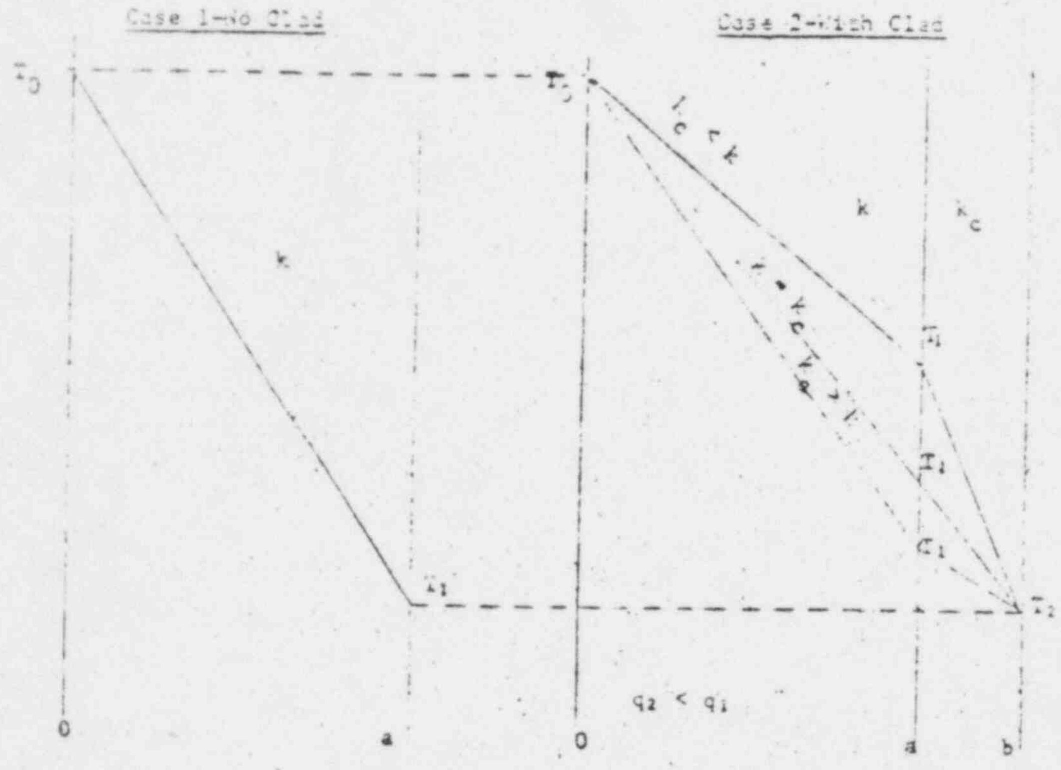


Figure 6

Only in the case where $k_c = \infty$ (infinity) or in other words $R = 0$ would the heat transferred in case 2 be equal to case 1 (assuming same value of a and same temperature limits). Of course if $R = 0$ then T_1 would equal T_2.

Now let us look at an example by considering the case of the radiator again. Suppose we take the same radiator as in our previous example, but let's assume that your wife has been nagging you to paint "that dirty old thing". Thus, you slap on a coat of the nearest paint available. Let's assume that the resulting coat of paint is 1/32" thick and has a thermal conductivity of 4 BTU/hr/°F-ft. What is the new value of q and q/A?

To solve this, let's use Equations (18), (19), and (20). From Equation (18)

$$U = \frac{1}{\frac{a}{k} + \frac{b-a}{k_c}} = \frac{1}{\frac{1}{4x12x30} + \frac{1}{32x12x4}} = 743 \text{ BTU/hr-ft}^2\text{-°F}$$

From Equation (19)

$$q = 743 \times 4 \times (212-68) = \underline{428{,}000 \text{ BTU/hr}}$$

and from Equation (20)

$$q/A = 107{,}000 \ \underline{\text{BTU/hr-ft}^2}$$

Thus we can see that the coat of paint has reduced the rate of heat transfer and heat flux by about a factor of two. This reduction has been brought about by the increase of thermal resistance due to the coat of paint. Or, stated otherwise, the overall heat transfer coefficient has been reduced. For example, in the case of the unpainted radiator

$$U = \frac{1}{a/k} = \frac{k}{a} = \frac{30}{\frac{1}{4\text{x}12}} = 1440 \text{ BTU/hr-ft}^2\text{-°F}$$

and in the painted case $U = 743$ BTU/hr-ft^2-°F, which is approximately 50% of the initial value.

It is of further interest to note what the effect of the paint has been on the temperature drop through the radiator itself. In the simple case we assumed that the outside of the radiator was at room temperature of 68°F. Let us now see what the coat of paint has done to this temperature.

To calculate this, we will utilize the fact that the heat rate, as well as the heat flux, is equal to 428,000 BTU/hr and 107,000 BTU/hr-ft^2 respectively. In other words, the amount of heat which is conducted through the radiator is the same as that which is conducted through the paint. Thus utilizing Equation (12)

$$T_1 = T_0 - q/A \ \ a/k = 212 - 107{,}000 \times \frac{1}{4\text{x}12\text{x}30}$$

$$T_1 = 212-74.2 = 137.8 \text{ °F}$$

The interface temperature between the paint and the metal radiator has been driven up to approximately 138°F by the paint. Another way of stating this would be to say that it was necessary to have a temperature drop of 138-68 = 70°F to enable 107,000 BTU/hr-ft^2 to be conducted through the paint.

Once again, it is mentioned that we have thus far neglected to consider the effect of the air on the outside of the radiator and what effect this might have on the actual outside temperature of the radiator.

Also, in discussing the effects of cladding we have assumed that the thermal contact resistance at the metal to cladding interface was negligible; thus that a good metallurgical bond was obtained.

In the case where we painted the radiator, we might assume that if we adequately prepared the surface of the radiator that we would have obtained a good bond between the metal and the paint. However, what about the effects of temperature on the paint with time. If the paint begins to blister or flake with time, then we would loose the bond at these spots and perhaps entrap insulating pockets of air which would cause the contact resistance to increase.

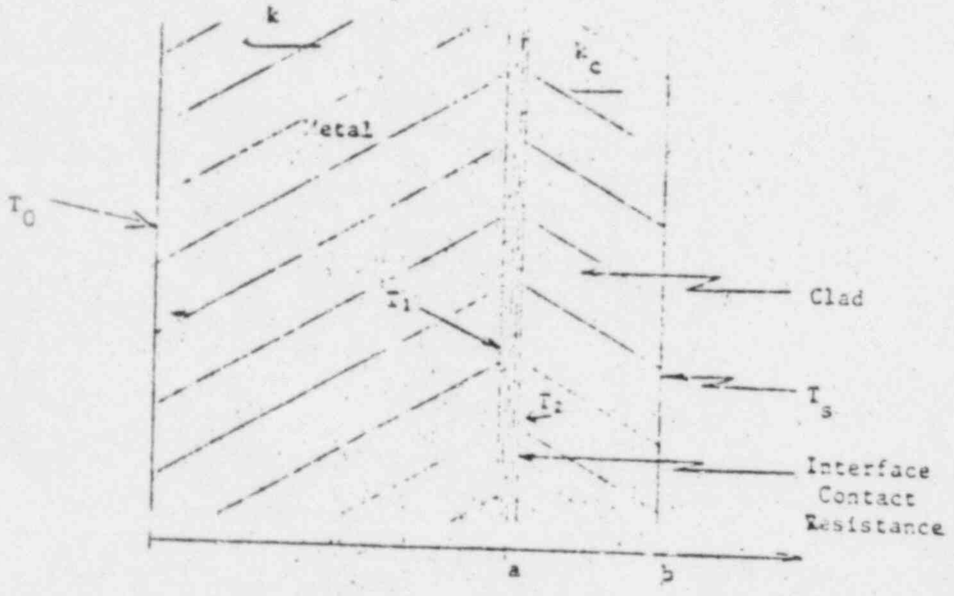


Figure 7

If the value of the contact resistance is known, this gives us no particular calculational problem, because we know that we need only to sum the resistances present in the series heat conduction path.

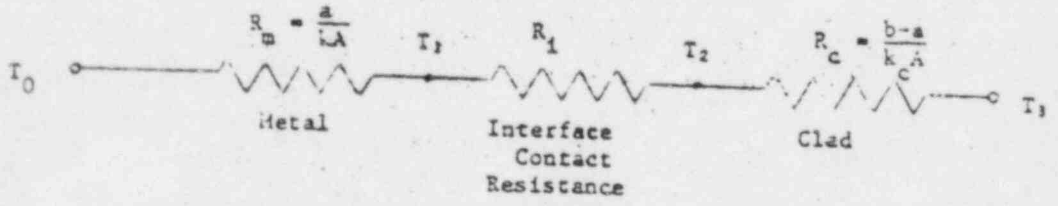


Figure 8

Thus

$$R = \frac{a}{kA} + R_1 + \frac{b-a}{k_c A}$$

and the overall heat transfer coefficient could be found from

$$UA = \frac{1}{R} = \frac{1}{\frac{a}{kA} + R_i + \frac{b-a}{k_c A}} \tag{21}$$

However, the resistance to the flow of heat between two solid surfaces in simple mechanical contact is highly unpredictable. Estimates can be made by calculating the thermal conductance of a gas film of the thickness indicated by the roughness of the surface, but the results are very uncertain. Experience indicates, for example, that a thermal conductance of between 5000 and 10,000 BTU/hr-ft^2-°F results at the surface of contact between uranium metal rods and various cladding materials. However, in the case of uranium dioxide fuel, which cracks during use, the contact conductance is typically on the order of 1000 BTU/hr-ft^2-°F.

Further, in uranium oxide fuel elements there is generally a gas space provided between the oxide and the cladding to provide space for fission gases and to allow for differential expansion between the fuel and the cladding. In this case, the two surfaces may not be in intimate contact and thus the resistance will be relatively high at this point. Sometimes this small space is filled with helium gas. As fission products collect in this space, the effect is to decrease the conductivity (increase the resistance) with time.

8.2.3 Coolant Film Temperature Drop

In our discussions to date, we have assumed that the surface of the material through which heat is being conducted is at the same temperature as the surrounding coolant. Of course, anyone who has touched a hot material such as a hot iron or a hot radiator knows that this is not the case. The flow of heat from the surface of a solid to a fluid is a complex phenomena which is a function of many factors, including the shape and dimensions of the interface between the two, the surface roughness of the solid, whether the fluid is stagnant or flowing, the direction and velocity of the flow, etc. For example, anyone who has blown on his soup or on hot solder has instinctively utilized the fact that the transfer of heat is improved by fluid motion.

Although convective heat transfer to or from a fluid is a complex phenomenon, for our discussion we will look at it in a purely macroscopic sense. For example, let us define the following equation for the convective transfer of heat

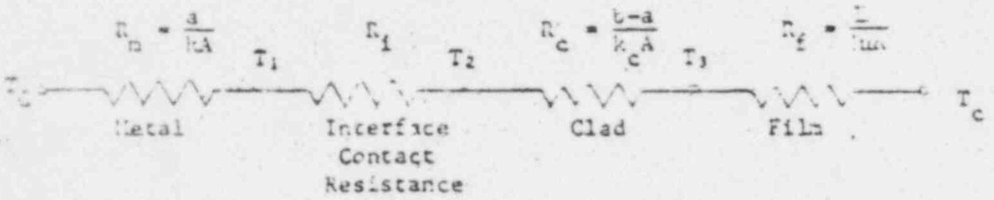
$$q = h\,A\,\Delta T \text{ (BTU/hr)} \tag{22}$$

where q (BTU/hr) and ΔT, the temperature difference between the solid and the fluid in degrees Fahrenheit, are as previously defined. The new parameter h is called the heat transfer coefficient (or sometimes unit thermal conductance) which has the same units as U, the overall heat transfer coefficient, (i.e., BTU/hr-ft^2-°F). The heat transfer coefficient is thus a measure of the heat transferred per square foot of heat transfer area per degree temperature difference between the heat transfer surface and the fluid.

Actually, if we re-arrange Equation (22) into the form

$$q = \frac{\Delta T}{\frac{1}{hA}} \tag{23}$$

we see that the quantity 1/hA expresses the thermal resistance to convective heat transfer caused by the film of somewhat stagnant fluid next to the heat transfer surface. Thus, if we now look at the case of the slab of metal through which heat is being transferred, and if we assume that the slab is clad on one side and heat is transferred to a coolant from the clad, we have the following situation as shown in Figure 9, where the overall thermal resistance can be seen as



$$\text{Thus,}\quad UA = \frac{1}{R} = \frac{1}{\frac{a}{KA} + R_i + \frac{b-a}{k_c A} + \frac{1}{hA}} \tag{24}$$

If there is similarly convective heat transfer at the inside surface such that a heat transfer coefficient, h_i (i for inside), can be defined, then U would be determined from

$$UA = \frac{1}{\frac{1}{h_i A} + \frac{a}{KA} + R_i + \frac{b-a}{k_c A} + \frac{1}{hA}} \tag{25}$$

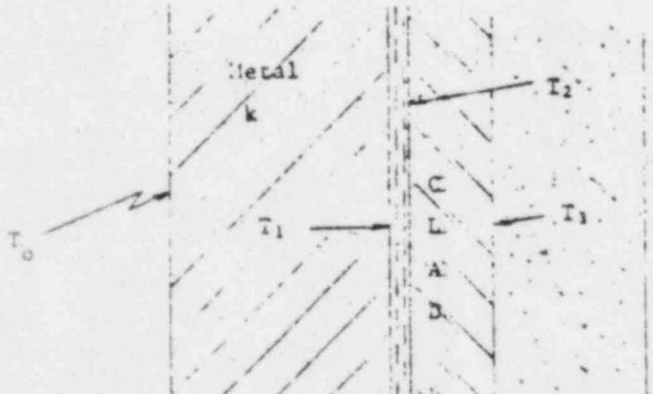


Figure 9

Thus the effect of additional resistances to heat flow on the overall heat transfer coefficient could be determined as has been already demonstrated, and then the rate of heat transferred would be determined from

$$q = UA\ (T_i - T_c) \qquad (26)$$

where T_i is the temperature of the hot fluid.

Let us now return to the case of the painted radiator and determine what effect the resistance of the film has on the temperature profile through the radiator, the rate of heat transfer and the heat flux.

We will assume that the resistance on the inside of the radiator is quite small. This should be a good assumption because steam will be condensing on the inside surfaces and this condition usually results in high film coefficients. Further, we will assume that the painter did a good job with high quality, high temperature paints so that we can ignore any contact or air gap resistances between the paint and the radiator. Further, we will assume that the heat transfer coefficient due to natural convection cooling by the room air flowing over the radiator is 1.5 BTU/hr-ft^2-°F. In order to solve the problem, let us first find the overall heat transfer coefficient, U. For this case, from Equation (25)

$$UA = \frac{1}{\frac{a}{kA} + \frac{b-a}{k_c A} + \frac{1}{hA}}$$

or

$$U = \frac{1}{\frac{a}{k} + \frac{b-a}{k_c} + \frac{1}{h}} = \frac{1}{\frac{1}{4\times12\times30} + \frac{1}{32\times12\times4} + \frac{1}{1.5}}$$

or

$$U = 1.497 \approx 1.5\ \text{BTU/hr-ft}^2\text{-°F}$$

Thus, we see that because of the large thermal resistance of the film (or low heat transfer coefficient, h, of the film) the overall heat transfer coefficient, U, is essentially equal to h in this case. Thus, the heat transfer rate will be controlled or limited by the film. In other words, the film is acting as a good insulator in this case.

We can now determine q and q/A

$$q = UA\ \Delta T = 1.497 \times 4 \times (212-68) = 862\ \text{BTU/hr}$$

and

$$q/A = 215.5\ \text{BTU/hr-ft}^2$$

The rate of heat transferred and the heat flux have been reduced by a factor of 495 by the outside film (as can be seen by the comparison).

You might be interested in the fact that heating and ventilating engineers have determined that typical steam radiators, operating under standard conditions (steam at 215°F and room air at 70°F, thus a 145 degree difference), transfer between 210 and 270 BTU/hr-ft^2. For calculations, they assume an average of 240 BTU/hr-ft^2. As can be seen then, our somewhat simplified approach has given us quite realistic results.

One should also note that the effects of the paint on the rate of heat transfer is not nearly as great, once we also consider the effect of the film of air on the outside.

Let us now look at the temperature profile, once again remembering that the rate of heat transfer through the metal is the same as that through the paint as well as that through the film. This must be true in the steady state or we would be storing or gaining heat somewhere.

Thus,

$$T_1 = T_0 - q/A\ a/k = 212-215.5 \times \frac{1}{4\times12\times30} = 212-0.1495 = 211.85°F$$

$$T_2 = T_1 - q/A\frac{b-a}{k_c} = 211.85-215.5 \times \frac{1}{32\times12\times4} = 211.85-.141 = 211.71°F$$

and as a check

$$T_c = T_2 - q/A\ 1/h = 211.7-215.5/1.5 = 211.7-143.7 = \underline{68°F}$$

Let us now look at Figure 10 where there are several interesting observations one can make. One is that in Case 2 (clad present) the rate of heat transfer can be seen to have been reduced by about a factor of two over that in Case 1 (no clad present). However, because of the effects of the relatively stagnant layer of coolant in Case 3 (clad and film present), the rate of heat transfer in this case has been reduced by almost 1000 times over that of Case 1. There is not too much we can do, in a practical sense, to reduce the resistance of the film. Of course, we could install a blower on each radiator to blow air over the radiator. This would tend to break up the film somewhat and give a higher value of h. This is, of course, not too practical in the home, but is used industrially.

The temperature profile through the film in Case 3 is shown as a dashed line, because we do not know its exact shape. However, it is not linear other than for the extremely thin layer next to the metal, where the heat transfer is primarily by conduction to the air molecules rather than by convection.

Although our examples have been primarily that of a radiator, because this is something we are all familiar with, what has been said is applicable to heat removal in a reactor. For example, the metal could just as easily be the fuel in a

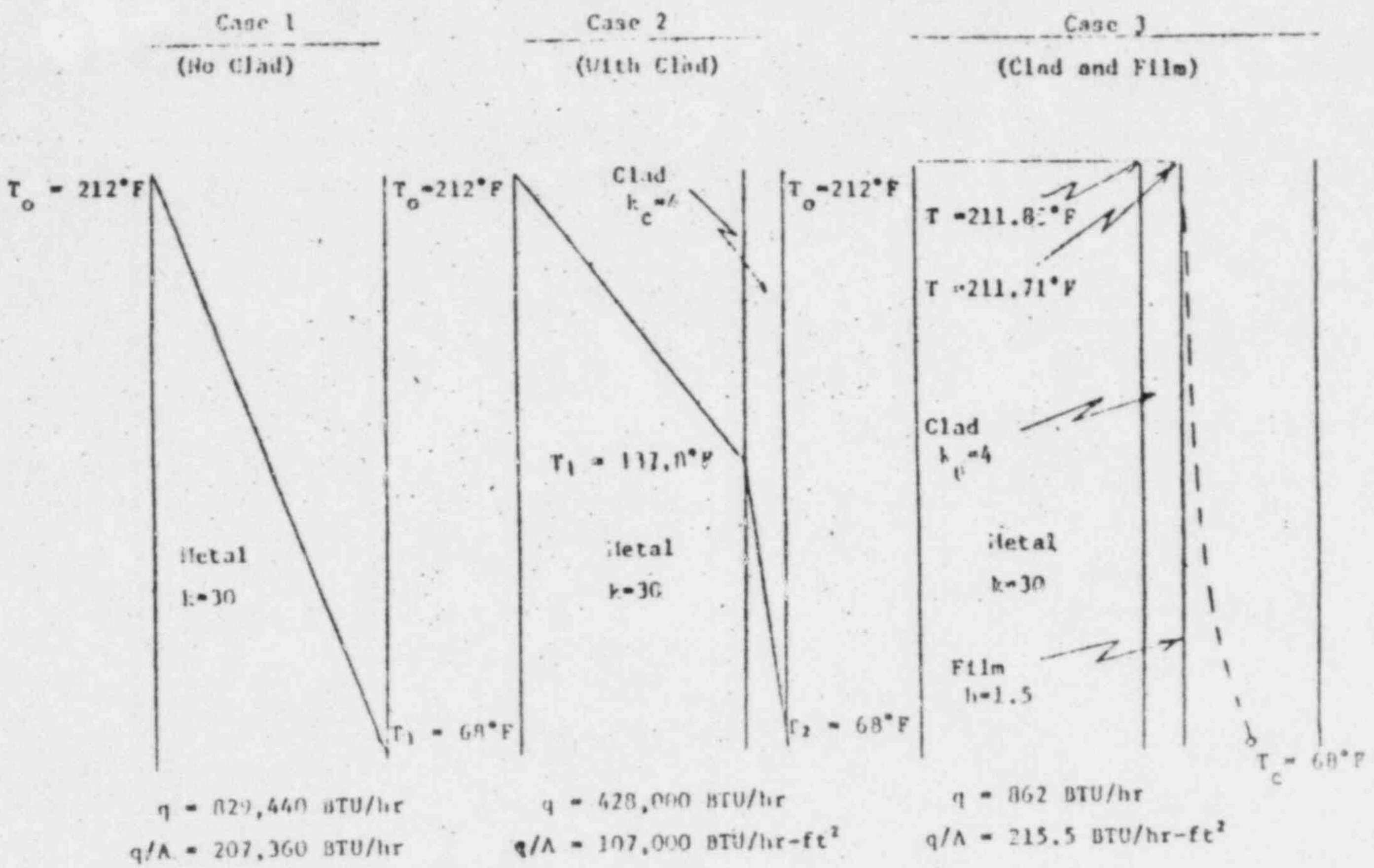

Case 1
(No Clad)
Case 2
(With Clad)
Case 3
(Clad and Film)
T_o = 212°F
T_o=212°F
Clad
k_c=4
T_o=212°F
T =211.8_°F
T =211.71°F
Clad
k_c=4
Metal
k=30
T_1 = 68°F
T_1 = 137.0°F
Metal
k=30
T_2 = 68°F
Metal
k=30
Film
h=1.5
T_c= 68°F
q = 829,440 BTU/hr
q/A = 207,360 BTU/hr
q = 428,000 BTU/hr
q/A = 107,000 BTU/hr-ft²
q = 862 BTU/hr
q/A = 215.5 BTU/hr-ft²

Figure 10

fuel element (although heat would be internally generated in the fuel as well as conducted through it). The paint on the radiator could just as easily be the cladding on the fuel which prevents fission product release and corrosion of the fuel (however, h_c for most cladding would be higher than for paint). Further, the coolant could be water under either natural or forced convection conditions, except that the heat transfer coefficient in that case would be higher than that for air under natural convection conditions.

We should also remember that we have been looking at the simplest possible geometry situation, that of slab or one dimensional cases. For example, the radiator might be more suitably considered as hollow cylinders rather than a slab. However, as soon as one goes to other geometries such as cylindrical geometry and spherical geometry, the resulting equations become more complex and many times mask out the effects of physical phenomena. Because our purpose is to introduce the topic of heat transfer and fluid flow, with emphasis on concepts, definitions, units, phenomena, etc., we have attempted to simplify the mathematical complexities. One should be aware of the fact, however, that for handling the effects of heat transfer in pin or rod type fuel elements, pipes, etc., the form of the equation would be somewhat different than those we have discussed. The general effects of cladding, film coefficients, etc., would be quite similar, however.

8.2.4 Effects of Uniform Internal Generation on Temperature Profile

In our discussions to date, we have been assuming that the heat that is being transferred has been produced external to the material and has actually been introduced through one of the boundaries of the material and has been conducted through it to the other boundary which is being maintained at a cooler temperature.

However, in a reactor, there are numerous components that receive their heat through internal generation caused by fission (in the case of fuel elements) or through the absorption of radiation (which also releases heat in the material). For example, in Table 1 in Section 5.3 on Reactor Physics, we saw that the total amount of energy released from fission is approximately 207 Mev. Of this, approximately 12 Mev appears in the energy of neutrinos which escape from the reactor system removing their energy with them. In addition to the remaining 195 Mev from fission, there is approximately 3 Mev of energy released as a result of parasitic capture of neutrons in the various materials of the core. Thus approximately 198 Mev of energy is available as heat to the reactor system. Of this, approximately 90% is released as heat in the fuel elements (178 Mev), 5% (10 Mev) is released in the moderator and 5% in the reflector and shields.

The heat produced in the fuel from fission will be distributed in a manner proportional to the flux, because it is the neutron interactions in the fuel which produce fission. Thus, if the flux is not uniform over the core, neither will be the heat generation rate. However, for our initial look at the effects of heat generated in a material, let us assume that we do have uniform internal generation.

For our discussion, let us assume that we have a thin slab of fuel in which heat is being uniformly generated due to the fissioning of uranium. This condition is shown in Figure 11. For simplicity, we will assume that heat is being transferred from left to right due to the fact that the left hand side is well insulated and the right side is kept at a lower temperature, T_1, by suitable cooling.

We can see that this is a case similar to that shown in Figure 4, but in which instead of hot gases introducing heat on the left surface, this surface is insulated so that no heat enters or escapes from this side. All heat is produced uniformly inside the fuel and flows out the right hand surface maintaining it at temperature T_1.

Now, if we derived the equations for the heat transferred in the slab, we would find that t(x), the temperature at any position x in the fuel, would be

$$T_x = T_0 - \frac{Q x^2}{2 k_f} + \left(\frac{T_1 - T_0}{a} + \frac{Qa}{2k_f}\right) x \qquad (27)$$

where Q is the <u>internal heat generation rate in BTU/hr-ft^3</u>.

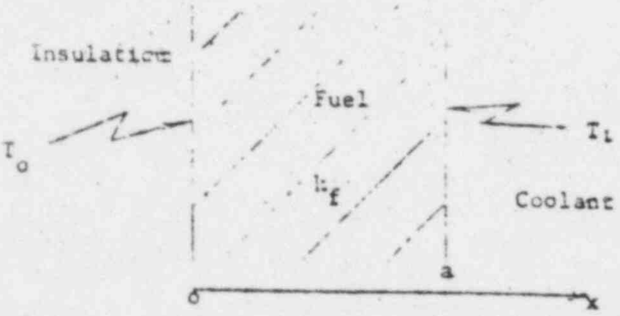


Figure 11

Further, we would find that the heat flux would no longer be constant in the x direction (as we would move from left to right, the heat flux would increase due to the fact that all heat generated to the left would be transferred to the right) and could be expressed as

$$q/A = \frac{Q}{2}(2x-a) + \frac{k_f(T_0 - T_1)}{a} \qquad (28)$$

If we now look at the case where x=a (i.e., the case at the right hand surface), then

$$q/A = \frac{Qa}{2} + \frac{k_f(T_0-T_1)}{a} \qquad (29)$$

Now, in order that we can compare these results with Equation (10), let us assume that the rate of internal heat generation, Q, is such that the heat flux on the right hand surface is the same as that considered earlier in the derivation of Equation (10).

Thus $Q = \frac{q}{Aa}$ BTU/hr-ft^2

Then from Equation (29), $q/A = \frac{qa}{2Aa} + \frac{k_f(T_0-T_1)}{a}$

or

$$\frac{q}{2A} = \frac{k_f(T_0-T_1)}{a}$$

or finally

$$q/A = \frac{2k_f(T_0-T_1)}{a} = 2k_f\frac{\Delta T}{\Delta X} \qquad (30)$$

or multiplying through by A

$$q = 2\,k_f\,A\,\frac{\Delta T}{\Delta X} \qquad (31)$$

Thus, if we compare Equation (31) with Equation (7) or compare Equation (30) with Equation (10), we see that if we have a slab conducting heat and which has the same thickness, conductivity, and temperature drop as a second slab which has uniform internal heat generation, the heat transferred in the case with internal generation will be twice that of the straight conduction case. Another way of expressing this would be to say that for the same rate of heat transfer, a slab with uniform internal generation would require only half the temperature drop as the slab not having internal generation. Actually, this is not too surprising when you consider that with uniform internal generation, the heat has to travel, on the average, only half the distance between x=0 and x=a.

Although we will not take the time to work an example of this case, you should note that the expression for the temperature drop across the slab of fuel is no longer as simple as was the case for simple conductive heat transfer. The temperature drop is no longer linear as previously was the case.

8.2.5 Temperature Profile in Clad Plate-Type Fuel Element

Let us now look at the plate-type fuel element depicted in Figure 12.

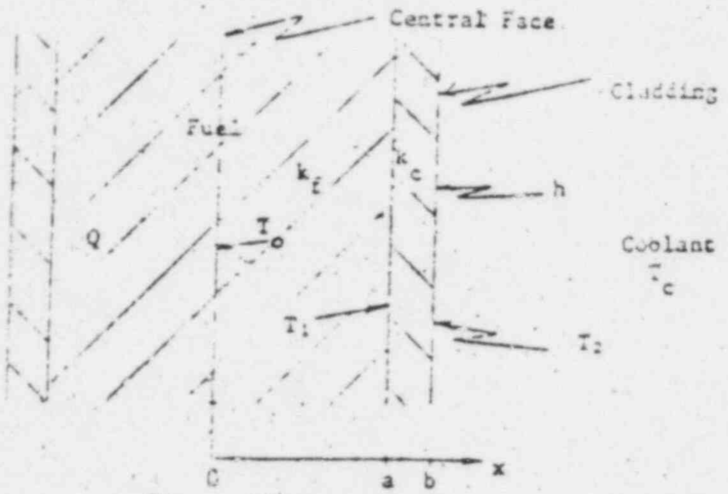


Figure 12

One can see in Figure 12 that the fuel has a thickness of 2a. The temperature at the center face is assumed to be T_0, that at the fuel-cladding interface is T_1', at the cladding-coolant boundary, T_2, and the coolant temperature is T_c. Actually this situation is quite similar to the physical situation we have previously considered, except that internal generation is present in the fuel, and also cladding and coolant are now present on both sides of the fuel. However, by taking advantage of symmetry and the fact that if the heat generation rate is uniform, we need consider only one half the fuel element, and what we find there will also apply to the other half. Therefore, let us work with the +x direction.

We know that at the interface the heat flow in the positive and negative x direction equal each other and therefore there is no net heat flow. That is

$$\text{at } x=0 \quad q=0 \quad q/A=0$$

If we substitute this into Equation (28) we find

$$T_0 - T_1 = \frac{Qa^2}{2k_f} \tag{32}$$

Also, using the method previously used for the cladding where we will assume that no heat is generated internally (some will be from absorbing neutrons, gammas and betas)

$$q = \frac{A\, k_c (T_1 - T_2)}{b-a}$$

But q for this case = QAa so that $(T_1 - T_2) = \frac{Qa(b-a)}{k_c}$ (33)

and similarily

$$T_2 - T_c = \frac{Qa}{h} \tag{34}$$

Upon adding Equations (32), (33), and (34) we obtain

$$T_0 - T_c = \frac{Qa^2}{2k_f} + Qa\left(\frac{b-a}{k_c} + \frac{1}{h}\right) \qquad (35)$$

One should note that in Equation (35), we have assumed that the contact resistance between the clad and the fuel is negligible. If this is not the case, the resistance can be handled as previously shown in Equations (24) and (25).

Also, because the temperature profile in the fuel is no longer linear, one must obtain a separate expression for the temperature profile. This can be obtained by substituting Equation (32) into Equation (27), obtaining an expression for the temperature at any position x in the fuel as

$$T(x) = T_0 - \frac{Qx^2}{2k_f} \quad \text{(in fuel only)} \qquad (36)$$

In order to work an example, let us assume that we have a plate type fuel element, in which the fuel is UO_2 and is 1/2 inch thick and which is clad on each surface with 0.020 inches of zircalloy. We will assume that the coefficient of heat transfer, h, between the coolant and cladding is 3000 BTU/hr-ft^2-°F and the rate of internal heat generation to be 10^7 BTU/hr-ft^3. If the thermal conductivity of the fuel is assumed to be 1.2 BTU/hr-ft-°F, k_c for the zircalloy to be 7 BTU/hr-ft-°F and the average temperature of the coolant be 500°F, let us find temperatures T_0, T_1 and T_2.

From Equation (34), (remember that a = 1/2 x 1/2 x 1/12 = 1/48 ft)

$$T_2 = T_c + \frac{Qa}{h} = 500 + \frac{10^7}{48 \times 3000} = 500 + 69.4 = \underline{569.4\ °F}$$

From Equation (33)

$$T_1 = T_2 + \frac{Qa(b-a)}{k_c} = 569.4 + \frac{10^7 \times 0.020}{48 \times 7 \times 12} = 569.4 + 49.6 = \underline{619°F}$$

From Equation (32) or (36)

$$T_0 = T_1 + \frac{Qa^2}{2k_f} = 619 + \frac{10^7}{2 \times 1.2 \times 48 \times 48} = 619 + 1805 = \underline{2424°F}$$

In order that you might more readily visualize the temperature profile obtained in the above problem, take a piece of graph paper and plot the temperatures. Remember that the temperature profile in the fuel is no longer linear, so at least three other temperatures (i.e., at x = 1/16, x = 1/8, x = 3/16 inches) should be obtained.

Note that in our earlier discussion of the radiator, the radiator was a good heat conductor, but because cooling was by natural air convection the film coefficient was low. Thus, we saw a relatively small temperature drop through the metal, but a relatively large drop through the film. In this fuel element example, the opposite is the case. Uranium dioxide (which is a ceramic and thus a poor heat conductor) has a low thermal conductivity and thus requires a large temperature drop to cause appreciable heat flow. In contrast, the film coefficient was large (indicating forced convection of high velocities) and as a result the temperature drop through the film is small. It is a feeling for the possible magnitude and shape (or profile) of the temperature distribution that we want you to get from these discussions in contrast to memorizing or working with the equations.

8.2.6 Coolant Temperature Rise and Exit Conditions — Uniform Generation

In our discussion to date, where we have considered the presence of a coolant, we have assumed that the coolant remained at some average temperature which we designated as T_c. However, we know that if we add heat to the coolant, its temperature will rise (unless it is at its saturation temperature). For example, from the definition of the unit of heat, the BTU, we know that one pound of water will rise in temperature one degree Fahrenheit with the addition of 1 BTU of heat energy.

In the programmed learning portion of these lectures you worked with the temperature increases or decreases of various materials with the addition or loss of heat. Also, you learned that one can calculate the rate of heat transfer if you know

a) the coolant flow rate, W(lb/hr)
b) the temperature rise, ΔT (°F)
c) the specific heat c (BTU/lb-°F)

or in other words

$$q = W\ C\ \Delta\ T\ \text{(BTU/hr)} \qquad (37)$$

where q is the rate of heat transferred to or from the coolant. Now, let us look at Figure 13 where we have attempted to depict a coolant channel between two adjacent fuel plates. Fuel plates, rather than fuel pins or rods, are pictured once again because of the simplicity of working with and visualizing slab geometry. However, if one has fuel elements in the form of pins or rods with the coolant flowing axially up past the rods, the phenomenon is the same, but the mathematics slightly more complicated.

As the coolant flows up through the coolant channel, it is increased from the inlet temperature, T_i, until it flows out the top of the channel at temperature T_o. (Do not confuse this T_o with that previously used in our discussion of temperature profiles inside the fuel.)

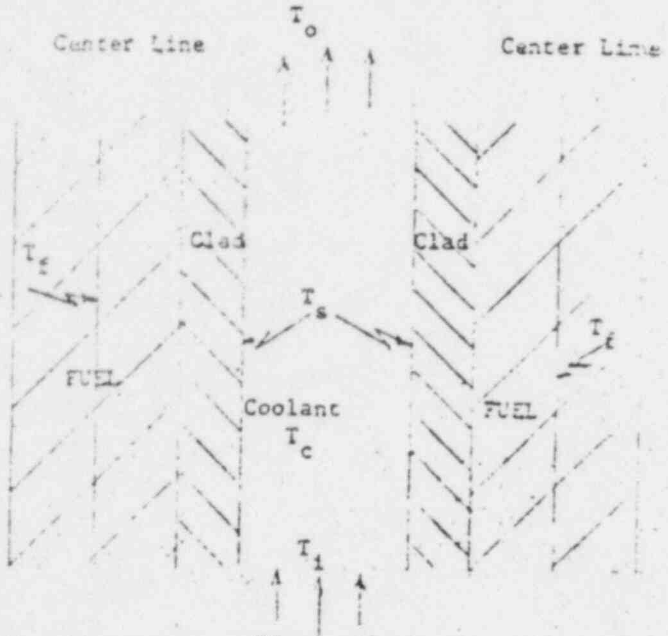


Figure 13

If the rate of internal heat generation in the fuel adjacent to the cooling channel is uniform along the whole length of the coolant channel, then the heat added to the coolant per foot of travel up the channel will also be constant, causing a temperature rise which would be proportional (or linear) to the distance up the channel. Thus the temperature rise of the coolant as a function of position in the channel would be as shown in Figure 14.

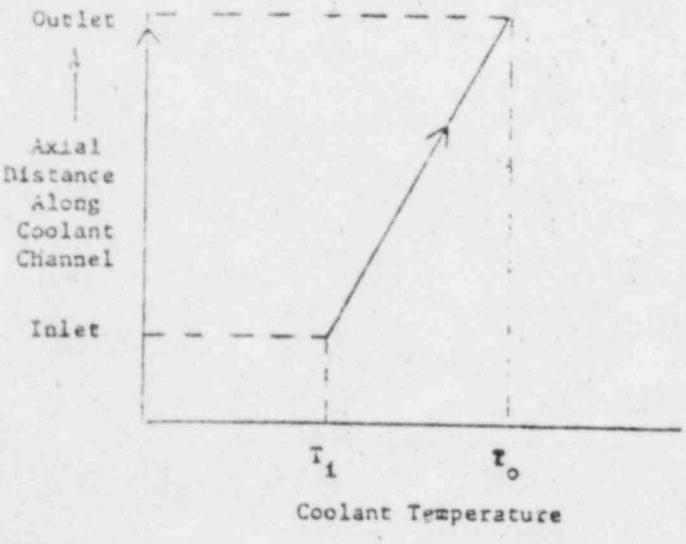


Figure 14

But what does this do to the surface temperature, T_s, of the cladding and what does it do to the temperature of the fuel element at its centerline, T_f?

Well we should know from our previous discussions that, because the same amount of heat is being generated at each position up the channel, a uniform amount of heat must be transferred through the fuel, the cladding and the film. Thus the temperature drop between the surface of the cladding and the coolant (i.e., the film temperature drop) $T_s - T_c$ will be a constant. This is also true for the temperature difference between the fuel centerline and the surface of the cladding, $T_f - T_s$ (although one would expect that $T_s - T_c$ would be different in magnitude than $T_f - T_s$, as seen by our previous examples).

Let us now sketch what might happen to the surface temperature of the cladding, T_s, and the centerline temperature of the fuel with the assumption that heat is uniformly generated in the fuel, as is shown in Figure 15.

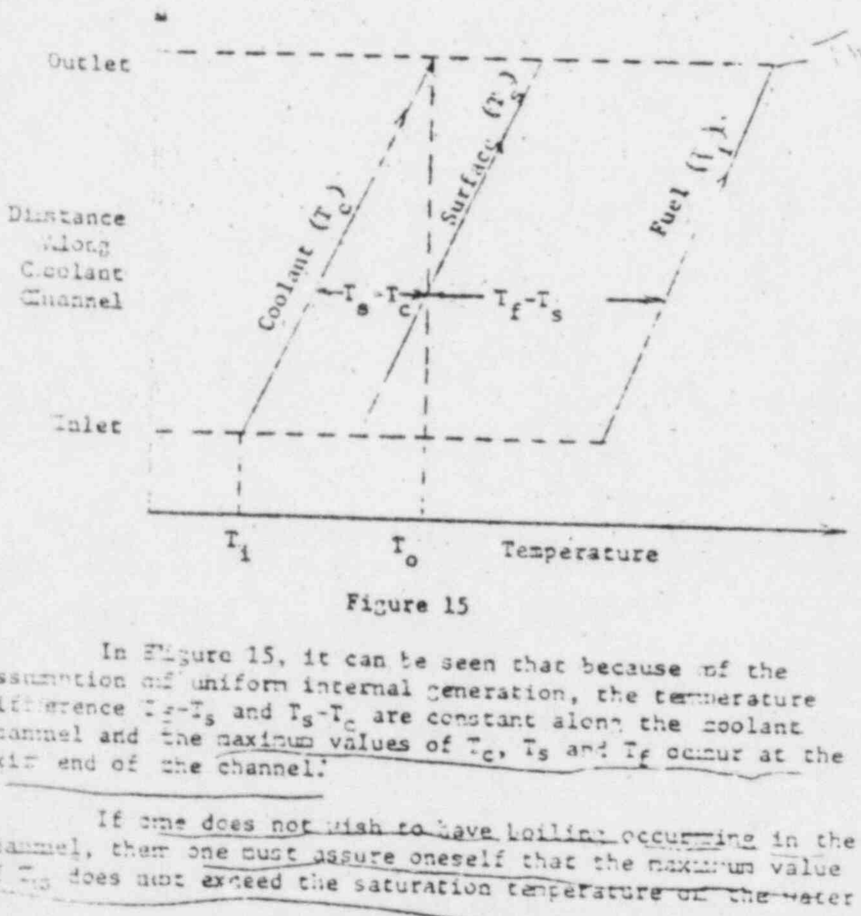


Figure 15

In Figure 15, it can be seen that because of the assumption of uniform internal generation, the temperature difference T_f-T_s and T_s-T_c are constant along the coolant channel and the maximum values of T_c, T_s and T_f occur at the exit end of the channel.

If one does not wish to have boiling occurring in the channel, then one must assure oneself that the maximum value of T_s does not exceed the saturation temperature of the water for the pressure conditions at that point. Keeping in mind the fact that to cause water to flow up the channel, there is, by necessity, a need for a pressure drop from the inlet to the outlet to cause the appropriate flow rate. One should also note that from Equation (37) it can be seen that for a given amount of heat transferred to the water in the channel, the temperature increase will be less if the coolant flow rate in the channel, m, is increased. Thus, if the channel we are considering is the hottest channel in the core and if the coolant exit temperature or the maximum surface temperature is too high, one could cause these temperaturesto be less if more coolant could be caused to flow through the channel. For this reason one finds many times that orifices are placed in the channel to control the flow rate through the channel.

As one attempts to push the temperatures of the fuel and the coolant exit to a maximum to gain maximum thermal efficiencies and reactor power outputs, one becomes extremely interested in knowing that one does not exceed the maximum temperatures without damaging the materials or causing excessive boiling, etc. In the case of uniform internal heat generation, the problem might not be too extreme, because at least we would know that the maximum temperatures would occur at the exit. Therefore, we could look at these temperatures more closely and perhaps even install instrumentation at that point to measure the temperatures.

8.2.7 Non-Uniform But Symmetrical Internal Generation

Unfortunately, the rate of internal generation of heat inside a reactor is not uniform throughout the reactor. This is due to the fact that the neutron flux which causes fission, is not uniform throughout the core. As one would expect, neutrons leak out of the outside surfaces of a reactor core and are lost to the reactor. Just as in any other physical phenomenon in which there is a net outflow from a system (water flow, electricity flow, heat flow) there is a gradient or potential necessary to cause the flow. In fact, we have just spend considerable time discussing the fact that temperature gradients (or drops) are necessary to cause heat to flow.

Therefore, the neutron flux in a reactor core is not uniform, but is a maximum in the middle and decreases according to mathematical relationships, which are dependent on the overall shape of the reactor core. Therefore, the neutron flux (and thus the internal heat generation rate) in a reactor might conceivably have the general shape shown in Figure 16.

In reactor cores in which the shape of the core approximates that of an upright cylinder, the theoretical shape of the neutron flux (ignoring control rods, etc.) in the axial direction is that of a cosine function as shown in Figure 16.

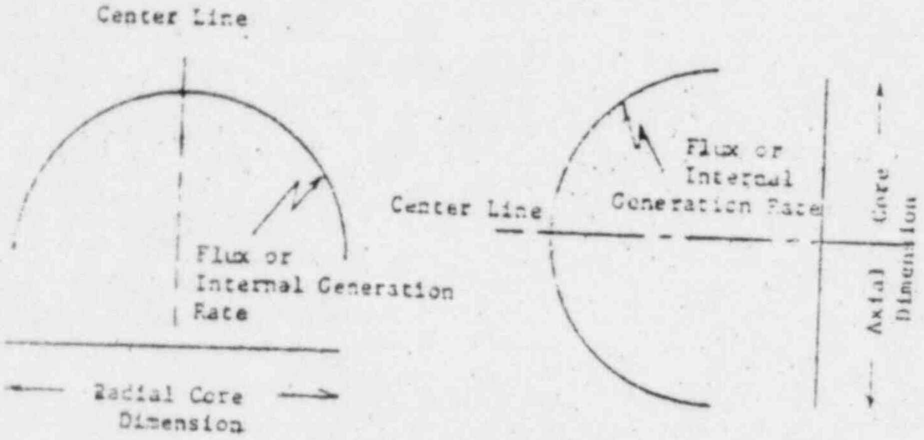


Figure 16

Thus, instead of heat being generated uniformly along the axial length of the coolant channel, more heat is generated (and thus transferred to the coolant) at the axial centerline than is generated elsewhere. Therefore, the rate of temperature increase of the coolant would be greater at the center of the core than it would be near the inlet and outlet. Thus the temperature rise of the coolant would be as shown in Figure 17 in contrast to that shown in Figure 14.

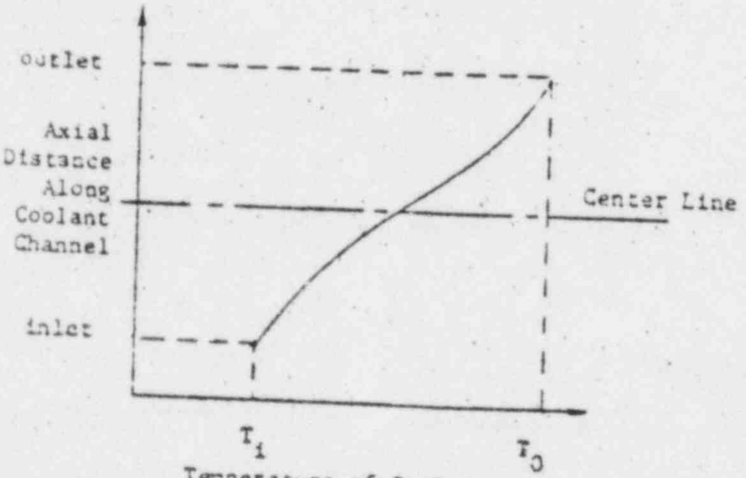
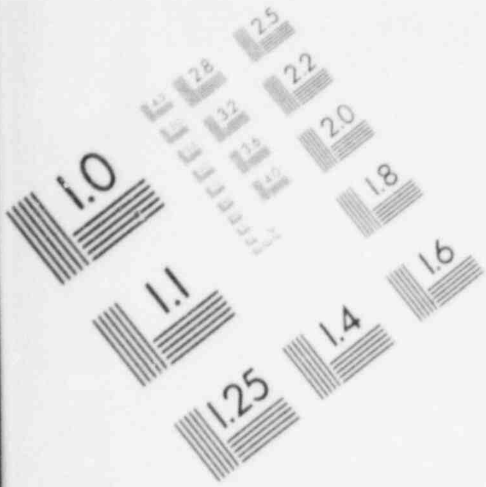
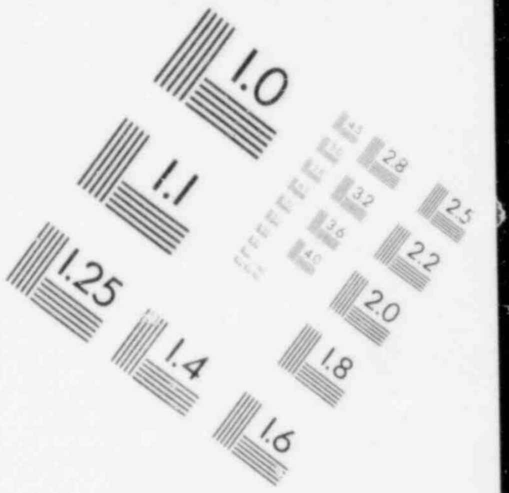


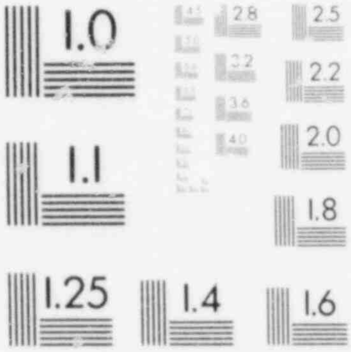
Figure 17

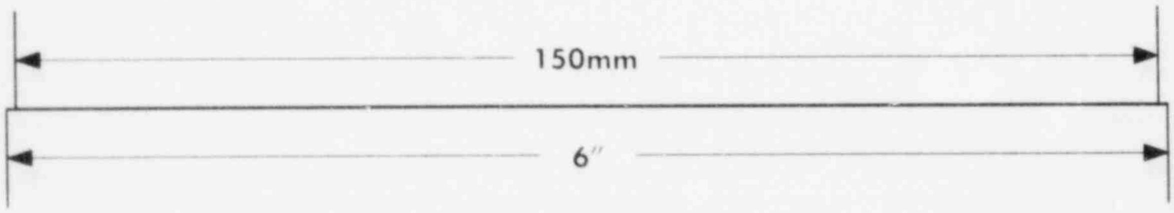
From the above figure it can be seen that the water does not take on much heat (thus there is not a large temperature rise) at the inlet to the coolant channel. However, as one approaches closer to the axial centerline, more and more heat is gained and thus the rate of temperature rise per

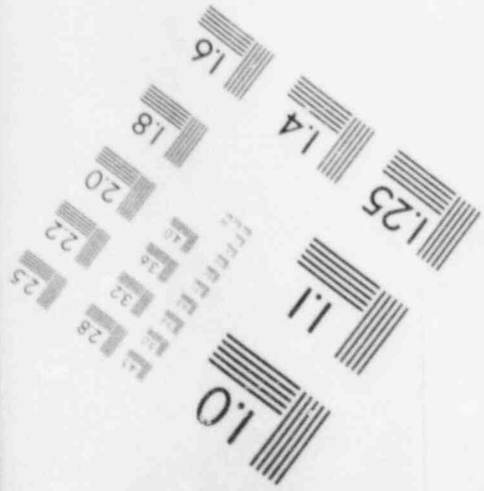
IMAGE EVALUATION
TEST TARGET (MT-3)

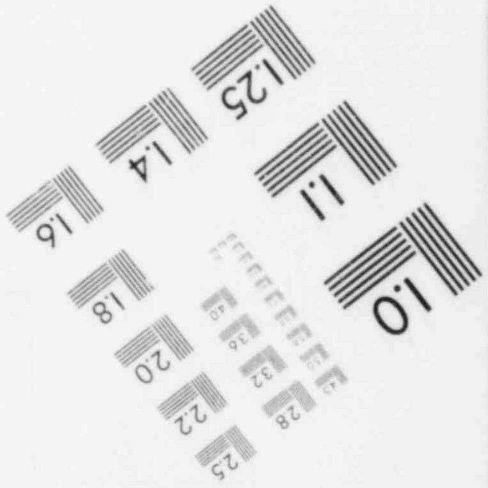












unit length of channel increases until a maximum rate is reached at the centerline. Beyond this point the rate of temperature rise once again decreases, but of course the temperature continues to increase because heat is still being added.

Let us now look at what effect this has on the surface temperature of the cladding and on the fuel element centerline temperatures. Because the rate of internal heat generation is no longer uniform along the axial length of the channel, the amount of heat being transferred through the fuel and through the cladding to the coolant will vary along the channel. Further, because the temperature drop through the fuel and cladding and through the coolant film will be proportional to the heat being transferred through them, the temperature difference $T_f - T_s$ and $T_s - T_c$ will no longer be constant along the channel. However, the temperature difference $T_f - T_s$ and $T_s - T_c$ should be of the same general shape as the axial neutron flux or internal generation rate shown in Figure 16.

As a result of these considerations, the axial shape of the cladding surface temperature and the fuel centerline, temperatures would be of the general form shown in Figure 18.

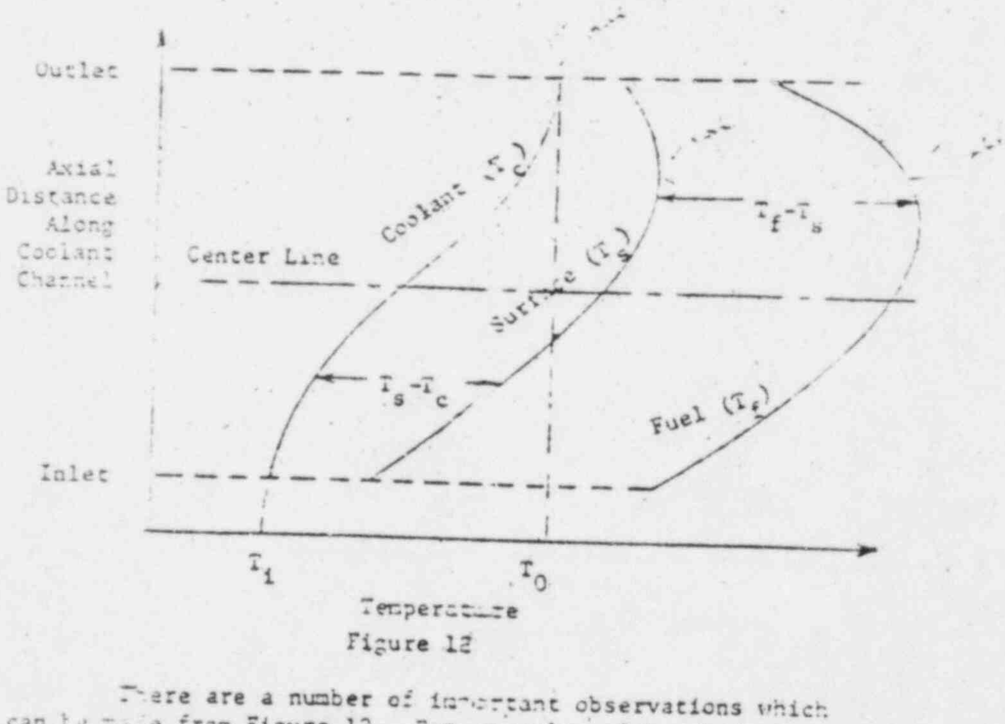


Figure 18

There are a number of important observations which can be made from Figure 18. For example, although it is difficult to see because of the critical effects, the temperature differences $T_f - T_s$ and $T_s - T_c$ have the shape of the axial flux in Figure 16. We also note that the maximum cladding surface temperature, T_s, and maximum fuel centerline temperature, T_f, no longer occur at the exit of the channel. In fact, the fuel surface temperature increases from the inlet condition until it reaches a maximum and then decreases. Thus the point at which the maximum surface temperature occurs is not readily defined. This also occurs for the fuel centerline temperature. It increases as one moves axially up the channel, reaches a maximum and then decreases, so that its point of maximum is also not readily defined. Although it is perhaps not too easy to see, the maximum fuel temperature occurs at a point slightly closer to the channel inlet than does the surface temperature.

Note also that because the fuel temperature is not constant along the length, one might expect that heat would flow axially in the fuel from the point of maximum temperature to the fuel at lower temperatures. This is true, but because the axial length is long compared to the heat flow path to the coolant, the assumption that heat flows only in the one direction (from fuel to coolant) is a reasonable one.

3.2.6 Actual Internal Generation Conditions

Actually, if the internal generation rate was of a cosine shape in the axial direction, the problem of determining the point at which the maximum temperatures occurred and how high these temperatures are would not be a major mathematical problem for the engineer. However, unfortunately life is not even this simple. We know for example that in many PWRs the control rods enter the reactor vessel at the top and especially at the beginning of core life are inserted into the reactor at varying distances. These distances depend on a number of considerations including power levels, fuel burn-up, amount of xenon present, reactor temperature, etc.

These control rods which contain materials, which are heavy absorbers of neutrons, distort the shape of the neutron flux quite appreciably. For example, if the control rods are partially inserted, they depress (or decrease) the neutron flux where they exist. By doing this they might distort the neutron flux in a manner similar to that shown in Figure 19.

As seen from Figure 19, the effect of the control rods partially inserted into the top of the reactor is to depress the flux or heat generation rate below what it would be if they were not present. This then causes the flux to "belly-out" in the lower regions of the core. If one were to study the effect of this type of heat generation distribution, one would find the temperature profile of the coolant, cladding surface and fuel centerline to be more distorted than was the case in Figure 13. Further, the point at which a particular temperature would be a maximum would be quite impractical to determine because it would be different for each position of the rods, coolant temperature, etc. Thus, if we wanted to measure the maximum temperature we would not know where to locate the thermocouple.

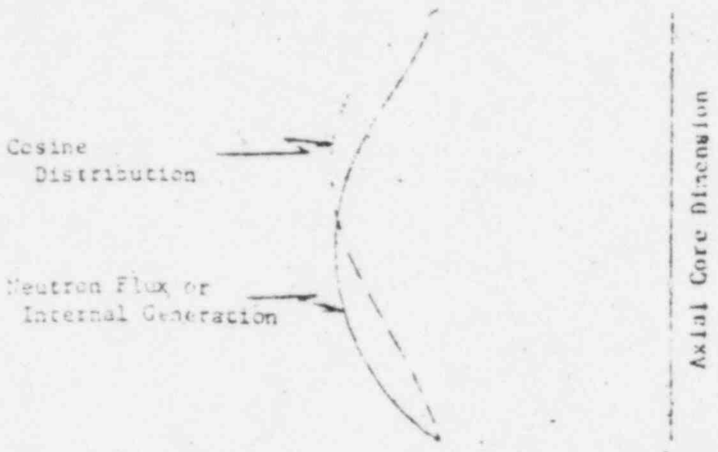


Figure 10

8.2.9 Effects of Non-Uniform Radial Heat Generation

We have discussed the effects of non-uniform axial heat generation. Unfortunately, as indicated in Figure 16, the flux or heat generation in the radial direction is also not uniform. Thus, what we have discussed so far about the axial temperature distribution would not be the same for each channel, but would be different for each coolant channel in the core, depending on its radial position. The effect of rods on the radial distribution of flux is also important. For example, if one inserted the control rods (from the top) in one side of the core, but left the rods on the other side out of the core, the radial distribution would be distorted in a manner perhaps similar to that shown in Figure 10 for the axial condition.

Thus, by being able to predict the point in the reactor where the fuel is the hottest or the surface temperature is the hottest, so that certain limits will not be exceeded, is an extremely difficult problem. As a result, one is presently faced with putting an extremely large number of nuclear detectors throughout the core to determine the approximate shape of the neutron flux, because one can then estimate how the heat generation, heat fluxes, fuel temperatures and surface temperatures are distributed to make sure that they do not exceed prescribed license or safe limits.

UNCERTAINTIES

8.2.10 Uncertainities and Hot Channel Factors

From our previous discussions you should now be aware of the fact that the internal heat generation rates are not really uniform throughout the reactor core or in fact in any one particular fuel element because of radial, axial or local neutron flux perturbations. Because these neutron flux and thus internal heat generation rates can also vary as a function of time (due to fuel burnup, poison build-in and burnout, control rod movements, etc.), it should be apparent that each coolant channel or fuel element is potentially different from each other one in the reactor core. Therefore, some will be hotter channels than others and thus will be closer to the point of exceeding cladding material or fuel melting temperatures. The design engineer must make sure that the hottest channel does not exceed safe temperature limits and the operator must be aware of the operating restrictions placed on the reactor so that the reactor is operated in a manner consistent with the design engineer's asumptions.

In the previous sections an attempt was made to demonstrate that it is difficult to know which channel or fuel element is the hottest, or, in a particular channel, at what location the maximum cladding surface temperature or maximum fuel element centerline temperature is reached.

Actually, there are numerous factors in addition to the radial and axial variations in heat generation rates that affect where the hottest channel or the hottest spot in a channel occurs.

For example, if we refer to Figure 20 we will see depicted several possible sources of uncertainities which might result in the creation of hot spots in the fuel elements or cladding. Some of these are as follows:

a) Irregular contact between fuel and cladding. At specific spots the fuel and cladding might be in intimate contact and thus provide for good heat flow from the fuel to the cladding. However, at other spots there may be gaps between the two, thus increasing the thermal resistance and producing a bigger temperature drop across the gap in order for the heat being generated in the fuel to be driven out. As a result of these irregularities, more heat might flow out in one direction than in another.

b) Irregularities in cladding thickness. The cladding, due to manufacturing tolerances or installation damage might be thicker or thinner in spots than the average. The thicker portions act as increased resistances and the thinner portions reduce thermal resistance. Further, the bulge in the cladding could conceivably reduce the flow area for the coolant and thus restrict the amount of coolant flowing through the channel. This would mean that the smaller amount of coolant would get hotter than expected as it flowed through the channel thus affecting the cladding surface temperatures and fuel temperatures along the length of the channel.

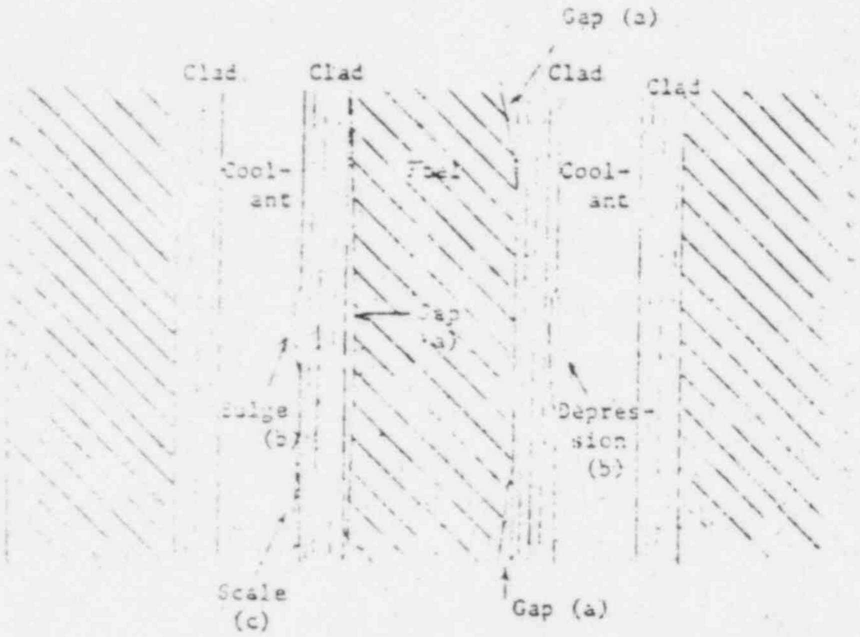


Figure 20

The reduced flow in the channel could also reduce the film coefficient of heat transfer in the channel thus resulting in a higher temperature drop across the film which would result in higher cladding and fuel temperatures.

c) Corrosion and deposits of scale and crud. The deposition of scale and crud can result in additional thermal resistances and increased temperatures of the cladding and the fuel. This could drive the heat out other locations nearby which are not similarly fouled. The added resistance to coolant flow could so reduce the flow that coolant temperatures would rise above expected levels.

In addition to the above, there are two other minor complications that we might mention:

a) Irregularities of coolant flow to individual channels resulting from design manufacturing tolerances or installation errors or due to bowing of fuel pins, etc., thus causing different flow rates than expected in the channels.

b) Non-uniformity of fuel enrichment or of fuel distribution in the fuel elements thus resulting in variations in fission rates and heat generation other than anticipated.

In order to cope with these uncertainties, the design engineer frequently establishes a set of "hot channel" or "hot spot" factors. The hot channel concept is based on the assumption that a reactor having solid fuel elements with coolant passages (or channels) between them will have one channel in which a combination of dimensions or heat generation will produce a cladding temperature or a fuel temperature above that existing at any other place in the reactor core. This temperature can thus represent a design limitation which can be used as a design criterion. One design criterion frequently used is based on the establishment of a maximum permissible heat flux in the hot channel which is a specific fraction (normally about 50 percent) of the heat flux calculated to cause fuel burnout.

The hot channel or hot spot factors relate to some important parameters such as the temperature rise of the coolant as it passes through a coolant channel, or the heat flux at the surface of a fuel element or the temperature drop across the film between the cladding and the coolant.

The hot channel or hot spot factor is in reality, for example, the ratio of the maximum rise in coolant temperature in the hot channel to the temperature rise of the coolant in the average channel. For example, if we refer to Table 12.3 in the Glasstone and Sesonske textbook, Nuclear Reactor Engineering, we find a list of three typical hot channel factors for the Yankee Reactor first core. These three hot channel factors are:

$F(T_o - T_i)$ = Ratio of maximum to average coolant temperature increase

$F(q/A)$ = Ratio of maximum to average heat flux

$F(\Delta T)$ = Ratio of maximum to average film temperature drop

The value of these three typical hot channel factors as a function of several variables is shown in Table 1.

The hot channel factors can thus be utilized much as a factor of safety is used in other engineering design to place limitations or design criterion on the average channel conditions permitted in the reactor core.

The reactor operator should be aware of this information so that he does not alter operating procedures which perhaps have been established so that conditions do not arise in the reactor which would cause temperatures and heat fluxes to exceed design limitations. Otherwise the reactor could be damaged and/or lives of individuals endangered.

Table 1

Hot Channel Factours*

Fuel Rod Characteristics	$F(\Delta T_y - T_i)$	$F(q/A)$	$F(\Delta T)$
Pellet Diameter	1.002	1.003	1.003
Pellet Density	1.027	1.05	1.05
Pellet Enrichment	1.011	1.022	1.022
Rod diameter, pitch and bowing	1.007	-	1.134
Coolant Flow Characteristics			
Flow Distribution in Pelnum	1.07	-	1.07
Flow Distribution Due to Boiling	1.05	-	1.05
Product of Engineering Factors	1.28	1.08	1.37
Nuclear Factors			
Local Power Peaking	1.3	1.3	1.3
Overall Maximum to Average	1.[illegible]	2.9	2.9
Product of Nuclear Factors	2.[illegible]4	3.8	3.8
OVERALL HOT CHANNEL FACTOR	3.[illegible]1	4.1	5.2

*Data obtained from Table 12.3 of Nuclear Reactor Engineering, by Glasstone and Sesonske, D. Van Nostrand Company, Inc., 1967.

8.3 Heat Transfer Coefficients

In the programmed learning section of our discussion, and at other times in our discussions of heat transfer from a surface to a coolant, it was indicated that the coefficient of heat transfer through the coolant film was a function of the velocity of the coolant. Thus, in general, heat transfer coefficients in forced convection are higher than those of natural convection. One might then logically assume that the problem of assuring low film temperature drops and high film heat transfer coefficients is only a matter of pumping the coolant, for example water, through the core at high velocities.

Actually the film coefficient is also a function of numerous other variables. Some of these variables are the thermal conductivity, viscosity and specific heat of the coolant, all of which change as the temperature of the coolant changes. In addition, the film coefficient is also a function of the size and physical shape of the coolant channel.

We will not discuss the relationship between the film coefficient of heat transfer and these variables in greater detail. However, it is expected that you will remember that 'h' increases with increasing velocity in the channel. For example, if the pump were to stop not only would the water in the channels get quite hot and boil but at the same time 'h' would decrease, causing the cladding surface temperatures and the fuel temperatures to rise sharply to drive the heat out through the added resistance of the film. One hears quite frequently about the provisions that are made to assure adequate cooling in the case of a "loss of flow" accident.

Before proceeding, perhaps I should also point out that one cannot just increase the coolant velocity without limit. As discussed within the programmed learning section, the pressure drop and thus pumping costs increase quite rapidly with increased velocities. Also, as discussed in the programmed text, the forces exerted on core components by the resulting pressure differences can cause vibrations and/or structural damage. Efforts to prevent structural deflections or damage can result in increased fabrication costs due to the need for heavier structural components. Therefore, one must eventually trade off the possible advantages of higher coolant velocities with other design and cost considerations.

8.3.1 Boiling Heat Transfer

In our earlier discussions it was indicated that water reactors could be subdivided into two major categories: those in which a significant amount of boiling is permitted — the BWR, and those in which boiling is, in general, suppressed — the PWR. It was also mentioned that in the more recent PWR's some nucleate boiling was permitted in an effort to increase average water outlet conditions from the reactor.

Let us now qualitatively discuss the problem of removing heat from the surface of a fuel element upon which boiling is taking place. We have already discussed the fact that a microscopically thin and relatively stagnate film tends to insulate the heat transfer surface from the coolant, resulting in a temperature difference between the surface and the coolant if heat is being transferred.

For our discussion, let us consider that initially the surface of the fuel element and the coolant are essentially at the same temperature and that essentially no heat is being transferred. However, as the power level of the reactor is increased so that heat is produced in the fuel element, the temperatures in the fuel element will increase to drive the heat from the element and the temperature drop across the film will increase correspondingly. If we assume that during this stage of the process the film coefficient of heat transfer remains relatively constant, we know from our previous discussion of conductive and convective heat transfer that the temperature drop across the film will increase proportionately with the amount of heat (or the heat flux) that is being transferred through the film. (See Region I of Figure 21 where this effect is depicted. Note that the plot is on a log-log scale).

However, if we continue to raise the power level, the point will eventually be reached where the temperature drop through the film is such that surface temperature of the fuel cladding will reach the saturation temperature of the water, although the bulk of the coolant itself might be appreciably below the saturation temperature. As this condition is reached, tiny steam bubbles will form on the cladding surface nucleating at imperfections of the surface or around small gas bubbles entrapped on the surface, etc. These bubbles will grow to the point where because of their buoyancy they will break away from the surface, go into the coolant and collapse. As more heat is added, more and more of this <u>nucleate boiling</u> will occur. The formation of these tiny bubbles as well as their motion as they grow and break away causes a kind of scrubbing or agitation of the film. This breaks up the film and results in improved heat transfer. This condition can be seen in Region II of Figure 21 where the heat flux is plotted versus temperature difference between the surface and the fluid.

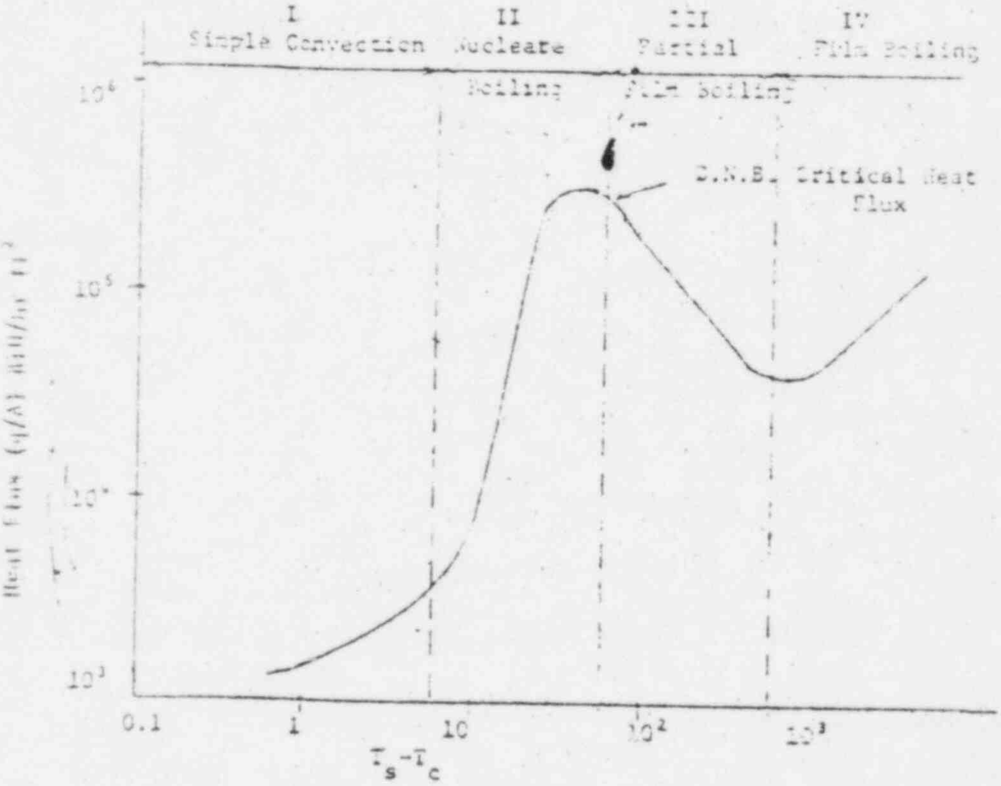


Figure 21

The rapid increases in heat flux with relatively small increases in temperature drop across the film continues throughout a great portion of Region II as the extent of the nucleate boiling increases. However, as the power level is increased and the heat flux thus increases, eventually one reaches the point that the bubble formation on the surface of the cladding is so dense that the bubbles begin to coalesce and form a continuous film of vapor over sections of the surface. This film of steam vapor serves as an insulating blanket. As a result it begins to require greater film temperature differences to increase the heat flux a small amount. This effect can be seen near the right-hand side of Region II and the left-hand side of Region III of Figure 21.

Initially, this film blanketing is unstable: first unstable forming on one section, then breaking away and perhaps forming on a nearby section. However, if the amount of heat attempting to escape from the element continues to increase, eventually the film blanket covers the surface and heat transfer through the blanket of steam must take place by conduction and radiation neither of which is very efficient at these temperatures. However, as the fuel element attempts to accommodate more heat, the temperatures soar upward, both through the cladding and through the fuel element. Eventually, the temperatures become so great that heat transfer through the film by conduction and radiation becomes more effective and the curve begins to rise again. <u>However</u>, the temperatures of cladding and the fuel have become so high by this time that there is the possibility that the melting points have been reached, leading to <u>burn-out</u> of the fuel element. Burn-out may result in the rupture of the cladding, thus enabling the release of large quantities of radioactive solids and gases into the coolant.

As a result, the point of maximum heat flux in the nucleate boiling region is a point of considerable interest to the reactor designer and operator. To exceed this point can readily lead to burn-out. (This does not say that burn-out could not occur sooner if the particular cladding in use had a low enough melting point that this temperature was reached before the maximum point was attained.) This maximum point is referred to by a number of different names by reactor personnel. Some of these are:

<u>Departure from Nucleate Boiling</u> (D.N.B.)
<u>Critical Heat Flux</u>
<u>Maximum Heat Flux</u>
<u>Burn-out Heat Flux</u> (not that burn-out will necessarily occur at that point, but that exceeding the point would probably lead to burn-out)

If the fuel element is undergoing nucleate boiling near the D.N.B. point, a slight increase in heat will cause a sudden change to film boiling, which usually results in burn-out. Thus one does not wish to operate the reactor under conditions which would cause the heat flux to approach too close to the D.N.B. point. Normally, therefore, a factor of safety equal to the ratio of the D.N.B. heat flux to the design heat flux is usually used in design and in operation. This factor of safety is commonly called the <u>burn-out ratio</u>.

Thus:

$$\underline{\text{Burn-out Ratio}} = \frac{\text{D.N.B. Heat Flux}}{\text{Design Heat Flux}}$$

Generally in the technical specifications for a reactor the minimum burn-out ratio is specified at a given power level (frequently at the level where the reactor will be scrammed automatically -- such as 125% of full power) and is normally in the reange of 1.5 to 3.0.

In a number of power reactors a large number of in-core instruments (such as small fission chambers) are distributed throughout the core and calibrated to read percent of maximum permissible heat flux so that the operator can assure himself that these values are not being exceeded during operation.

INSTRUMENTATION AND CONTROL

9.0 Introduction

The first element in the reactor instrumentation system and the one that is probably most unique to a reactor plant system is the radiation detector. A device which can be exposed to reactor radiation levels and produce a signal proportional to them which can be used to activate indicating, recording, and control devices is required. Before proceeding into a discussion of specific types of detectors and the manner in which they work, it is appropriate to review briefly some of the basic types of nuclear radiation which must be sensed and the general manner in which they may be detected.

9.1 Radiation

9.1.1 The Alpha Particle

The alpha particle is a fast moving stable group of two protons and two neutrons, which results when certain types of radioactive nuclei decay. It is essentially the same as the nuclei of a helium atom, however, it differs in origin. As an alpha particle released in a radioactive decay event slows down in passing through matter, it will ultimately acquire two orbital electrons and become a normal helium atom. The two protons and two neutrons which comprise the alpha particle each have a unit mass of 1. As a result, the total relative mass of the alpha particle is 4, or 4 times that of the hydrogen atom. Since the alpha particle consists of two neutrons and two protons, the neutrons having zero charge and the protons having a unit charge of +1, the total charge of the alpha particle is a +2. In the instrumentation normally used for operation of a nuclear plant, there is ordinarily no detection of the alpha particle. However, it might be noted that this particle is produced by normal decay of the fissionable material which serves as the raw material for energy production in a nuclear plant. When such material is lost, the necessary search may well be conducted with an alpha detector.

9.1.2 The Beta Particle

A beta particle is actually an electron; however, the term is generally used to refer to an electron which has been ejected, with considerable kinetic energy, in the decay of a radioactive nucleus. The beta particle has an electrical charge of -1 or +1 and a mass which is negligible with respect to a neutron or alpha particle.

The detection of the beta particle is not ordinarily utilized for operational purposes in a reactor instrumentation system. However, there is one circumstance in which beta detection can become of considerable importance to reactor operational personnel. If a misfortune occurs, and the fission products normally contained within the fuel element cladding are released into the reactor primary system, or perhaps from it, a substantial portion of the resulting contamination and radiation levels will involve beta radiation.

One practical problem which has arisen in the process of cleanup from such a release occurs when radiation levels are so high that personnel working time in an area is limited. This working time commonly is controlled by making periodic observations of the indication on a self reading dosimeter. However, this reading does not reflect the beta radiation exposure since beta radiation does not penetrate the metal walls of the dosimeter. As a result, when working time is limited in this fashion, the results from the development of the film badge insert, which may be received a week later, may indicate an undesirable exposure. To avoid a problem, the ratio of beta to gamma radiation can be determined, with the best instrumentation available, and used to produce a safety factor which can be applied to the reading observed on a self reading dosimeter.

9.1.3 Gamma Radiation

Gamma radiation is electromagnetic in nature, and is therefore similar to light and radio waves. However, it is of much higher frequency (shorter wave length) and indeed is at the upper extreme of the electromagnetic spectrum.

A substantial fraction of the energy released when the fission event occurs appears in the form of gamma radiation. The intensity of this radiation is thus proportional to the rate at which fission is occurring, and hence reactor power level. Therefore, it would seem possible to use a measurement of gamma radiation level in or near the reactor core as a measure of reactor power level, and indeed it can be done. However, the measurement is complicated by the fact in addition to the gamma radiation released instantaneously, as a result of fission, many of the resulting fission products themselves decay, at various rates, with the release of sizable amounts of gamma radiation. Now this component of the overall gamma field is not proportional to the instantaneous fission rate, and reactor power level, but instead is a complicated function of previous operating history of the core. Since the overall gamma field has this component which is not directly related to the current power level of the reactor, its value as an indication of reactor power level is severely handicapped.

Regardless of the foregoing, gamma radiation level can be, and is, used as a measure of reactor power level. Its use for this purpose, however, is limited to the region between, perhaps 10 and 100% of full power for the reactor in question. In this region, the gamma level resulting from decay of previously created fission products tends to be relatively small. An example of this is the "% Power Channel" of the Penn State TRIGA Reactor. However, even with the reactor operating at its full rated power level, there is an uncertainity of perhaps 5% in this channel. In a power generating reactor, which must be operated at a level as close as possible to some rated level for economic purposes, and which must, at all cost, be maintained below that level for safety reasons, such an uncertainty would be undesirable. While the gamma radiation level produced by a reactor is not generally measured for operational purposes, its existence is still of great importance to the reactor operator. This is due to the fact that the neutron sensitive systems which are used for reactor power level determination are inevitably sensitive to gamma radiation. Therefore, as will be seen, rather elaborate schemes are used to screen out the gamma signals produced in such systems.

Gamma radiation becomes of importance to reactor operating personnel in another way which might be mentioned, although it is somewhat secondary to the reactor operating problem itself. Much of the radioactivity produced in materials exposed to neutron irradiation in or near the reactor core is gamma activity. As a result, the reactor operator frequently finds himself concerned with the shielding and handling of such material.

9.1.4 The Neutron

The neutron is one of the basic ~~building~~ building blocks of the atomic nucleus. It has a unit mass of 1 and an electrical charge of 0. The neutron is perhaps the particle of most interest to the reactor operator, since its absorption by the nuclei of certain of the heavy elements and the resulting fission of the element is the process which releases the energy which a power reactor is designed to produce. As a result, neutron detection is a matter of paramount concern to the reactor operator.

9.1.5 Other Types of Radiation

In the reference material associated with this program, one can find a variety of other particles and suspected particles described but these are not sufficiently obvious in a reactor environment to be of interest to the operator.

9.2 Ionization

Consider a fast moving charged particle such as an alpha or beta particle moving through some material. It will occasionally approach closely enough to an orbital electron of one of the atoms or molecules of the material to create forces which will cause the electron to be dislodged from the atom or molecule. What is left of the atom, after the electron has been stripped from it, is referred to as a positively charged ion since it has lost the negative charge due to one electron. In addition, there is now a free electron moving about the system. These two particles, the remains of the atom with one electron removed, and the free electron moving about the system are generally referred to as an ion pair. The ability of a charged particle to produce this ionization is expressed in terms of a number called specific ionization which is specifically the number of ion pairs formed per centimeter of path traveled in a given material. One can visualize that specific ionization will tend to increase with the charge of the particle since it will thus be able to exert more force on the orbital electron. Also, assuming particles with the same kinetic energy, those which have higher mass will move more slowly and will spend more time in the vicinity of a given orbital electron, and the probability of electron removal will be enhanced. For these reasons, the specific ionization produced by an alpha particle is considerably greater than that caused by a beta particle. The specific ionization due to alphas may be 50,000 to 100,000 ion pairs per centimeter of travel, whereas a beta particle of similar energy may produce only 50 to 300 ion pairs per centimeter.

Since the neutron possesses no charge, it is not capable of producing ionization in the manner of beta and alpha particles. However, fortunately for the reactor operator who must detect neutron level, it is not unusual for neutrons to produce ionization by secondary means. Various materials have a strong tendency to absorb or capture neutrons in their nuclei, which leaves the nuclei in an "excited state," or a condition of excess energy. It commonly releases this energy by emitting a charged particle, which produces ionization.

Gamma photons resemble neutrons in that they possess no charge and do not produce ionization in the manner described for alpha and beta particles. However, gamma radiation can interact with matter to produce ions in any of three processes: the photoelectric effect, Compton scattering, and pair production. Without attempting further explanation of these processes, it will be stated that, as a result, gamma radiation is strongly ionizing.

9.3 Electrical Fields

The production of ions in material by various types of radiation has been discussed. However, to use this phenomena to detect and measure radiation, it is necessary to create with it an electrical signal. Consider Figure 1, which shows two electrodes enclosed in a gas-filled container. If a voltage is connected across the two electrodes as shown, a potential gradient will be created between them. Any ions between the two electrodes will be exposed to a force towards one electrode or the other. The positive ions will be attracted to the negative electrode and the negative ions to the positive electrode. When a positive ion reaches the negative electrode, it will absorb an electron. When a negative ion reaches the positive electrode, it will give up an electron. In this process an electron will have been transferred between the two electrodes, and effectively through the external circuit. If sufficient ions are available, a measurable current can be produced through the external circuit. If the supply of ions is continuously replaced by ionization due to radiation entering the container of gas, the size of the current produced will be related to the radiation level.

9.4 Potential Effects

The very basic system shown in Figure 1 is, while crude and simple, nevertheless representative of a large percentage of practical radiation

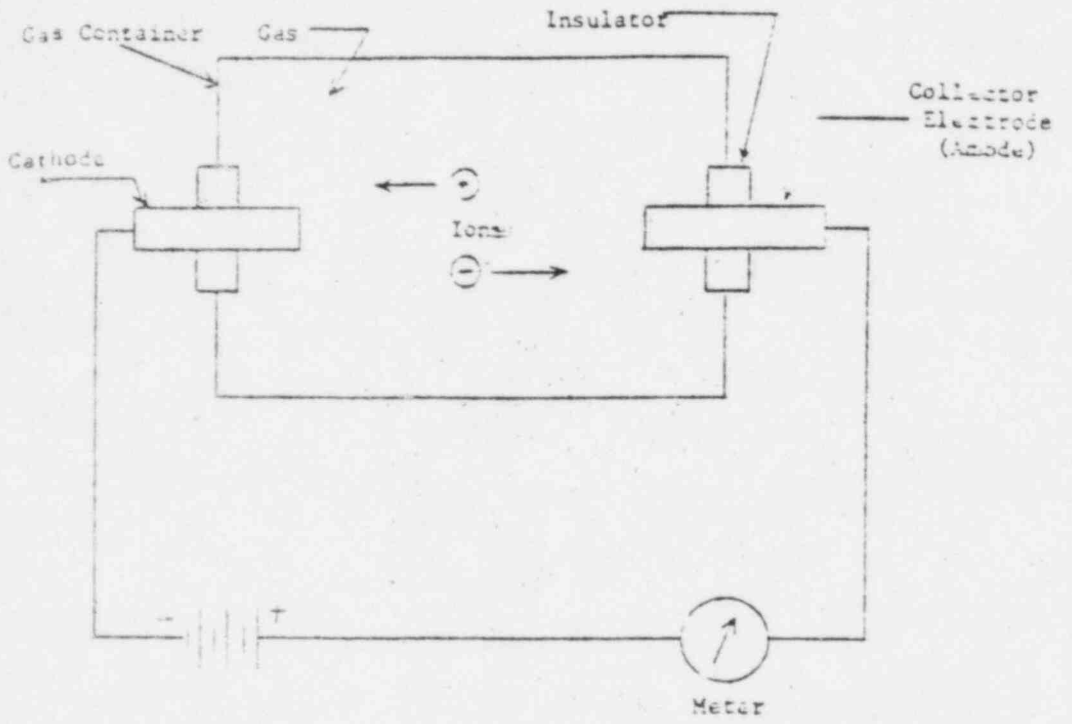


Figure 1

Basic Radiation Detection System

detectors in use. As might be expected, the operating characteristics of such a device vary considerably with the voltage applied. Figure 2 is a plot of relative ion collection vs voltage, as the voltage is varied from 0 to several thousand volts. It is customary to divide this characteristic curve into the regions shown in Figure 2 and consider each region in considerable detail.

In region I, the voltage is too low to impart the necessary separation velocity to all of the ions produced and since they do not move quickly far enough apart, they have the possibility of recombining. For this reason, this is called the recombination region. This is the lowest output region of the curve. It is not generally used in any practical radiation detector. Increasing the voltage further, region III is encountered which is referred to as the ionization region. In this region, the potential applied is sufficient to prevent any recombination of ions after formation and they all travel to the electrodes. Since the number of ion pairs formed is a function of the intensity of the radiation to which a device is exposed, the output current will be proportional to the level of the radiation at the point where the detector is located. Another important point concerning this region which is of significance to reactor operations is the fact that there is little variation in charge collected (or current) with variations in the supply voltage. Therefore, if the voltage is established at some point

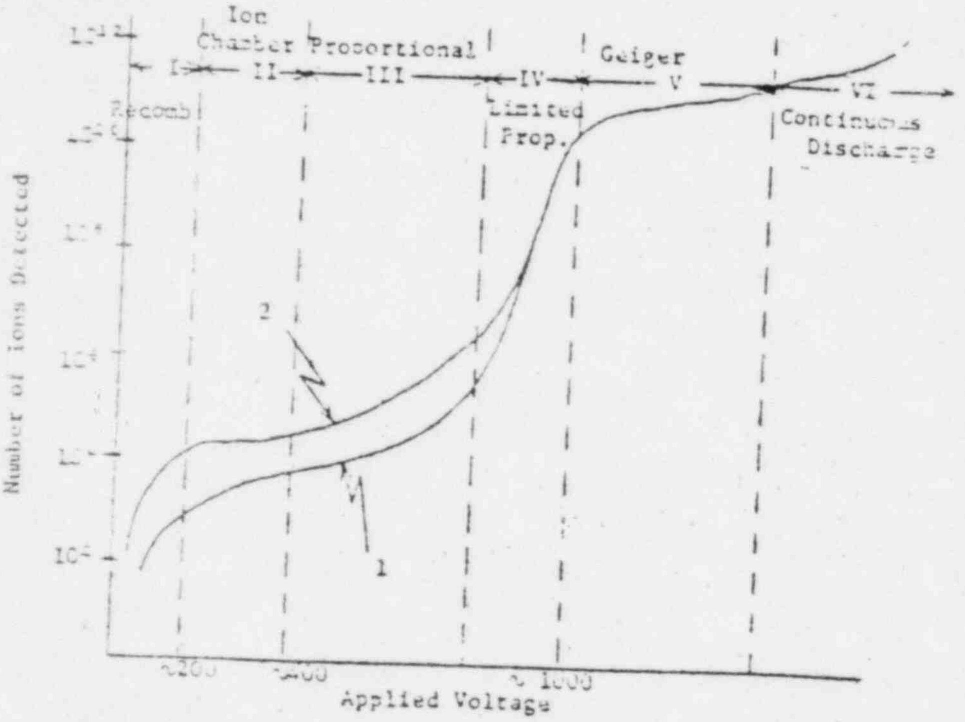


Figure 2

Number of Ions Detected vs Voltage Applied

On A Gas Filled Detector

(within this region) minor variations in the voltage have relatively little effect on system calibration. Devices which operate in this region are of considerable importance in reactor instrumentation.

Region III is called the proportional region where it can be observed that the ion collection gradually increases as the voltage is increased. The voltage in this region is large enough to cause a high potential gradient between the two electrodes. This potential gradient causes the electrons released by ionization to obtain high velocities in the gas which can cause additional ionization by collision with the other atoms of the gas. The first ionization produced releases an electron which begins moving towards the positive electrode at such velocity that it can remove other orbital electrons. As a result of these multiple collisions, soon a very large number of electrons are rushing toward the anode (positive electrode). This effect is called gas amplification and is also sometimes referred to as the Townshend Avalanch. The amplification in this region, meaning the number of free electrons eventually produced by one initial ionizing event, may vary from 2 to as high as 10^6.

As the voltage is further increased, Region IV is entered which is known as the limited proportional region. The reason for this can be seen from the curve: the relationship between ion collection and particle energy is beginning to disappear. In this region, the increased field strength caused by the increase of voltage across the detector causes a further increase in electron velocities. These electrons are capable of releasing electrons by additional mechanisms photoelectrically from the cathode and also by photo ionizing the gas in the detector. These processes tend to contribute to the electron avalanche and tend to make the curve swing upward as the voltage increases.

In Region V, referred to as the Geiger region, the field strength produced by the potential difference has increased to a point such that once ionization has occurred, the avalanche continues to build up until it is limited by the build up of positive ions surrounding the central anode. The pulse size, in this case, is independent of the type of radiation causing the ionization. The voltage at the beginning of this region is called the <u>geiger threshold</u>. The advantage of operating in this region is the large output obtained, which is on the order of volts. This large output allows the production of an adequate current for the operation of an indicating device, such as a meter or earphones, with a minimum amount of amplifying equipment. For this reason, detectors operating in the Geiger region are often used in lightweight, rugged portable radiation detecting systems for both "health physics" type surveying, and uranium prospecting.

In Region VI, the acceleration of ions due to the impressed voltage is so great that once ionization has been produced and the discharge begins, it continues even though no further ionization is produced from external sources. A detector operating in this mode is said to have broken down.

9.5 Detector Output Systems

Given a device which will produce electric current as a result of exposure to radiation, there are two basic techniques for handling the signal.

One common scheme is to design the system so that the current produced by an ionizing event in the chamber decreases to zero before another ionization takes place. With an arrangement of this type, the output is actually a series of pulses whose frequency is determined by the rate at which radiation is arriving at the detector. To use this type of output the pulses are counted by an electronic counter after suitable amplification and shaping.

The second type of system is designed such that the speed of transfer of charge through the external circuit is slow with respect to its production so that in effect an average current is produced which is proportional to the radiation reaching the detector. With this type of system indication is by means of a current measuring device, which in some cases may be a simple d'Arsonval type micro or even milliameter. However, in most applications the current is sufficiently small that electronic amplification is necessary.

9.6 Neutron Sensitivity

Previously the process where ionization takes place in a radiation detector has been described, while noting that the neutron does not directly produce ionization. It has been stated that neutrons can produce ionization by secondary means, as a result of neutron capture by a nucleus which ultimately releases a charged particle, causing ionization. A nuclear reaction of this type which is used extensively in practical neutron detection involves the use of the element boron, specifically its B-10 isotope. The reaction used is:

$$_0n^1 + {}_5B^{10} \longrightarrow {}_3Li^7 + {}_2He^4 + 2.78 \text{ Mev}$$

The lithium recoil nucleus and the alpha particle (helium nucleus) tend to share the 2.78 Mev energy release. Both of these particles produce ionization as they move through the chamber in which the B-10 was originally incorporated.

Various schemes have been employed to introduce B-10 into a radiation detector. To achieve the desired result, it is necessary that the B-10 be incorporated in some fashion that will allow the charged particles produced as a result of the interaction with a neutron to enter the sensitive, gas filled volume of the chamber, and produce ionization. Since the range of the particles produced is rather limited in solid matter, the choices available are either to fill the chamber with a gaseous compound of boron, or to use a thin layer of boron on the interior of the chamber. In general, the use of a gaseous boron compound, such as BF_3, gives greater sensitivity; the use of solid B-10 on the walls of a chamber gives a more stable instrument with greater life.

Another very powerful technique for producing a neutron sensitive detector is to actually include fissionable material in the construction of the chamber, usually U-235. For its operation, the chamber depends upon the ionization produced by the fission fragments which are produced when a U-235 nuclei is fissioned by an absorbed neutron. The large amount of ionization produced by this event, relative to that produced by gamma radiation in the chamber, makes discrimination against the gamma radiation relatively simple.

9.7 Specific Neutron Detectors

From the material presented so far, it should be apparent that there is quite a variety of choices available to the designer of a neutron detector for a specific application. The various voltage-collection regions, the several means of producing an output signal, and the alternative methods for producing neutron sensitivity, all combine to give a large number of possible combinations. However, as might be expected, through the years of design and use of neutron detectors, a relatively small number of the possible combinations has been found to be feasible to build and desirable to use. The next objective of this program will be to describe the type of neutron detection systems which may be encountered in a reactor, particularly a power reactor. Specification sheets for a typical detector of each class discussed are contained in Appendix I of this section.

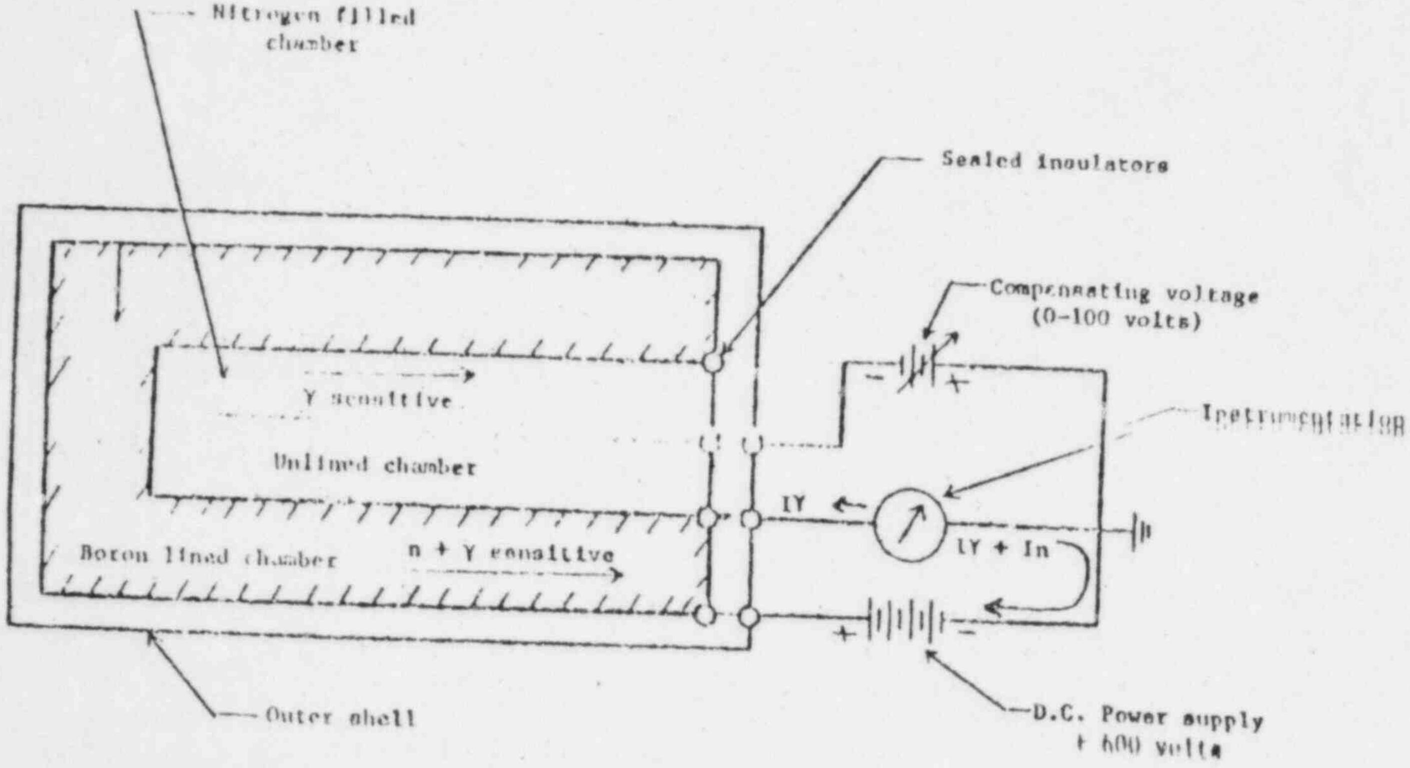
9.7.1 Uncompensated Ion Chamber

An uncompensated ion chamber does not differ much in basic construction and operation from the hypothetical detector considered earlier. Very basically it consists of two electrodes enclosed in a chamber or can which is filled with some gaseous material. It may be designed concentrically, with the can serving as one electrode. The two electrodes are maintained at a substantial difference in electrical potential or voltage and if the chamber is subjected to ionizing radiation ion pairs are formed in the filling gas which tend to migrate to the appropriate electrodes. This is used to produce a current in the external circuit which is proportional to the magnitude of the radiation level. However, to use this device to record the neutron level adjacent to the reactor, a problem arises from the fact that neutrons do not produce ionization to any substantial extent in the filling gas. As previously suggested, the solution is to introduce into the chamber some material which will absorb neutrons and in the process produce radiation which can cause ionization. In this particular type of chamber, the material generally used is the Boron-10 isotope of boron. As previously described, Boron-10 absorbs neutrons coming into the chamber with the conversion of the Boron-10 to Lithium-7 and the release of an alpha particle which produces ionization in the filling gas. Such a chamber is commonly positioned adjacent to the reactor core, usually outside the reactor pressure vessel, and is used to measure neutron flux in what is referred to as the power range, say from 10% to 100% of full power. A problem develops if such a chamber is used for indication of the neutron level in the reactor core at lower power levels. This problem results from the fact that the chamber is sensitive to gamma radiation as well as neutron radiation. Note in Appendix I, for a Westinghouse WL 8075 chamber, that the gamma sensitivity is on the order of 5×10^{-11} amperes per roentgen per hour. Consider what this gamma sensitivity might mean in several typical operating situations. The detector may be located at a point outside the reactor pressure vessel with a neutron flux at full power of 2.5×10^{10} nv. Based on the sensitivity given on the specification sheet, the output on the chamber in this situation can be seen to be 1×10^{-3} amps or 1 milliamp. Now the reactor is producing gamma radiation as well as neutrons and if the neutron flux at the detector is 2.5×10^{10} as postulated previously, the intensity of the gamma radiation might be on the order of 100,000 roentgens per hour. Based on previously mentioned gamma sensitivity of 5×10^{-11} amps per roentgen per hour, an output current of 5×10^{-6} or 5 microamperes is produced due to the gamma radiation. This is less than 1% of the current produced by the neutron flux so it can be concluded that the chamber output in this situation is essentially representative of the neutron level, which is the objective. If, after a period of power operation under these conditions the control rods drop into the reactor core and cause it to shut down, the neutron level in the reactor will exhibit almost a step negative change, lowering the neutron level down to perhaps 10% of what it was during reactor operation. Subsequent to this sharp drop, the neutron level will continue to decrease on the characteristic 80 second period and with the end of the neutron chain reaction, production of gammas from the fission process will also cease. However, there will be a strong residual gamma radiation level produced by the high concentration of fission products which exist in the core following reactor operation and as the neutron level drops, a point will be reached at which a major portion of the signal produced by the chamber is due to the gamma radiation from the residual fission products rather than from the neutron level. In this circumstance, the chamber output becomes very unrepresentative of the neutron level. If an operator were to attempt to use this chamber output as a guide in restarting the reactor after such a shutdown, he might encounter difficulty. The change in neutron multiplication and neutron level produced by withdrawal of control rods can be completely overshadowed by the component of the signal due to the residual gamma radiation. The reactor operator, seeing no response in his power level instrumentation might continue to withdraw additional increments of control rod. The reactor, of course, will respond to this increase in reactivity and the neutron level will increase. However, it may not become apparent to the reactor operator that the neutron level is increasing until the absolute neutron level reaches a point where it is appreciable relative to the gamma-produced signal. Depending on how rapidly the rods have been withdrawn, at this point it is quite possible that sufficient reactivity may have been added to the reactor core to place the reactor on an extremely short period, possibly producing a period scram. If one can anticipate this situation, it is possible operationally and procedurally to avoid the problem by prescribing a pull and wait procedure in which the rods are withdrawn a rather limited amount and an ample amount of time is allowed between rod movement to allow any neutron level increase produced by rod withdrawal to climb out of the background produced by the gamma radiation. However, it would be much more desirable to utilize a chamber in which this sensitivity to gamma radiation is considerably reduced. Thus a need for a gamma corrected or compensated chamber arises. Before leaving the uncompensated ion chamber, it should be stated that it is an extremely useful and simple device and is commonly used to serve as a power range detector in a power reactor system. However, for the region below this level and above that covered by the source or startup range instrumentation, a "compensated" ion chamber is generally provided.

9.7.2 Compensated Ion Chamber

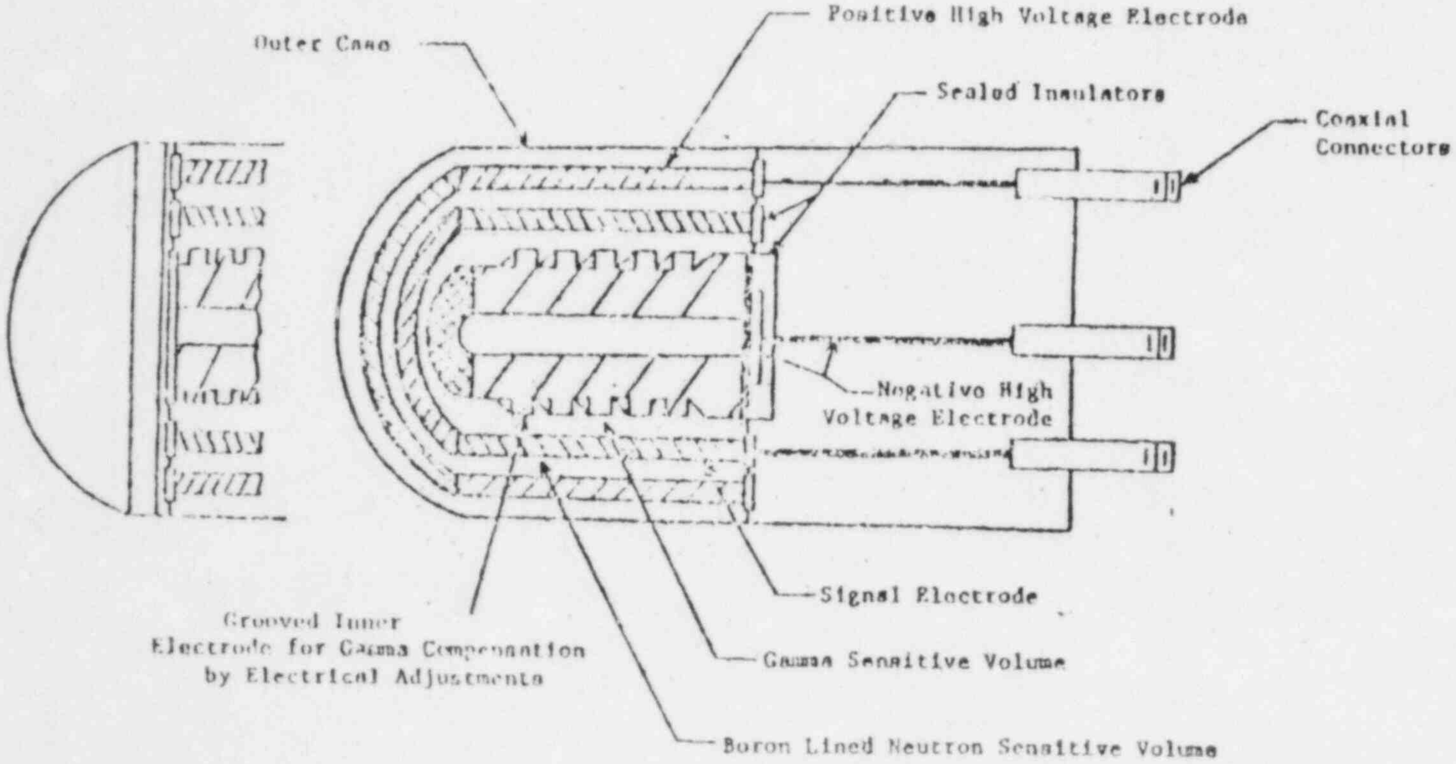
The compensated ion chamber is actually two chambers in one, or two sections in one chamber. One section is sensitive to gammas. The second section is made sensitive to both neutrons and gammas by applying a coating of Boron-10. The outputs of these two chambers are connected in opposition, electrically speaking, as shown in Figure 3. Mechanically, the two chambers are generally assembled concentrically as shown in Figure 4. Since the outputs of the two chamber sections are connected in opposition electrically, one of the signals is subtracted from the other. If the output signal coming from one section of the chamber is proportional to a neutron plus gamma signal and the signal coming from the other section of the chamber is proportional to gamma radiation alone, it can be seen from the expression given below that when the sections are electrically connected so they subtract from one another, a signal is obtained that is proportional to the neutron radiation alone.

$$(n + \gamma) - \gamma = n$$



AN ELECTRICALLY ADJUSTABLE GAMMA COMPENSATED ION CHAMBER

Figure 3



AN ELECTRICALLY ADJUSTABLE GAMMA COMPENSATED ION CHAMBER
Figure 4

This is the manner in which the device works and it works surprisingly well but there is one important qualification in the description so far given and that is the assumption that gamma sensitivities can be made equal for both sections of the chamber. Of course, the manufacturer determines this to some extent by selecting the sizes of the two sections of the chamber, but he cannot manage to make it completely constant under all conditions of reactor operating history for all situations of gamma radiation gradients across the chamber. For this reason, the chambers must always be equipped with some scheme for adjusting the constants associated with one of the two sections. One obvious way to do this is to make the volume of at least one of the chambers adjustable. This was the scheme adopted in the early "Oak Ridge" type chamber, which has been used extensively in reactors such as the BSF (Bulk Shielding Facility), the Materials Testing Reactor at the Idaho testing site and others. This type of chamber was used in the original Penn State research reactor which was started in 1955. However, this type of chamber has one major disadvantage which is the fact that to make the adjustment one must insert a screw driver into a slot in the chamber assembly and rotate it to produce a change in volume of the chamber. In the case of the Penn State reactor, the screw driver turned out to be 20 or 25 feet long. Another scheme attempted to solve this problem was to build into the chamber an electrical motor which could be used to drive the necessary threaded arrangement to produce a change in the volume of one of the chambers. However, the reliability of the electric motor in the radiation fields to which it was exposed was not outstanding. To cope with this problem, Westinghouse personnel came up with a rather ingenious scheme shown in Figure 4. Note that the negative electrode which comprises one side of the inner sensitive volume is grooved or serrated. As the voltage is changed on this particular electrode, the electrical field surrounding it tends to change in such fashion as to, if you will, fill up the grooves. This produces an effective change in chamber volume without any mechanical operations. Thus, it is possible to produce a compensating type of adjustment without actually requiring any mechanical adjustment of chamber size. These compensating voltage adjustments are used to correct for variations in manufacturing tolerances and variations in positioning of the chamber adjacent to the reactor.

In actual practice, additional problems develop, even if the chamber is installed and adjusted for a particular set of conditions with one particular ratio of neutron radiation to gamma radiation. The adequacy of the compensating adjustment changes with reactor operating or non-operating history. One reason for this can be easily given since the chambers are assembled concentrically. This being the case, the outer chamber tends to shield the inner chamber from gamma radiation. As long as the gamma spectrum remains the same, the proper adjustment of the compensating voltage can cope with the situation. However, following shutdown the situation changes. The gamma spectrum changes as the short half-life fission products decay with material penetrating powers changing substantially with gamma spectrum. As a result, during transient operating conditions, it is sometimes necessary to make careful adjustments of the compensation. There is a certain amount of art as well as science involved in doing this and it should be done by someone who has a substantial amount of experience in this operation. However, one general guide line which might be suggested and which the author has found to be of use is that, do not ever allow the signal as displayed on the output of instruments associated with the compensated ion chamber to cause the indicating instrument to peg downscale. As long as it is somewhat upscale, it will provide some evidence of neutron level and the changes which might be taking place in it.

One way in which to learn more about the problems of chamber compensation is to consider the manner in which improper compensation, either over-compensation or under-compensation, may manifest itself. One common situation in which such evidence is available is when the reactor is shutdown following a period of operation at high power level. If the chamber is reasonably well compensated, the neutron level as displayed on a channel actuated by a compensated ion chamber will show initially the sudden sharp drop in the neutron level changing into the 80 second delayed neutron shutdown period. This shutdown curve should be evident well toward the lower end of the range of the channel in question. If the chamber is under-compensated, one will observe a drop of several decades after which the indication tends to level off and show no further drop. This lack of further drop is an indication of the signal produced by gamma decay of the fission product inventory in the core and indicates that the channel is responding to this more than to neutrons. If the chamber is over-compensated, as the reactor power level drops, it may tend to suddenly drop towards zero, deviating in this fashion from the straight line ordinarily displayed on the log-arithmic type instrument. This indicates that the channel is over-compensated and is responding too strongly to the gamma radiation produced by the fission product decay.

Other possible operational manifestations of chamber over or under compensation can be described. Assume the reactor is operating at full power and the lead supplying the compensating voltage breaks. What effect does the operator see on the indication for that particular channel? This question has been observed on the AEC reactor operator's examinations. If the reactor is operating at a substantial power level, above its normal shutdown condition, and the ratio of neutron sensitivity to gamma sensitivity is typical, one would expect to see nothing or very little at any rate.

Another situation can be postulated which is perhaps more informative. Assume that the reactor is operating at some lower power level (perhaps 10 watts to 1000 watts, depending on the residual level common to the reactor being considered), but at any rate assume that the power level is not far above the normal shutdown level. Also, assume that there is a substantial amount of gamma radiation being released by residual fission products. Assuming these conditions, what happens if the lead to the compensating section of the chamber opens? If the chamber is under-compensated, say radically under-compensated, one will see nothing. There will be no change in the indication. If the chamber is over-compensated, implying that the gamma or residual radiation from the core is producing an excessive chamber output, due to the gamma sensitivity of the chamber, one will tend to see a large increase in the indication coming from this particular chamber. If the chamber is over-compensated, the actual output will tend to be depressed relative to the true neutron level in the core. This being the case, if the compensating voltage disappears, one will see a step increase in indication. Such an increase might have sufficient size and speed to produce a period trip in reactor safety systems. If the chamber is properly and adequately compensated, one will still tend to see a positive increase in indication from the channel. Since the cancellation effect of the compensating channel will disappear, the resulting signal will be due to the neutron signal plus the gamma signal from the one section of the chamber.

Compensated ion chambers are ordinarily used to feed the channels which provide indication of reactor neutron level between the low or source range and the range of power operation. In this particular range, it is common to develop a signal which will cause dropping of the rods and reactor shutdown in case the rate of increase of reactor power level becomes greater than desired. This is one of the reasons that considerable attention is being devoted to this problem. The need to understand the reactor instrumentation channel which is being fed by a compensated ion chamber is important at all times. However, it is perhaps least of a problem under the conditions when a reactor is being started or restarted from a situation of low residual gamma radiation such as the first startup or after a refueling. Circumstances under which the proper understanding of the operation of the compensated ion channel become most interesting and most important are those following a sudden shutdown of the reactor. This is a situation that can conceivably be of considerable interest in utility operations since returning a plant to a line following an unscheduled shutdown could be of major importance to the entire system. Under these circumstances, compensation can be a problem. The gamma flux level and the gamma flux spectrum are changing at a rate which may require continual adjustments of compensation. However, it is important to make them correctly.

9.7.3 BF_3 Detectors

There is some difficulty in producing ion chambers for power reactor use which possess the sensitivity required to detect the levels of neutron flux existing when the reactor is shutdown. However, it is possible to construct a boron trifluoride gas filled counter for operation in the proportional region which has adequate sensitivity to thermal neutrons for use in the source or shutdown region. It can be useful in fluxes lower than 10^{-1} nv. A typical BF_3 counter may have a sensitivity of 20 counts per second per nv, although it can vary from approximately 5 counts per second per nv to perhaps [illegible] counts per second per nv. The operating range of such a detector may typically be from one count/sec to 10^5 counts/sec, corresponding to a neutron flux of 10×10^{-2} to 10×10^3. This detector, produces pulses as a result of ionization in the detector volume. This ionization can be produced by gamma rays or, due to the fact that the chamber has filled with boron trifluoride, it can be produced by the alpha particles resulting from absorption of neutrons in the boron. Therefore, a situation exists which is somewhat analogous to that in the ion chamber in which response from both gamma rays and neutrons is obtained whereas the objective is to measure neutrons and neutron level only. In this type of system, as in the ion chamber system, a scheme is available which allows discrimination against the gamma ray signal. In the case of the BF_3 proportional counter, this scheme is based on the fact that the gamma ray has a much lower specific ionization than the alpha particle from the neutron alpha reaction and therefore much smaller pulses will be produced. By transmitting these pulses through an appropriate electronic circuit, the gamma ray induced pulses can be screened out. This is accomplished by discriminating against all pulses which are below a certain height. The very simplest form of a discriminator circuit may be developed by using a bias diode. This is illustrated in Figure 5.

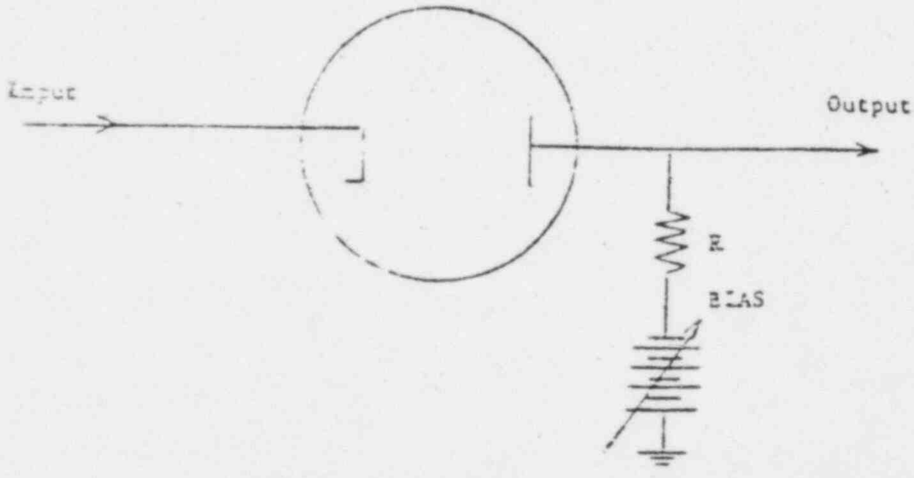


Figure 5

The signal input is a negative pulse applied to the cathode of the diode. When the pulse height is high enough to drive the cathode below the potential of the biased plate, a current flows through the diode and the resistor R, thus producing an output pulse. In the case of the output from a BF_3 proportional counter, the bias can be adjusted so that the input voltage produced by a gamma ray signal from the chamber is not sufficient to effect the output but that produced by the neutron-alpha reaction is high enough to cause the diode to conduct and produce a pulse across the output load resistor. This, of course, is only the simplest form of discriminator circuit and those in practical use are generally considerably more complicated and more sophisticated.

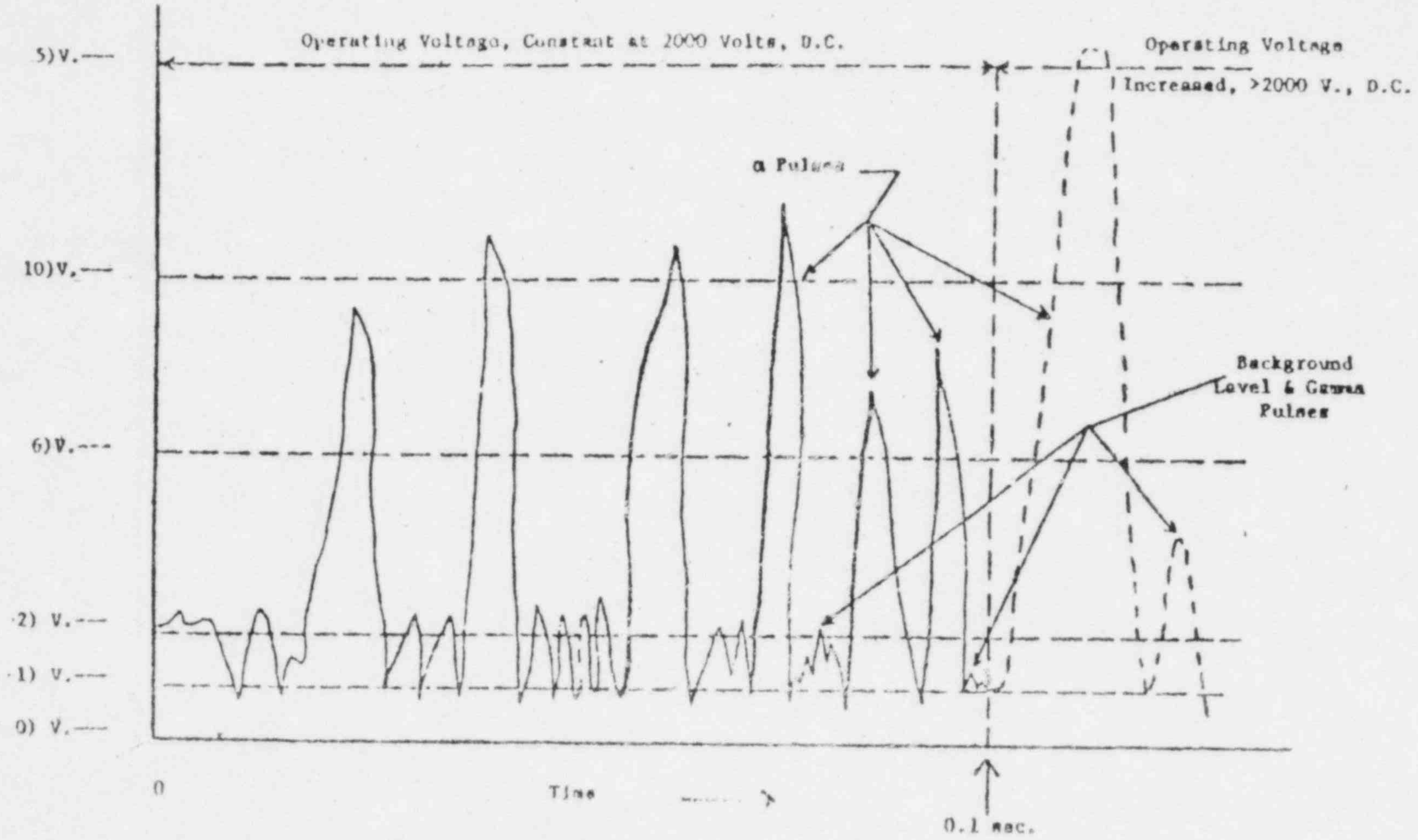
Figure 6 is a sketch showing a typical train of pulses which might be produced by a BF_3 detector. Two sizes of pulses are shown; a large pulse such as might result from the alpha ionization, as well as more numerous smaller pulses typical of gamma ionization. This sketch also attempts to give some idea of the effect of various discriminator voltage settings.

Determination of the proper discriminator setting is one of the chores involved in preparing a reactor instrumentation system for reactor startup. The technique involves the following steps. The detector is placed in a constant thermal neutron flux (perhaps produced by a calibration source) which should be strong enough to give a counting rate of several hundred counts per second. Then the operating voltage on the detector is adjusted to an appropriate value. Following this, the discriminator setting is varied and the count rate measured for each discriminator setting. This data is then plotted as shown in Figure 7. Note that as the discriminator voltage setting is increased in negative fashion from -2 volts, a rather sizeable decrease in counting rate is obtained. This represents the action of the discriminator in screening out noise and gamma pulses. In the case of the curve shown, it levels off perhaps in the region from -4 to -10 volts and it is somewhere in this region that the discriminator should be set.

In the adjustment of the circuits associated with a BF_3 proportional counter, there is another adjustment that must be made to assure proper operation. This is the determination of proper operating voltage. As covered previously, the pulse height produced in the proportional region is a function of the voltage applied to the electrodes in the chamber. As a result of low voltage, neither neutron nor gamma pulses, can pass through the discriminator. On the other hand, if the voltage is high, the pulses produced from both gammas and neutrons may be high enough to pass through the discriminator. As a result, it is customary to prepare another characteristic curve for a BF_3 channel which will indicate the change in response of the channel with changes in applied voltage. A typical plot of this information is shown in Figure 8 (Curve A).

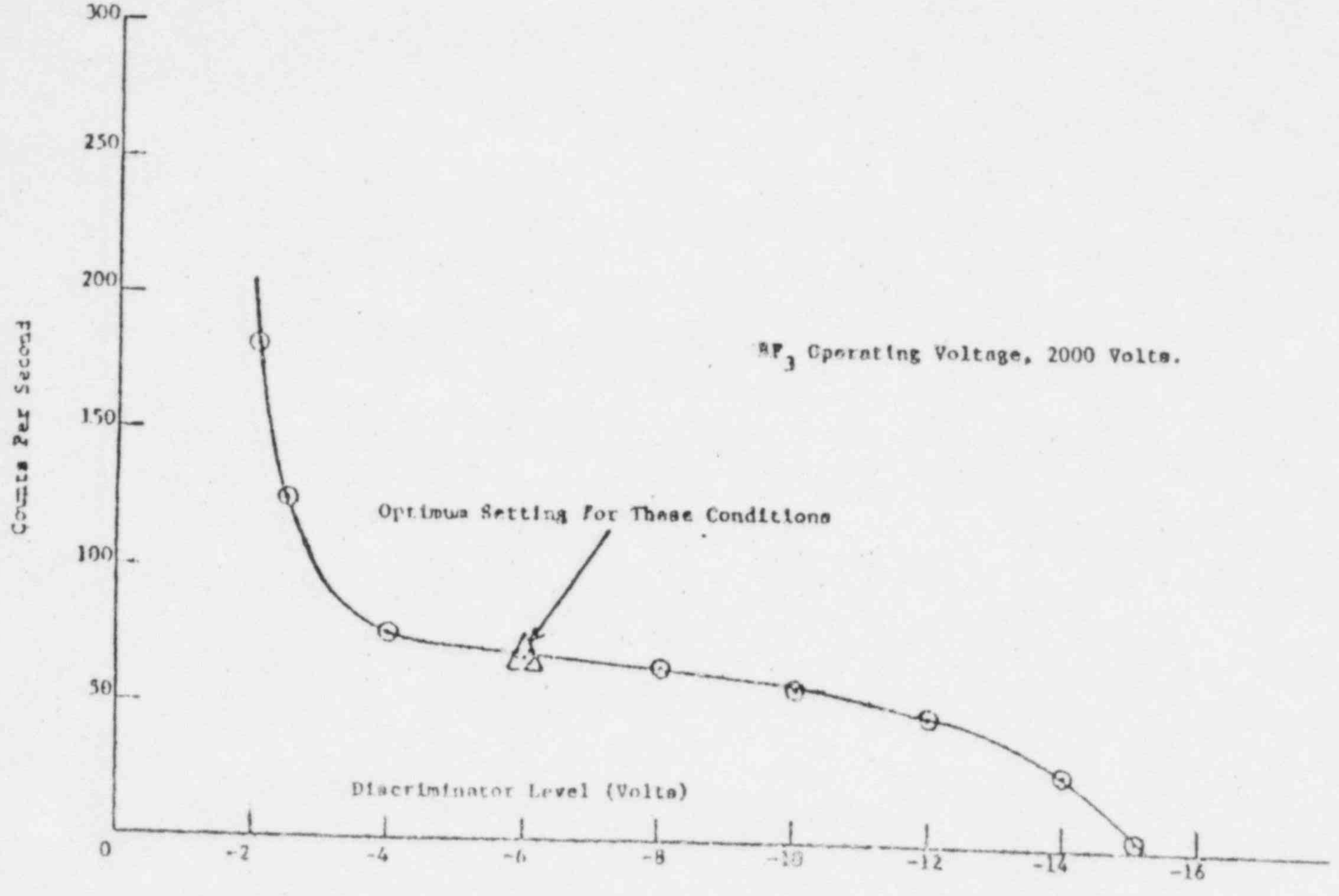
Another important characteristic of the BF_3 which is indicated in Figure 8 is the tendency of the voltage plateau to become more narrow and to shift in high gamma radiation fields. Because of this, as can be seen from Figure 8, it is advisable to select the voltage operating point out towards the knee of the curve.(Curve B).

On aspect of BF_3 operation which always concerns the designers of reactor instrumentation systems is the fact that they are susceptible to permanent damage if exposed to radiation fields above thier normal



PULSE HEIGHT vs. TIME AT THE INPUT OF THE DISCRIMINATOR CIRCUIT

Figure 6

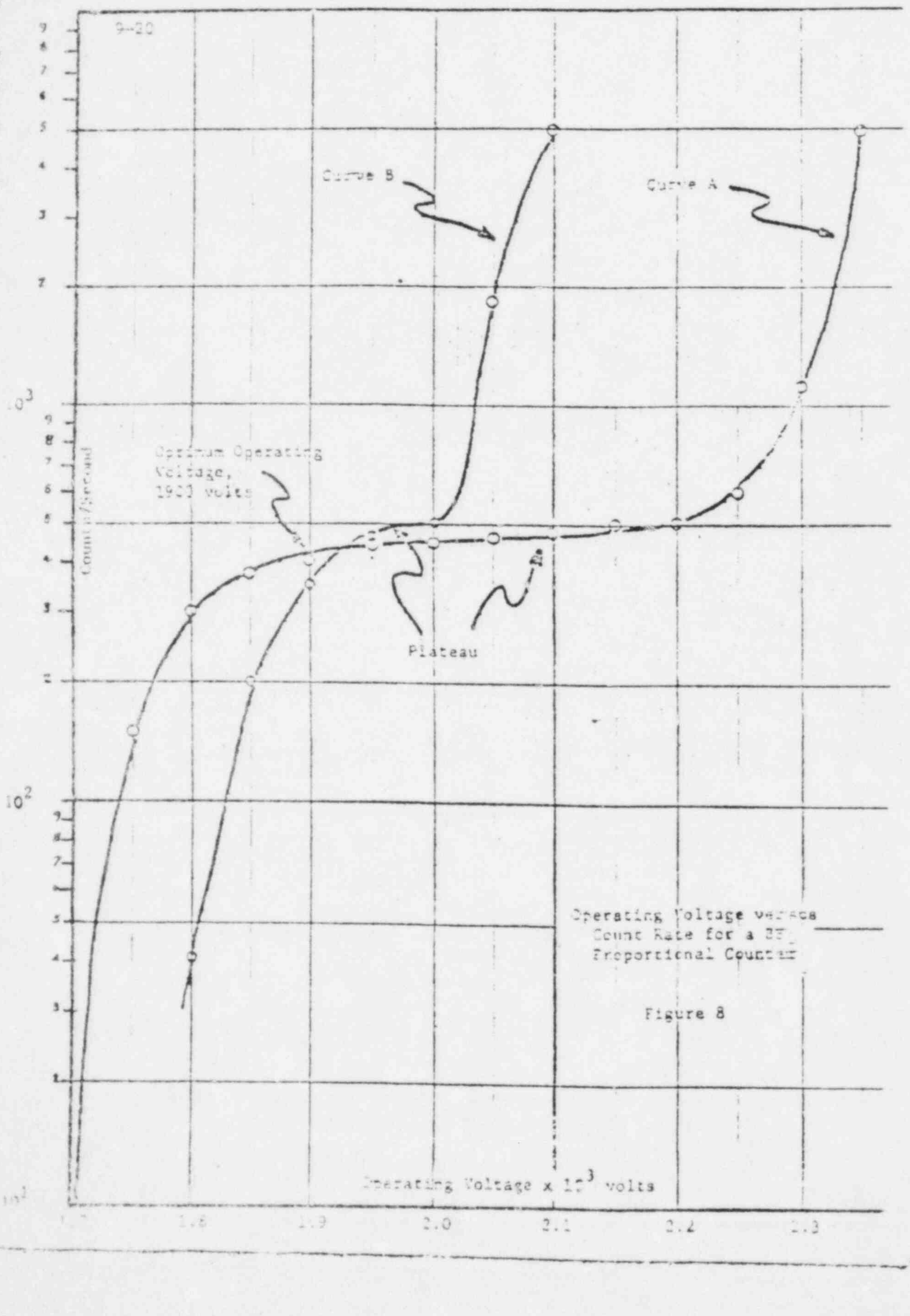


DISCRIMINATOR SETTING vs. COUNTS PER SECOND

Figure 7

(INTEGRAL BIAS CURVE)

1093 0347



Operating Voltage versus
Count Rate for a BF$_3$
Proportional Counter

Figure 8

range with voltage applied. As a result of this, all reactor instrumentation system, which use BF_3 detectors, include provisions for disconnecting the high voltage supply from the source range detectors (BF_3) as the reactor power level is increased. This discrimination is usually done manually by the operator. The customary symptom of detector damage is a tendency for the voltage plateau to disappear.

9.7.4 B-10 Lined Proportional Counter

A comparatively recent development is the high sensitivity proportional counter in which neutron sensitivity is produced by lining the detector volume with solid B-10. This method of producing neutron sensitivity has always been attractive since the counter, and its plateau are much more stable than with a BF_3 gas system. However, in general, it has been difficult to produce sensitivities comparable to the BF_3. Current developments, however, are solving this problem, and it might be anticipated that this type of detector will be utilized in the future to provide source range neutron information. The stability of this detector is such that it is being included in systems which have no provision for removing the high voltage supply as reactor power level increases. Since this detector, like the BF_3, depends on the (n,α) reaction in B 10, the need to establish a voltage plateau and discriminator curve remains essentially the same as for the BF_3 detector.

9.7.5 Fission Chamber

Another type of detector which is used in the source range and also used as a power range detector is the fission chamber. This is an ionization chamber which is constructed with a layer of enriched uranium on the electrode surfaces. Thermal neutrons, of course, are absorbed by the 235 and fission occurs. The fragments resulting from the fission having considerable energy create ionization in the filling gas that produces pulses. Chambers in which ionization is produced in this fashion have been operated in two different manners to cover two different ranges of reactor operation. Fission chambers have been used for source range detection in which the chamber is used to produce pulses. In this type mode of operation, the fission chamber is generally used with a discriminator type circuit as described for the BF_3 counter previously. In this case, the discriminator is to be used to discriminate the pulses produced by the fission fragments which are large, from those produced by the alpha particles from the natural decay of the fissionable material involved which are considerably smaller. The fission counter is also sometimes used as a power range detector in which region it is used as a current-producting device.

The fission chamber has not generally been used for source range measurement in power reactors due to the fact that for a reasonable content of U-235, its sensitivity is considerably lower than that of the BF_3. However, it has been rather universally used in research reactor applications, because in this case, it is usually feasible to locate the chamber in close proximity to the reactor core.

9.8 Other Neutron Detectors and Techniques

The neutron detectors previously considered are those most likely to be used as primary elements in a reactor instrumentation system. However, there are a variety of other schemes for sensing neutron flux. Some of these are used to supplement the basic neutron detection in a reactor, and are worthy of mention.

9.8.1 Activation Techniques

One simple technique for determining neutron flux at a particular point is to insert at that point a thin foil, or wire, of a material of known neutron capture cross section. After a pre-selected period of time the foil, or wire, is removed and its activity determined using standard counting techniques. If the information listed below is known, the flux at the point of irradiation can be calculated using relationships previously developed in this program.

a. Cross section of material
b. Dimensions and weight of foil or wire
c. Time of irradiation
d. Time between end of irradiation and beginning of counting
e. Activity at time of count

One distinct advantage of this technique is that it utilizes small bits of material which can be inserted into points in a reactor core which are inaccessible to larger detectors so far described. For this reason it has been widely used to make detailed studies of flux distributions in research and power reactor cores. In the case of a power reactor, this is the type of work which is generally done in a critical facility before the core is installed, or perhaps even constructed. This technique, of course, has the disadvantage in that it does not, in general, provide timely information concerning flux. As a result it is not feasible to use it in routine operation of a reactor.

Several common examples of this technique are described below. Small, thin gold foils have been used extensively for flux plotting. Due to their relatively high cross section, measurements can be obtained at reactor power levels sufficiently low that the resulting activity in the fuel elements is essentially negligible. Many research reactors are so constructed that it is not possible to obtain an absolute power level calibration by thermal means. In such cases it has been common to use gold foils to measure flux at many points in the core, at a low level, and integrate this data over the total core volume to obtain an absolute power level. This value is then extrapolated to the full power region and used in the adjustment of the normal instrument channels.

Another similar technique is the use of wire made of cobalt-aluminum alloy. In this case, the percentage of cobalt is made quite low, and as a result, these wires can be exposed to high fluxes without becoming too radioactive to count. Such wires have been used to measure flux distributions in power reactors at full power. With the rapidly increasing size of power reactor cores, such information is becoming of increasing importance because in a large core an undesirable local disturbance can develop which cannot possibly be sensed by the detectors outside the pressure vessel. As a result, some power reactors have been provided with facilities for inserting flux wires without shutdown or disassembly of the reactor system. Information from such a system can be used to supplement the out-of-vessel detectors with a reasonably short time lag.

One final word on activation techniques. Such techniques are commonly used to obtain information on the spectrum (or energy distribution) of the flux as well as its intensity. For example, the gold foil previously mentioned can be covered with cadmium, which will filter out the "slow" neutrons. The flux data obtained from the foil will then give the "fast" neutron flux.

9.8.2 In-Core Detectors

It may previously have been implied that it is not possible to provide a detector with a continuous electrical output for insertion directly into the reactor core. However, the increasing need for detailed flux distribution information in power reactor cores has stimulated the development of devices which are sufficiently small and radiation resistant for such service. The Appendix contains information describing the manner in which one device of this type operates. Such devices are not usually used to feed the basic reactor operating instrumentation. Instead, their output is analyzed (perhaps by a computer) and used to determine changes required in plant conditions such as rod position, to optimize core performance.

A typical PWR might use 52 assmeblies each utilizing 7 detectors of this type. These will give flux information over the length of the core at various radial positions. The outputs will be connected to the on-site computer.

9.8.3 Neutron Sensitive Thermopile

This device is not currently of great importance, but it is included because it detects neutrons by a mechanism different from any so far described.

If a thermocouple circuit (hot and cold junctions) is inserted into a reactor environment, both junctions will be heated equally and no thermal EMF will be produced. Now, if one of the junctions is coated with a material with a high neutron cross section, such as Boron-10, this junction will be heated by the energy released by neutron absorption. As a result of this, a temperature difference will develop, and a resulting thermal EMF. This EMF will be proportional to neutron flux. The sensitivity of such a device can be increased by using a thermopile, or an assembly of a number of junctions.

One specific disadvantage of this device is that its relatively large heat capacity results in a long time constant (greater than 5 seconds).

9.9 Reactor Nuclear Instrumentation Systems

The detectors which have been discussed represent the initial or primary components of a nuclear instrumentation system. Their output is not of the proper magnitude or form for the actuation of the indicating, recording, and control devices needed for plant operation. Therefore, they must be followed with amplifiers to increase the signal level to the point required to operate such devices. Computing devices must also be inserted to change the form of the signal to that which will be most useful in reactor operation.

One aspect which is perhaps unique to reactor instrumentation is the need to monitor signal levels over a tremendous range. It should be realized from the consideration of reactor physics and reactor kinetics that it is extremely important to follow variations in reactor neutron level in going through the critical region in a start-up. From this point it is necessary to follow variations in neutron level and, rate of change of neutron level, all the way up to above the full power point. This range can involve a factor of 10^{12} or more. It is difficult, if not impossible, to assemble a single instrument system which can cover such a range adequately. As a result, it has become customary to instrument a reactor with a number of nuclear detection systems, each covering a different portion of the total reactor neutron level or power range. When this is done, the instruments assigned to cover each range of reactor level can be designed for optimal performance in their assigned range. It has generally been found feasible to cover the normal power reactor operating range by dividing it into three different ranges. The first of these is designed to cover that range of neutron level in the reactor in which the reactor start-up source is influential. For this reason, this range has come to be referred to as "the source range" or "the start-up range". As shown in Figure 9, the source range typically covers a range of reactor power from 10^{-10} percent of full power to 10^{-4} percent. The output of this channel is usually displayed in terms of counts/second, and over the range indicated for this channel the variation would be a factor of 10^6. In the shutdown condition of the reactor, such a channel might have a count rate of something greater than 1 count/sec. so the overall calibration of the associated instruments would vary from 1 count per second to 10^6 counts per second. Referring again to Figure 9, it is shown that sometime before the upper limit of the range on the source range another set of instruments becomes operational. This may be designed to be operational from 10^{-5}% to perhaps 10% of full power of the reactor. Proceeding upward, the power range is encountered which generally replaces the intermediate range for reactor power levels between perhaps 10% and 100% of full power. This range, of course, must be responsive to signals which are actually over 100% of full power so that it can be used to produce necessary safety shutdown-type control actions. Following sections will consider each of these ranges in further detail.

9.9.1 Source Range

The source range typically uses a counter type detector, which

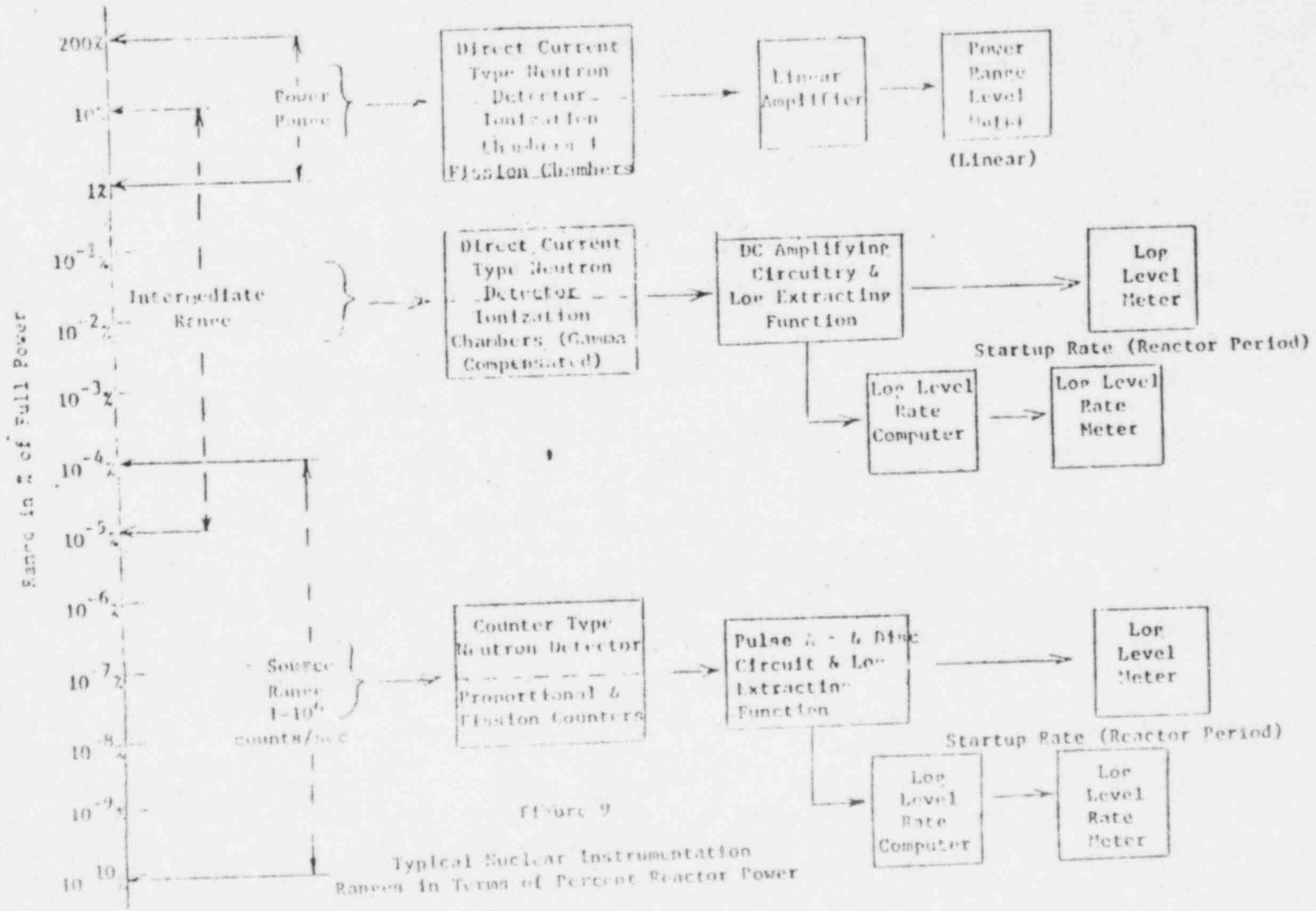


Figure 9

Typical Nuclear Instrumentation Ranges in Terms of Percent Reactor Power may be either a proportional counter or a fission counter, which is followed by the necessary electronic devices to amplify the pulses and discriminate against extraneous pulses which may occur in the system. Another feature which customarily is incorporated into the source range channel is that of log extraction. Since this channel covers such a broad range of reactor power level, variations at the low end would be almost imperceptible if it were displayed in linear fashion and of course it is these variations at the low end which reveal what is actually occurring in the reactor during the transit through criticality. To solve this problem, it is customary to take the logarithm of the count rate output of the chamber and display this signal. This provides increased sensitivity at the low end of the range covered and, as a result, decreases sensitivity at the upper end.

In addition to using the output of this chamber to provide an indication of neutron level, it is also customary to produce by computing circuits an indication of the speed or rate at which the reactor power level is increasing. Now it is a characteristic of reactors, at least before temperature and other feedback effects become apparent, that the power level will increase in exponential fashion following the insertion of a certain amount of reactivity. It is a characteristic of exponentials that the rate of change of the variable in question, whether it be reactor power level or anything else, divided by the instantaneous value of the variable, gives a constant which characteristic of the exponential. It is, therefore, convenient in reactor control systems to compute this characteristic constant. This turns out to be rather simple since the logarithm of the source range signal has already been taken for convenience in indication. A relatively simple electrical differentiating circuit can now he used to take the derivative of this log signal and produce this constant which is related to the rate of power level increase. The mathematics of this situation can be examined assuming that a step insertion of reactivity has been made and that the reactor power level is increasing according to the law

$$P = P_o e^{t/T}$$

If the natural logarithm of both sides of this expression is taken, we obtain

$$\ln P = \ln P_o + \frac{t}{T}$$

If the derivative of both sides of this expression is taken, the result is:

$$\frac{\partial(\ln P)}{\partial t} = \frac{1}{T}$$

Thus, a signal which is representative of the rate at which reactor power level is increasing relative to its absolute value has been obtained. This is a quantity which is extremely important in the reactor start-up situation. This signal which can be developed, related to the characteristic exponential on which the reactor power level is increasing, can be displayed on a meter calibrated in various units. In research reactor systems, it has been rather common to calibrate this meter in terms of T, the number of seconds required for the reactor power level to increase by a factor of e. This quantity is known as reactor period. A typical start-up period used in a research reactor might be 20 seconds. When indicated on the meter, this would indicate that every 20 seconds, the reactor power level is increasing by a factor of e. However, the practice which appears to be developing in the nuclear power industry is to calibrate this meter in terms of decades per minute (referred to as startup rate). To compare these two modes of indication, if the reactor period is 20 seconds and the reactor power level is increasing by a factor of "e" or approximately 2.7 every 20 seconds, in a minute it would increase 2.7^3. Therefore with a 20 second period, the reactor power level is increasing by a factor of 19.7 every minute or about 1.2 decades per minute ($10^x = 19.7$, $\log_{10} 19.7 = x \approx 1.2$).

To sum up the typical source range system, it is activated with a detector primarily designed to produce a pulse output. These pulses are amplified and fed to an electrical circuit which computes the number of pulses per second and produces an output signal which is proportional to the number of pulses per second and therefore the neutron level. The logarithm of this signal is then taken to provide a convenient indication for the reactor operator. Subsequent to this step, the derivative of the logarithm is taken to give a signal related to the rate of change of reactor power level. This "period" signal also is displayed.

When the intermediate range is considered, it will be found that this signal produced by differentiating the log of the level signal is used to produce a signal which can cause a reactor trip. The use of such a trip has been attempted in the source range but it has tended to be rather unsuccessful since the statistical nature of the detector output tends to cause rather sizeable fluctuations in the output signal. Taking the derivative of this signal tends to exaggerate the fluctuations. Thus, the period meter in the source range tends to vary rather widely and it is difficult to adjust the time constants in such a system so as to give proper indication and still avoid the possibility of a spurious trip signal.

9.9.2 Intermediate Range

The primary sensing element in the intermediate range is almost universally the compensated ion chamber which has been discussed in some detail. Following the ion chamber, we have an amplifier designed so that its output is proportional to the logarithm of the input signal. The situation is analogous to that in the source range in which the range of neutron level covered by the channel is very large. To obtain a reasonable indication at all, it is desirable to observe the logarithm of the signal rather than the signal directly. Again, as in the source range channel, the derivative of the log N signal is taken to obtain a measure of the rate of change of reactor power level. In this case, the signal is actually used to produce a reactor trip if the rate of change becomes excessive.

9.9.3 Power Range

As the intermediate range approaches the upper level of its capability, the power range instrumentation should take over and begin to provide the operator with the necessary information for operation of the reactor. In moving into this range, the region is entered where heat generation by the reactor becomes significant. As a result, small variations in reactor power level become of increasing importance to the operator. For this reason, the electronic equipment associated with the power range is designed to produce a linear indication of reactor power level on which small variations can be more easily seen throughout the range than with the log type of indication provided in the source or intermediate range. The detector which serves the power range of the primary sensing element may be an uncompensated ion chamber or it may be a fission chamber operated in the mean-level mode. Power range instrumentation is generally designed to cover the reactor power level between approximately 10% of reactor full power up to and including 150% of full power. Power range instrumentation is designed to provide signals which can be used for automatic shutdown of the reactor in the event that the reactor power level reaches some undesirable level above the 100% point. This trip setting may be anywhere from 105% to 120% depending on the design of the reactor and the assumptions made in its safety analysis.

9.9.4 Overlap

Figure 9 exemplifies a typical three range system. The regions of reactor power level covered by the three ranges of instrumentation and the resulting output currents and the flux levels are shown. These flux levels are flux levels at the detector rather than flux levels in the core. Note that in this layout an overlap of approximately a decade between the various chambers is shown. This is extremely important because as the power level is increased, during a start-up for example, as the upper limit of usefulness of one range is reached, it is necessary that the next range become active to allow continued proper knowledge of conditions in the reactor. Provision of this overlap is initially the responsibility of the designer of the reactor control system who must select chambers, instrumentation, and positions for location of the chambers so that this overlap exists. However, to actually guarantee overlap in reactor operation, it may be necessary to adjust chamber position in the process of initial start-up of the reactor. Changes can occur in the primary detectors, the associated electronic instrumentation, or in the reactor itself, which can cause this overlap to be decreased. It is extremely important that the operator be wary of such a situation. As the reactor power level is increased, he must be prepared to stop this power level increase if it appears that the power level reading is going to go off scale in one range before adequate information is received from the next range.

9.9.5 Redundancy

If the reactor instrumentation system is designed so that a proper channel is provided for each range of operation, the operator will theoretically have adequate information to carry out his duties. However, if he were provided with only one source of information in each range, a problem could develop if a failure should occur in that range channel. To reduce the probability of this happening, it is customary to provide at least two, and in some cases more, channels of instrumentation for the various ranges. For example, a typical power reactor control system as shown in Figure 1[illegible] and Figure 11 will have two identical channels of start up source range instrumentation and two identical channels of intermediate range instrumentation. When the power range is reached, it is customary to provide three or even four channels of instrumentation. Thus, in any case, one can reassure oneself of the adequacy of the information being received by comparing one channel against the other.

It is not completely obvious why two parallel channels are considered adequate for the startup and intermediate ranges, when three or more are generally supplied for the power range. After all, if two channels of instrumentation are contradicting one another, how does one tell which is correct? In the case of a three channel system, if one channel disagrees with the other two, one can assume with reasonable probability that it is indicating incorrectly.

There are several answers to this question. One is that, if in a two channel range, the two channels are not in agreement, the operator may be able to identify the one which is correct simply by his knowledge of plant conditions. For example, if reactivity has been inserted by withdrawing control rods, one should expect the reactor power level to be increasing. If one channel does not indicate this increase, one might conclude that it is the channel which is malfunctioning.

One reason why additional channels are more essential in the power range is that this is the region in which the reactor is going to spend the major portion of its life. It should therefore be possible for any one of the power range channels to fail wihtout producing a shutdown of the reactor. It should be possible to remove any one of the power range channels from service for maintenance without requiring reactor shutdown.

9.9.6 Calibration

Ordinarily, absolute calibration of either the source or intermediate range in terms of power level is not attempted. The basic requirements are that these channels indicate proper changes in neutron level and rates of change in neutron level and provide suitable overlap between ranges. The absolute level in terms of say, kilowatts, is not too significant. However, when the power range is reached the situation is somewhat different. In this range, the reactor is releasing energy in measurable quantities and it is important that the indication produced by power range be calibrated rather precisely in terms of energy released from the reactor. The fact that these power range channels are used to produce reactor trip

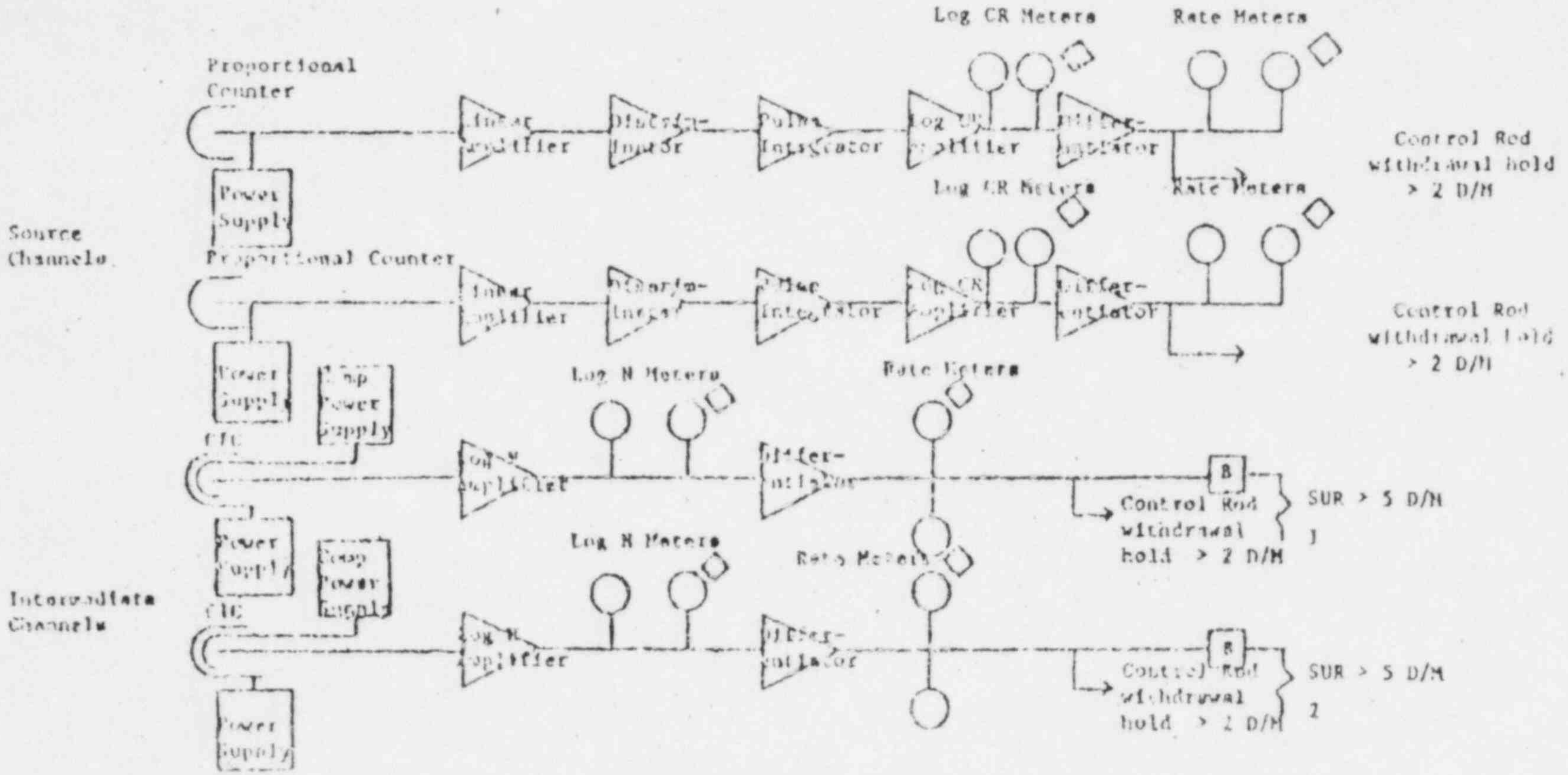

Log CR Meters
Rate Meters
Proportional Counter
Power Supply
Linear Amplifier
Discriminator
Pulse Integrator
Log CR Amplifier
Differentiator
Control Rod withdrawal hold > 2 D/M
Log CR Meters
Rate Meters
Source Channels
Proportional Counter
Linear Amplifier
Discriminator
Pulse Integrator
Log CR Amplifier
Differentiator
Control Rod withdrawal hold > 2 D/M
Power Supply
Comp Power Supply
CIC
Log N Meters
Rate Meters
Log N Amplifier
Differentiator
B
Control Rod withdrawal hold > 2 D/M
SUR > 5 D/M
1
Power Supply
Comp Power Supply
Log N Meters
Rate Meters
Intermediate Channels
CIC
Log N Amplifier
Differentiator
B
Control Rod withdrawal hold > 2 D/M
SUR > 5 D/M
2
Power Supply

Figure 10

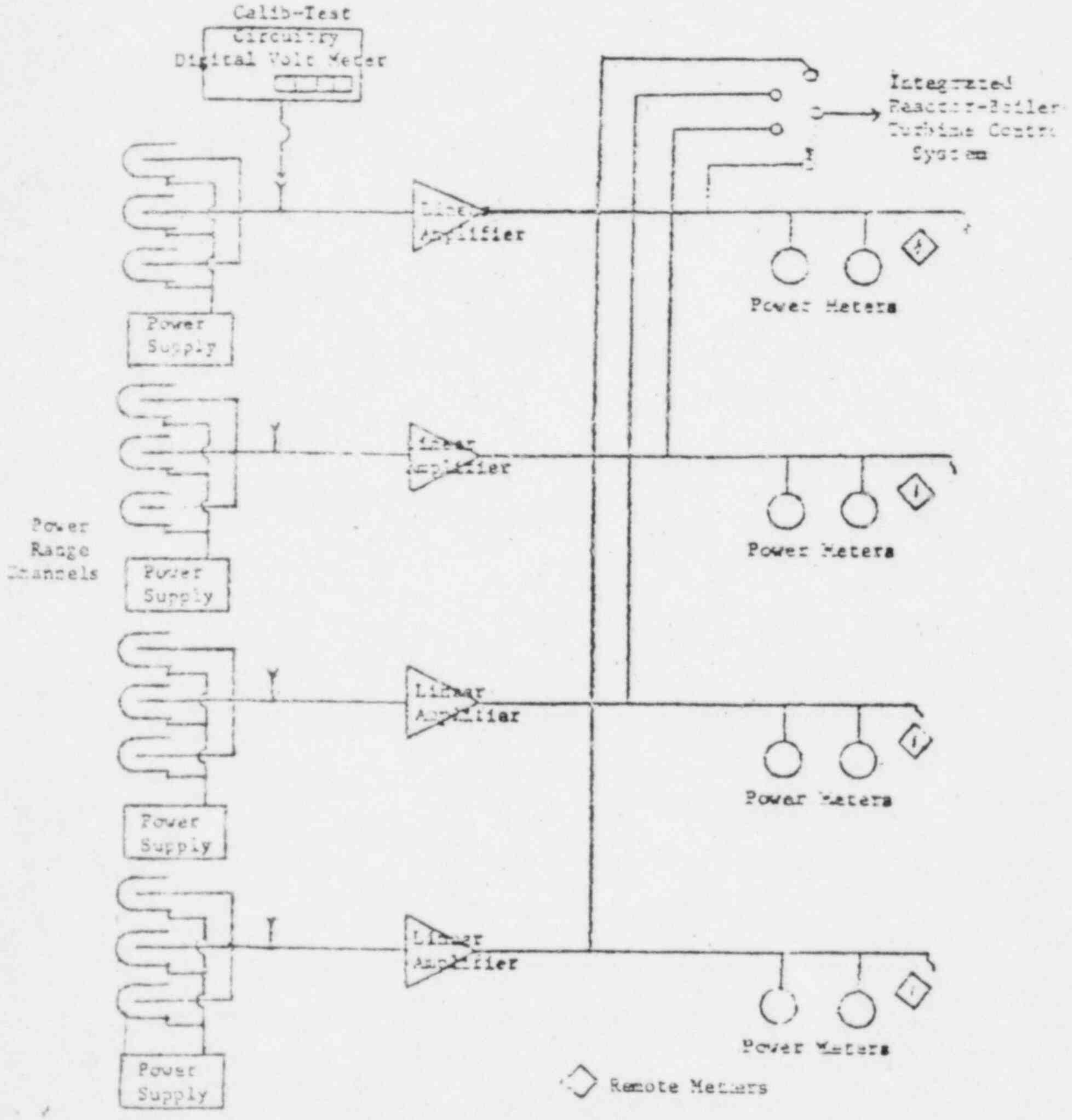

Calib-Test
Circuitry
Digital Volt Meter
Integrated
Reactor-Boiler-
Turbine Control
System
Linear
Amplifier
Power Meters
Power
Supply
Linear
Amplifier
Power Meters
Power
Range
Channels
Power
Supply
Linear
Amplifier
Power Meters
Power
Supply
Linear
Amplifier
Power Meters
Power
Supply
Remote Meters

Figure 11

signals in the event of an over-power situation increases the importance of their calibration. Power range instruments are usually adjusted on the basis of measurements made of the thermal power output of the plant. The thermal output of the reactor can be determined by multiplying the flow through the core by the temperature difference across it by the appropriate constant to obtain the rate of power production. Another mode of determining the actual operating power level of the reactor is to make a measurement of steam generated. However, this involves measurements and sometimes assumptions regarding quality of steam produced which tends to reduce somewhat the accuracy available. At any rate, in some fashion, a knowledge of reactor power level is always derived from the thermal output of the core and this value is used to readjust the power range neutron detecting channels until they coincide. This may be done either by adjusting the position of the chambers relative to the core, or more commonly by manipulating an adjustment potentiometer in the power range circuit.

9.10 Electronic Circuitry

It is not intended in this program to attempt a detailed or comprehensive coverage of the many different electronic circuits employed in reactor control systems. For one thing, many of these use standard electronic techniques which are not peculiar to reactor control systems. Also, there is not too much point in covering the circuit details, as there is a wide variation from one plant to another. The transition from vacuum tubes to transistors, microcircuits, etc., has made the rate of obsolescence of nuclear electronics gear particularly rapid. However there are a limited number of topics in the area of circuitry which should be discussed in any program for reactor operations personnel.

9.10.1 Input Coupling

The section on detectors has described the fashion in which radiation produces ionization, ion-pairs, and charge (q in Ampere-seconds or coulombs) in a detecting device. This charge goes to produce a voltage across the capacitance formed by the chamber and its associated leadwire. At this point it might be noted that typical chamber capacitance ranges from 100 μμf to over 500 μμf. For the capacitance of the cable connecting the detector to the remainder of the system, a typical number often used is 10 μμf/foot. To obtain a useful signal from the detector, the charge obtained in this fashion must be drained off through a load resistance so that a useful current or voltage signal can be produced. As might be imagined, the manner in which this is done will be somewhat different depending on whether a voltage pulse or a continuous current is required from the chamber.

9.10.1.1 Pulse Input Coupling

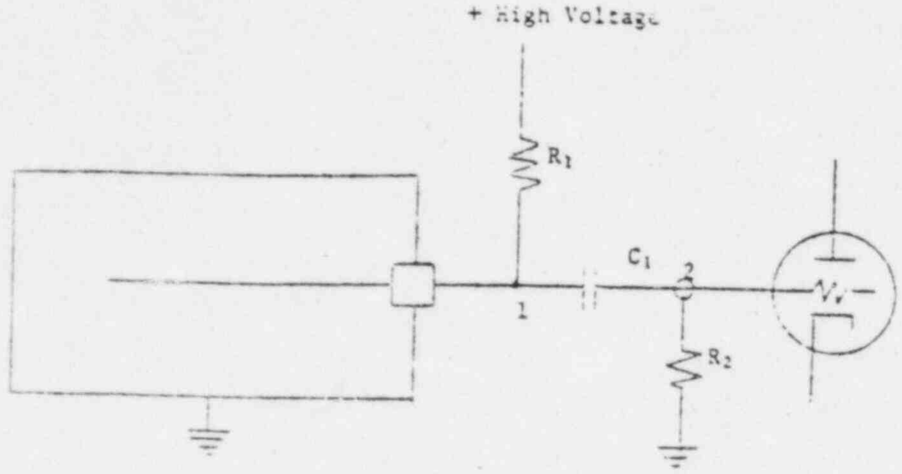
It is common in reactor source range systems to use the detector output to develop voltage pulses which are amplified and shaped before being fed on through the system. A typical input circuit is shown in Figure 12a. When a current pulse flows through R_1, as a result of ionization in the detector, the voltage at point 1 is lowered, causing the capacitor to discharge. This will cause a current to flow from ground through R_2, producing a negative voltage pulse at (2). Note that the time constant, R_2C_1, should be short relative to the reciprocal of the maximum counting rate anticipated. In other words, the currents which flow as a result of an ionizing event in the detector should be over well before the next event. In addition, R_2 must be large enough to allow a voltage of adequate size to develop across it.

9.10.1.2 Current Input Coupling

The charges produced in the chambers used in the intermediate and power ranges are generally allowed to flow through an external current measuring device. The number of events taking place per second is so great that the total effect is a DC current (with some noise voltage) flowing. Such a circuit is shown in Figure 12b. The current developed by ionization in the chamber flows through resistor R_1, developing a voltage which can be fed into an amplifier. Since the current might be very small, perhaps 10^{-11} amperes, the resistor required to give an adequate 1 millivolt signal for amplification would be large such as 10^8 ohms. A more desirable approach is to use a current feedback type system as shown in Figure 13. Note that if the gain G is very large, E_o approaches $-E_{sig}$. In this case, the voltage drop seen by the chamber current is zero, and the effective steady state resistance is zero. In practice, it might amount to several thousand ohms.

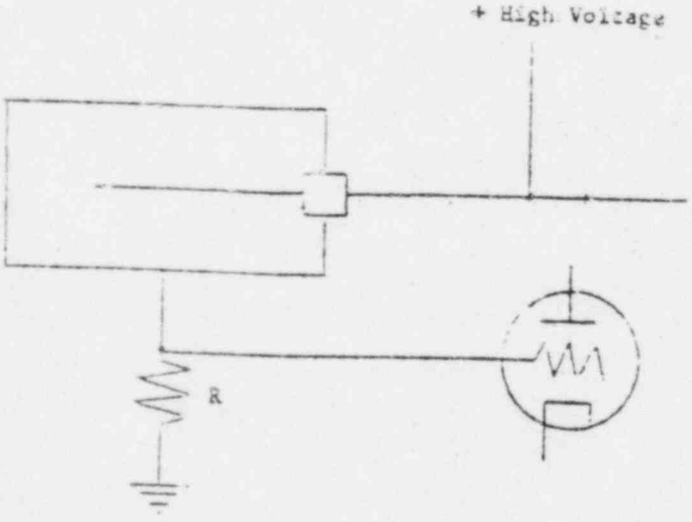
9.10.2 Preamplifiers

The length of cable required between the location of the neutron detectors (near the reactor vessel) and the remainder of the instrumentation system (in the control room) may be 300 feet or more in a large reactor power plant. If the chamber capacitance is 300 uuf and the cable capacitance is 10 uuf/foot, a run of this length could increase the total input capacity from 300 uuf to 3300 uuf, or a factor of 11. If it is assumed that the output pulse voltage is determined by the charge produced by one ionizing event and the total input capacitance (Q/C = V), an increase of 11 in capacitance will produce a corresponding reduction in the size of the pulse. To avoid this possibility, a preamplifier is often installed as close as possible to the detector. Such an amplifier may serve only as an impedance changing device, with a voltage gain which is low, perhaps even unity. Such an amplifier, while it may produce no increase in voltage, by virtue of its low output impedance, can supply the current required to charge the cable capacitance.



PULSE INPUT COUPLING CIRCUIT

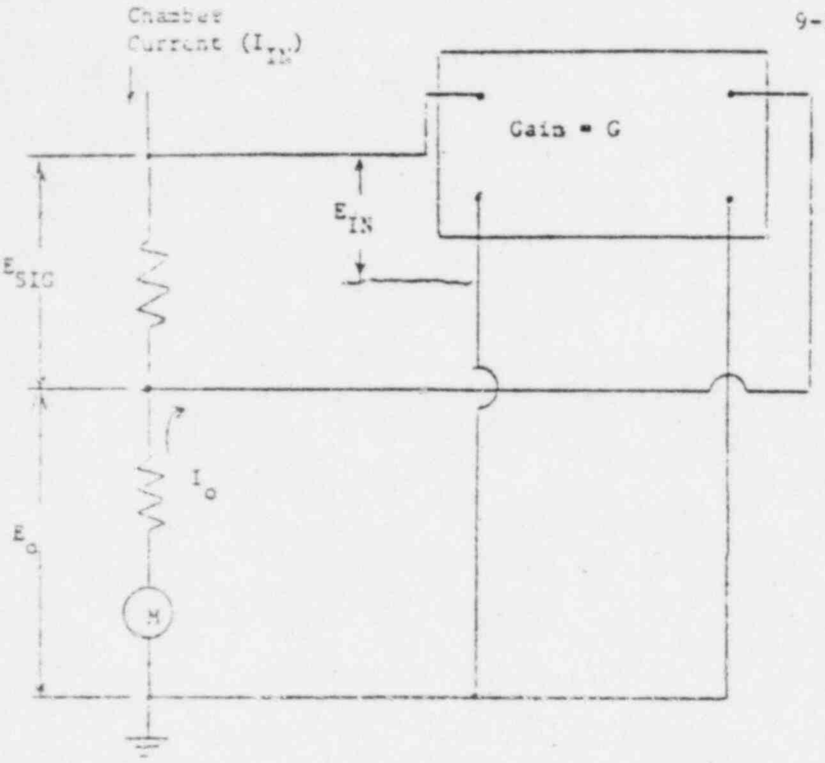
A



CURRENT INPUT COUPLING CIRCUIT

B

Figure 12



$$E_{IN} = E_{SIG} + E_o$$

$$E_o = -GE_{IN}$$

Solving this for E_{IN} and substituting this in the first equation yields:

$$E_o = \frac{-E_{SIG}}{1 + \frac{1}{G}}$$

If $G >> 1 \quad E_o \approx - E_{SIG}$ and $\frac{I_{IN}}{I_o} \approx \frac{R_o}{R}$

CURRENT FEEDBACK ELECTROMETER

Figure 13

Preamplifiers are ordinarily used only in the pulse output, source range channels. In the intermediate and power range channels with current type transmission of signals, the cable capacitance does not affect the steady state flow of current once the capacitance has been charged. Of course it will still reduce the response of the system to transients.

9.10.3 Count Rate Computation

The train of pulses which exists in a pulse type channel after amplification, discrimination, shaping, etc., is still of no use to the reactor operator unless he has some means of determining their rate of arrival. One simple way to do this is to use the pulses (with the necessary amplification) to drive a mechanical register. Of course the pulse rate is usually too fast for a mechanical device so, prior to the register, an electronic "scaling" circuit is ordinarily used which totalizes pulses until 1000 have been received and then transmits a pulse to the register. Thus each count on the register represents 1000 actual counts.

Now this system still gives only the grand total of pulses received, and not the pulse rate. To obtain a measurement of pulse rate, the operator must observe the change in total counts recorded over a known period. This is rather awkward for normal operational use, although early reactors made use of this type of count rate determination. It is still used at times when high precision is required, such as maintaining a check on the approach to criticality while fuel is being loaded. If a scaling type device is counting while the reactor is in normal operation, a by-product results which some operators have found useful. That is, the constant clicking of the register gives the operator an audible indication of reactor power level. Surprisingly small changes in power level (and hence counting rate) can be detected in this fashion.

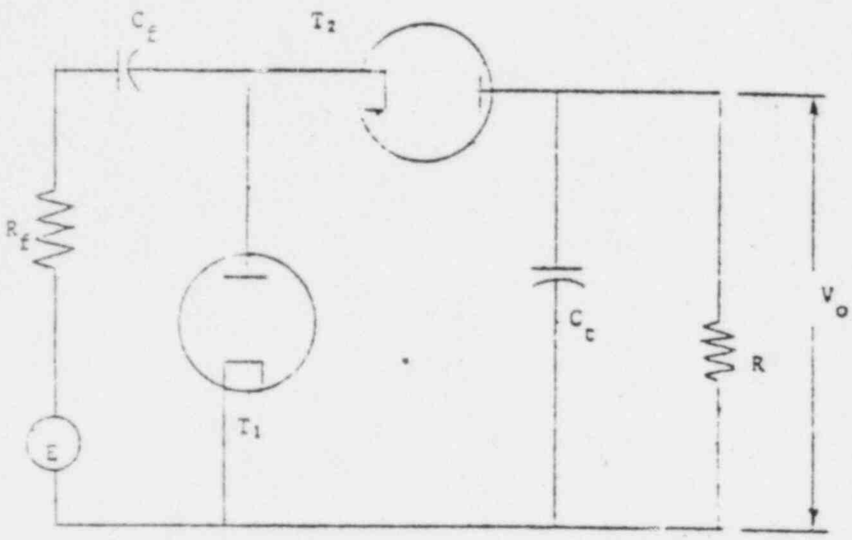
Regardless of the foregoing, something more convenient is required in modern reactor control systems. Therefore, the circuit known as a counting-rate circuit is used. (See Figure 14)

A counting-rate circuit is generally arranged so that each input pulse feeds a known charge into a capacitor (C_t) shunted by a resistor (R). The voltage across the capacitor builds up to an equilibrium value at which the rate of loss of charge through the resistor equals that being supplied to the capacitor. When this occurs, the voltage, V_o, across R is:

$$V_o = r\,q\,R$$

where r is pulse rate and q is charge per pulse. From this expression it can be seen that V_o is proportional to pulse rate and independent of C_t.

Assume E in Figure 14 with internal resistance R_f produces pulses of voltage V and duration T. Capacitor C_f charges through R_f



COUNT RATE CIRCUIT

Figure 14

and T_1 to almost V. When V returns to zero, C_f discharges through T_2 transferring a charge of V C_f per pulse to C_t.

To insure that the charge is completely transferred, C_f must be very much smaller than C_t, and V_o must be very much smaller than V. To allow equilibrium to be attained between pulses, $1/r - T$ must be greater than 5 R_f C_f.

The output of such a circuit is essentially a DC signal, although statistical fluctuations will exist in it. These can be minimized by selection of component values, but decreases in response time to transients will result.

It might be observed that the portion of the circuit formed by C_f and R can be compared to the equivalent circuit of a chamber operating in a current output mode, and its load resistor. In this case, however, the designer has considerable control over the charge contained in the pulse and as a result it is possible to produce a usable analogue (DC current or voltage) type signal even when the count rate is low, such as one count per second.

9.10.4 Logarithm Computation

As mentioned previously, coverage of the range of reactor power which is ordinarily assigned to the source and intermediate range system is not feasible with a single range linear readout. One solution to this problem is to use a single range system in which the range is switched (usually manually, but sometimes automatically) as the power level increases. An example of this can be seen in the Linear Level channel of the Penn State TRIGA Reactor.

However, designers of power reactor systems generally prefer to avoid the need for this type of switching if at all possible. One method of doing this is display the logarithm of a level signal, rather than the signal itself. This type of presentation can be seen on the Log N channel of the Penn State TRIGA Reactor.

Several schemes have been found for extracting the logarithm of a signal. One which is very simple but which has been used extensively will be described.

The voltage across certain diodes such as a 9004 is, over certain ranges, proportional to the logarithm of the current through the diode. Figure 15A is a circuit of this type. Note that a high resistance is shown in series with the voltage signal The reason for this is that the current through the diode must come from what is effectively a current source. Otherwise, the voltage across the diode would feed back and affect the current through it. The 9004 diode (which is a vacuum tube diode), if operated with a heater voltage of 3 volts, will provide a logarithmic voltage output for a current range of 8 decades.

9.10.5 Start-Up Rate (or Period)

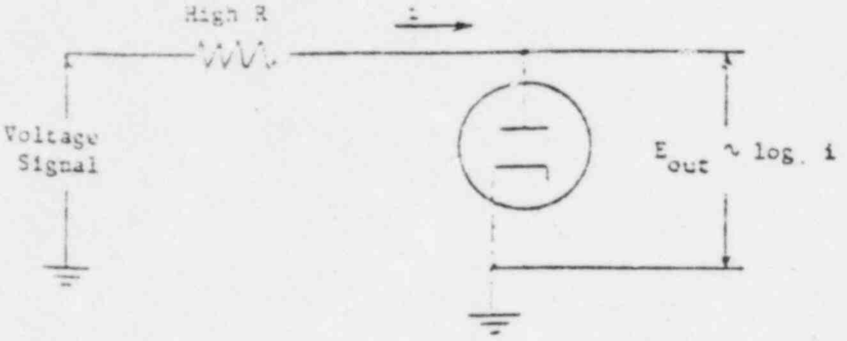
Since the reactor is basically an exponential rather than a linear system, the time required for the power level to change by a factor of e gives a very convenient description of its rate of change at any given time.

Consider the circuit of Figure 15B. If the input voltage is changing, a current will flow into or out of capacitor C. This current is proportional to the rate of change of the input voltage if R and C are properly selected. The output voltage will then also be proportional (i R) to the rate of change. Such a circuit is called a differentiating circuit. In the case of an exponential increase in reactor power level, if a log P (or log N) signal is supplied to the circuit:

$$\log_e P = \log_e R_o + \frac{t}{T}$$

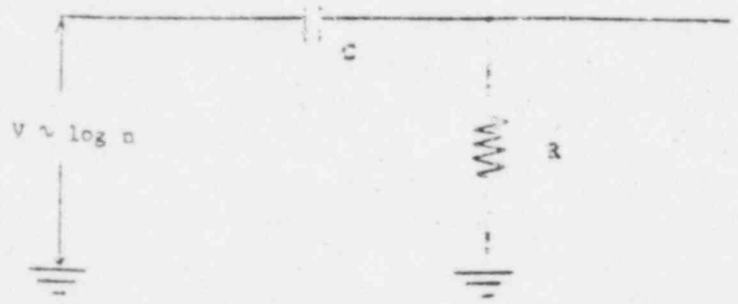
$$\frac{d(\log_e P)}{dt} = 0 + \frac{1}{T}$$

Therefore, the output voltage is proportional to the inverse of the period T. The relationship between T and Startup Rate (SUR) has been described.



LOG CIRCUIT

A



B

Figure 15

9.11 Reactor Safety Systems

The first designers of nuclear reactors recognized that these systems possessed energy release capabilities greater than any device which had ~~not~~ yet been created. They also realized that any control scheme devised for a reactor, whether manual or automatic, had some finite failure probability. As a result, it became customary to incorporate additional equipment which would automatically terminate the chain reaction if certain undesirable conditions developed. Such conditions were most commonly an excessive neutron flux or an excessive rate of change of neutron flux. As the years have passed, such equipment has become continually more sophisticated. It is the intent of this section to discuss some of the thinking involved in this development, to define some of the concepts involved, and finally, to examine certain current safety systems.

9.11.1 Typical Reactor Safety System

A typical reactor safety system is one which supplies the energy, usually electrical, to energize magnets, clutches, or other devices which support neutron absorbing material (safety rods) in an out-of-core position. The system is ordinarily arranged so that if the reactor instrumentation detects an unsafe condition in the reactor plant, power is interrupted to the rod supporting magnet, or clutch, allowing the neutron absorbing material to be inserted into the reactor core by whatever energy storage system may be in use.

In many of the early reactors (particularly test and research reactors) the inputs to the safety system were primarily signals from neutron detecting instrumentation such as neutron level or period. With the development of the large power reactors the need for economics of operation make it necessary to operate close to the thermal limits of the system, and it becomes necessary to include some of the coolant system measurements in those that produce shutdown (scram) signals.

9.11.2 Failure Mode

It is impossible to construct any of the necessary components for a safety system without recognizing some possibility for failure. Therefore, it is customary to design equipment so that it will be "fail safe", meaning that any, or at least the most probable, failures will cause the safety action the equipment is intended to produce.

As a result of study in this area, actually three types of failure have been identified, as defined below:

a. Safe Failure - this is as described above. A simple example is failure of the power supply for rod supporting magnets.

b. Unsafe Failure - this means that the failure actually produces a positive action causing a possible hazard to the reactor. An example of this is an open circuit in the feedback loop of an automatic control system which will cause a continual rod withdrawal.

c. "Sterile" Failure - a failure which causes a component or system to become inoperative, with no indication of this, is considered to be "sterile". An example of this might be an open circuit in a thermocouple where the associated instrumentation is not designed to do anything specific in this event.

9.11.3 Redundancy

As described above, an effort is made to design reactor safety systems so that failures will not cause the reactor to be unprotected. However, there always will remain some probability of unsafe or sterile failures. To further protect the reactor, two or more complete channels are provided for each safety function. This, in conjunction with design or safe failure, produces a safety system whose probability of not taking the desired action is extremely small.

9.11.4 Coincidence

Previous sections have discussed the extreme lengths to which designers have gone to produce reactor shutdown in case of indications of trouble (real or otherwise) in the reactor safety system. It can be readily imagined that systems designed on the basis of the ideas presented so far would not permit a reactor to operate very long without a shutdown due to some malfunction in the safety system. This situation might be acceptable in a research reactor, or a small demonstration power reactor, but with the development of 1000 MWe plants, an unnecessary shutdown can not be taken lightly.

One of the major defenses against an unnecessary shutdown is the use of coincidence. This means simply that the shutdown signals produced by the various safety channels are considered by a logic system which calls for a reactor shutdown only if two out of three (or some other combination) of the safety channels produce a shutdown signal. This produces a great increase in plant reliability with a relatively small reduction in safety.

In addition to directly improving reliability, the coincidence systems provide a secondary reliability benefit. This is the capability of removing a channel from service for test and repair. This removal would ordinarily cause a trip signal to be produced by the channel concerned. However, if two or more trip signals are required to produce a shutdown, one channel can be taken out of service.

9.11.5 Reliability and Test Schedule

The term "reliability" as used in the missile and space industry, generally means the probability that system will survive to complete its mission. The mission length is generally short (hours or days), and no provision is made for testing during the mission. The situation in a power reactor is much different. The mission or life time of the plant is relatively long (perhaps 40 years). The probability of survival for any particular component over this period of time is low. However, periodic testing is more feasible, and indeed, essential. In this case reliability is considered the probability of survival from one complete test and overhaul to the next.

9.11.6 Independence

Another aspect of safety system design which is receiving considerable attention is the relative independence of the various channels. All the effort which goes into applying redundance and coincidence can easily be defeated if a single failure in a system common to all channels can occur. If such a failure is "safe" then reactor reliability is jeopardized. If the failure is "unsafe" then reactor safety is at stake.

One obvious lack of independence would be the supplying of instrument power from a single source. Thus, several quite independent sources of power are provided. Possibly a less obvious source of dependence is the influence of instrument room environment. For example, could a change in ambient temperatures produce a calibration shift in key components in a number of channels? Current designs are considering such points.

9.12 Nuclear Plant Control

The control of a nuclear power plant can be divided into three areas:

a. Control of reactor neutron level
b. Control of primary coolant conditions to meet turbine demands.
c. Control of turbine generator systems.

In this section a rather simple treatment of (a) will be given, and (b) will be considered. (c) will not be discussed. As in other sections, discussion will be related specifically to pressurized water reactors.

9.12.1 Control of Reactor Neutron Level

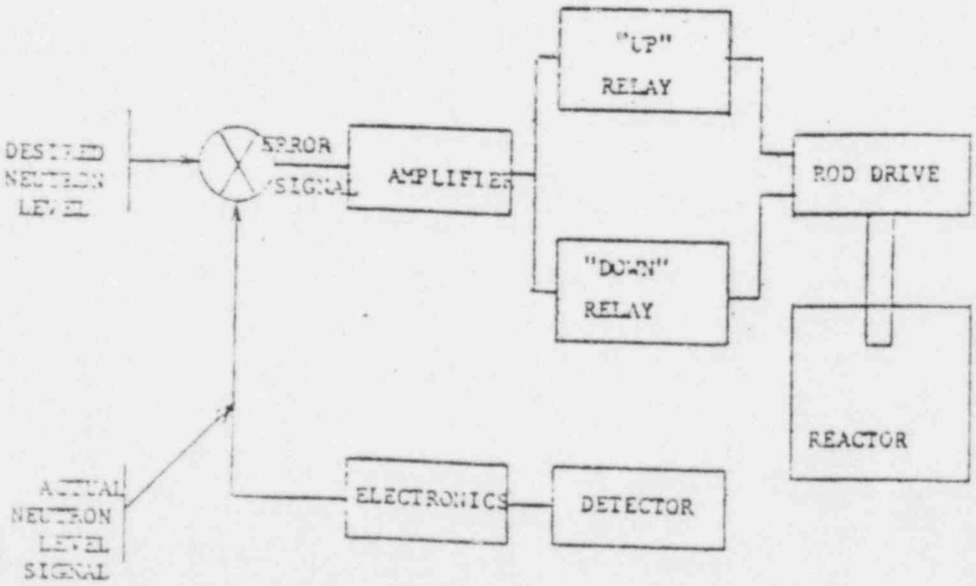
The concepts covered in Chapter 6, Reactor Kinetics, define the manner in which reactor neutron level (or flux) is controlled. Insertion or removal of a neutron absorbing material such as a control rod, or a boron compound in the coolant-moderator will change reactivity and cause the neutron level either to increase or decrease. If the reactor is operating in a low power region where effects such as temperature coefficient feedback are negligible, the neutron level changes resulting from a reactivity change will be exponential. The automatic control system required to maintain a desired neutron level in this region needs to be carefully designed and might involve some rather sophisticated computing circuitry. This is particularly true if it is expected to operate over a number of decades, which would be the case if automatic start-up were required.

However, a nuclear power plant is generally designed for manual start-up and control until the power range is reached. After all, if it operates with the hoped for reliability it will be shutdown and restarted only once a year or even less often. Therefore, the desire for an automatic control system is generally limited to the rather narrow range between 10 and 100% of full power.

Now in this region the typical behavior of the reactor departs from the exponential or logarithmic response which has been studied in considering reactor kinetics without regard to heat production and core temperature. When operating in this region an increase in power level (neutron level) produces an increase in core temperature which, in turn, results in a reduction in reactivity due to the negative temperature coefficient. (This discussion assumes a negative temperature coefficient.)

Thus, the sequence which developes when reactivity is inserted by rod withdrawal is as follows. First the power level increases, producing a core temperature increase. The temperature increase causes a reactivity reduction which eventually cancels out the original insertion. The final result is that the reactor levels off at a new and somewhat higher power level. Thus, in this region the reactor tends to behave more like the "linear" devices which are more common, such as valves, furnaces, etc. In other words, in this region the reactor is inherently stable, as an increase in input (reactivity) causes a compensating reaction.

Due to the characteristics described in the previous paragraph, reactor neutron level control can be accomplished with a relatively simple automatic control system. (Actually a number of power reactors are operated safely and efficiently with no automatic control). Such a system is shown in Figure 16.



SIMPLE NEUTRON LEVEL CONTROL SYSTEM

Figure 16

N 6609

1093 087

In Figure 16, a signal coming from a neutron detector is compared with a neutron level demand signal (from a source to be explained later). The difference is amplified and used to actuate an "UP" or a "DOWN" signal to the selected rod drives. As a result, rod motion at a constant speed begins and continues until the neutron demand signal is satisfied. This form of control has been called "slow speed floating control."

The adequacy and safety of this mode of control depends on:

a. Operation within the range where the negative temperature coefficient is effective

b. A rod drive speed which is sufficiently slow.

To insure item (a), an automatic cutout system is usually provided for low neutron level demand.

9.12.2 Primary Coolant Conditions

The pattern that the temperature, pressures, and flows throughout the plant assume as a function of power output is called the program. Figure 17 indicates a basic plant, and the symbols that will be used are

T_c = inlet coolant temperature to the reactor

T_h = outlet coolant temperature from the reactor

T_{av} = average coolant temperature = $(T_c + T_h)/2$

F = primary coolant flow

F_s = steam flow

P = absolute steam pressure

T_s = steam temperature

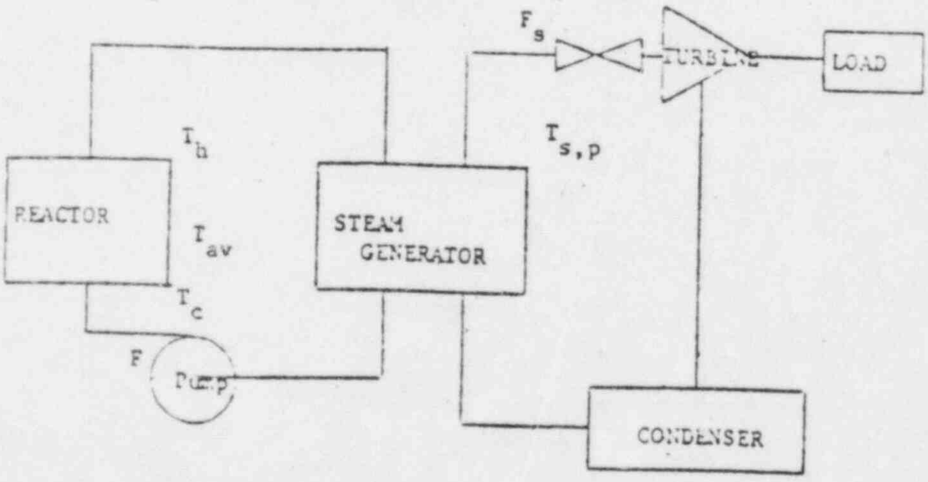


Figure 17

A distinct relationship exists between the temperature and the pressures of the primary and secondary sides of the plant. The difference between the average temperature of the primary coolant and the steam temperature is directly proportional to the power being transferred from the primary coolant to the steam side.

$$P = K\ (T_{av} - T_s)$$

where P is the total power and K is a proportionality factor which depends on the heat-transfer characteristics of the steam generator. The value of K varies slightly with coolant flow rate, thermal power, and boiler level but may be assumed constant for a first approximation. The values of T_{av} and T_s at full power can be determined from saturation data from steam tables when the steam pressure for 100 percent power is specified. As zero output power is approached, the difference between T_{av} and T_s converges to zero and the value at which the temperatures converge may be arbitrarily located on a temperature scale, assuming the presence of a control system to enforce the selection. The point to be made is that once the constants of the primary side of the plant are known, the secondary constants are automatically specified and vice versa. For example, if one desires a plant in which the average temperature of the primary coolant is constant with power level changes, then the temperature program of the steam is automatically specified as dropping off as the power increases. Conversely, if one desires a given program of steam temperature as a function of power level, the average temperature program of the primary coolant is immutably fixed as long as we assume a constant K.

9.12.3 Constant-T_{av} Program

One type of desirable program for a nuclear power plant is the program whereby the average temperature of the primary coolant is constant regardless of the power output. This type of program is shown in Figure 18 for constant primary coolant flow. For reactors having a negative temperature coefficient this is the natural program of the reactor and the one that requires the least amount of external control.

Let us examine the simple plant shown in Figure 17, making the followng assumptions. The steady-state program desired is the one of Figure 18 with constant-coolant flow. The reactor has a negative temperature coefficient, and the plant is operating so that a steady output is being supplied to the load. Now suppose that more output is required by the load. This greater loading causes more heat to be extracted from the heat exchanger, and for a short period of time, the heat capacity of the heat exchanger and coolant can usually supply the additional load. However, this extra energy extracted from the system requires that the temperature of the coolant into the reactor must drop. If the reactor has a negative temperature coefficient, the dropping of the inlet temperature causes a drop in the average temperature, and consequently the reactor will possess more reactivity. If the reactor was initially in a critical state, it now temporarily becomes supercritical. The output temperature of the coolant rises, and more energy is then available from the reactor. Finally, in the steady state the reactor returns to its critical condition with the average coolant temperature the same as it was initially. It will be noted that with-

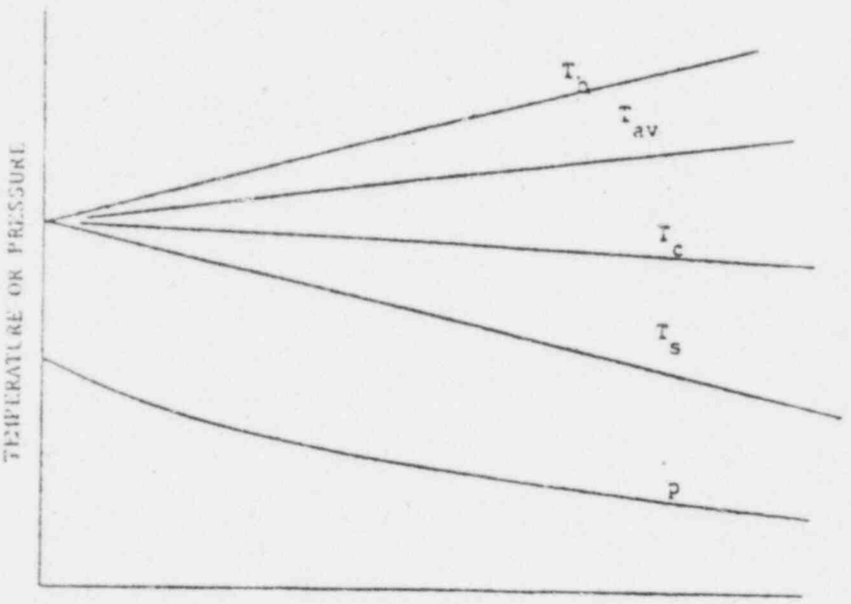


Figure 18

out any control mechanism whatever, the reactor system has stabilized itself about a given average temperature and automatically supplies a reasonable demand placed upon it. In other words, if the plant programming calls for the T_{av} to be constant, no control-rod motions or no external control means are necessary to handle changes in power demand.

The advantages of the constant-average-temperature program are important. If the effects of aging and poisoning could be ignored, no external reactor control would be necessary during normal operation. The reactor would simply follow throttle changes in a stable manner by means of the negative temperature coefficient. Another significant advantage of the constant-T_{av} program concerns the pressurizer. If T_{av} is constant, the volume of coolant in the primary loop is essentially constant and the required size for the pressurizer is a minimum.

However, there are also some disadvantages to this program. The principal one is the large change in steam pressure over the power range, as seen in Figure 18. A wide pressure range means larger and heavier steam piping, automatic throttling devices, special turbines, and boiler-feed pump problems. Furthermore, since the effects of aging and poisoning cannot be ignored, there always must be a feedback control loop to hold T_{av} constant even in a simple plant. This control loop, however, can be very slow and might be manual.

9.12.4 Plant Control Systems

If the analysis of the basic plant indicates that the plant is inherently sufficiently stable, the automatic control problem is simplified. The plant effectively controls itself supplying whatever load is imposed. Any difference between reactor power and steam load results in a rate of change of average coolant temperature. The reactivity change caused by the temperature change then alters reactor power, thus resetting the temperature and the heat balance. The speed of transient response obtained may in itself be adequate for the system application, so that in many cases automatic control of the power level for improved transient performance may not be required. The use of nuclear measurements and the generation of a neutron power demand signal become unnecessary. Thus the role of the automatic system is reduced to that of maintaining the steady-state temperature program.

The requirements for a plant that is inherently sufficiently stable suggest a control system of the type shown in Figure 19. Here the temperature error is controlled directly by insertion and withdrawal of reactivity by control-rod motion. The inherent stability of the plant compensates for any change in reactivity by shifting the temperature level until the net reactivity is again reduced to zero. Thus by moving the control rods a given distance, the temperature shift to a new constant value, and direct control of the program is accomplished with no neutron information being required.

In summary, if the plant has sufficient natural stability, direct control of plant parameters may be attempted. If the plant has poor inherent transient performance, neutron-level control should also be used.

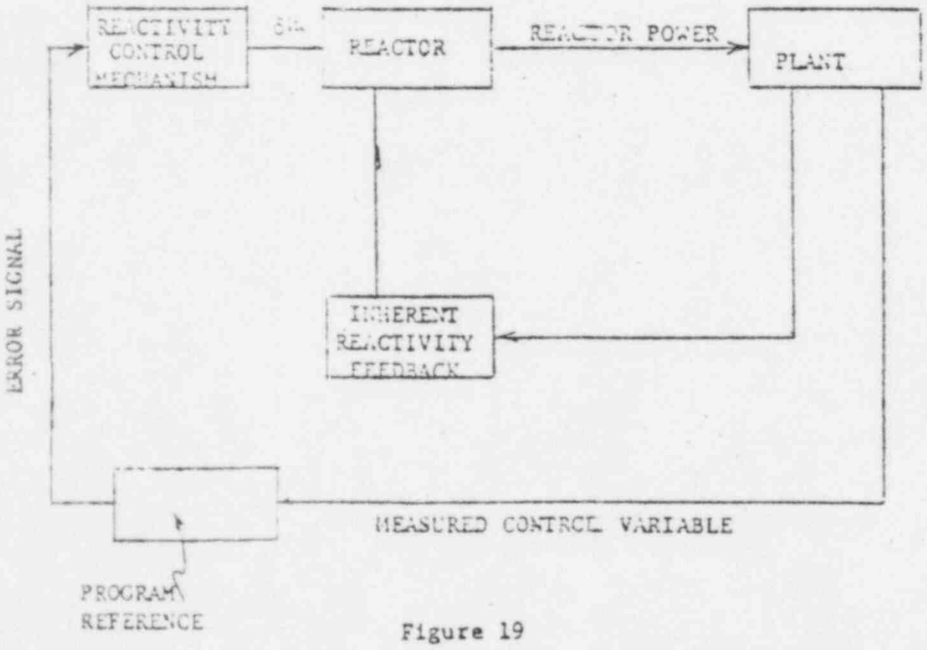


Figure 19

9.12.5 Control Systems for Plants with Insufficient Inherent Stability

The most obvious type of control program is the constant-T_{av} program. As has been pointed out, this is the natural program of the reactor. A few types of control system setups for the constant-T_{av} program will be examined.

Figure 20 indicates two types of control systems. Figure 20A represents a scheme whereby the power demand signal n_o is generated according to the equation

$$n_o = K \quad (T_{av,ref} - T_{av}) + \frac{1}{\tau} \quad (T_{av,ref} - T_{av})\,dt$$

The n_o signal generated in this manner is then compared with reactor power, assumed proportional to the neutron flux level, and the resultant error signal actuates the control rods. The integral of T_{av} error is necessary to provide a steady state n_o signal, since no T_{av} error exists in the steady state. This system, while simple, is at a disadvantage because a poor transient response results from throttle changes. As the throttle opening is increased, steam flow increases immediately, thus withdrawing more power from the primary coolant and lowering the ave. coolant temperature. The reactor power is increased to supply the demand, preventing further depletion of the reservoir of energy stored in the coolant. The sooner reactor power equals the steam demand, the shorter the duration of the transient. The control system of Figure 20A requires that an error in T_{av} exists for a length of time following a throttle change, in order to furnish the necessary change in demand signal n_o, since the accumulated integral of the T_{av} error is the only contributor to n_o in the steady state. In other words, this system behaves pretty much in the same manner as would the natural plant stability by itself. The T_{av} signal ensures that the long-term depletion and poisoning changes are wiped out.

Figure 20B shows a control scheme that does not have the limitations just described. In this system it is again assumed that the enthalpy of the steam generator is essentially constant throughout the power range. Since power at any level is then the product of steam flow and enthalpy the thermal power being delivered is directly proportional to steam flow F_s. The generated demand signal then becomes

$$n_o = K_1F_s + K_2 \quad (T_{av,ref} - T_{av}) + \frac{1}{\tau} \quad (T_{av,ref} - T_{av})\,dt$$

Here the demand signal is nearly directly proportional to steam flow and a change in steam demand immediately changes n_o. The reactor power is adjusted much more rapidly than in the system of Figure 20A, where a temperature error was required for a length of time. The integral of T_{av} error is included to supplement the steam-flow signal in regions where the steam-flow indication may be in error. The value of K_1 should be very close to unity; that is, K_1 = 1 percent reactor power demand per 1 percent steam flow. The scheme of Figure 20B may be called the direct method of controlling n_o; that of Figure 20A the indirect method.

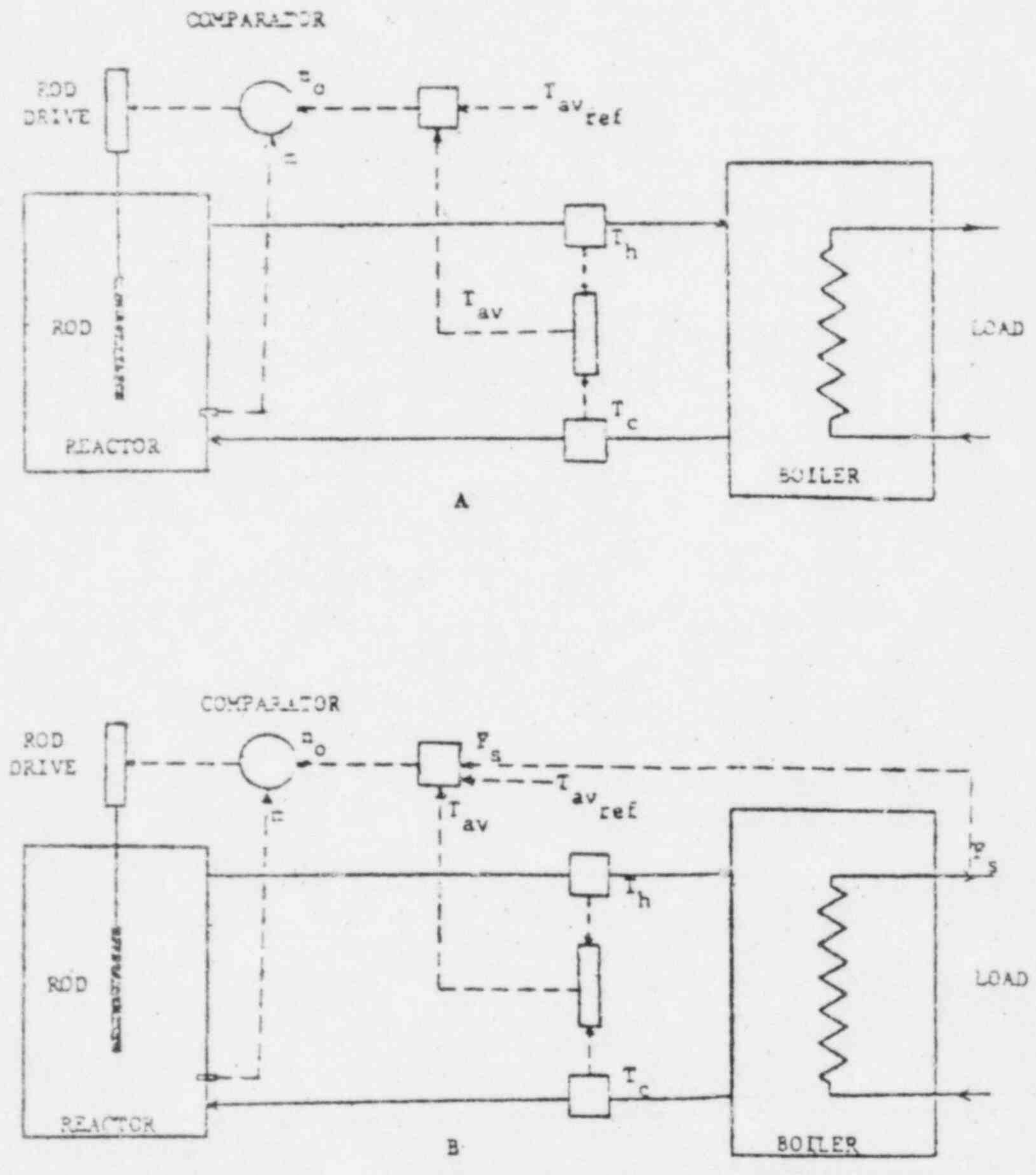

COMPARATOR
ROD DRIVE
n_o
n
$T_{av_{ref}}$
T_h
T_{av}
ROD
REACTOR
T_c
BOILER
LOAD
A
COMPARATOR
ROD DRIVE
n_o
n
F_s
$T_{av_{ref}}$
T_{av}
T_h
F_s
ROD
REACTOR
T_c
BOILER
LOAD
B

Figure 20

In a number of reactor plants planned, the scheme of Figure 203 is used with the change that MW demand is used instead of the steam flow signal (F_s). This is basically the same system except that its transient response is probably somewhat improved, since the lag involved in the reflection of a change in MW demand through the turbine-generator to the steam generator is eliminated.

The preceding discussion of plant control has been completely restricted to a discussion of systems to give constant T_{av}. Other systems, however, are possible. For discussions of these, refer to texts such as Schultz, "Control of Nuclear Reactors and Power Plants."

9.13 Rod Drive Mechanisms

If a reactor instrumentation and control system is studied systematically beginning with the neutron detectors and proceeding through the system, eventually the rod drive mechanisms are reached, they are therefore the last component in the system. (This section will assume that the control rods themselves, or at least their absorber sections, are outside the control system proper).

In discussing rods and rod drives it has been customary to divide them into several different types, depending on their function. In reactors where such a division of function has been incorporated into the design, a difference in the design specifications for the rods and their drives has sometimes resulted. (Such distinctions seem to be disappearing in current power reactor designs, with all rods and drives being identical). Types often considered are:

a. Safety Rod - If a rod or rods are moved to a ready or cocked position where they can be released to rapidly make $k < 1$, the rods are called safety rods. When safety rods are used, there is always a secondary means of control (such as shim rods or boron in the coolant) so that the reactor k will not be equal to or greater than unity when the safety rods are at their position of least effect on k. One or more such safety rods are very common in low power or critical experiment work and are pulled to ready or the cocked position before fuel or moderator is added. A stored energy source is always provided and controlled by a magnet latch so that, upon loss of power, or a shutdown signal, the rod will run rapidly to the minimum k position.

   In practice most rods in all reactors have been provided with rapid insertion capability using stored energy.

b. Shim Rod - shim rods are used to make relatively large adjustments in reactivity. They are used during start-up to bring the reactor to critical, and during operation to offset changes in fission-product poisoning and fuel depletion to keep $k = 1$. For this use, speed is not needed and these rods move slowly.

c. Regulating Rod - there may be one or more rods similar to shims but with limited reactivity control. Any rod so used is called a regulating rod even though its general design may be that of a shim rod. When feedback control is used to regulate flux or power and the designers desire a high performance system, the mechanisms attached to this rod may be very carefully designed, as are servo-mechanisms, for a specified performance.

d. Shim-Safety Rod - space is seldom available for more than one set of rods and the features of the quick release for safety rods and the shim rod control of reactivity are incorporated into one system. This is by far the most practical system and is widely used. When the mechanism must perform both the functions of shim and safety, the design obviously must be more complex.

There is one important advantage of this delsgn which the engineer should recognize. When shim rods are in position, the reactor core is always just critical (i.e., $k = 1$). Even a slight movement of a bank of these rods can quickly reduce k to less than unity. This point cannot be overstressed. Thus, the important safety feature in a shim-safety rod system is the time it takes the rod to move a short distance, say 3 to 6 in.

This chapter will deal with the design of shim-safety rod drive mechanisms. The features of safety rod drive mechanisms are similar.

9.13.1 Basic Drive Requirements

Four principal design requirements are (a) speed, (b) positioning, (c) position indication, and (d) safety release.

9.13.1.1 Speed

The mechanism must move the rod in a direction to increase or decrease reactivity at about 10^{-4} δk per second at the position in the core where the rod is most effective. The drive speed v in inches per second can be estimated from

$$V = \frac{L}{FK} \times 10^{-4} \text{ (in/sec)}$$

where L = core length in inches, K - total rod effectiveness in δk, F = a factor which reflects the rod effectiveness at the opening point as compared with the average effect. The factor 10^{-4} has dimensions δk/sec.

If a curve of rod effectiveness vs rod position is available, the factor F can be determined exactly. A designer might arbitrarily use a conservative value of F=4. This simply says that, when a single rod projects into the core without other rods present, the rod will produce most of its effect when it has traveled 1/4 of its stroke beyond the bank.

Another value to keep in mind is that in a water-moderated power reactor, each group is usually worth about 2% δk. Thus, for a 12-ft stroke, one would get

$$V = \frac{144}{4} \times \frac{1}{0.02} \times 10^{-4}$$

$$V = 18 \times 10^{-2} \text{ in/sec}$$

$$V = 10.8 \text{ in/min}$$

9.13.1.2 Safety Release

A stored-energy source such as gravity or a compressed spring must act on the drive rod during the quick return stroke which reduces reactivity. A magnetic clutch or latch should be inserted between the drive mechanism and the rod which, when de-energized, allows the least amount of mass of the rod system to move freely under the effect of the stored energy. Friction in guide bearings and seals must be considered, and to depend entirely upon gravity is undesirable but widely accepted. When the rod travel is several feet, a positive driving force to reduce reactivity is often used. A pneumatic piston is well suited to this service. Where application of the return force throughout full travel is impractical, a compressed spring, several inches long, can be used to give the rod an initial acceleration of about five times gravity.

Once the rod is set in motion, it must be stopped at its most effective position. The rod can easily be broken in this action; the most severe loading occurs during deceleration. To obtain smooth deceleration, a dashpot is often used. The principal problem is to carry out the deceleration in a distance of less than about 10% of the rod stroke.

9.13.1.3 Positioning

The mechanism must position the control rod accurately and this position must be repeatable. The mechanism should be irreversible so that forces acting upon the control rod cannot move the mechanism. Motors used to power the rod drive mechanisms should stop the drive within about 0.2 sec after the power is turned off. Mechanical tolerances should provide a position repeatability of 2×10^{-5} δk. Repeatability R in inches can then be computed from

$$R = \frac{2L}{FK} \times 10^{-5}$$

9.13.1.4 Position Indication

An essential part of the drive mechanism is a position pickup for use with a remote position indicator located at the reactor control console. A popular device for this service is a synchro transmitter and receiver pair. The accuracy of position indication should equal the position repeatability given above. Limit switches or equivalent indicators such as magnetic detector coils are required at least in the position corresponding to the reactor's minimum reactivity. One or more intermediate indicators are desirable and a maximum-reactivity position indicator is also desirable to show when the rod has ceased to be useful for inserting positive reactivity.

9.13.2 Specific Considerations

Some more detailed points to be considered in drive design are covered in the material to follow.

9.13.2.1 Drive Rod

Control rod mechanisms generally are located on the outside of the reactor pressure vessel. As a result it is quite a distance from the reactor core and the actual absorber or poison section. A "drive rod" is usually used to make the connection. The drive rod must withstand the forces developed during rapid reactivity reduction (or scram), especially the force of deceleration as discussed above. Joints in this section are necessary for assembly and disassembly. Threaded joints are often used but the joined sections must be locked or guided so that they cannot twist and work loose.

9.13.2.2 Mechanism Travel

The ultimate mechanical limits of the drive mechanism should allow about 1 to 5% greater travel than that rod stroke specified by the physics of the core. The exact stroke limits can be set after installation with reference to the control-rod position in the core by adjusting limit switches in the motor control circuit.

9.13.2.3 Motors

Any type of motor--pneumatic, hydraulic, or electric-- can be used for a rod drive but the easiest to apply is often an electric motor.

Direct current motors offer two advantages: simple speed control, and dynamic braking by simply shorting armature terminals. In reactor applications to obtain these advantages, dc motors are operated with the field continuously energized, control switches being used in the armature circuit only. Direct current motors are easily incorporated into feedback control systems because of the adjustable speed feature. The chief disadvantages are brush maintenance and the necessity of providing dc power; both disadvantages are easily overcome.

Two-phase ac motors also provide the advantage of speed control and are especially suitable for use in a feedback control system. In addition, commercial two-phase servo motors offer a high ratio of torque to armature-inertia ratio which makes a mechanical brake unnecessary. The gear ratio between motor and drive is made high and if there is a reasonable friction loss in this gearing, no mechanical brakes are necessary. This type of motor has been used for experimental reactors where power requirements for the rod drives are less than 1/4 hp. On power reactors the drives require larger motors; three-phase motors are more economical, as long as speed variation for three-phase motors is not required. Three-phase ac motors have a low torque-to-inertia ratio as compared with two-phase motors so that a solenoid-operated friction brake on the motor shaft is almost mandatory to stop the rod within a reasonable time.

The chief advantage of both types of ac motors is that the maximum speed is fixed by the line frequency and not by the applied voltage. Thus, with constant line frequency we can guarantee the maximum speed under any conditions.

9.13.2.4 Position Measurement

Position measurement is important to safe reactor operation. Both limit switches and continuous position indicators should be attached to the rod to ensure that some position indication will always be available. In an emergency it is most important to know where the rods are. Any position indicator signals must be carried to the control room.

In attaching a pickup unit for rod position indication, the most direct method is preferred, for example, a rotating shaft geared directly to the rod drive shaft. The transmitter of a synchro pair attached to this rotating shaft should be geared so that 350 degrees or less of rotation at the transmitter represents full rod stroke thus eliminating ambiguity at the receiver. The typical accuracy of position indication for this arrangement, for example, is ± 1/350th of the rod stroke. This is normally not accurate enough for experimental and research reactor operation and a second fine-position transmitter is used which turns 360 degrees for each 1 to 10 inches of rod movement. Provision should be made on the position-indicating shaft to install several cams to operate limit switches. Control-circuit interlocking often requires not only switches operated at the travel limits but also at some intermediate rod positions.

9.13.2.5 Magnetic Holding Devices

Magnetic latching of the mechanism with the current "on" to hold the rod in position provides desirable fail-safe rod release. Holding magnets are often used on both shim safety and safety rod drives to provide for fast-action safety release of the rod from the drive mechanism. By employing mechanical advantage, a solenoid latch with a holding force of 40 to 80 lb can be used for rods weighing up to 1000 lb. Roller latches reduce friction and dc magnets provide a more certain holding force. The latch linkage should be designed so that the return force on the rod acts to open the latch even if the latch spring should fail. Holding magnets are specially designed dc powered and usually clad with stainless steel for protection of magnetic material and lead wires. The magnet wire is often covered with glass insulation. Holding force of the magnet is about 150 lb. Mating faces on rod and drive members are designed for quick release.

The magnets previously described are generally used in that portion of a rod drive train where the motion is linear. In cases where the linear rod drive motion is derived initially from the rotary output of a motor of some type, with shafting connecting the motor to a rack and pinion or comparable mechanism, another possibility exists. This is to install int he shafting a conventional friction clutch, electrically operated. The design is such that when the clutches are de-energized for emergency shutdown the motor and gear head are disengaged allowing stored energy from a spring or gravity to drive the rod.

9.13.2.6 Shock Absorber

As discussed earlier, a shock absorber is necessary to stop control rods. Specially designed, they can use air or oil where stopping action is applied to the rod drive mechanism, or use reactor coolant as compressive medium where the rod itself is released from the drive. Commercial units are available with a variety of force vs distance characteristics suitable for stopping rods. To ensure sufficient fast rod travel, the shock absorber action usually is limited to about 10% of rod stroke.

9.13.2.7 Penetration of High Pressure Systems

The aspect of drive mechanisms which has caused the most design effort is the need to penetrate the high pressure vessel of power reactors. A mechanism suitable for a pool type research reactor operating at room temperature and pressure can be a very simple rack and pinion drive, driven by a small electric motor. The problems involved in producing control rod motion

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inside a PWR operating at 2000 psi are much more interesting. Several approaches to this problem will be described.

9.13.2.7.1 Magnetic Coupling

One method of producing rod motion without any actual break of the pressure vessel is to create a movable magnetic field external to the vessel which will exert a force on a magnetic armature which is inside the vessel and attached to the control rod itself. A simplified sketch of such a system is shown in Figure 21.

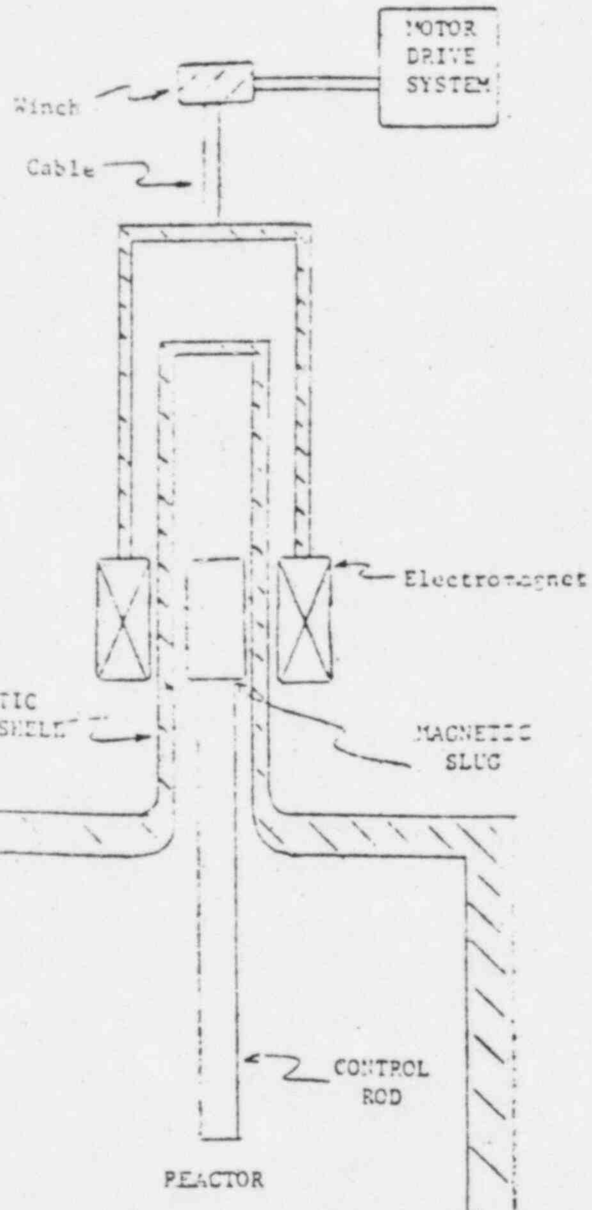
9.13.2.7.2 Linear Shaft Seals

The rod which attaches to the absorber section can be brought directly through the reactor head (top or bottom) if suitably linear motion seals are provided. If this is done a variety of drive mechanisms can be selected with relative freedom. One point that must be considered is that in such a design there is generally an area (such as that of the drive rod) upon which the system pressure acts and attempts to force the rod out of the reactor. This force must be restrained by the rod drive mechanism.

There has been a tendency to avoid the use of such seals in reactor systems due to the possibility of leakage of primary coolant. However, their use is increasing. If a number of such seals are installed in series along the shaft the leakage may be made very small, and may be collected if desired. To be even more conservative an external pressure higher than the system pressure may be applied to the external end of the seal. This will cause any leakage to be into the system rather than out of it.

9.13.2.7.3 Rotary Shaft Seals

Seals similar to those referred to in the previous case can be used with a rotating shaft. If motion is transmitted inside the pressure vessel in this fashion, the device which converts rotary motion to linear motion must be located inside the pressure vessel. Other considerations for rotary seals are similar to those for linear seals.



MAGNETIC ROD DRIVE

Figure 21

9.13.2.7.4 "Internal" Drive Motors

Motors have been designed which are capable of operating in the water conditions of a PWR, using techniques similar to those used in the design of canned primary coolant pumps. With such a motor, a drive mechanism can be installed completely <u>inside</u> the primary coolant system. Thus, the only penetration necessary is for the control and power wiring to the mechanism.

9.13.3 Typical Drive Mechanisms

To illustrate some of the points which have been discussed, several current designs will be described.

9.13.3.1 Yankee-Rowe

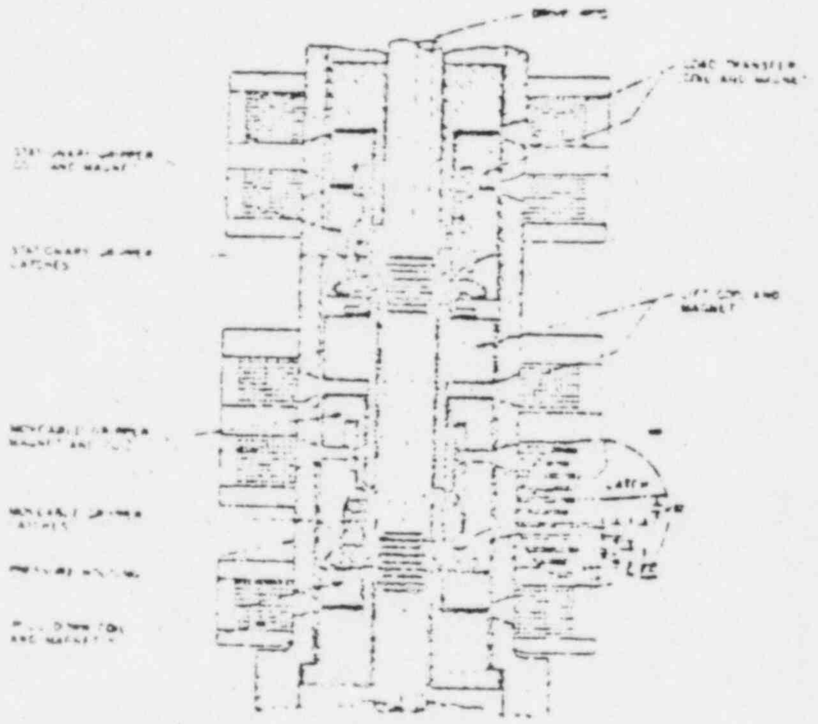
The control rod drives are located on top of the reactor vessel. A positive-grip type magnetic jack mechanism designed and developed by Westinghouse is used to drive the control rods. The general arrangement of the mechanism is shown in Figure 11.

In the positive-grip, magnetic jack mechanism, the components which operate in the high pressure, high temperature main coolant system are the drive shaft that connects to the control rods, the movable and stationary grippers, and the solenoid plungers that actuate the grippers. All of these parts are enclosed within a pressure housing which is seal-welded to the head adaptor on the reactor vessel head.

The electromagnetic coils that actuate the solenoid plungers and, in turn, the grippers are mounted on the outside of the pressure housing. There are five sets of coils: the stationary gripper coil, the movable gripper coil, the lift coil, the pull down coil, and the load transfer coil. The action of these coils is programmed so that the stationary gripper holds the driven shaft while the movable gripper is relocating to raise or lower the control rod through one 3/8 inch step.

The relative positions of the movable and stationary grippers are such that the latches engage and disengage the drive shaft grooves without load.

The operating cycle is programmed by a motor-driven cam-switch arrangement. A drive with a fixed speed limits the control rod rate of travel.



Yankee-Rowe Drive Mechanism

Figure 22

9.14 Process Instrumentation and Control

It is the intent of this section to discuss the measurement and to a very limited extent the control of certain "non nuclear" reactor plant parameters. This discussion will, in general, be limited to requirements which might arise in the primary system of a pressurized water reactor, although much of the content is applicable to other plant systems, and other reactor types.

This area of reactor plant instrumentation and control is sometimes referred to as "process instrumentation and control."

9.14.1 Temperature Measurement

A nuclear power reactor is an energy producing system which releases heat to the working fluid (such as water in a BWR or PWR). This absorption of heat by the working fluid results in a temperature increase. As a result, accurate temperature measurements are extremely important in the evaluation of reactor performance. Such measurements are equally important in determining efficient and safe operating conditions in all of the plant systems.

Before proceeding to discuss methods of temperature measurement, the basic units of temperature measurement will be reviewed.

9.14.1.1 Temperature Scales and Units

A change in the temperature of a substance will occur when energy is lost or gained by the substance. This change of temperature can be detected by an instrument such as the familiar glass thermometer, which is filled with an opaque liquid whose density changes with temperature, or by a thermocouple whose electrical output changes with temperature.

For most common uses the familiar glass thermometer is used. It has a bulb, filled with mercury or alcohol, connected to a fine tube into which the fluid can flow as it expands when the temperature increases. The scale of a thermometer may be one of several types. In this country, the Fahrenheit scale is generally used for ordinary purposes, but the most common scale in scientific work is the Centigrade. On the Centigrade scale, the 0° mark is the point where the column of fluid stands at the freezing point of water, and the 100° mark is where it stands at the boiling point of water under one atmosphere of pressure. On the Fahrenheit scale, the freezing point of water is 32°, and the boiling point of water under one atmosphere of pressure is 212°. In the Fahrenheit scale, there is a difference of 180° (212°-32°), and in the Centigrade, a difference of 100 degrees (100°-0°) between base points. Thus, one Centigrade degree is equal to 1.8 Fahrenheit degrees. To convert readings from Centigrade to Fahrenheit or vice versa, it must also be remembered that the two scales begin at different reference points. On the Centigrade scale, the freezing point is zero degrees, but on the Fahrenheit scale, the freezing point is 32 degrees above zero. It should be obvious then that in order to obtain a relationship between the two scales related to the size of degrees, it is necessary to subtract or add 32 degrees to obtain the point where the respective scales begin.

Keeping these facts in mind, it is possible to write the following equations. To convert Fahrenheit to Centigrade, the equation

$$°C = (°F - 32)/1.8$$

is used, and conversely to convert Centigrade to Fahrenheit another form of the same equation,

$$°F = 1.8\ °C + 32$$

may be used. Therefore, to convert a temperature of 43°C to °F:

$$\begin{aligned} °F &= (43° \times 9/5) + 32° \\ &= 77.4° + 32° \\ &= 109.4° \end{aligned}$$

Two different, but related, temperature scales should be mentioned. These are the Rankine and Kelvin scales. These are derived from the Fahrenheit and Centigrade scales, respectively. The difference is that to either a Fahrenheit or Centigrade reading a constant is added which refers the temperature to "absolute" zero which is the temperature level at which all molecular motion ceases, and which is considered the lowest theoretical temperature possible. To convert to these scales:

°K (degrees Kelvin) = °C + 273°

°R (degrees Rankine) = °F + 460°

While the Kelvin and Rankine scales may seem to be of academic interest to the power reactor operator, there are several reasons why they are of considerable importance in the nuclear industry.

For one thing, temperature must usually be expressed on either the Kelvin or Rankine scales when being used in thermodynamic laws and equations. Another, more restricted, need is in the work related to development of nuclear rocket engines. In this case, both the temperature of the major propellant (liquid H_2) and the environmental temperatures involved in outer space are not far above absolute zero, and the use of these scales becomes convenient.

Now, to proceed to power reactor temperature measurement. It is usually not feasible to use a common liquid filled glass thermometer. System pressures are often high for glass, and many points where temperature measurements are required are quite inaccessible, and the measuring device must be capable of transmitting a signal for a considerable distance (100 to 1000 feet).

9.14.1.2 The Thermocouple

In 1821, Seebeck observed that if, in a circuit consisting of two wires of dissimilar metals joined at both ends, one junction is at a different temperature than the other, a voltage difference will exist between the two wires causing a circulating current to flow. For certain metals at least, this voltage difference is closely proportional to the difference of the temperatures of the hot and cold junctions. In the Appendix of this section, a variety of information concerning the various common types of thermocouples is given including tables giving output vs temperature.

9.14.1.2.1 Thermocouple Construction and Installation

The term "thermocouple" is used rather loosely to identify various portions of a thermoelectric measuring system but for now assume it means the first element in the system, the hot junction and its immediate assembly. This assembly, as found in a reactor plant system generally consists of two lengths of wire of the materials desired, long enough to reach from the point of measurement (perhaps the center of a primary coolant pipe) to a point suitable for terminal facilities (the outside of the pipe). At the in-pipe end the two wires are welded together to form the hot junction. The wires leading from the hot junction are insulated by magnesium or aluminum oxide with an outer stainless steel sheath. To withstand pressure in the pipe, the sheath must either be welded to it, or provided with some type of pressure fitting.

The hot junction may be exposed, included in the sheath. If it is included in the sheath, it may be either insulated from it or grounded to it. The latter arrangement gives better mechanical and thermal properties, but can complicate the electrical connection of the thermocouple.

The entire assembly previously described may be arranged to slip into a well which is permanently installed in the pipe. This makes replacement of the thermocouple much simpler, but can introduce a time lag into the response. This lag should be minimized by making certain any annular space is filled with a good conductor of heat.

Another means of measuring the fluid temperature within a pipe is to simply weld or clamp the hot junction assembly to the outside of the pipe. This gives a measurement which is not quite representative but eliminates the problem of penetrating the pipe.

9.14.1.2.2 Thermocouple Extension Wire

Ideally the thermocouple wires would be carried all the way to the measuring instrument without any joints or discontinuities. However, this may become costly depending on the length of the run and the price of the wire. Also, it may introduce excessive resistance into the circuit. To avoid these problems, "extension wire" is available for some types of thermocouples, which is made of less expensive material, but is similar thermoelectrically (at least over a specific range).

9.14.1.2.3 Cold Junction Compensation

As previously mentioned, the signal obtained from a thermocouple system is inevitably a function of the difference between two temperatures. If a measurement is made with normal equipment, the operator must measure the cold junction temperature and correct for it. The cold junction may also be controlled at some known temperature such as in an ice bath. This simplifies the correction (or eliminates it if thermocouple data based on the controlled temperature is used).

Where continuously indicating and/or recording equipment is involved, the equipment usually includes a means for continuously adjusting the zero of the instrument to correct for changes in cold junction temperature. This is commonly by means of a temperature sensitive resistor (perhaps nickel) at the proper point in the circuit. The need for this can be eliminated by locating the cold junction at a point of known, unchanging, temperature. One advantage of this technique is that if there is such a point near the hot junction, plain copper conductor can be run from this region to the instrument.

A final observation on cold junction compensation. In many cases considerable effort is expended in either correcting manually for cold junction temperatures or in designing automatic correction into a circuit. This effort may be unnecessary in some thermoelectric systems. For example, if a temperature of 2000°F is being measured, with the cold junction located in a controlled temperature instrument room, normal variations in cold junction temperatures will be invisible in the measurement.

9.14.1.2.4 Thermocouple Instrumentation

Thermocouple output, and therefore temperature, in many cases is measured by a rather simple millivoltmeter type instrument with the proper sensitivity and calibration. Such a system is adequate for many applications although it has several drawbacks. One is that the sensitive meter movement is deficient in developing force to operate control mechanisms. Also, it requires current from the thermocouple and therefore calibration is affected by IR drops in the thermocouple and its extension wire.

A more sophisticated approach is to use a "potentiometer" or "null-balance" type instrument. A much simplified schematic of such an instrument is shown in Figure 23A. The thermocouple voltage is compared with that from an adjustable slidewire, and the slidewire is adjusted by a servo amplifier and motor until the two are equal. The mechanical motion produced can be used to move an indirect pointer or a recorder pen. Since in the steady state condition this circuit does not draw current from the thermocouple, lead length does not affect calibration.

Another type of circuit which is coming into use is shown in Figure 23B. This circuit is often drawn so that it looks considerably different from that of the traditional self balancing potentiometer but they are basically similar. As shown in Figure 23, the basic difference is that instead of using the amplifier to drive a motor-slidewire unit to balance the input voltage, it supplies current directly to a resistor which generates the balancing voltage. This circuit can be compared to that shown in Figure 13, Page 9-35, for current measurement. That could be considered to be a current feedback system, whereas this is a voltage feedback system.

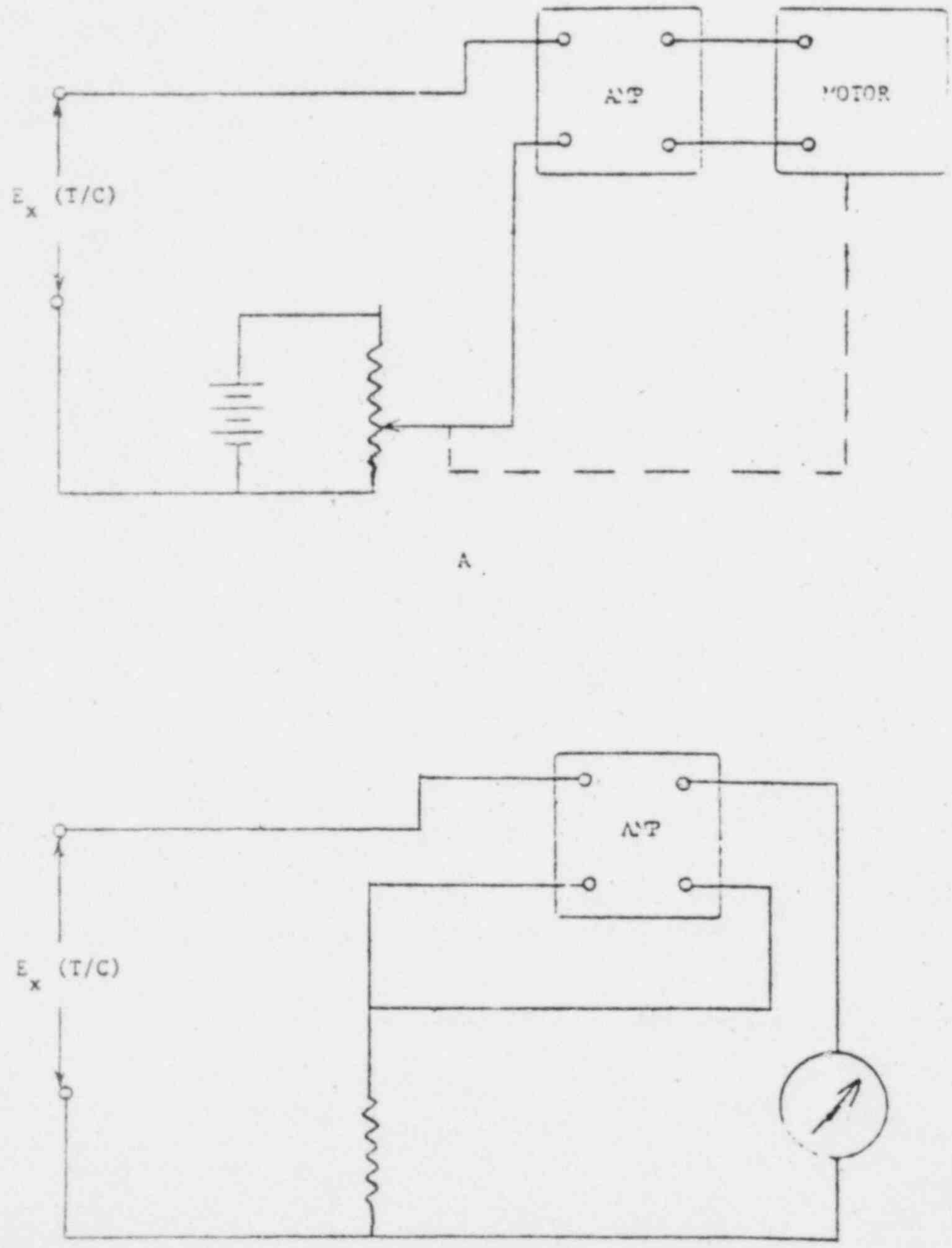


Figure 23

9.14.1.2.5 Thermocouple Computing Circuits

Two or more thermocouples can be connected in various ways to perform simple arithmetic operations.

a) they can be connected in series adding to sum two temperatures

b) they can be connected in parallel to average two temperatures

c) they can be connected in series opposition to take the difference of two temperatures

The last application is important to reactor systems because the difference of reactor inlet and outlet temperature is a function of reactor power outlet (assuming constant flow). These computing operations are generally not possible if the function is grounded.

9.14.1.3 The Resistance Thermometer

A resistance thermometer is a length of wire, usually coiled to reduce its size, which is made of a material whose resistance varies substantially, reproductibly, and if possible, linearly with temperature. Typical materials are nickel, and platinum. To obtain a measure of temperature it is necessary only to measure the resistance of the thermometer. No cold junction problems are involved. No extension wire problems exist, except that depending on the type of measuring circuit used it may be necessary to keep the lead resistance below some value. Considerations on installation are similar to those given in Section 9.14.1.2.1 for thermocouples. Resistance vs temperature data for several types of resistance thermometer are included in the Appendix of this section.

9.14.1.3.1 Resistance Thermometer Instrumentation

The resistance of a resistance thermometer is generally measured using some form of bridge circuit. This may be the familiar laboratory type Wheatstone bridge but as encountered in a reactor instrumentation system it is more apt to be a continuously "self balancing" circuit analogous to the potentiometer circuit shown in Figure 23A. Such a circuit is shown in Figure 24. Note the manner in which three conductors are used to connect the resistance thermometer to the measuring instrument. This places the lead resistance of one conductor in both sides of the bridge circuit. This tends to correct for variations in lead resistance.

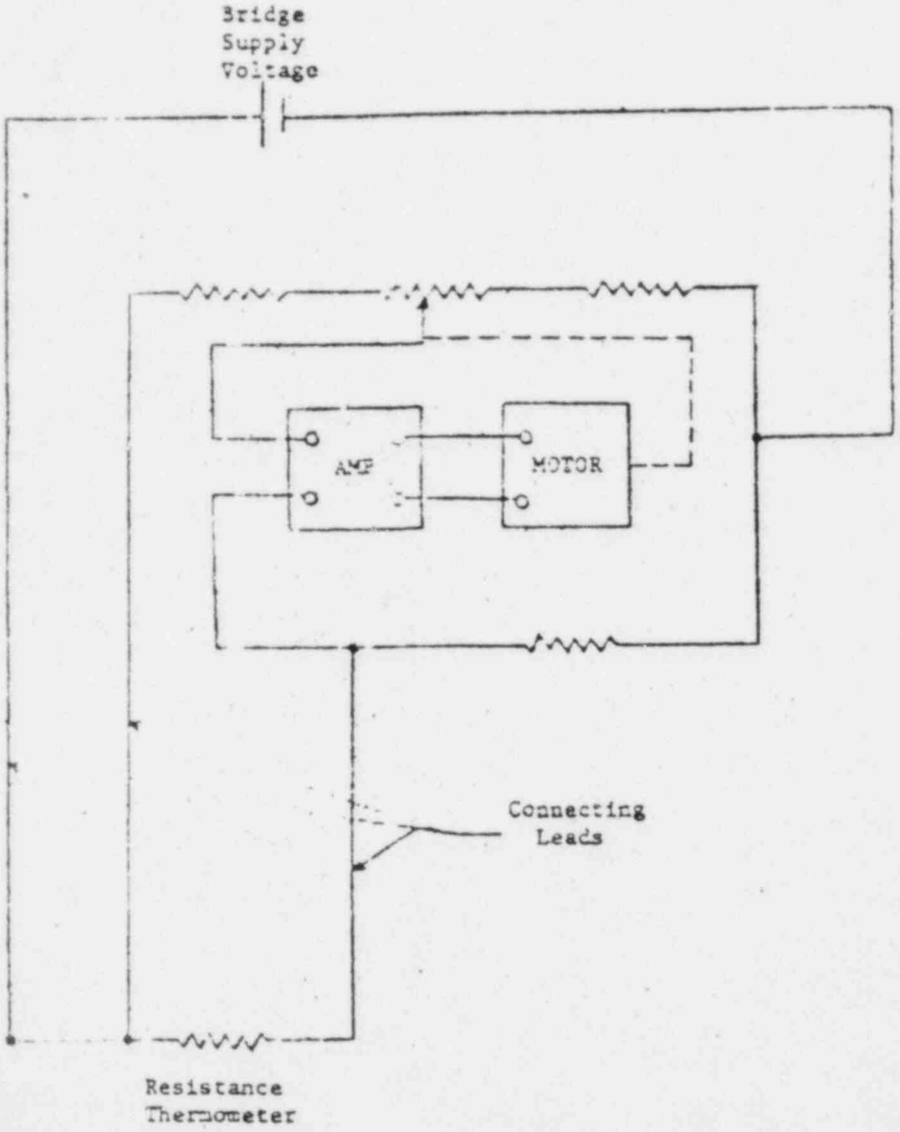

Bridge Supply Voltage
AMP
MOTOR
Connecting Leads
Resistance Thermometer

Figure 24

Another point might be noted concerning this circuit. It involves only the balancing of resistance and, therefore, changes in the value of the bridge supply do not affect the calibration. This is not true in the EMF measuring circuits shown in Figure 23. If it is desired to use a resistance thermometer system to measure differential temperature, it can be accomplished by installing a thermometer in two of the legs of the bridge circuit.

Resistance thermometer circuitry can also be incorporated into non-mechanical type measuring systems, analogous to that shown in Figure 23B.

9.14.1.4 Other Methods

A variety of other methods of temperature measurement may be encountered in the operation and maintenance of a reactor plant. However, these covered are the most likely to be encountered in the normal control room instrumentation.

9.14.2 Pressure Measurement

Pressure is force applied to an area. In a reactor, many different forces are at work. It is necessary to measure the pressures of gases and liquids. The reactor is essentially a closed system and therefore subject to internal pressure changes. Liquids are forced, under pressure, throughout reactor systems, and it is important that these pressures be observed and recorded. Helium pressures, air pressures, and others are also measured. As do temperatures, pressures keep the reactor operator informed as to the status of the various reactor systems. They provide a method of observing many internal changes, and also may serve to operate safety devices should extreme conditions arise.

9.14.2.1 Pressure Scales and Units

In measuring pressure, two different scales are in general use. These are the gage system and the absolute system. The only difference between the two systems is the reference point for zero pressure. In the absolute system, zero pressure is a complete vacuum, whereas in the gage system zero pressure is equal to the pressure of the earth's atmosphere. The absolute system of measurement is usually employed in scientific work, particularly when a high degree of accuracy is required.

Since there are several units of force and several units of area, a variety of units of pressure have evolved. One common unit of pressure is "pounds per square inch". In scientific work it is convenient and accurate to measure pressure in terms of the height of a column of fluid which can be supported by a given pressure. One of the simplest instruments used is a barometer. A barometer may be constructed by filling a glass tube, about 80 cm long (and closed at one end), with mercury, and then inverting it and thrusting the open end into a well of mercury. The mercury column falls until its weight equals the weight of the column of air of cross section equal to that of the mercury column and extending from the surface of the mercury in the well to the top of the atmosphere. As the weight of the air changes, owing to changes in the atmospheric pressure, the level of the mercury in the tube moves up or down. The height of the mercury column, read from a meter stick placed alongside, thus serves as a measure of the atmospheric pressure.

9.14.2.2 Pressure Instruments

The manometer is one of the simplest pressure-measuring devices in use. Basically, this device is usually a U-shaped glass tube that is partially filled with mercury, water, or a suitable material. If a pressure is applied to one side of the column, the level in that leg drops, and rises on the opposite side. The difference between these two levels is directly proportional to the applied pressure. Considering the applied pressure as difference in the height, the device tells us that a given pressure is capable of supporting a column of inches, centimeters, or other units of height. The column may then be calibrated to read in terms of the desired pressure units. Such a pressure will by necessity be read in the gage system, as one leg of the U tube is subjected to the atmospheric pressure. To provide for readings in the absolute system, it would be necessary to close one end of the tube, and seal it off from the atmosphere. This closed end must be completely evacuated or the pressure of the trapped air must be taken into consideration in the determination of the applied pressure. The pressure itself may be computed by multiplying the density of the liquid by the difference in height.

A second method of measuring pressure employs the use of a diaphragm mechanism. This unit consists of sealed box or case, pressurized positively, and containing a thin piece of corrugated stock, made from a material such as brass or stainless steel. The corrugated stock forms the actual diaphragm, and when attached to a pressurized system, the diaphragm will bulge either inward or outward, as the pressure varies with respect to atmospheric pressure. This unit is capable of measuring either positive or negative pressures. The bulging of the diaphragm may be related to a pressure reading by installing appropriate mechanical linkage between the diaphragm and a pointer and scale. As a force is applied, the diaphragm moves and the motion is translated to the pointer, which moves across a scale calibrated to read in terms of pressure.

A third pressure-measuring device is a bellows. This unit operates in much the same manner as the diaphragm. The device is enclosed in a sealed case, and a spring mechanism is used to control the amount of expansion or contraction with changes in pressure. The spring provides a greater flexibility in the range of pressures that may be measured, as it may be adjusted to make the desired range either high or low. As either a positive or negative pressure is applied, the bellows move in or out, and the movement is translated to a mechanical device, which moves a pointer across a calibrated scale. The bellows has a greater mechanical advantage of motion than does the diaphragm. Pressures are normally read in the gage system, but may also be measured in the absolute system, by enclosing the bellows in a case from which all air has been evacuated.

A final pressure-measuring device to be considered is the Bourdon tube. Basically, this device consists of a spiral tube, normally elliptical in shape, and appropriate mechanical equipment. The tube is closed at one end and then curved into a spiral form. When a pressure is applied to the open end of the Bourdon tube, the spiral shape tends to unwind. A spring mechanism may be attached to a pointer, which translates the unwinding of the tube to a scale, which is calibrated to read in terms of pressure. The pressure is read in the gage system, and the Bourdon tube may be used to measure pressures both above and below atmoshperic.

9.14.2.3 Differential Pressure

As was true of temperature measurement, differential pressure can provide certain important information on reactor behavior. Three of the devices just discussed, namely, the diaphragm, manometer and bellows, adapt readily to this type of differential measurement. In the case of the manometer, the two pressures to be measured are connected to the two ends of the U tube. The difference in height of the two legs then becomes an indication of pressure differential.

In considering the diaphragm and bellows, the situation is somewhat different. If the diaphragm is used, the two pressures are applied to the opposite sides of the diaphragm. The same arrangement will work with the bellows, but generally a separate bellow is used for each of the two pressures to be measured.

9.14.2.4 Transmission of Pressure Measurements

Methods of using pressure or differential pressure to produce a mechanical motion have been described.

If the point at which indication is desired is near the point of measurement, small lines containing the fluid can be run to one of these pressure sensitive devices and the resulting mechanical motion used to deflect a pointer on a scale. However, in a nuclear power plant distances between the point of measurement and the control room are too great for such a simple system. In actual systems it is generally desired to convert the motion produced by the pressure sensitive element to an electrical signal.

One method of doing this is to utilize the change in resistance in a length of wire when it is stretched or contracted (by the movement of the pressure resistance element). If four such wires are activated by the pressure sensitive element, an electrical bridge circuit can be connected to give an unbalance signal proportional to the applied pressure. This forms what is usually known as a strain gage type transducer.

Another method of producing an electrical signal from the motion of a pressure sensitive device is to use it to move a metallic slug which will vary the coupling between two or more transformer coils. By its nature such a system is limited to use with an AC system, which makes it susceptible to errors produced by inductance and capacitance in the wiring.

9.14.2.5 Importance of Pressure Measurement

Accurate measurement of pressure, and the resulting control actions are of paramount importance in a PWR. If the pressure increases beyond a certain point, certain irreversible relief devices are provided such as safety valves and rupture discs, which while protecting the primary piping produce an inevitable shutdown of the reactor. If pressure decreases the possibility exists that boiling will begin, which may produce a sufficient change in the heat transfer from the core to cause damage to fuel.

In addition to this direct importance of pre pressure measurement, it will be seen that the differential pressure measurement plays an important part in the most common methods of measuring both flow and liquid level, two other parameters of great importance in reactor operation.

9.14.3 Flow Measurement

Flow is a parameter of great interest in reactor operation, particularly the flow in the primary coolant system. A knowledge of this flow is essential in making one type of colorimetric determination of reactor power level. A decrease in primary coolant flow below some level without a corresponding decrease in reactor power generation will always cause damage to the reactor core. While such a decrease will be reflected in temperature and other measurements, the flow measurement generally responds first. For this reason the primary flow instrumentation usually provides one or more signals to the reactor safety system, to cause reactor shutdown or power reduction.

The aspects of flow measurements to be discussed will all relate to liquid flow, as found in the primary cooling system of a PWR, although some of the techniques and devices may be applicable to gas and vapor flow. All of the devices to be discussed will provide a measurement of volumetric flow, in units such as gallons per minute. Thus, if a knowledge of mass flow rate in units, such as pounds per hour, is desired, a correction for specific gravity or density of the fluid is required. It might be noted at this point that the change in density of water over the range 70°F to 600°F is substantial.

9.14.3.1 Flow Measurement by Head Meter

Head meters are based on the fact that a fluid stream under pressure possesses energy, or "head," in several forms. This energy can be converted from one form to another (and back). Thus, if a restriction is placed in a line, the fluid velocity or velocity head increases while the pressure or pressure head decreases. Once the restriction is passed the energy will return to its original distribution except for friction type losses due to turbulence.

If, in a line with a restriction, a pressure tap is placed at the high pressure (low velocity) point prior to the restriction and another at the low pressure (high velocity) point within the restriction, and the differential pressure between the two is measured, a measure of the flow rate will be obtained. The relationship involved is:

$$Q = K\sqrt{H}$$

where Q is the flow rate, H is the differential pressure, and K is calibration constant. The square root relationship results from the fact that the velocity head (kinetic energy) is proportional to the square of the velocity.

The actual devices used to measure and transmit the differential pressure produced by the various devices to be discussed will not be included since the measurement of differential pressure has been considered in Section 9.14.2.3. Note that if one of these devices is used to measure flow, the calibration of the receiving instrument will be non-linear in square root fashion unless a square root extracting element is included in the system.

In comparing the various head meter devices the distinction between "pressure drop" and "pressure loss" should be made. Pressure drop is the pressure differential across the device most of which is recovered. Pressure loss is that portion of the drop which is not recovered. Pressure loss can be of concern in a system if it is appreciable with respect to the total driving pressure.

Head meters are the least expensive means of measuring flow, particularly where the line size and flow are large. As a result, they are usually what is found in a reactor primary (and secondary) system.

Various devices specifically designed to produce a pressure drop will be discussed. However, any component in the loop which represents a restriction can be used to measure flow if suitably calibrated. This applies to the steam generator or the reactor core itself.

9.14.3.2 Orifice Plate

An orifice plate is a thin disc with a hole which is located between flanges in a line, as shown in Figure 25A. This is the simplest, least expensive, and most commonly used of head meter devices. Its major drawback is that the turbulence downstream results in the greatest pressure loss of any of the head meters.

9.14.3.3 Venturi

This is shown in Figure 25B. As might be expected from its contours, it produces the least turbulence of any of the head meter devices and therefore gives the lowest pressure loss. However, it is quite expensive and is generally avoided.

9.14.3.4 Flow Nozzle

The flow nozzle is shown in Figure 25C. It is essentially the front half of a Venturi. It provides a desirable compromise between a Venturi and an orifice plate, and is used frequently.

9.14.4 Liquid Level Measurement

The need to measure liquid level arises at many points in a nuclear power plant. However, the system which is specifically being considered, the primary coolant system of a PWR, is completely filled with liquid water and no level measuring problem exists, with one extremely important exception. This is the pressurizer vessel, in which electrical immersion heaters are used to boil the water and create a steam bubble. The pressure of the system is determined by the pressure in this vessel, and is controlled by controlling the heat supplied to the immersion heaters. It is essential to maintain the level of the steam-water interface in the pressurizer within certain limits. If the level were to fall too low, the heaters would be uncovered and would overheat. The steam filled volume in the upper portion of the pressurizer vessel is relied upon as a pressure transient absorber. If this volume were to become too small, as a result of high liquid level in the pressurizer, a pressure surge might trip the safety valves causing blowdown of the system and requiring shutdown of the reactor.

As a result of these undesirable consequences of extreme variations in pressurizer level, the instrument system which measures it generally initiates the maximum available corrective action. In the case of high level, a letdown system is generally provided to allow liquid to bleed from the primary coolant system, which will result in a drop in pressurizer level. In the case of low level, makeup pumps are called upon to add liquid to the system.

9.14.4.1 Liquid Level by Differential Pressure

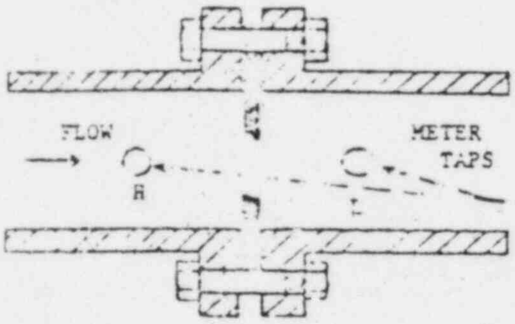
A common method of pressurizer level measurement involves the use of a differential pressure transducer, previously described.

In an unpressurized vessel, open to the atmosphere, the gage pressure at the bottom of the tank is directly related to the height and density of the liquid in the tank. Thus a convenient way of measuring the level is simply to make a pressure measurement at the bottom of the vessel and calibrate it in units of level, such as feet or inches. To do this, of course, the density of the liquid must be fixed and known, or a density correcting device must be added.

In the case of a pressurized vessel, the problem becomes more complicated. The static pressure in the system can easily be so large that pressure fluctuations at the bottom of the vessel due to level changes are negligible. This is indeed the case in a PWR pressurizer where the static pressure is greater than 2000 psi, while level changes on the order of fractions of a foot (which means tenths of a psi) are of interest.

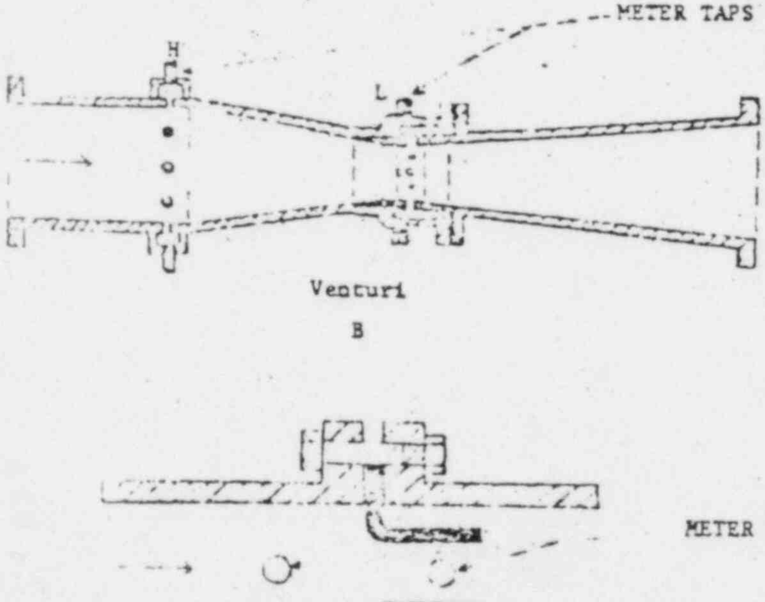
To overcome this problem, the differential pressure between a tap at the bottom of the vessel and one entering the steam bubble at the top of the pressurizer is measured. The upper tap is exposed to only the static system pressure, whereas the tap at the bottom has system pressure plus that due to liquid height. If the difference between these is measured, the resulting signal is proportional to liquid height only.

To assemble the system previously outlined, equipment might be assembled as shown in Figure 26A. However, certain problems exist in such a system. If the differential pressure transducer is located at a different level than that of the lower tap into the pressurizer, it will produce a measurement



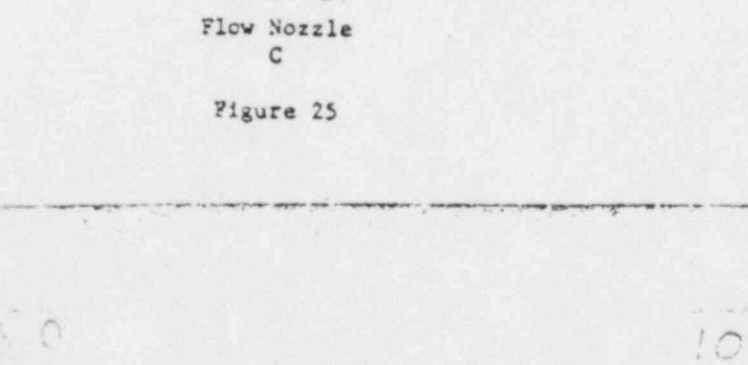
Orifice Plate

A



Venturi

B



Flow Nozzle

C

Figure 25

9-76

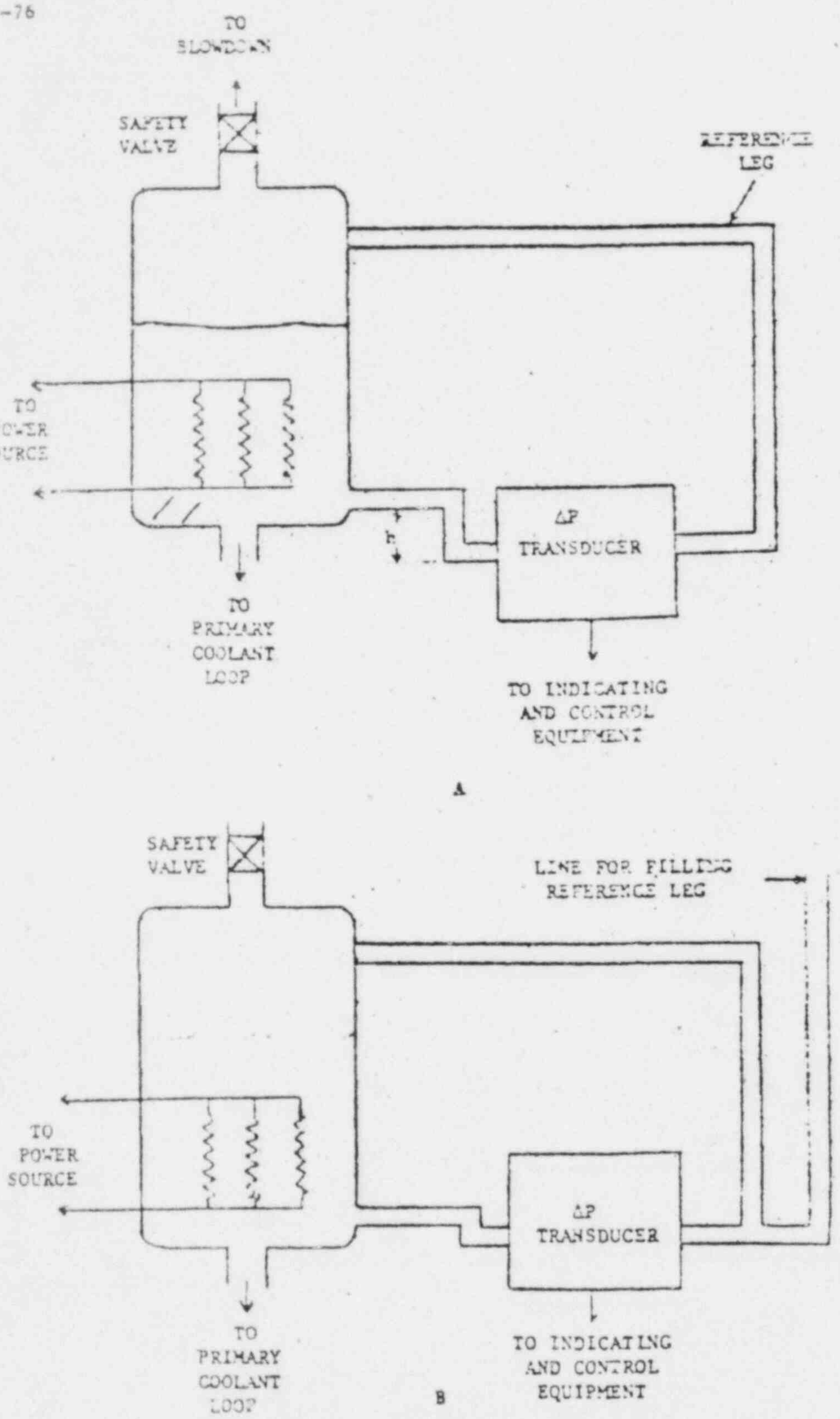


Figure 26

of level difference between its elevation and the pressurizer interface, rather than that between the lower tap and the interface. This discrepancy is indicated by the distance "h" indicated in Figure 21A.

Another problem concerns the piping identified as the "reference leg" in Figure 21A. The adequacy of this connection as a reference depends on its remaining free of liquid, so that it senses system static pressure only. However, since this leg may be physically external to the pressurizer and ordinarily supplied with heat, it may well run at a temperature substantially lower than the pressurizer. As a result, condensation may occur within it, causing it to fill with liquid water. If this happens, the meaning of the measurement disappears.

A modification of this scheme, shown in Figure 21B, solves these problems. Note that in this system a line is provided for filling the reference leg with water. This eliminates the condensation problem. In addition, the elevation of the transducer is no longer critical, since any pressure due to liquid in the input lines is equal on both sides.

Note that with this system the differential pressure becomes zero when the vessel is full, and is at a maximum when the vessel is empty. To allow normal calibration of receiving instruments, the transducer is ordinarily set up to be reverse acting. In other words, zero transducer output is given by maximum differential pressure input and vice-versa.

9.14.5 Measurement Transmission

Before concluding this section, a word might be said on the general problem of transmission. Many years ago, particularly in the chemical and petroleum industries, it was recognized that there was considerable desirability in converting all measurements to a "standard" signal at or near the point of measurement for transmission. The system that was used almost universally until recent electronic developments was one in which all measurements were converted to a 3-15 psig pneumatic signal. This was accomplished by using measuring devices which in some fashion produced a mechanical motion. This motion was then used to vary the position of a flapper in front of a nozzle to which compressed air was supplied. The back pressure in the nozzle then becomes a measure of the variable.

The measuring devices used with this system were those which most directly produced a mechanical motion. The pressure measuring devices described in this section of course do this. For temperature measurements the thermocouple and resistance thermometer described in this section were not quite so convenient. It was more common to use a bulb filled with some liquid which would expand with temperature increases and cause deflection of an element similar to a Bourdon tube.

Such systems have been used in many reactors; however, since World War II, a variety of electrical and electronic systems have become available which convert variables to a uniform electrical signal. Typical systems involve the transmission of 4-20 milliamp DC (Honeywell) or 1-5 milliamp DC (Leeds and Northrup).

Westinghouse

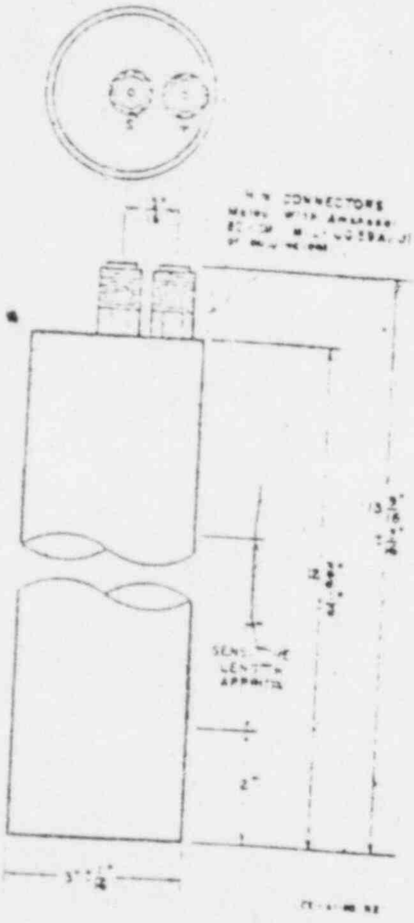
WL-8075

November 1, 1964

UNCOMPENSATED IONIZATION CHAMBER TYPE WL-8075

The WL-8075 is a guard-ring ionization chamber designed to detect thermal neutrons in the range from 2.5×10^{3} to 2.5×10^{10} neutrons/cm²-second. The guard-ring construction combined with the use of high-purity alumina ceramics provides a design which minimizes leakage currents usually detrimental to low current operation. The detector is extremely rugged and the high alumina insulation used throughout permits operation to 300°F and minimized deterioration due to radiation damage.

The rugged construction makes the detector especially useful for Safety and Power range reactor instrumentation where mechanical shock and vibration present problems. The internal design provides a relatively high unperturbed thermal neutron sensitivity of about 4.4×10^{-14} amperes/neutron/cm²/second in a compact device. The gamma sensitivity is approximately 5×10^{-11} amperes/R/hour.



MECHANICAL

Maximum Diameter	3-1/16	Inches
Maximum Overall Length	13-3/4	Inches
Approximate Sensitive Length	7	Inches
Net Weight	2-1/2	Pounds
Shipping Weight	10	Pounds

MATERIALS

Outer Case	Aluminum
Electrodes	Aluminum
Insulation	Hi-Purity Alumina Ceramic
Neutron Sensitive Material	Boron Enriched to 95% in B-10
Gas Filling	Argon-Nitrogen Mixture

IMPEDANCE

Resistance:		
Signal Electrode to Case (minimum)	10^{12}	Ohms
H.V. Electrode to Case (minimum)	10^{12}	Ohms
Capacity:		
Signal Electrode to Case (Note 3)	250	pf
H.V. Electrode to Case (Note 4)	170	pf

MAXIMUM RATINGS

Voltage Between Electrodes	1500	Volts
Temperature	300	°F
Internal Pressure (Note 2)	180	Pounds/Inch²
Thermal Neutron Flux	10^{11}	nv

TYPICAL OPERATION

Operating Voltage (Note 1)	200 to 1000	Volts
Saturation Characteristics		Figure 2
Thermal Neutron Flux Range	2.5×10^{3} to 2.5×10^{10}	nv
Thermal Neutron Sensitivity (Approx.)	4.4×10^{-14}	Amperes/nv
Gamma Sensitivity (Approx.)	5×10^{-12}	Amperes/R/hr

NOTES

1. The voltage necessary to produce saturation varies with the neutron flux level. Either polarity may be used. See Figure 2.
2. The pressurizing atmosphere must be dry and non-corrosive.
3. With H.V. electrode grounded to case.
4. With signal electrode grounded to case.

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SUPERSEDES WL-8075-2/1/62
Neutron & Radiation Detector Section
WESTINGHOUSE ELECTRIC CORPORATION, ELECTRONIC TUBE DIVISION, ELMIRA, NEW YORK

TYPICAL CONNECTION DIAGRAM

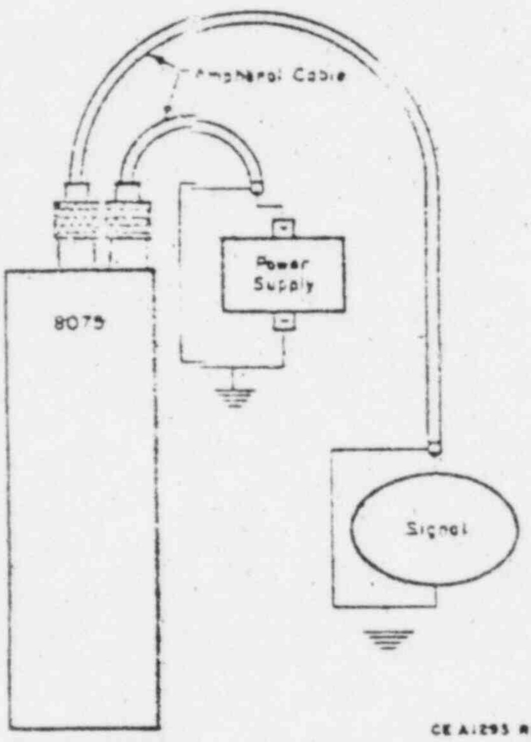


FIGURE 1

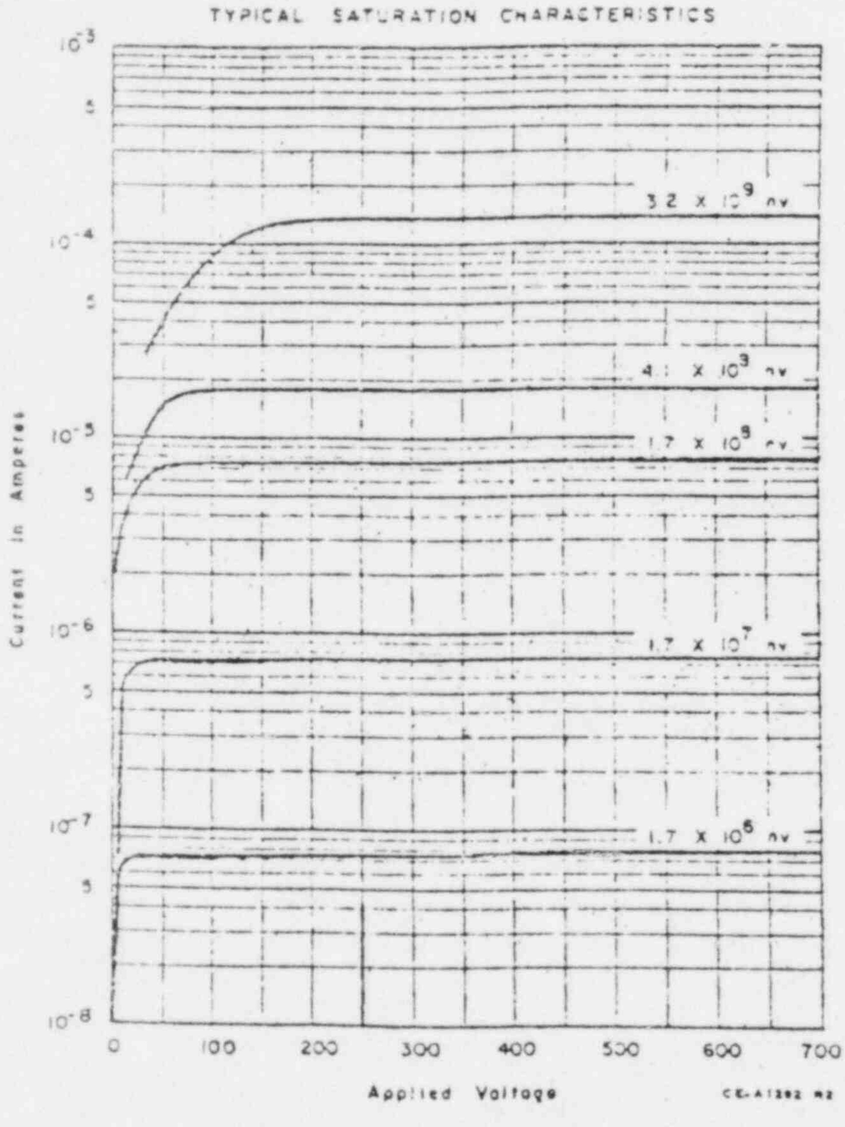


FIGURE 2

76647

[illegible] 0900

Westinghouse

6377

May 1960

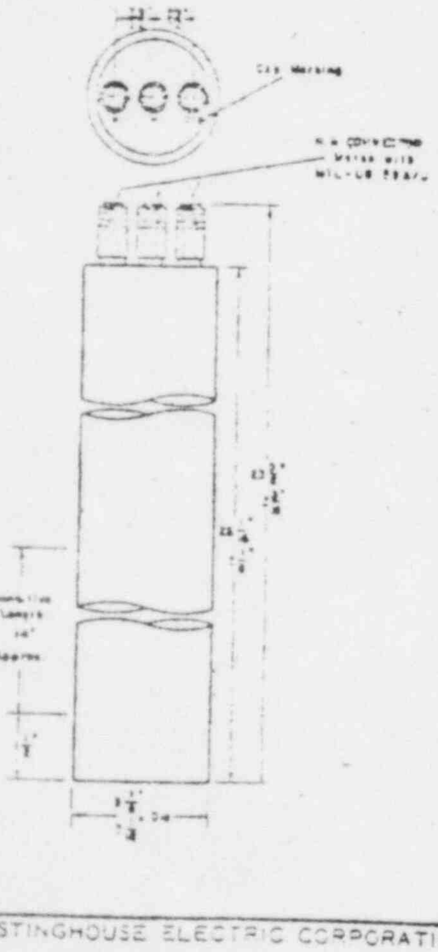
COMPENSATED IONIZATION CHAMBER TYPE 6377

The 6377 compensated ionization chamber is designed to detect thermal neutrons in the range from 2.5×10^4 to 2.5×10^{10} neutrons/cm² second, in the presence of very high gamma radiation fields. The detector is extremely rugged in construction, meeting MIL-S-901 for shock and MIL-STD-167 Type I for vibration, and may be operated in any position at temperatures up to 175 F. The 6377, including the connectors, is constructed of magnesium alloy, with high stability crosslinked polystyrene insulation. The use of this latter material assures completely noise free performance of the detector, even in the lowest decade of operation.

The 6377 incorporates two outstanding features. The first is the use of guard ring construction throughout to minimize the reduction in signal currents due to electrical leakage of the insulators. The second is the provision for continuously variable electrical compensation. This feature provides any desired degree of reduction of the signal caused by gamma radiation, including complete cancellation.

The thermal neutron sensitivity of the 6377 is approximately 4×10^{-14} amperes/neutron/cm² second. The gamma sensitivity, when operated uncompensated, is approximately 3×10^{-11} amperes/R/hour.

The 6377 is similar to the 7353, differing only in outline dimensions.



MECHANICAL:		
Maximum Diameter	3-3/16	Inches
Maximum Overall Length	23-15/16	Inches
Approximate Sensitive Length	14	Inches
Net Weight	5-3/8	Pounds
Shipping Weight	19	Pounds
MATERIALS:		
Outer Case	3% Al, 97% Mg alloy	
Electrodes	3% Al, 97% Mg alloy	
Insulation	Stabilized Polystyrene	
Neutron Sensitive Material		
Coating	Boron enriched in B10	
Thickness	[illegible] mg/cm²	
Gas Filling	Nitrogen	
IMPEDANCE		
Resistance (Note 2)		
Signal Electrode to Case (Minimum)	10^{14}	Ohms
H.V. Electrode to Case (Minimum)	10^{12}	Ohms
Compensating Electrode to Case (Minimum)	10^{12}	Ohms
Capacitance (Note 1)		
Signal Electrode to Case (Approx.)	275	μμf
H.V. Electrode to Case (Approx.)	315	μμf
Compensating Electrode to Case (Approx.)	125	μμf
MAXIMUM RATINGS:		
Voltage Between Electrodes (dc)	1500 max.	Volts
Temperature	175 max.	Degrees F
External Pressure (Note 3)	180 max.	Pounds/inch²
Thermal Neutron Flux	5×10^{11} max.	nv

Neutron & Radiation Detector Section

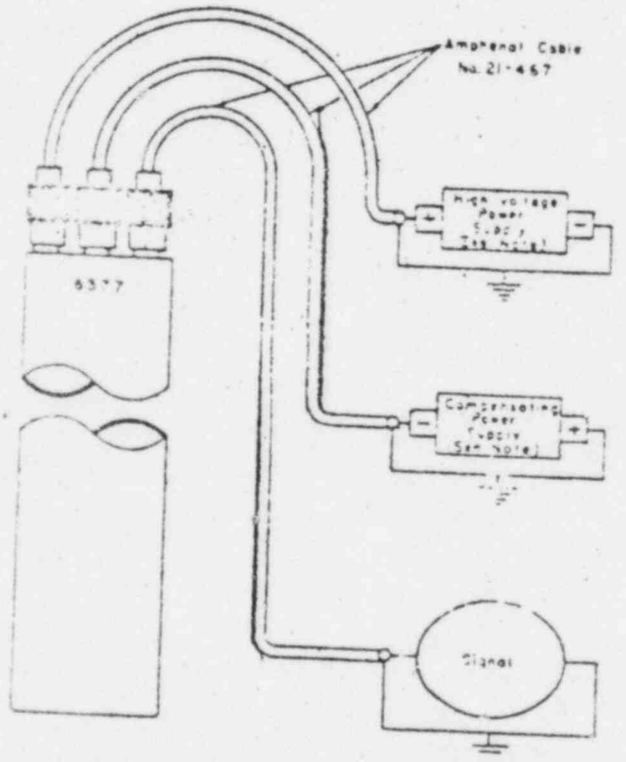
WESTINGHOUSE ELECTRIC CORPORATION, ELECTRONIC TUBE DIVISION, ELMIRA, NEW YORK

TYPICAL OPERATION:

Typical Connection	See Figure 1	
Operating Voltage	300 to 800	Volts
Compensating Voltage (See Figure 2)	−10 to −40	Volts
Saturation Characteristics	See Figure 2	
Thermal Neutron Flux Range	2.5×10^2 to 2.5×10^{10}	nv
Thermal Neutron Sensitivity	4×10^{-14}	Amperes/nv
Gamma Sensitivity:		
Total Compensation		zero
Uncompensated	3×10^{-11}	Amperes/R/hour

1. Capacitance is measured between an electrode and case, with all other electrodes grounded.
2. The detector may not be immersed directly in water, and high humidity environments should be avoided as they will impair performance.
3. The pressurizing atmosphere must be dry and non-corrosive.

TYPICAL CONNECTION DIAGRAM



Note: Permissible power supply regulation and ripple will depend upon the particular application. See Section entitled "Ionization Chamber Operation."

FIGURE 1

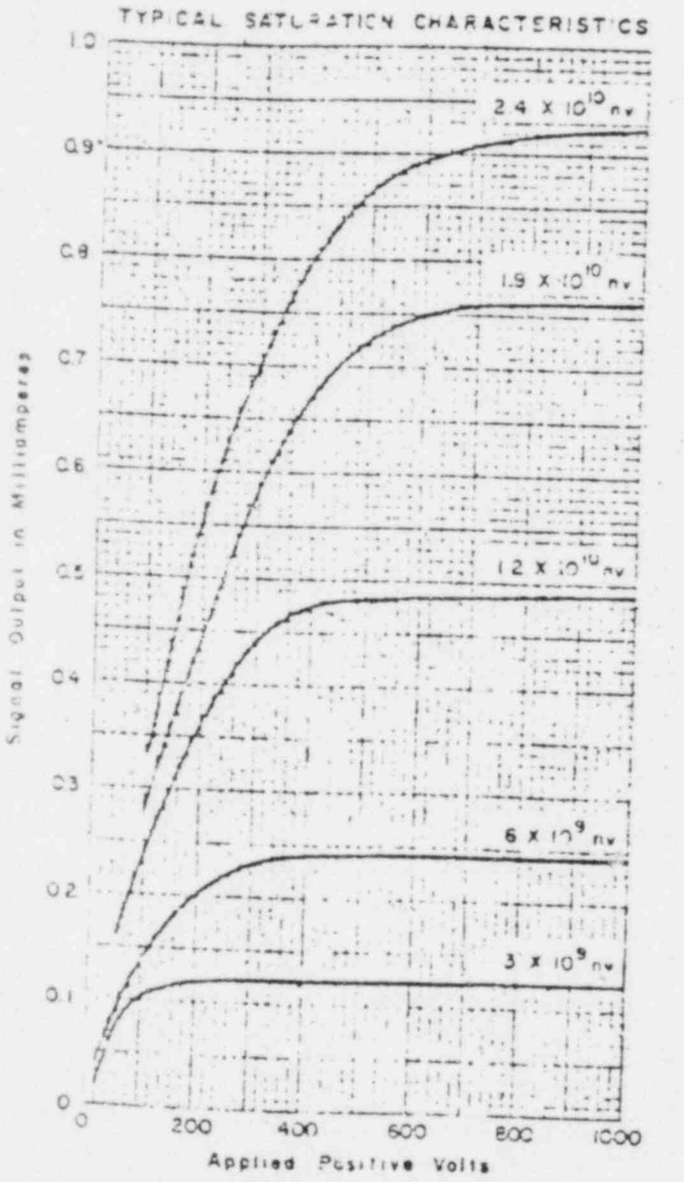

TYPICAL SATURATION CHARACTERISTICS
Signal Output in Milliamperes
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
2.4 X 10¹⁰ nv
1.9 X 10¹⁰ nv
1.2 X 10¹⁰ nv
6 X 10⁹ nv
3 X 10⁹ nv
0
200
400
600
800
1000
Applied Positive Volts

FIGURE 2

TYPICAL COMPENSATION CHARACTERISTICS

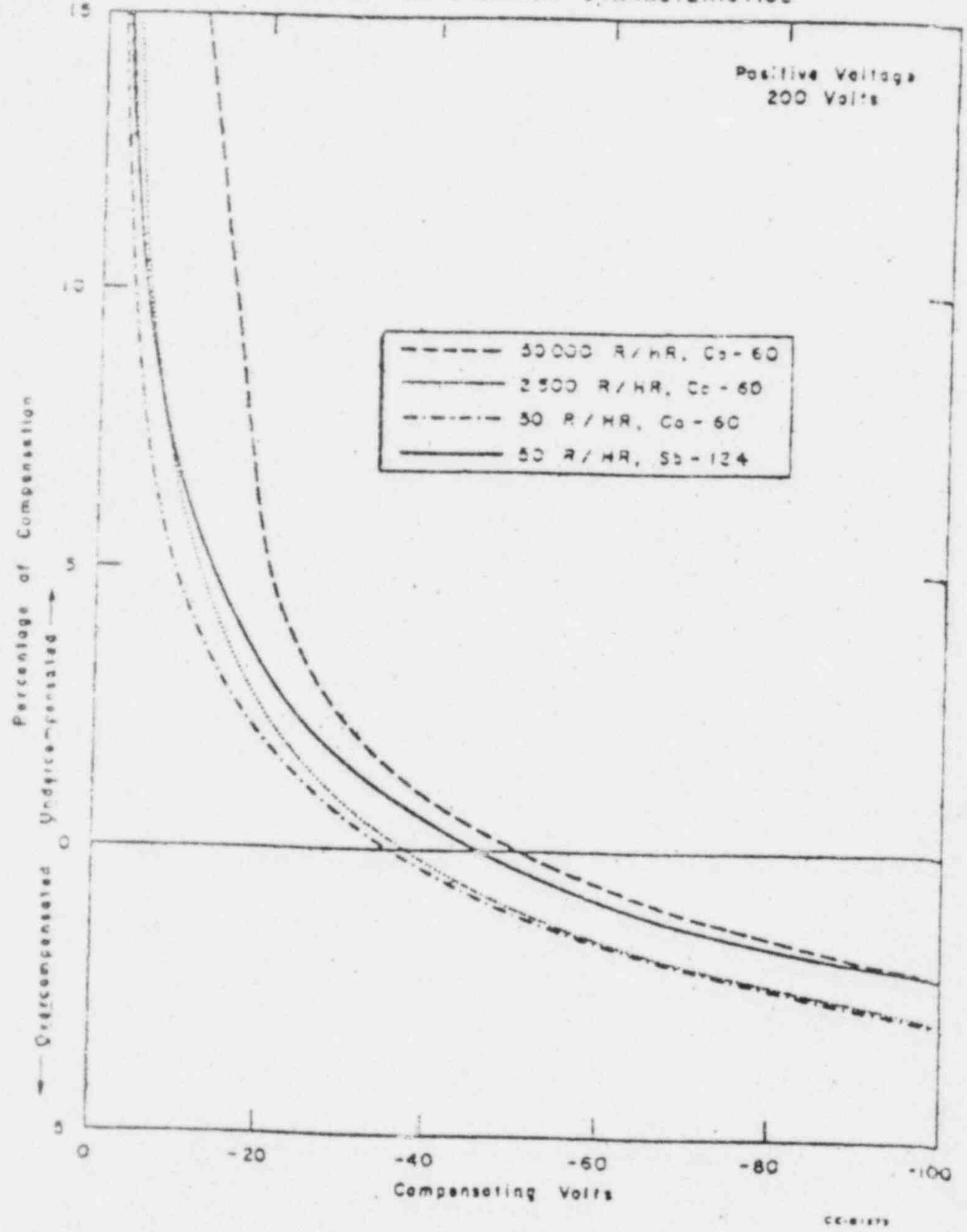


FIGURE 3

Westinghouse

WL-7087
WL-7087A

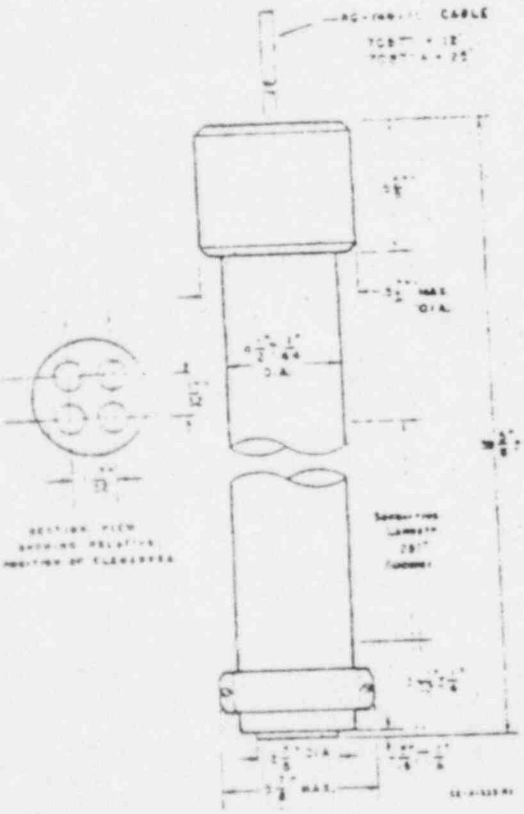
September 4, 1962

HIGH SENSITIVITY BF_3 PROPORTIONAL COUNTER
TYPES WL-7087 AND WL-7087A

The WL-7087 proportional counter is designed to detect neutrons of thermal and intermediate energies in the range from 1.25×10^{-1} to 1.25×10^{4} neutrons/cm²/second. It is a multiunit BF_3 proportional counter in which the outer case is electrically insulated from the elements, thereby avoiding ground loop problems in circuitry. The detector is provided with an integral 12 foot length of RG-149/U coaxial cable. The cable and outer case are hermetically sealed, permitting operation in high humidity environments. The WL-7087 is rugged in construction, meeting MIL-S-901 for shock and MIL-Std-167 (type 1) for vibration, and will operate in any position at temperatures up to 175°F.

The detector consists of four type WL-7317 proportional counters surrounded by a heavy walled aluminum outer case. The materials used in the detector have been selected for low activation properties, thereby facilitating handling after exposure to neutron fluxes. The thermal neutron sensitivity of the WL-7087 is approximately 40 counts/neutron/cm^{-2} at an operating voltage of 2000 volts.

The WL-7087A is similar to the WL-7087, differing only in having a cable length of 25 feet. Except for outer diameter and weight, the WL-7087 and WL-7087A are identical to the WL-7429 and WL-7429A respectively.



MECHANICAL:		
Maximum Diameter	5-7/8	Inches
Maximum Overall Length	35-3/4	Inches
Approximate Sensitive Length	26	Inches
Net Weight	36-3/4	Pounds
Shipping Weight	100	Pounds
Cable Length:		
WL-7087	12	Feet
WL-7087A	25	Feet
MATERIALS:		
Outer Case		Aluminum
Element Body		EC Grade Aluminum
Electrodes		0.001 Inch Dia. Tungsten
Insulation:		
Elemental Detectors		Alumina Ceramic
Cable Connectors		Polystyrene
Cable		Polyethylene
Moderator		Polyethylene
Neutron Sensitive Material:		
Content		Boron trifluoride enriched to 96% in Boron-10
Pressure		55 Cm of Hg
IMPEDANCE:		
Resistance (minimum)	10^{11}	Ohms
Capacitance:		
WL-7087	344	µµf
WL-7087A	670	µµf
MAXIMUM RATINGS:		
Voltage (Between Electrodes)	2500	Volts
Temperature	175	Degrees F
External Pressure (Note 3)	180	Pounds/Inch²
Operating:		
Thermal Neutron Flux	1.25×10^{4}	nv
Non-Operating:		
Thermal Neutron Flux (Note 1)	10^{10}	nv
Gamma Flux (Note 1)	10^{5}	R/hr

Neutron & Radiation Detector Section

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TYPICAL OPERATION (Note 2)

Operating Voltage	2150	Volts
Thermal Neutron Flux Range:		
With Conventional Circuitry 2.5×10^{-2} to 2.5×10^{3}		nv
With Special Circuitry (Note 4) 1.25×10^{-2} to 1.25×10^{4}		nv
Sensitivity	46	CPS/nv
Plateau Characteristics:		
Length (minimum)	200	Volts
Overall Slope (maximum)	4	Percent
Output Pulse Characteristics:		
Amplitude (at cable end, approx. unloaded)		
WL-7087	15	Millivolts
WL-7087A	1.5	Millivolts
Inherent Background (Note 5)	0.25	Counts/second

1. When in radiation levels exceeding the maximum operating levels the high voltage should be removed and the tube shimmed through a maximum of three megohms. See the section entitled "Proportional Counter Operation".
2. These characteristics will vary depending upon the associated circuitry. For application details see the section entitled "Proportional Counter Operation".
3. The pressurizing atmosphere must be dry and non-corrosive.
4. An extension of operating range of approximately two magnitudes indicated may be obtained with the use of low-noise, high resolution circuitry.
5. In radiation free areas, and with low-noise circuitry, values of the order of 12 to 15 counts per minute have been observed.

Westinghouse

WL-22791

December 1, 1963

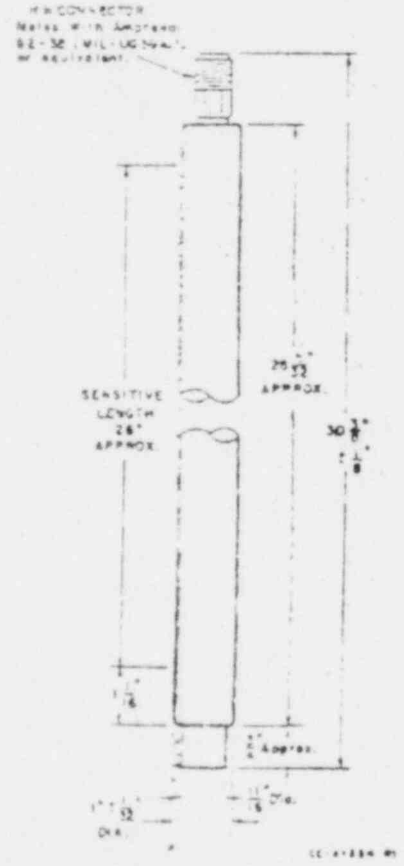
B10 LINED PROPORTIONAL COUNTER TYPE WL-22791

The WL-22791 proportional counter is designed to detect neutrons of thermal energy in the range from 10^{-1} to 5×10^{4} neutrons/cm²/second. This counter is extremely rugged, designed to meet military specification MIL-S-901 for shock and MIL-Std-167 (type 1) for vibration, and will operate in any position at temperatures up to 310° F.

The materials used in the construction of the detector, including the connector, have been selected for low activation properties, thereby facilitating handling after exposure to neutron fluxes. The thermal neutron sensitivity of the WL-22791 is approximately 16 counts/neutron/cm² at an operating voltage of 800 volts.

MECHANICAL

Maximum Diameter	1-1/32	Inches
Maximum Overall Length	30-1/2	Inches
Approximate Sensitive Length	26	Inches
Net Weight	12	Ounces
Shipping Weight	5	Pounds



MATERIALS

Body	EC Grade Aluminum
Electrode	0.001 Inch Dia. Tungsten
Connector	Aluminum
Insulation:	
Electrode Supports	Alumina
Connector	Polystyrene
Neutron Sensitive Material:	
Coating	Boron Highly Enriched in Boron-10
Gas Fill	Argon 95% - CO_2 (5%)

IMPEDANCE

Resistance (minimum)	10^{12}	Ohms
Capacitance (average)	20	μμf

MAXIMUM RATINGS

Voltage Between Electrodes	1500	Volts
Temperature	310	°F
External Pressure	180	Pounds/Inch²
Operating:		
Thermal Neutron Flux	4×10^{4}	nv
Nonoperating:		
Thermal Neutron Flux (Note 1)	5×10^{10}	nv
Gamma Flux (Note 1)	10^{6}	R/hr

TYPICAL OPERATION (Note 2)

Operating Voltage	800	Volts
Thermal Neutron Flux Range	10^{-1} to 5×10^{4}	nv
Sensitivity	16	cps/nv
Plateau Characteristics:		
Length (minimum)	150	Volts
Overall Slope (maximum)	0.3	%/Volt
Output Pulse Characteristics:		
Amplitude (average, unloaded)	80	Millivolts
Inherent Rise Time (average)	2×10^{-7}	Seconds
Inherent Background (Note 3) (approx.)	0.2	Counts/second

NOTES

1. When in flux levels exceeding the maximum operating level, the high voltage should be removed, and the tube shorted through a maximum of 3 megohms.
2. These characteristics will vary depending upon the associated circuitry. For application details see the section entitled "Proportional Counter Operation".
3. In radiation free areas, and with low-noise circuitry, values of the order of 4 to 5 counts per minute have been observed. High humidity environments should be avoided since they may impair performance.

Neutron & Radiation Detector Section

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TYPICAL OPERATION AS A COUNTER: (Note 1)

Operating Voltage	300	Volts
Operating Voltage Plateau (See Figure 2)	200 to 300	Volts
Thermal Neutron Flux Range	1.4 to 1.4 x 10^5	nv
Sensitivity (Note 1)	0.7	CPS/nv
Output Pulse Characteristics:		
Amplitude (Unloaded)	2 x 10^{-4}	Volts
Maximum Rise Time, Average	2 x 10^{-7}	Seconds

TYPICAL OPERATION AS A CHAMBER:

Operating Voltage (Note 3)	300 to 1000	Volts
Saturation Characteristics	See Figure 3	
Thermal Neutron Flux Range (Note 4)	5 x 10^3 to 1.4 x 10^{10}	nv
Thermal Neutron Sensitivity	1.4 x 10^{-13}	Amperes/nv
Gamma Sensitivity	4.2 x 10^{-11}	Amperes/R/hour

1. The sensitivity is 0.7 counts/neutron/cm^2 when the alpha background counting rate of the natural radioactive uranium is adjusted to 5 counts/second. By varying the pulse height selector on the associated circuitry other sensitivities are available. See Figure 1 and the section entitled "Fission Counter Operation".
2. The pressurizing atmosphere must be dry and non-corrosive.
3. The minimum voltage required for saturation is dependent upon the incident neutron flux level. See Figure 3.
4. The lower limit of operating range is determined by an inherent alpha background current of approximately 1.4 x 10^{-9} amperes.

COUNTER SENSITIVITY AS FUNCTION OF PULSE HEIGHT SETTING

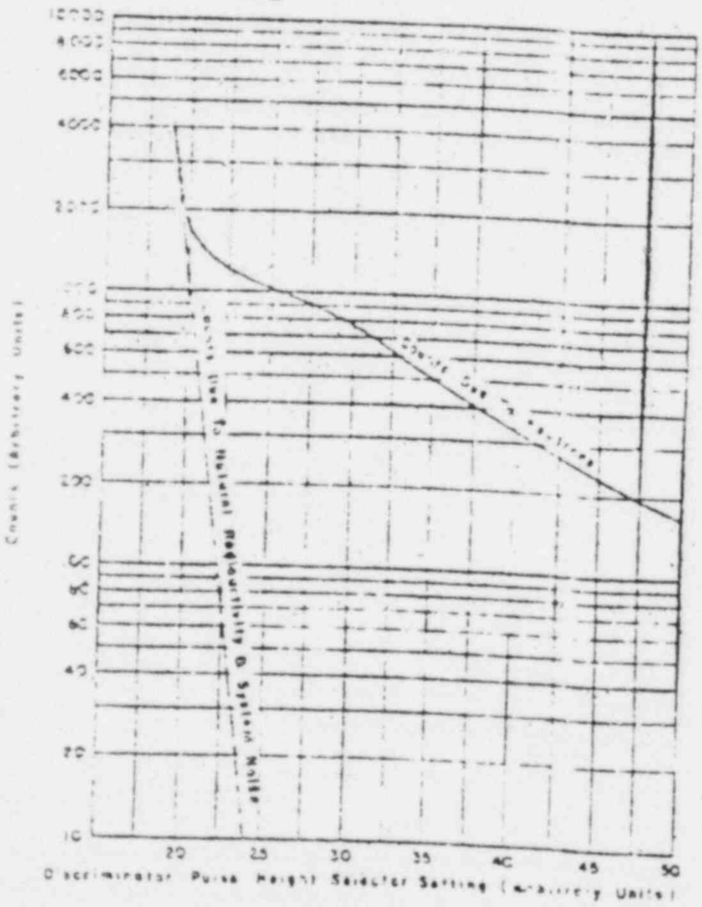


FIGURE 1

Westinghouse

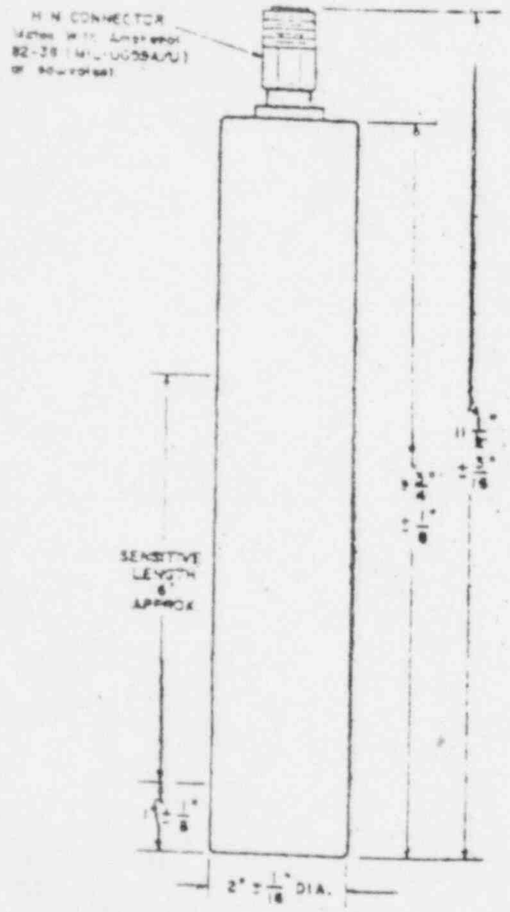
6376
6376A

March 15, 1960

DUAL RANGE FISSION CHAMBER TYPES 6376 AND 6376A

The 6376 fission chamber is designed to detect thermal neutrons in the range of 1.4 to 1.4 x 10^5 neutrons/cm^2/second when operated as a counter, and in the range of 5 x 10^5 to 1.4 x 10^{10} neutrons/cm^2/second when operated as a chamber. The detector is extremely rugged in construction, meeting MIL-S-901 for shock and MIL-STD-167 (type I) for vibration, and may be operated in any position at temperatures up to 175°F. The 6376A is identical to the 6376 except being rated to 300°F.

The 6376 is constructed of aluminum, with high purity alumina ceramic insulators throughout, including those used in the type HN coaxial connector. In typical operation as a fission counter the 6376 has a thermal neutron sensitivity of approximately 0.7 counts/neutron/cm^2 throughout a wide range of applied voltage. When connected for service as an ionization chamber, the thermal neutron sensitivity is approximately 1.4 x 10^{-13} amperes/neutron/cm^2/second with a gamma sensitivity of approximately 4.2 x 10^{-11} amperes/R/hour.



MECHANICAL:

Maximum Diameter	2-1/16	Inches
Maximum Overall Length	11-11/16	Inches
Approximate Sensitive Length	6	Inches
Net Weight	1-3/4	Pounds
Shipping Weight	12	Pounds

MATERIALS:

Outer Case	Aluminum
Electrodes	Aluminum
Insulation	Alumina Ceramic
Neutron Sensitive Material:	
Content	U_3O_8 Enriched to more than 90% in U-235
Thickness	2 mg/cm^2
Total Quantity	1.72 Grams
Gas Filling	Argon-Nitrogen Mixture
Gas Pressure	76 cm of Hg

IMPEDANCE:

Resistance (minimum)	10^9	Ohms
Capacitance	190	μμf

MAXIMUM RATINGS:

Voltage Between Electrodes	1000	Volts
Temperature:		
6376	175	Degrees F
6376A	300	Degrees F
External Pressure (Note 2)	150	Pounds/Inch^2
Thermal Neutron Flux	3 x 10^{10}	nv

Neutron & Radiation Detector Section

WESTINGHOUSE ELECTRIC CORPORATION, ELECTRONIC TUBE DIVISION, ELMIRA, NEW YORK

N 6656

1093 0918

COUNTING RATE CHARACTERISTIC

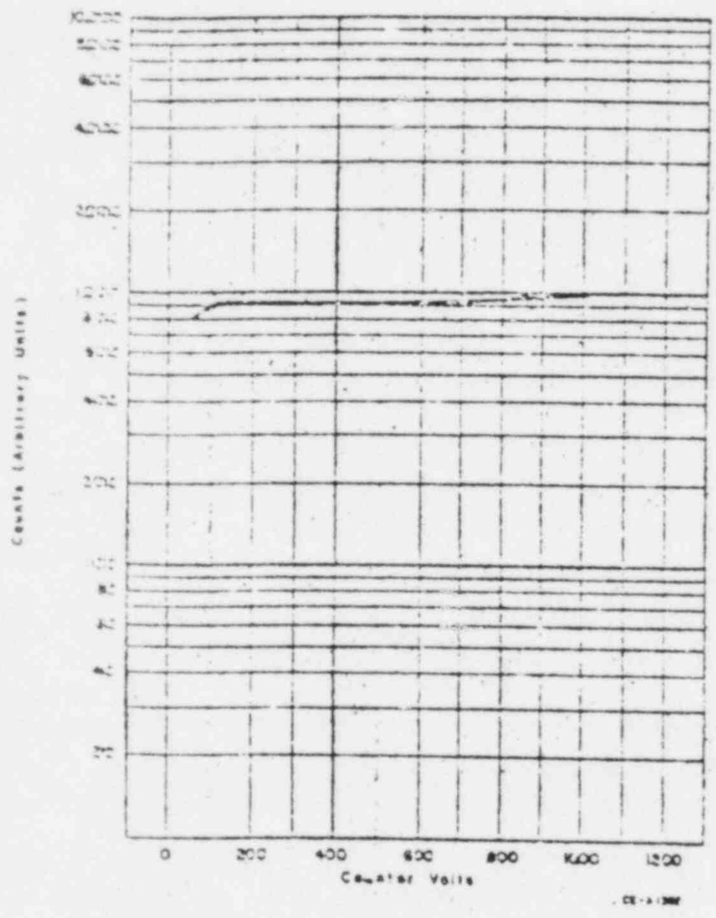


FIGURE 2

TYPICAL SATURATION CHARACTERISTICS
Ion Chamber Operation

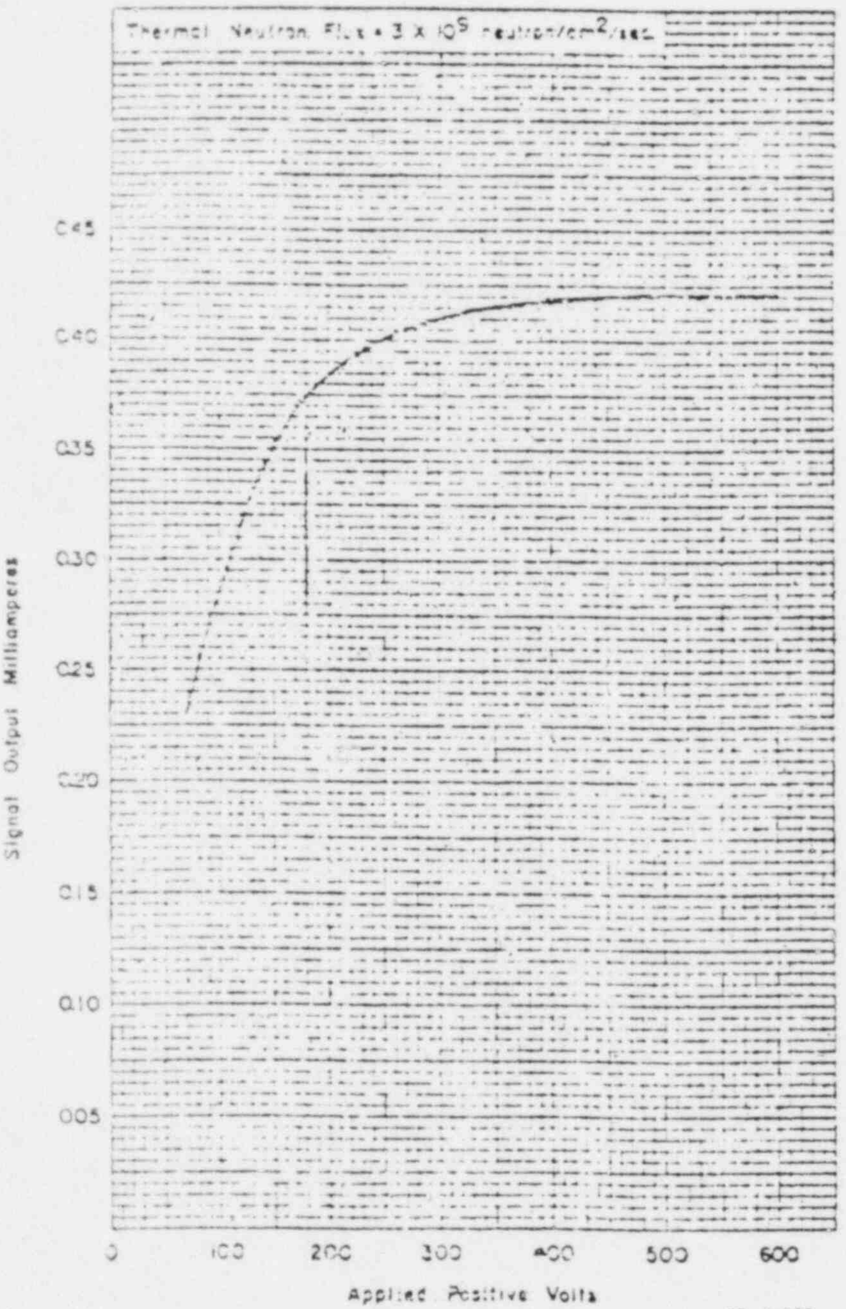


FIGURE 3

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TABLE OF FIGURES

REUTER-STOKES AND THE SELF-POWERED DETECTOR

NTRODUCTION

Since operating data on the self-powered detector were first reported in [illegible], a variety of designs has been thoroughly tested and critically evaluated. Self-powered detectors can be designed to be selectively sensitive to either neutrons or gamma rays. They are particularly useful for continuously measuring the flux distribution in a reactor core over a period of months or years. In power reactors they can be used for flux mapping and controlled power flattening. In test reactors they can be used for measuring flux in test assemblies or reactor loops. During commissioning, they can be used for calibrating other flux and power measuring instruments over the range [illegible] n/cm²-sec. If average flux over an extended length or volume is required, self-powered detectors can be made in the form of long flexible cables.

The main advantages of self-powered detectors are:

Low cost.
Simplicity of operation and read-out instrumentation.
Accuracy and reliability.
Low burnup and long life.
Continuous operating capability in high flux, high temperature environment.
Reproducibility of detector output characteristics.

PRINCIPLES OF OPERATION

The principle of operation of the self-powered detector is conversion of the incident radiation on the detector emitter material to energetic electrons which penetrate the solid insulation and come to rest on the collector or its surroundings. The deficiency of electrons in the emitter results in a positive charge on the center conductor of a coaxial cable attached to the emitter. If the center conductor of the cable is connected to the outer sheath of the cable by a resistor, the rate of positive charge production can be measured as a voltage or current. This voltage or current is directly proportional to the rate at which radiation is being absorbed by the emitter.

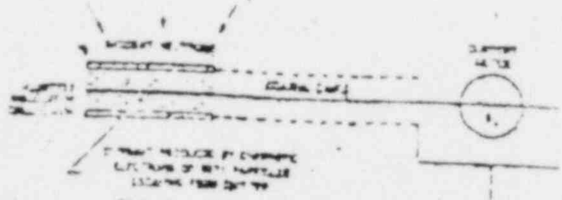


FIG. 1 SELF POWERED NEUTRON DETECTOR OPERATION

There are three primary mechanisms by which the incident radiation is converted to energetic electrons:

I Neutron Capture and Beta Decay

Neutron capture in the emitter produces a capture product which decays through beta emission. Some of the beta particles are energetic enough to escape from the emitter and insulator, resulting in a positive charge on the emitter as described above.

The emitter material must have a reasonably large neutron capture cross section, and the capture product must be a beta-emitting isotope with a short half-life. An example is Rhodium 103 which has a 150 barn cross section. Its neutron capture product Rhodium 104 emits 2.4 MeV beta particles (maximum) and has a 42 second half-life. The response time of a detector to a change in neutron flux is governed by the half-life of the neutron capture product. There is no beta-emitting isotope combining high cross section with very short half-life which would permit manufacture of a practical prompt-response detector.

II Neutron Capture Gamma Conversion for Prompt Response Detectors

Neutron capture in the emitter produces capture gamma rays which are partly absorbed in the emitter itself, producing Compton and photo electrons. Some of these are energetic enough to escape from the emitter and surrounding insulator resulting in the required positive charge on the emitter.

The emitter material must have a reasonably large neutron cross section, but in this case must not produce any interfering beta-emitting capture products.

In actual practice, however, a material with a very long-lived beta-emitting capture product can be used. Cobalt 59 is an example of a suitable material. When neutrons are absorbed, the gamma rays emitted within 10^{-14} seconds after capture are converted to Compton and photo electrons. The time response of a detector which operates on this "prompt" principle is limited only by cable capacitance and electronics. In general, the prompt response detector is less sensitive than the beta current detector.

III Gamma Ray Absorption → Compton – Photo Electron Emission

Incident gamma rays absorbed by the emitter produce Compton and photo electrons which, as before, can generate a measurable voltage or current. This process, used in gamma ray detectors, requires an emitter material with very low neutron cross section and high gamma absorption coefficient such as [illegible] and [illegible]. Self-powered gamma detectors are prompt-response devices and in this respect are similar to detectors using mainly prompt gamma rays.

Cable Interference

The three electron-producing mechanisms described above are also functioning in the cable attached to the detector. Cable materials and dimensions must be selected to minimize the background signal.

GENERAL CONFIGURATIONS

The most common configuration for a self-powered detector is a cylindrical emitter surrounded by an annular insulator and an outer outer sheath. In fact a self-powered detector can be made from a length of mineral insulated cable which has a portion of its center conductor replaced by emitter material. The emitter is completely surrounded by mineral insulation and a metal sheath. Detectors with annular or flat emitters are also available. Flexible detectors can be used too, or they can be wound [illegible] to provide high sensitivity in small volume configurations such as flux monitoring or effluent monitoring.

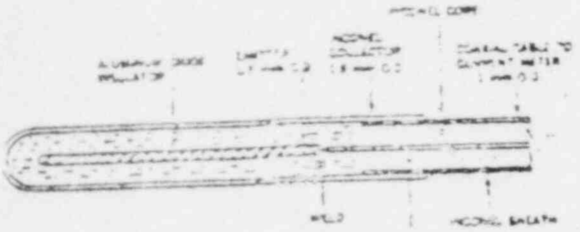


FIG. 2
SELF POWERED NEUTRON
DETECTOR CONSTRUCTION

4. STANDARD APPLICATIONS

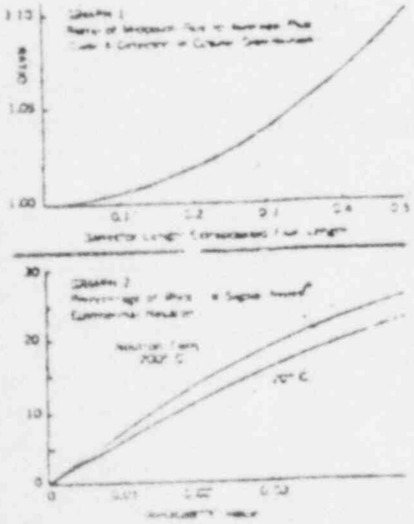
For self-powered detectors at fluxes greater than 10^{12} nv or 10 [illegible] R/hr, the following considerations apply. Detectors for power reactor operation, research and energy reactors and out-of-core experiments are described below.

5. EMITTER CHOICE

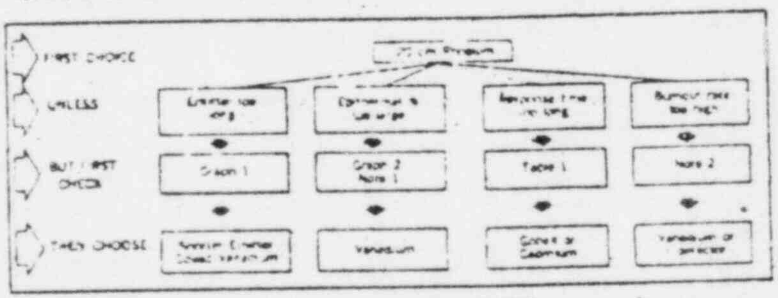
Figures 3 and 4 have been prepared as a guide in the selection of proper emitter material. Sensitivity is of prime importance. The higher the signal, the fewer the problems to be encountered with background, noise and instrumentation. The best emitter for general use is rhodium, because it combines high sensitivity with all the other advantages of self-powered detectors. However, for special applications V, Co, or Cd may be preferable. In any case, the longest emitter consistent with other requirements should be used. If necessary the low sensitivity detectors can be made in long lengths and coiled to provide high sensitivity in short axial lengths.

Our experience indicates that the problems associated with use of fairly low sensitivity detectors are not as severe as was originally anticipated. For example a 120 cm long cobalt detector (sensitivity 3×10^{-22} amp/nv) has been used in $5 \times 10^{?}$ nv. Careful installation and modern measuring techniques can extend the application of such detectors into the range previously considered unsuitable.

FIG. 3 EMITTER SELECTION GUIDE



FIRST RULE OF SELECTION: Design for highest sensitivity. From [illegible] the first choice would be a 20 cm Rhodium emitter. It is not, however, a universal choice. If your requirements include any of the special considerations listed below, refer to the references for an alternate choice.



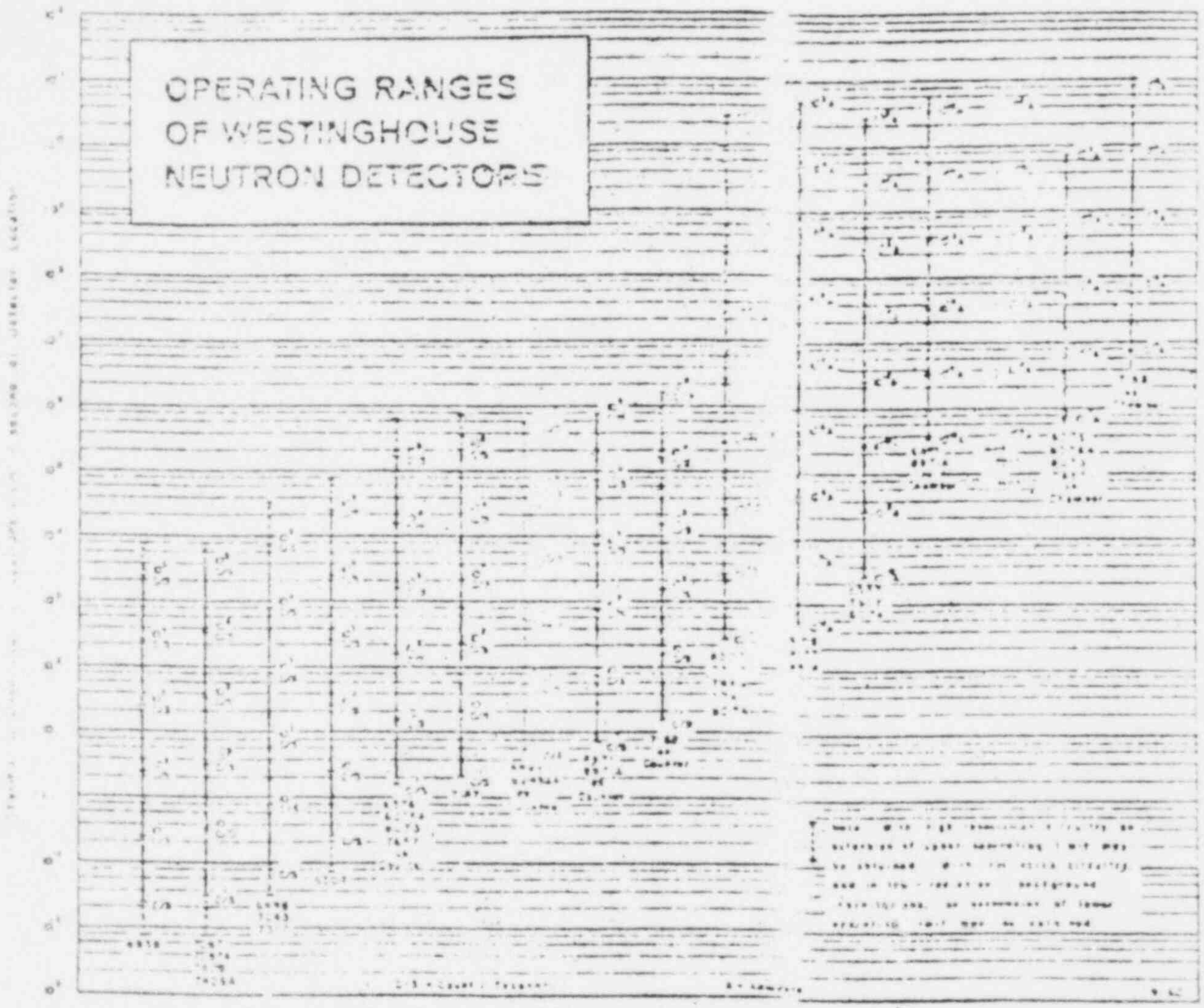
NOTE 1 Experimental [illegible] can result in calibration errors if the neutron spectrum changes significantly during irradiation. If not, the [illegible] signal can be compensated [illegible].

NOTE 2 The burnout rate of Rhodium (0.23%/mo at 10^{13} nv) is much lower than that of [illegible] but [illegible] is 0.07, vanadium is 0.01 [illegible] selections are [illegible].

TABLE 1 EMITTER	T½
Rhodium	42 s
Vanadium	3.8 m
Cobalt	prompt
Cadmium	prompt
	prompt

Graph 1 is reprinted from Appendix A in AECL report [illegible] by [illegible] and C.W. [illegible]. Reprinted with the permission of Atomic Energy of Canada Ltd.

BROADEST CHOICE OF OPERATING RATINGS AVAILABLE FROM A SINGLE SOURCE



available now...for far less than you might expect

The neutron detector [illegible] chart pictured above shows more than 75 different devices in regular production at Westinghouse. Contrary to what you might suppose [illegible] these detector performance capabilities, their prices [illegible] within the [illegible] of industrial reactor [illegible]. Since most are [illegible] detectors, you can purchase now from Westinghouse.

Detector Type	Length (Inches)	Diameter (Inches)	Method of Oper.*	Neutron Sensitive Material	Type of Insulation	Max. Oper. Temp (°F)	Neutron Sensitivity (per nv)	Operating Ratings Thermal Neutron Flux (nv max.)	Operating Volts	Impedance	
										Resistance (minimum Ohms)	Capacitance (pf)
WL 6307	12¼	1½	P.C.	$B^{10}F_3$	Alumina	175	4.5 Cps	$10^{[illegible]}$	2000	$10^{[illegible]}$	10
WL 6376A	11¼	2⅝	I.C./F.C.	U^{235}	Alumina	300	$1.4\times10^{[illegible]}$ A	$3\times10^{[illegible]}$	300 to 800	$10^{[illegible]}$	190
WL 6377	24	3¾	I.C.	B^{10}	Stab. Poly	175	$4\times10^{[illegible]}$ A	$2.5\times10^{[illegible]}$	300 to 800	$10^{[illegible]}$	275
WL 6937A	13¼	3¾	I.C.	B^{10}	Alumina	300	$4.4\times10^{[illegible]}$ A	$2.5\times10^{[illegible]}$	200 to 800	$10^{[illegible]}$	250
WL 6938	32½	2⅝	P.C.	$B^{10}F_3$	Alumina	175	45 Cps	$10^{[illegible]}$	2000	$10^{[illegible]}$	120
WL 6941A	11¼	2⅝	I.C./F.C.	U^{235}	Alumina	300	$2.6\times10^{[illegible]}$ A	$8.5\times10^{[illegible]}$	300 to 800	$10^{[illegible]}$	190
WL 6971A	11¼	2⅝	I.C./F.C.	U^{235}	Alumina	200	0.14 Cps	$3.5\times10^{[illegible]}$	300	$10^{[illegible]}$	190
WL 6998	29½	1½	P.C.	$B^{10}F_3$	Alumina	175	13 Cps	$10^{[illegible]}$	2000	$10^{[illegible]}$	20
WL 7087	29½	5¼	P.C.	$B^{10}F_3$	Alumina	175	40 Cps	$10^{[illegible]}$	2000	$10^{[illegible]}$	344
WL 7087A	39½	5¼	P.C.	$B^{10}F_3$	Alumina	175	40 Cps	$10^{[illegible]}$	2000	$10^{[illegible]}$	610
WL 7186	52½	⅞	F.C.	U^{235}	Alumina	125	0.001 Cps	$10^{[illegible]}$	250 to 400	$10^{[illegible]}$	55
WL 7187	11¼	2⅝	F.C.	U^{235}	Alumina	300	0.52 Cps	$2\times10^{[illegible]}$	300	$10^{[illegible]}$	190
WL 7188	11¼	2⅝	I.C./F.C.	U^{235}	Alumina	300	0.07 Cps	$10^{[illegible]}$	300	$10^{[illegible]}$	190
WL 7243	15¾	2⅛	P.C.	$B^{10}F_3$	Alumina	175	13 Cps	$10^{[illegible]}$	2000	$10^{[illegible]}$	30
WL 7317	29¾	1½	P.C.	$B^{10}F_3$	Alumina	175	13 Cps	$10^{[illegible]}$	2000	$10^{[illegible]}$	15
WL 7353	24½	3¾	I.C.	B^{10}	Stab. Poly	175	$4\times10^{[illegible]}$ A	$2.5\times10^{[illegible]}$	300 to 800	$10^{[illegible]}$	275
WL 7429	39½	4⅝	P.C.	$B^{10}F_3$	Alumina	175	40 Cps	$10^{[illegible]}$	2000	$10^{[illegible]}$	344
WL 7429A	39½	4⅝	P.C.	$B^{10}F_3$	Alumina	175	40 Cps	$10^{[illegible]}$	2000	$10^{[illegible]}$	610
WL 7606	13¾	3¾	I.C.	B^{10}	Alumina	500	$4.4\times10^{[illegible]}$ A	$2.5\times10^{[illegible]}$	200 to 800	$10^{[illegible]}$	250
WL 7657	13¾	3¾	I.C./F.C.	U^{235}	Alumina	500	0.7 Cps	$1.4\times10^{[illegible]}$	200 to 800	$10^{[illegible]}$	225
WL 7741	23¾	3¾	I.C.	B^{10}	Alumina	500	$4\times10^{[illegible]}$ A	$2\times10^{[illegible]}$	300 to 800	$10^{[illegible]}$	315
WL 8073	11¾	2¾	I.C./F.C.	U^{235}	Alumina	300	0.7 Cps	$1.4\times10^{[illegible]}$	300	$10^{[illegible]}$	190
WL 8074	23¾	3¾	I.C.	B^{10}	Stab. Poly	175	$4\times10^{[illegible]}$ A	$2.5\times10^{[illegible]}$	300 to 800	$10^{[illegible]}$	275
WL 8075	13¾	3¾	I.C.	B^{10}	Alumina	300	$4.4\times10^{[illegible]}$ A	$2.5\times10^{[illegible]}$	200 to 1000	$10^{[illegible]}$	250
WL 8105	10¾	3½	I.C.	B^{10}	Stab. Poly	175	$2\times10^{[illegible]}$ A	$2.5\times10^{[illegible]}$	300 to 800	$10^{[illegible]}$	130
WL 8137	10¾	3½	I.C.	B^{10}	Stab. Poly	175	$2\times10^{[illegible]}$ A	$2.5\times10^{[illegible]}$	300 to 600	$10^{[illegible]}$	110
WX 4344	11¾	2¾	F.C.	U^{235}	Alumina	700	0.7 Cps	$3\times10^{[illegible]}$	300 to 800	$10^{[illegible]}$	190
WX 4414	13¾	3½	I.C./F.C.	U^{235}	Alumina	700	0.25 Cps	$4\times10^{[illegible]}$	300	$10^{[illegible]}$	200

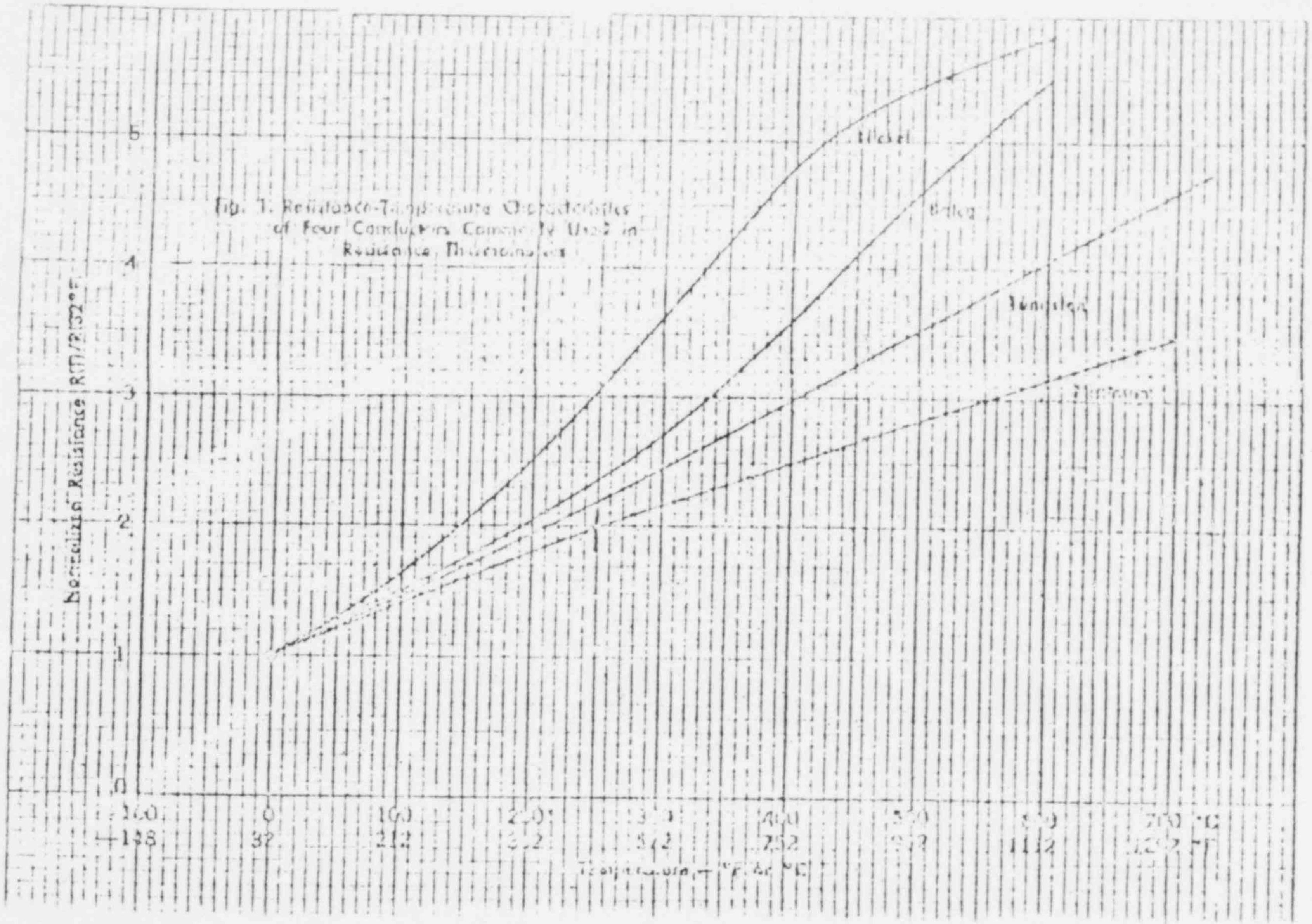


Fig. 1. Resistance-Temperature Characteristics of Four Conductors Commonly Used in Resistance Thermometers

SHIELDING

10.0 Introduction

A nuclear reactor in addition to producing a great deal of heat also produces a high level of radiation. The nuclear plant operating personnel, as well as the general public, must be protected from radiation in accordance with the limits as defined in the Code of Federal Regulations, 10 CFR 20. To accomplish this task, various shields are placed around the reactor and facility to attenuate the neutron and gamma radiation emitted from various sources such as the reactor core, coolant loops, auxiliary equipment, and spent fuel elements.

Shielding techniques and calculations become very complex and time consuming; therefore, we will cover only some of the basic concepts and shielding calculations which are necessary for a basic understanding of the problems of shielding. We will begin with a brief description of what type of radiation is emitted from a typical reactor core and then determine how the various types of radiation interacts with matter. Once this is understood we can then develop some simple shielding equations which can be used as a "rule of thumb" for effective shielding.

10.1 Radiation From A Nuclear Reactor

Radiation from a nuclear reactor consists of alpha particles, beta particles, gamma rays and neutrons. Therefore, adequate shielding must be utilized to attenuate the radiation to a safe level for nearby operating personnel and the general public.

Generally speaking, the sources of radiation from a nuclear reactor can be classified into two groups: radiation as a result of the fission process, and radiation from neutron and gamma ray interaction with materials outside the core (See Figure 1 and 2).

10.2 Radiation Interaction With Matter

10.2.1 Types of Interactions

The interaction of radiation with material can take on several forms such as : disintegration, scattering, absorption, and transmutation.

We can classify all types of radiation into two categories, either charged or uncharged particles. We find that charged particles interact with matter causing direct ionization and uncharged particles interact with matter and cause ionization by secondary means.

Charged Particles	Uncharged (Particles or Photons)
proton (${}_1P^1$)	neutron (${}_0n^1$)
beta (${}_1\beta^-$)	gamma (γ)
positron (${}_1\beta^+$)	x-ray
alpha (${}_2\alpha^4$)	

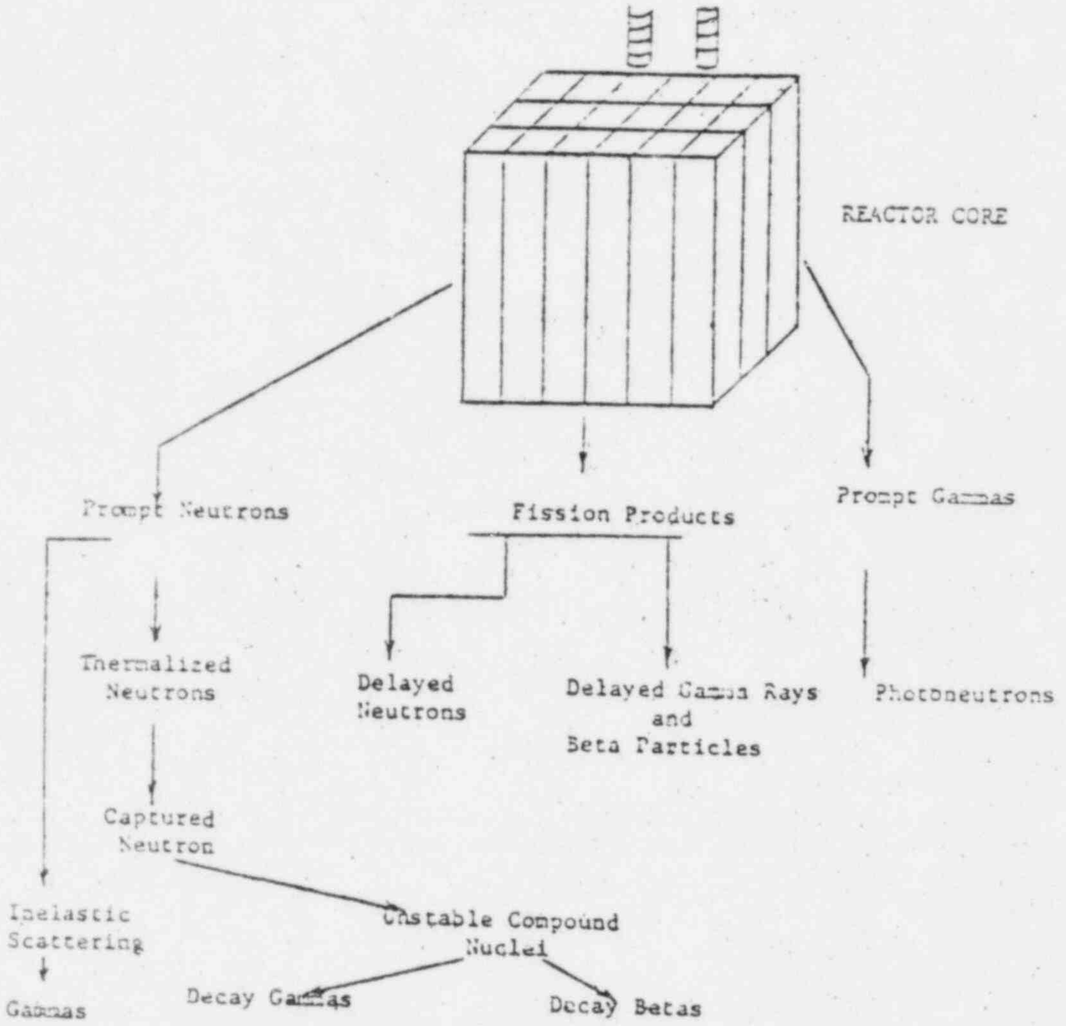


Figure 1

RADIATION PRODUCTION AS A RESULT OF THE FISSION PROCESS

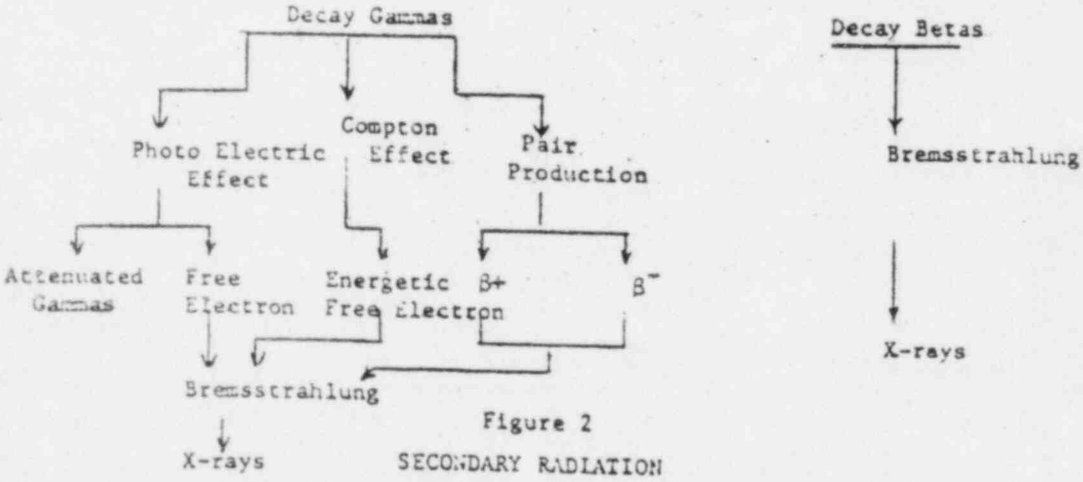
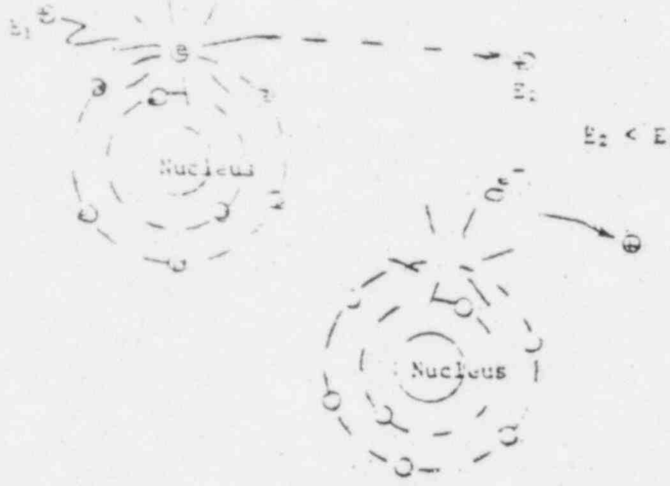


Figure 2

SECONDARY RADIATION

Charged particles, having an electric field surrounding them due to their charge, when passing through matter interact with the orbital electron (due to charge attraction or repulsion) in such a manner that the result is a de-acceleration of the charged particle and an increase acceleration of the electron in the atom's medium. In other words, there is an energy transfer from the charged particle to the atom's electrons. If this transfer of energy is large enough, an electron may escape from its parent atom creating an ion pair (See Figure 3).



Ion Pair

Figure 3

This results in the creation of a positive charged atom (since the atom has lost one electron) and a free electron.

Therefore, if we look at a large volume of material and the path of a charged particle through it, we would expect to see a trail of created ion-pairs until the energy of the charged particle was diminished to a point where it would recombine.

Uncharged particles, as we will show later, lose energy only by means of collisions and/or scattering effects.

With the above opening remarks we will now look more closely at the type of interactions which takes place with both charged and uncharged particles.

10.2.2 Alpha and Beta Shielding

You will recall from our earlier discussion that alpha particles interacting with matter have a definite range and that beta particles, although not having a definite range, do have a finite penetration limit in material in which all betas are attenuated. Therefore, to effectively shield against alphas or betas, all one needs to do is to interpose enough material equal to the range of that particle in that material between the radiation source and the radiation free area.

10.2.3. Range

The range of a charged particle is the distance that the particle can penetrate in matter while causing ionization. For example, a 1 Mev alpha or beta particle has the following range in air, water, concrete and lead:

	Range in Inches			
Particle	Air	Water or Tissue	Concrete	Lead
α	0.2	0.0002		
β	144	0.14	0.075	0.012

It is now obvious that one can shield against any beta or alpha particle by using a shield of some material having a thickness slightly greater than the range of that particle in that material. The range for each type of particle can be determined from the following equations.

10.2.3.1 Range of Alpha

A) $\text{Range (air)} = 0.318\, E^{3/2}$
where E = energy in Mev and
Range is in centimeters.

B) $\text{Range (solid)} = \dfrac{\text{Range (air)}\ 3.2 \times 10^{-4}\ A^{1/2}}{\text{Density } (\rho)}$

Example: Find the range of a 1 Mev alpha in water;
$\rho = 1\ \text{gm/cm}^3$

a) $\text{Range (air)} = 0.318\,(1)^{3/2} = 0.318\ \text{cm.}$

b) $\text{Range } (H_2O) = 0.318\,(3.2\times10^{-4})(18)^{1/2}/1$

$= 4.36\times10^{-4}\,\text{cm} = 1.72\times10^{-4}\,\text{in.}$

10.2.3.2 Range of Beta

A) $\text{Range (gm/cm}^2) = 0.546E - 0.16$
where E = Energy in Mev and
Range is in gm/cm^2

B) $\text{Range (cm)} = \dfrac{\text{Range (gm/cm}^2)}{\text{Density (gm/cm}^3)}$

Example: Find the range in centimeters of a 1 MeV beta in lead; ρ = 11.35 gm/cm³.

a) $R\ (\mathrm{gm/cm^2}) = 0.546(1) - 0.16 = 0.386\ \mathrm{gm/cm^2}$

b) $$R\ (\mathrm{cm}) = \frac{0.386\ \mathrm{gm/cm^2}}{11.35\ \mathrm{gm/cm^3}} = 0.034\ \mathrm{cm}$$

$$= 0.0134\ \mathrm{in.}$$

It should be noted that the above rule of thumb calculations very nearly approximate the experimental values mentioned earlier.

10.2.4 Neutron Attenuation

The neutron having no charge, interacts with matter by one of two methods, absorption or scattering.

10.2.4.1 Absorption

The absorption reaction is characterized by a neutron entering a nucleus resulting in other particles or photons leaving the nucleus.

10.2.4.1.1 Slow Neutron Absorption or Capture

When a slow neutron is absorbed usually one of the following photons or particles is emitted: gamma rays, alpha particles, or beta particles. For example, a process called radiative capture illustrates this type of absorption and is represented as follows:

$${}_{z}X^{A} + {}_{0}n^{1} \longrightarrow {}_{z}X^{(A+1)*} \longrightarrow {}_{z}X^{(A+1)} + {}_{0}\gamma^{0}$$

and is usually written $\boxed{X^{A}\ (n,\gamma)\ X^{A+1}}$

Two important (n,γ) reactions are:

(1) $${}_{1}H^{1} + {}_{0}n^{1} \longrightarrow ({}_{1}H^{2})^{*} \longrightarrow {}_{1}H^{2} + {}_{0}\gamma^{0}$$

(2) $${}_{48}Cd^{113} + {}_{0}n^{1} \longrightarrow ({}_{48}Cd^{114})^{*} \longrightarrow {}_{48}Cd^{114} + {}_{0}\gamma^{0}$$

The first reaction indicates what occurs whenever a hydrogenous substance (example - the human body) absorbs thermal neutrons. The resultant γ-ray which is emitted has at least 2.2 Mev of energy and therefore becomes a gamma shielding problem.

The second reaction is very important because cadmium has a high affinity for absorbing thermal neutrons and therefore is a good choice for reactor control rod composition.

10.2.4.1.2 Fast Neutron Absorption

Fast neutron absorption without resulting in fission is usually small and therefore will not be discussed any further other than to mention just one important type of absorption reaction as follows:

$$_{8}O^{16} + {_0}n^{1} \longrightarrow {_1}H^{1} + {_7}N^{16}$$

This reaction is very important in water cooled reactors because of the production of Nitrogen-16 which is highly radioactive and emits beta particles and a 6 Mev gamma ray.

10.2.4.2 Neutron Scattering

Neutron scattering is a process whereby a neutron collides with a target nuclei and transfers some or all of its energy to the nucleus but remains free after the reaction. There are two types of neutron scattering reactions described as follows:

10.2.4.2.1 Elastic Scattering

Elastic scattering is usually referred to as a "billiard-ball" type of collision where both kinetic energy and momentum are conserved.

$$({_0}n^{1})_a + {_z}x^{A} \longrightarrow {_z}x^{A+1} \longrightarrow {_z}x^{A} + ({_0}n^{1})_b$$

Note, that the neutron which enters the reaction is not the same neutron which leaves the reaction.

In this reaction a stationary nucleus is struck by a neutron traveling at some velocity. The amount of kinetic energy which is transferred to the nucleus depends upon the mass of the nucleus and the angle through which this neutron is scattered.

In the case of a head-on collision with a proton (Note: the proton has approximately the same mass as a neutron) the neutron could be stopped completely with the proton taking on all of the energy. If the neutron collides with a very heavy nucleus, the neutrons will glance off the heavy nucleus with only a slight decrease in energy.

Elastic collisions are the type of collisions which take place in the moderator of a reactor since the moderator elements are of low mass and will be most effective for reducing neutron energies of about 0.1 Mev down to 0.025 ev.

10.2.4.2.2 Neutron Inelastic Scattering

Inelastic scattering is the process in which a neutron is captured by a nucleus and then re-emitted at a lower energy.

$$({}_0n^1)_a + {}_zX^A \longrightarrow {}_zX^{A+1} + ({}_0n^1)_b + \gamma$$

where $E_{na} > E_{nb}$

and $E_{na} - E_{nb} = E_\gamma$

Note that when the neutron is absorbed in the reaction, the target nuclide is raised to a higher energy state and returns to a lower state by the emission of γ-ray photons.

Since the energy states of the target nuclei are of discrete values, there are only certain minimum energies required before an inelastic scattering reaction will be observed. The threshold is on the order of 0.1 Mev for heavy elements and greater than 1 Mev for lighter elements. Therefore, this means that neutrons with less than 0.1 Mev cannot lose energy by inelastic scattering..

10.3 Neutron Shielding

The method of shielding against neutrons involves two aspects: (1) the slowing down of fast neutrons (2) the capture of the slowed down neutrons. Each of the above methods ~~are~~ dependent upon the probability of interaction with nuclei and therefore are a function of the cross section for a particular reaction. It is therefore important that one understand how the cross section for a particular reaction varies as a function of neutron energy and the target material.

10.4 Types of Cross Sections

When a neutron interacts with a material it does so in two ways: (1) it can be scattered by the nucleus with the probability defined as the scattering cross section σ_s, (2) it can be absorbed by the target nucleus with the probability defined as the absorption cross section σ_a. Since the cross sections are probabilities one might expect to find and does find that there is a probability, however small, for each type (scattering or absorption) to occur in any reaction. Therefore, we shall define the sum of each probability to be the total cross section for a particular reaction.

$$\sigma_t = \sigma_a + \sigma_s$$

10.4.1 Scattering Cross Section

As mentioned earlier scattering and therefore the scattering cross section (σ_s) is the sum of the inelastic scattering cross section (σ_{ie}) plus the elastic cross section (σ_e).

$$\sigma_s = \sigma_e + \sigma_{ie}$$

You will see later that the scattering cross section for most target material will be in the range of 1 to 10 barns for thermal neutrons and will decrease as the energy of the neutron is increased. At high energies (>10 kev) the scattering cross section approaches the geometrical cross section of the target nucleus.

10.4.2 Absorption Cross Section

The absorption cross section (σ_a) is composed of the sum of the probability of capture or capture cross section σ_c plus the probability of fission σ_f.

$$\sigma_a = \sigma_c + \sigma_f$$

The absorption cross section for most target material is usually divided into three categories dependent upon the neutron energy. The three regions are:

1. [0.01ev to 100ev] The low-energy region where we will find that the cross section is inversely proportional to the neutron velocity (1/v region)

2. [100ev to 10kev] The intermediate - neutron energy or resonance region where one will find sharp increases in cross section at particular energies.

[>10 kev] 3. The fast-energy region is where the cross section decreases steadily with neutron energy.

10.4.3 Absorption Cross Section vs. Neutron Energy

10.4.3.1. Low Energy Region (0.01 ev to 100ev)

In this region we find that the cross section varies as 1/V where "V" equals the velocity of the neutron and represents the time spent by the neutron within a given distance of the nuclous. Therefore, the longer the neutron is in the vicinity of the nucleus the greater the chance of interaction between the two.

$$\sigma_a \;\alpha\; \frac{1}{V}$$

For example, a plot of the low energy region of manganese illustrated the 1/V Law for neutron energies between 0.015 ev to about 6 ev. Note, however, this curve represents the total cross section, but since the scattering cross section decreases at a regular rate and the only variation in the curve is due to the absorption cross section.

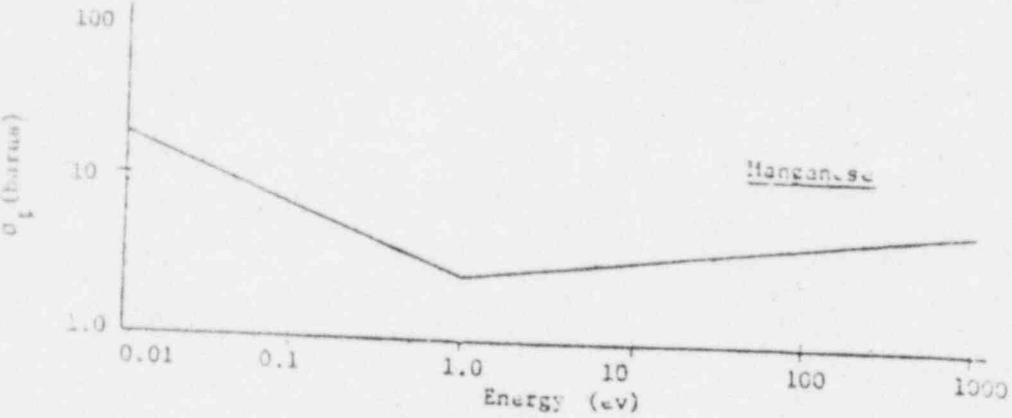


Figure 4

There are a few other low-mass nuclides which follow the 1/V low even well into the fast-neutron regions. Examples are B^{10} and Li^6.

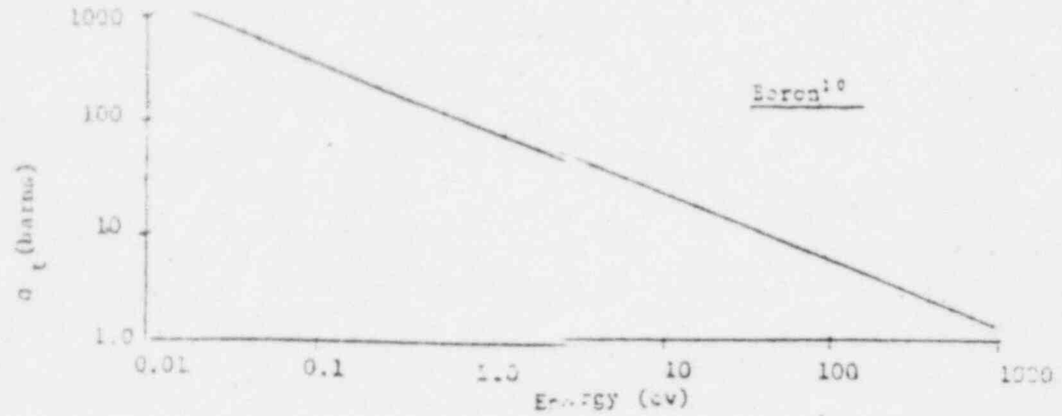


Figure 5

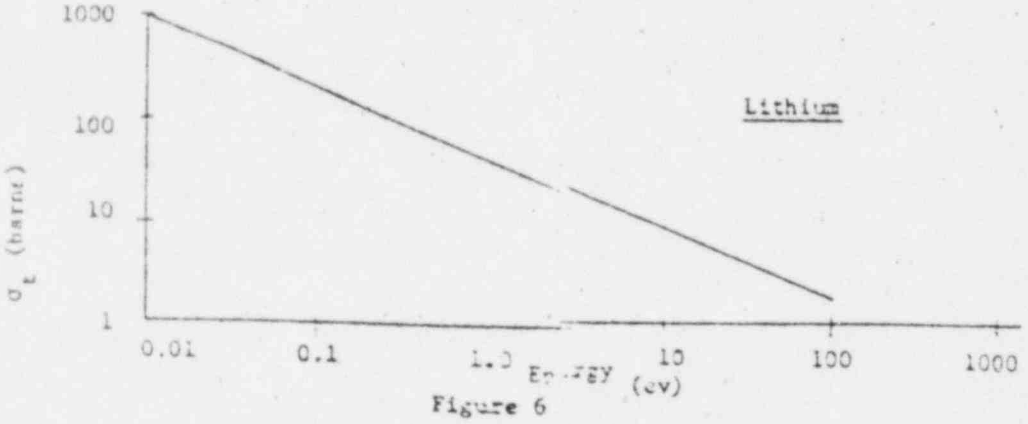


Figure 6

10.4.3.2 Intermediate Region (100 ev to 10 kev)

This region is known as the resonance and is characterized by the occurances of absorption peaks which rise sharply for some neutron energies and drop sharply between these resonance peaks. Manganese again is a good example to use to illustrate this region. (Figure 7)

We find that some resonance peaks are very large for example, Cadmium-113 exhibits a cross section of 2 x 10^4 barns for neutron energies of 0.17 ev and Xenon-135 exhibits an absorption cross section of 3.4 x 10^6 barns for neutrons of energy 0.7 ev.

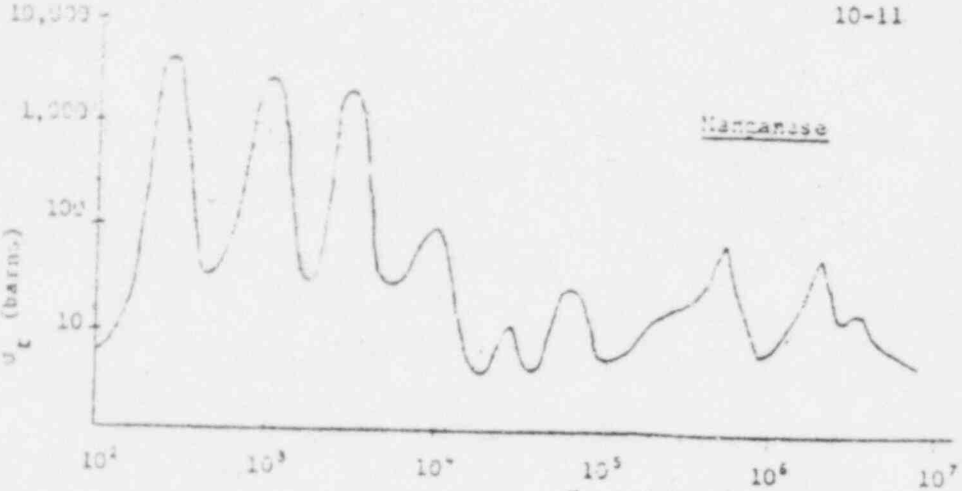


Figure 7

10.4.3.3 Fast Neutron Energy Region

In this region we find that scattering reactions predominate and therfore absorption reactions are insignificant.

At high energies we find that the elastic scattering cross section approaches the geometrical cross section of the target nuclide (πr^2).

We also find that the inelastic cross section also approaches the geometrical cross section of the target nuclide (πr^2).

Therefore, the total scattering cross section in this area is then:

$$\sigma_s = \pi r^2 + \pi r^2 = 2\pi r^2$$

where "r" (the radius of the nucleus) = 1.2×10^{-13} ($A^{1/3}$).

10.4.3.4 Scattering Cross Section for Water

The scattering cross section for hydrogen and therefore water can be represented as a function of neutron energy in the range of 1 to 12 Mev by:

$$\sigma_s = \frac{10.97}{E + 1.66} \text{ barns}$$

where E is the energy in Mev.

10.5 Neutron Shielding Equation

From the previous discussion of cross sectional variations you might expect and indeed do find that neutron shielding becomes very complicated and therefore will be discussed in general terms only.

The equation we will use for neutron attenuation is:

$$I = I_o e^{-N\sigma x}$$

where I_o = original neutron flux of single energy

I = attenuated neutron flux

N = atoms of target nuclei/cm^3

$*\sigma$ = total microscopic cross section for the reaction between a neutron of given energy and an atom of target material.

x = thickness of target material

*Note. the cross section is found either in the barn book of cross sections or approximated by the rule of thumb expressions discussed earlier.

10.6 Gamma Shielding

From our previous discussion of neutron reactions we are now aware of the fact that many neutron reactions are (n,γ type reactions. Therefore, when shielding against neutrons we must also then shield against gammas. We will now turn our attention in this direction.

10.6.1 Gamma Ray Attenuation

Gamma rays are absorbed exponentially in materials. For example, a narrow beam of parallel gamma rays of a single energy (monoenergetic), when passed through an absorber, can be expressed by the following relationship:

$$I = I_o e^{-ux} \qquad (1)$$

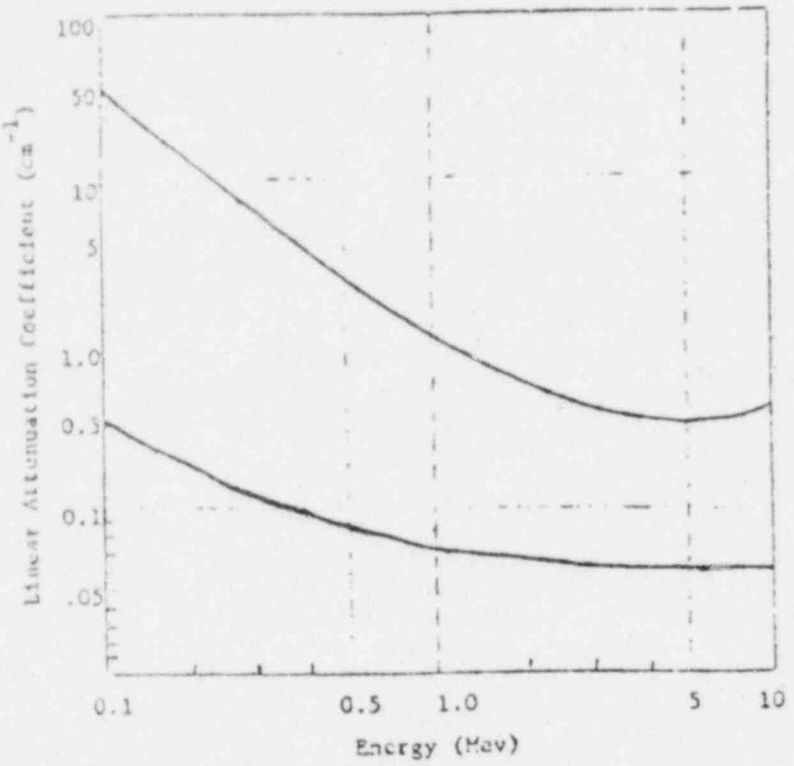
where I_o = original intensity of the source in units of intensity (mR/hr or R/hr)

I = intensity of the source after passing through a thickness of some material.

x = absorber thickness (cm)

u = the linear attenuation coefficiemt (cm^{-1})
$u = u_p + u_c + u_{pp}$

The linear attenuation coefficient (u) for some given material is the rate of reduction of the intensity of the beam through an absorber thickness. The attenuation coefficient for various materials is found through experimental analysis and is given in various published shielding manuals (See Figure 8).


Lead and Aluminum

Linear Attenuation Coefficient (cm^{-1})

Energy (Mev)	Water	Concrete	Al.	Iron	Lead
0.5	0.0966	0.204	0.227	0.651	1.64
1.0	0.0706	0.149	0.166	0.468	0.776
1.5	0.0575	0.121	0.135	0.381	0.581
2.0	0.0493	0.105	0.117	0.333	0.518
3.0	0.0396	0.0853	0.0953	0.264	0.477
4.0	0.0339	0.0745	0.0837	0.259	0.476
5.0	0.0301	0.0674	0.0761	0.246	0.483
8.0	0.0240	0.0571	0.0651	0.222	0.520
10.0	0.0219	0.0538	0.0613	0.231	0.554

Figure 8

It must be pointed out that the linear attenuation coefficient listed in Figure [illegible] is the total attenuation coefficient which includes the Compton effect, Photo Electric effect, and Pair Production.

$$u_{total} = u_c + u_p - u_{pp}$$

The attenuation coefficient is also a function of both the absorbing material and the energy of the gamma ray. It increases as the density, or atomic number, increases and decreases as the gamma ray energy increases (See Figure 8).

One should not confuse the term, "linear attenuation coefficient," with the term "mass attenuation coefficient", which is so often used in various shielding manuals. The mass attenuation coefficient is found by dividing the linear attenuation coefficient by the density of the material in question and is expressed in units of cm^2 per gram (See Figure 9).

The narrow beam concept is usually used for rough calculation and, therefore, is not very ideal for accurate calculations. The concept assumes that every scattering or absorption interaction within the medium will remove a photon from the original beam; and, since the detector is just large enough to detect the beam, it will only see the results of the attenuated original beam. In reality, this is not the whole story, for not all sources emit a narrow beam of gamma photons and not all materials used for shielding are very thin. It is also very probable that some of the scattered photons within the medium will re-enter the original beam, thus showing a higher detected intensity than that calculated by the narrow-beam concept. To account for this scattering effect, a new concept called "the broad beam concept" will now be introduced.

Mass Absorption Coefficients (cm^2/g)

Gamma Energy *	Iron	Air	Water	Lead	Al.
0.5	0.827	.090	.0966	0.144	.0805
1.0	0.596	.067	.0706	0.068	.0590
1.5	0.485	.0517	.0575	0.051	.0480
2.0	0.423	.0480	.0493	.0456	.0414
3.0	0.361	.0356	.0396	.0420	.0337
3.5	0.341	.0325	.0360	.0404	.0300
4.0	0.330	.030	.0339	.0392	.0296
5.0	0.312	.0262	.0301	.0426	.0270
10.0	---	.0216	.0219	.0452	.0218
20.0	----	.0162	.0180	----	.0201

Figure 9

*Gamma Energy in Mev.

10.6.1.1 Broad Beam Concept

The "Broad Beam Concept" is a more realistic approach for it takes into consideration the scattering events which add to the radiation received by the detector (See Figure 10).

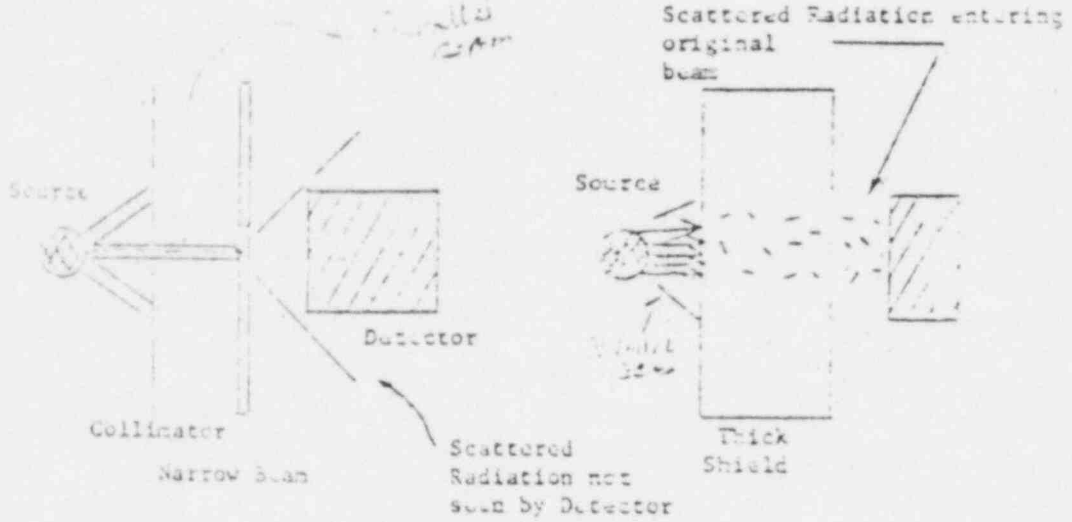


Figure 10

The equation for this concept is identical to Equation (1) except for the addition of a new term called the buildup factor (B).

$$I = I_o Be^{-ux} \qquad (2)$$

The buildup factor is a function of the shield material and thickness as well as the energy of the radiation. Its value, considering all of the aforementioned parameters, is strictly determined by experimental methods and is listed in various shielding manuals. Therefore, it is not uncommon to hear one express the buildup factor as a "fudge factor" for, indeed, this is really what it is - a means of making a wrong answer right.

Since the buildup factor is dependent upon the material used, its thickness, and the energy of the radiation source, we find that most buildup factor tables list the buildup factors as the product of the attenuation coefficient ~~times the~~ thickness of the material versus the energy of the radiation source (See Figure 11).

BUILDUP FACTORS

Material	Ux	Energy (Mev) 1	2	3	4	6	10
WATER	1	2.13	1.93	1.69	1.58	1.46	1.33
	2	3.17	2.77	2.42	2.17	1.91	1.53
	4	7.68	4.88	3.91	3.34	2.76	2.19
	7	16.2	8.46	6.23	5.13	3.99	2.57
	10	27.1	12.4	8.63	6.94	5.18	3.72
	15	50.4	19.5	12.8	9.97	7.09	4.30
	20	82.2	27.7	17.0	12.9	8.85	5.98
IRON	1	1.87	1.76	1.55	1.45	1.34	1.20
	2	2.89	2.43	2.15	1.94	1.72	1.42
	4	5.39	4.13	3.51	3.03	2.58	1.95
	7	10.2	7.25	5.85	4.91	4.14	2.99
	10	16.2	10.9	8.51	7.11	6.02	4.35
	15	28.3	17.6	13.5	11.2	9.89	7.54
	20	42.7	25.1	19.1	16.0	14.7	12.4

Figure 11

The product of ux is called the relaxation length and is defined as the thickness of absorber required to reduce the intensity of the incident radiation by a factor of "e" (2.718).

10.6.1.2 Half-Thickness Values

Sometimes it is convenient to know certain values or thickness of materials which will decrease the intensity of a gamma source by one half. This value is called the half-thickness for that material and can be derived from the following equation:

$$I = I_o e^{-ux}$$

if, x = the half thickness ($x_{1/2}$) then,

$$\frac{I_o}{2} = I_o e^{-ux_{1/2}}$$

$$\frac{1}{2} = e^{-ux_{1/2}}$$

$$2 = e^{ux_{1/2}}$$

$$\ln 2 = ux_{1/2}$$

$$0.693 = ux_{1/2}$$

$$x_{\frac{1}{2}} = \frac{0.693}{u} \qquad (3)$$

From the above Equation (3), we see that the half-thickness value for any material is simply found by dividing the ln of 2 by the attenuation coefficient for that material.

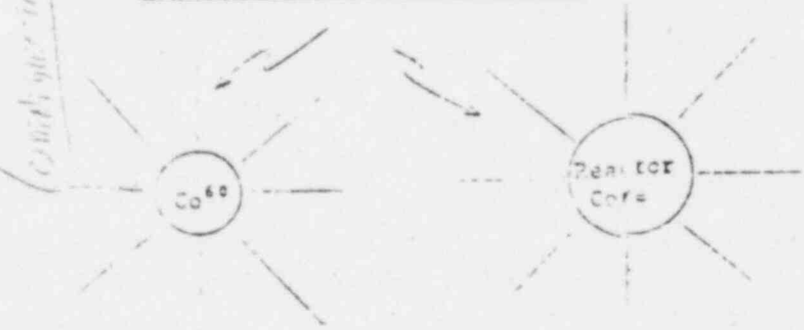
10.5.1.3 Point Source

In the previous concepts we were restricted to collimated beams, that is, narrow beams or broad beams. There was no allowance for the spreading of the radiation as one might expect to see from a point source.

The point source concept is used for the purpose of simplification when trying to determine the total number of photons or even neutrons that a source is emitting from its surface. Therefore, a point source is described as a source which emits S_p photons or neutrons per second in all spherical directions (See Figure 12).

(Omit figure [illegible]

Intensity Equal in All Directions



Point Source
Figure 12

If we now surround this source with a thin, spherical, non-absorbing shell of radius r_1, then S_p photons or neutrons per second will pass through the surface of the shell (See Figure 13).

The flux density ϕ_1, that is, the number of photons or neutrons passing through a cm^2 of the shell surface per second at radius r_1 is calculated from Equation (4) (See Figure 13).

If we now consider a second thin, spherical, non-absorbing shell of radius r_2 with $r_2 > r_1$ it is found that the flux density ϕ_2 may be calculated by Equation (5) (See Figure 14). It is assumed that attenuation of the medium is negligible.

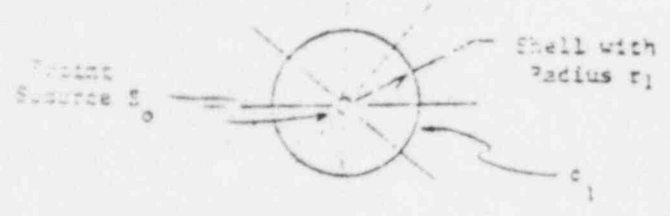


Figure 13

$$\text{Flux Density } \phi_1 = \frac{S_o}{4\pi r_1^2} \qquad (4)$$

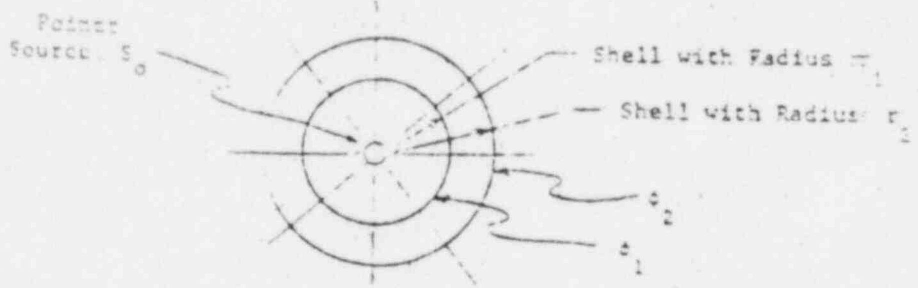
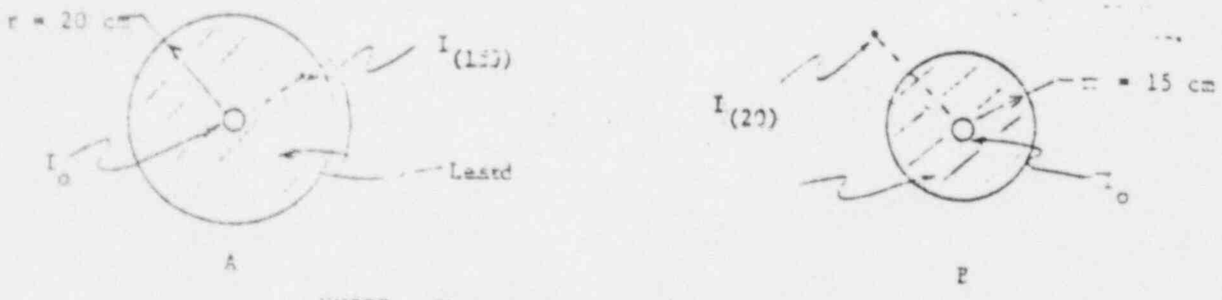


Figure 14

$$\text{Flux Density } \phi_2 = \frac{S_o}{4\pi r_2^2} \qquad (5)$$



NOTE: The student should investigate the situation depicted in Figure 15B which is similar to the example on Page 10-20

Figure 15

If we now solve Equations (4) and (5) for S_o, we have:

$$S_o = (\phi_1)(4\pi r_1^2) \quad \text{(Equation 4)}$$

$$S_o = (\phi_2)(4\pi r_2^2) \quad \text{(Equation 5)}$$

and since S_o in Equation (4) equals S_o in Equation (5), then by the use of simultaneous equations, we have:

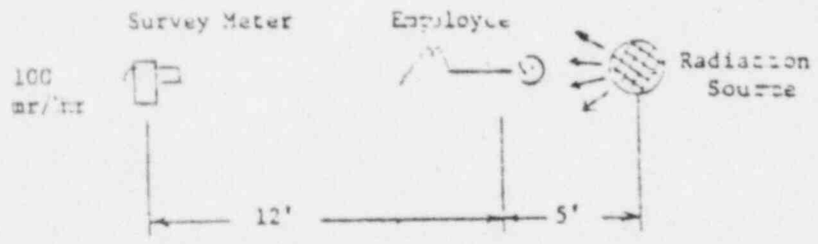
$$(\phi_1)(4\pi r_1^2) = (\phi_2)(4\pi r_2^2)$$

or

$$\frac{\phi_1}{\phi_2} = \frac{r_2^2}{r_1^2} \qquad (6)$$

Equation (6) is known as the "Inverse Square Law" which simply means that the intensity of the radiation from a source is inversely proportional to the square of the distance from the source. However, this law is only a "rule of thumb" and should only be used as a means of making rough estimate of the intensity of a source at some distance. We must next consider using the point source concept for various materials, but first look at the following problem:

Example 1 Upon re-entering the reactor control building after an emergency evacuation, you see an employee lying unconscious at the end of a long hallway. Assume that the employee is lying in direct line between you and a source of radiation and that your survey meter reads 100 mr/hr. What is the dose rate where the employee is lying (see figure below)?



Answer: $\frac{D_1}{D_2} = \frac{r_2^2}{r_1^2}$.

where: D_1 = dose rate detected by survey meter (100 mR/hr)

r_1 = distance from source to survey meter (17 feet)

r_2 = distance from source to employee (5 ft)

D_2 = dose rate at employee

Therefore,

$$\frac{100\ \text{mR/hr}}{D_2} = \frac{(5\ \text{ft})^2}{(17\ \text{ft})^2}$$

$$D_2 = 1.2\ \text{R/hr}$$

10.6.1.4 Radiation Intensity Through A Thick Shield

If, as we have learned, a source emits an intensity I_o, then the intensity at a distance "r" cm (disregarding attenuation in the material) will be:

$$I = \frac{I_o}{4\pi r^2}$$

where $4\pi r^2$ is the area of a spherical surface of radius r. To allow for the attenuation and buildup of radiation in the material, we will combine the above equation with Equation (2) and obtain the following:

$$I = \frac{I_o B e^{-ux}}{4\pi r^2} \qquad (3)$$

~~Example 1 Find the intensity of a 1 Mev gamma source, which is surrounded by a sphere of 20 cm of lead, at a distance of 15 cm. The intensity of the source is 1.5 x 10³ R/hr (See Figure 15)~~

~~Answer:~~

~~$I = \frac{I_o B e^{-ux}}{4\pi r^2}$~~

~~where, $I_o = 1.5 \times 10^3$ R/hr~~

~~$u = 0.776\ \text{cm}^{-1}$ for 1 Mev gamma~~

~~$x = r = 15$ cm~~

~~$ux = 11.64$~~

~~$B = 4.0$~~

~~Therefore, $I = \frac{(1.5 \times 10^3\ \text{R/hr})(4.0)(e^{-11.64})}{(4)(3.14)(15)^2}$~~

~~$I = 1.87 \times 10^{-5}$ R/hr/cm²~~

Not the correct formula

10.6.1.5 Dose Rate vs Source Strength

The source strength of various gamma emitting sources is usually given in units of curies where 1 curie equals 3.7×10^{10} disintegrations per second. Sometimes it is required to know the dose rate at some distance from such a source. This can be determined by use of the following equation:

$$\boxed{R/hr = \frac{6CE}{r^2}} \qquad (8)$$

where,
C = source strength in curies
E = sum total energy of photons per disintegrations in units of Mev
r = distance from the source in units of feet

Note: Equation (8) is only an approximation for gammas having an energy of between [illegible] and [illegible] Mev.

Probably the best application of this equation is in the calibration of portable survey instruments. By obtaining a standard source (Example Co^{60}) and then marking off distances from the source, a survey meter can be calibrated over its range by placing it at various marked off distances.

Example 1 Using a calibrated source of 5 curies of Co^{60}, determine what dose rate a portable ionization survey meter should read at 3, 6, and 10 ft. from the source.

Answer

$$R/hr = \frac{6CE}{r^2}$$

Since Co^{60} decays with two gamma photons having energies of 1.17 Mev and 1.33 Mev respectively, E = 1.17 + 1.33 or 2.5 Mev.

Then

$$R/hr = \frac{(6)(5\ \text{curies})(2.5\ \text{Mev})}{r^2}$$

$$R/hr = \frac{75}{r^2}$$

Therefore, (a) at 3 ft the dose rate is:

$$R/hr = \frac{75}{(3)^2} = \underline{\underline{8.34\ R/hr}}$$

(b) at 6 ft the dose rate is:

$$R/hr = \frac{75}{(6)^2} = \underline{\underline{2.08\ R/hr}}$$

(c) at 10 ft the dose rate is

$$R/hr = \frac{75}{(10)^2} = \underline{\underline{750\ mR/hr}}$$

10.6.1.6 Sample Problems

The concepts which have just been discussed are exemplified in the following example problems.

Example 1 A collimated beam of 5 Mev gammas having an intensity of 5 rem/hr is passed through a 10 inch sheet of lead. What is the intensity of the attenuated beam which emerges from the other side of the sheet?

Answer: $I = I_o e^{-ux}$

Where: $I_o = 5$ rem/hr

$u = 0.483\ cm^{-1}$ for 5 Mev gammas in lead

$$x = \frac{(10\ inches)(2.54\ cm)}{inch)} = 25.4\ cm$$

Therefore,

$$I = (5)\ e^{-(0.483)(25.4)}$$

$$I = 3.5 \times 10^{-3}\ rem/hr = 3.5\ mrem/hr$$

Example 2 Determine the half-thickness of water for 3 Mev gammas

Answer:

$$X_{1/2} = \frac{0.693}{u}$$

where, $u = 0.0396\ cm^{-1}$ for 3 Mev gammas in water

$$X_{1/2} = \frac{0.693}{0.0396}$$

$$X_{1/2} = \underline{\underline{17.5\ cm}}$$

Example 3 Consider a broad beam of 2 Mev gamma rays with an intensity of 160 rem/hr passing through a 10 inch sheet of lead. What is the intensity of the attenuated beam?

Answer: $I = I_o B e^{-ux}$

Where: $I_o = 160$ rem/hr

$u = 0.518\ cm^{-1}$ for 2 Mev gammas in lead

$$x = \frac{(10\ \text{inches})(2.54\ \text{cm})}{(\text{inch})} = 25.4\ \text{cm}$$

$$\mu x = 13.1$$

$$B = 3.5 \text{ for a 2 Mev gamma}$$

Therefore,

$$I = (160)(3.5)(e^{-13.1})$$

$$I = 1.11\ \text{mR/hr}$$

10.7 Shielding Materials

Shielding materials for different types of radiation can be classified into three distinct groups: high and medium "Z" materials, hydrogeneous materials, and capture materials, all of which serve an important function for the type of radiation shielding required.

10.7.1 High and Medium "Z" Materials

These elements are used to attenuate gammas and to slow down fast neutrons to approximately 1 Mev energy levels by inelastic scattering.

10.7.1.1 Iron and Steel

Such materials as iron and carbon steels for example, are excellent materials for use in the construction of reactor vessels for they tend to absorb the energy of gammas and fast neutrons which leave the reactor core. In fact, most of the reactor vessels built today for water cooled or moderated cores are constructed of multilayered sheets of carbon steel and stainless steel. Of course, steels are not only used for radiation shielding but also for their excellent structural properties and their ability to resist corrosion. Remember, when radiation is present, free oxygen is produced by the decomposition of water and, therefore, arises the problem of corrosion ($2H_2O \rightarrow 2H_2\uparrow + O_2\uparrow$).

10.7.1.2 Iron

Iron, having some of the same radiation attenuation qualities as steel, is unfortunately highly susceptible to corrosion and, therefore, would not be a very good material to use for a reactor vessel. On the other hand, iron would be an excellent choice for shielding purposes external to the reactor vessel. It is often used in the form of iron punchings mixed with concrete for the construction of walls and separators within the reactor containment vessel.

10.7.1.3 Lead and Iron

Lead and iron, having very high densities, are excellent gamma absorbers for gamma energies of about 2 Mev. However, at relatively higher and lower energies lead is appreciably better because of its high photo-electric absorption and pair production. Both can withstand exposures to radiation without significant damage and are also relatively inexpensive. Lead, with its high density, requires less total volume for gamma radiation shielding as compared to any other materials. The biggest disadvantage of lead is its poor structural qualities and its low melting point.

10.7.2 Hydrogeneous Materials

Materials containing hydrogen atoms are found to be excellent neutron shields and are graded as such by the number of atoms of hydrogen per volume of material.

10.7.2.1 Water

For example, water has a high hydrogen density of approximately 6.7×10^{22} atoms per cm^3 and therefore is a very good neutron shield and reactor moderator. Of course, heavy water (deuterium) is even better but is very expensive.

Neutrons which collide with hydrogen atoms decrease with energy by a large amount. In fact, a single inelastic collision would quickly reduce the energy of a fast neutron to thermal energy levels.

Although water is a good attenuator of neutrons, it is a very poor absorber of gammas. To compensate for this deficiency, it is often necessary to add small amounts of boron to the water to reduce the capture gamma ray production. Another disadvantage of water is its low boiling point at ordinary pressures, therefore being very susceptible to decomposition by radiation.

10.7.2.2 Masonite and Wood

Other hydrogeneous materials such as masonite and compressed wood having a hydrogen density of approximately 5×10^{22} atoms per cm^3 are also satisfactory neutron attenuators. When used as shielding material, it is usually found in layers between iron.

10.7.2.3 Polyethylene

Probably the most widely used hydrogeneous material for secondary or remote fast neutron attenuation is polyethylene (CH_2). It contains more hydrogen atoms per unit volume than does that of water and therefore is a slightly better attenuator. Another reason it is so widely used is because of its low temperature softening point (~240°F), and it can be molded very easily to any form and when cooled remains in a permanent set. Polyethylene like water can be borated, causing the production of secondary gammas due to the thermal neutron capture in hydrogen and carbon to be reduced and the thermal neutron absorption characteristics to be increased.

10.7.2.4 Concrete

Concrete is another excellent choice for shielding against neutrons, and in addition, it is structurally useful and very inexpensive. It has a hydrogen density of approximately 1.4 x 10^{22} atoms per cm^3 and a weight density of about 2.3 grams per cm^3. The macroscopic cross section of concrete for fast neutrons is just slightly less than that for water but is superior to water because of its great ability to also attenuate gammas.

Usually, concrete which is going to be used for shielding purposes is impregnated with some other material forming a heavy concrete mixture. For example, barytes concrete is often used for it contains the mineral barytes (barium sulfate) and has a density of 3.5 ~~g/cm^3~~.
gm/cm^3

10.7.3 Gamma Heating

One other point of interest with respect to reactor shielding materials is the effect of gamma heating which must be taken into consideration before choosing any material.

Radiation of such an intensity as that which leaves a reactor core, when absorbed in any shield, produces a great deal of heat within the shield due to the transfer of energy from both primary and secondary gamma rays in the shield. Engineers who design reactor shields must calculate this heating effect and make provisions for proper cooling, for failure to do so would lead to excessive thermal stresses, surface boiling, or even distortion of structural materials.

10.7.4 Plant Radiation Shielding

The biological protection against radiation from the reactor and its auxiliary equipment is usually classified into eight shielding categories:

1. The reactor thermal shield
2. The reactor vessel
3. The primary shield
4. The secondary shield
5. The containment vessel
6. The control room shield
7. The spent fuel shield
8. The auxiliary shields

We will now briefly discuss each of them.

10.7.4.1 The Reactor Vessel

The purpose of the reactor vessel is to house the core, control mechanisms, and coolant and to withstand the operating pressures within the vessel during operation (See Figure 3-45 Met Ed).

The vessel is normally constructed of layers of high strength stainless steel which also serves as an attenuator of radiation from the reactor core. However, direct neutron activation and gamma heating would be detrimental to the steel. To solve this problem, a thermal shield is inserted between the reactor coolant and the vessel, thus subjecting the vessel wall to an attenuated flux.

10.7.4.2 Thermal Shield

Since, as we have learned, all of the radiation which is absorbed in the reactor shield is from fast neutrons and gammas and which is eventually degraded into heat, a thermal shield is employed between the reactor core and vessel to protect the vessel wall from damage due to direct radiation (See Figure 3-46 Met Ed).

The thermal shield is usually constructed of a very dense material (Example - stainless steel cylinders) with a high melting point and, therefore, is a good attenuator of gamma radiation and an inelastic scatterer for fast neutrons. The heat which is absorbed by the thermal shield is eventually removed by the reactor coolant.

10.7.4.3 The Primary Shield

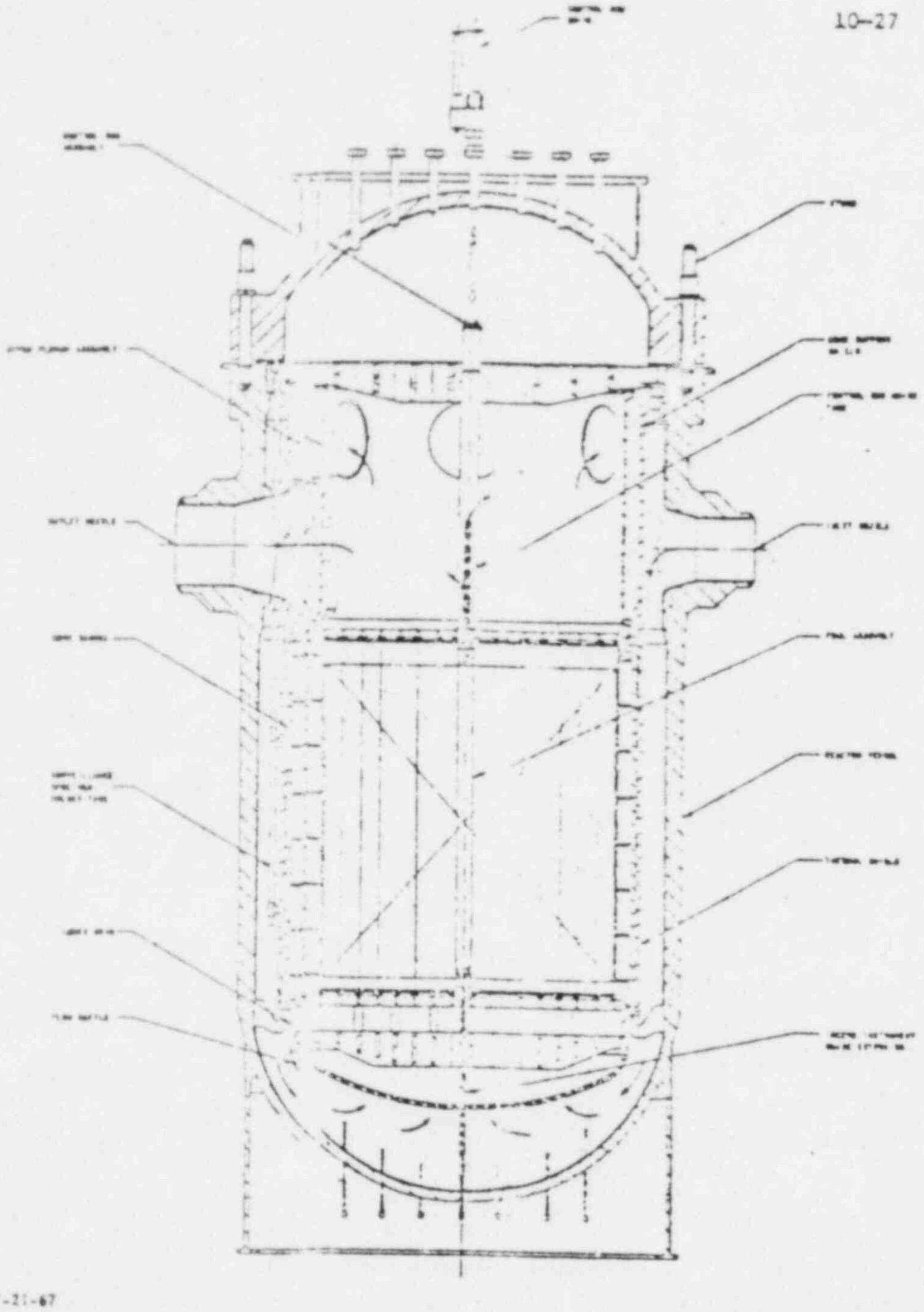
The primary shield or wall consists of some form of reinforced concrete having a density of approximately 2.3 gm/cm^3 and a wall thickness of about 5 feet. The wall surrounds the reactor vessel and attenuates the radiation escaping the reactor vessel (See Figure 1-8 Met Ed).

The amount of radiation attenuation by this wall is usually enough to reduce the radiation level enough to allow personnel to enter the containment vessel for limited periods of time during full power operation.

The primary shield also serves to reduce the neutron activation of the various components and structural materials within the containment vessel.

10.7.4.4 The Secondary Shield

The secondary shield or wall also is constructed of reinforced concrete (about 5 feet thick) and surrounds the various auxiliary equipment within the containment, such as coolant pumps, pipes and steam generators.



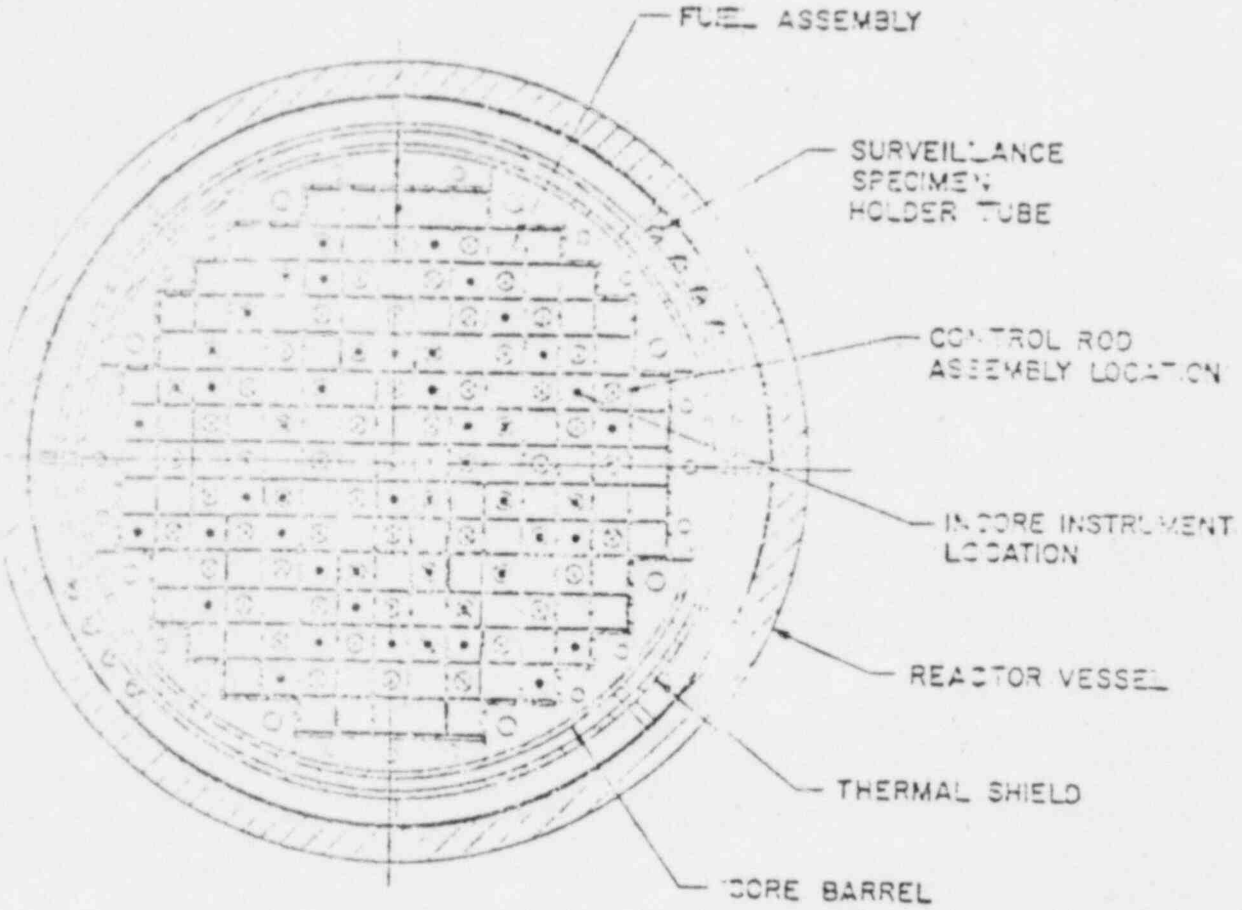
REV: 7-21-67
REVISED UPPER PLENUM
ASSEMBLY AND CONTROL DRIVE

REACTOR VESSEL & INTERNALS GENERAL ARRANGEMEN

FIGURE 3-45

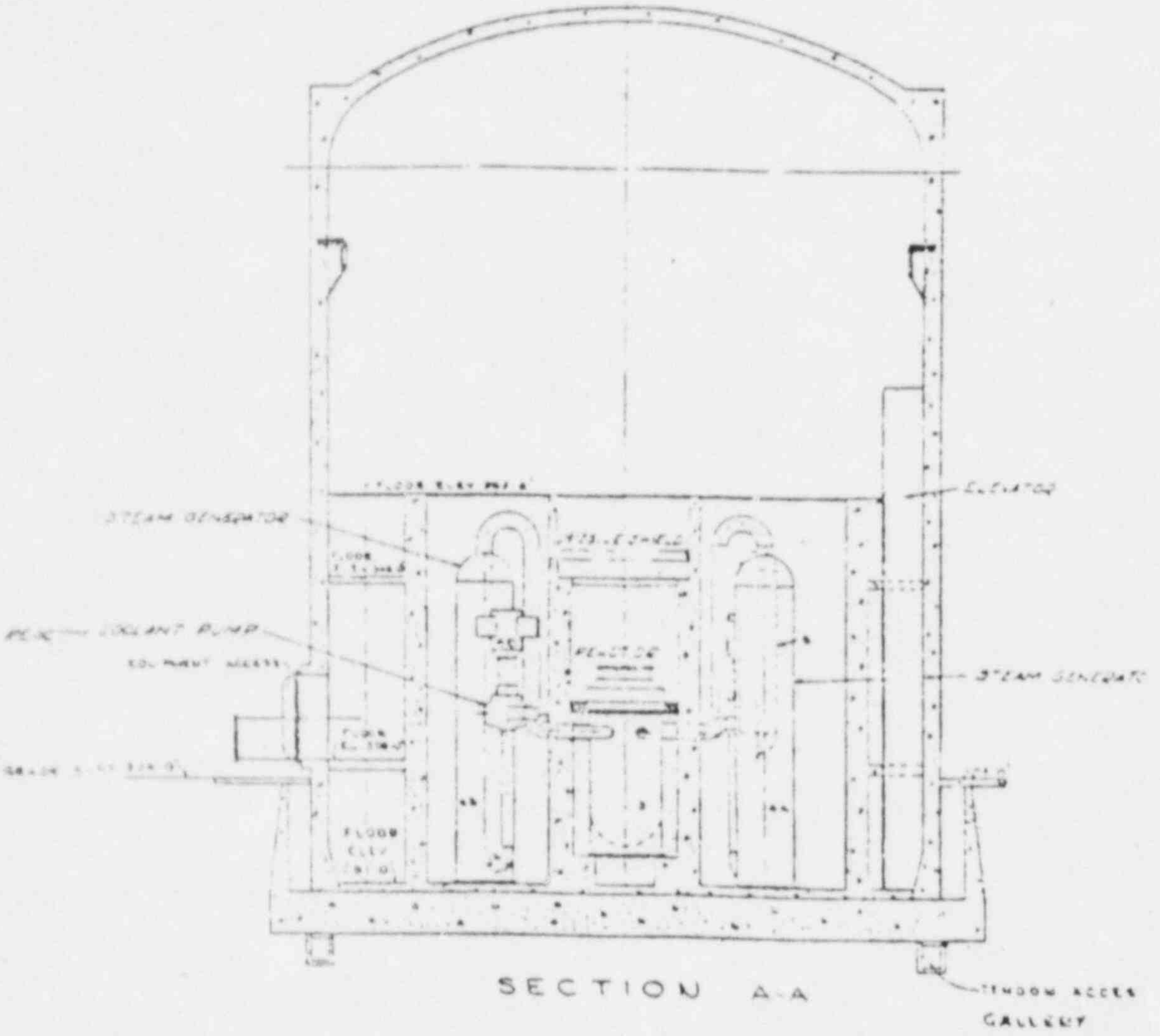
THREE MILE ISLAND NUCLEAR STATION

AMEND. 1 (7-21-67)



REACTOR VESSEL & INTERNALS GENERAL ARRANGEMENT
CROSS SECTION

FIGURE 3-46
THREE MILE ISLAND NUCLEAR STATION



SECTION A-A

SCALE

GENERAL CROSS SECTION A-A & B-B

FIGURE

THREE MILE ISLAND NUCLEAR STATI

AMEND. 6 (1-8-68)

Since all of the auxiliary equipment could contain activated coolant impurities and possibly fission products from leaky fuel elements, the secondary shield will protect personnel entering the containment vessel from direct radiation from these sources (See Figure 1-8 Mac Ed).

10.7.4.5 The Containment Vessel

The containment vessel, which surrounds all of the previous mentioned shields, is usually constructed of a reinforced, prestressed concrete (approximately 3.5 feet thick) and serves to further attenuate the secondary radiation from the primary and secondary shields to levels usually below 1 ~~Mr/hr~~ mR/hr.

10.7.4.6 The Control Room Shields

The amount of shielding surrounding the control room is based upon the maximum hypothetical accident. That is, considering the MHA for a reactor, the amount of shielding needed is based upon the calculation of shielding material needed to reduce the radiation level in the control room to a level less than the quarterly whole body exposure limits as defined in 10 CFR 20 (Code of Federal Regulations) during such an accident. In addition to the shielding requirements, special ventilation requirements are also needed. However, it is not necessary to discuss this requirement at this time.

10.7.4.7 Spent Fuel Shielding

During the removal of fuel for storage, shielding is needed to protect personnel performing the operation. The transfer of spent fuel is usually through specially designed transfer canals filled with borated water. In addition, the fuel elements are kept at minimum depths below the water surface to keep the radiation level at a minimum. Therefore, the combination of the concrete walls in the canal and the depth of the water is enough to insure a safe working exposure of individuals.

10.7.4.8 Auxiliary Shields

Many auxiliary shields are needed to attenuate the radiation from various components such as the purification systems, makeup systems, sampling systems, waste disposal systems, and others. Each shield would have its own characteristics and shape and, therefore, will not be pursued any further into our discussion.

GREEK ALPHABET

NAME	UPPER CASE	LOWER CASE	APPLICATION
Alpha	Α	α	Particle
Beta	Β	β	Particle, delayed neutron fraction
Gamma	Γ	γ	Photon, effectiveness factor
Delta	Δ	δ	Change, small amount
Epsilon	Ε	ε	Fast fission factor
Zeta	Ζ	ζ	
Eta	Η	η	Reproduction factor, efficiency
Theta	Θ	θ	Angle
Iota	Ι	ι	
Kappa	Κ	κ	
Lambda	Λ	λ	Wavelength, decay constant
Mu	Μ	μ	Micro, attenuation coefficient
Nu	Ν	ν	Frequency, neutrons/fission
Xi	Ξ	ξ	Energy decrement
Omicron	Ο	ο	
Pi	Π	π	Ratio of circumference to diameter
Rho	Ρ	ρ	Density, reactivity
Sigma	Σ	σ	Cross section, summation
Tau	Τ	τ	Fermi age, period
Upsilon	Υ	υ	
Phi	Φ	φ	Flux, fluence
Chi	Χ	χ	
Psi	Ψ	ψ	Poisoning
Omega	Ω	ω	Reciprocal time