CANDU Fundamentals

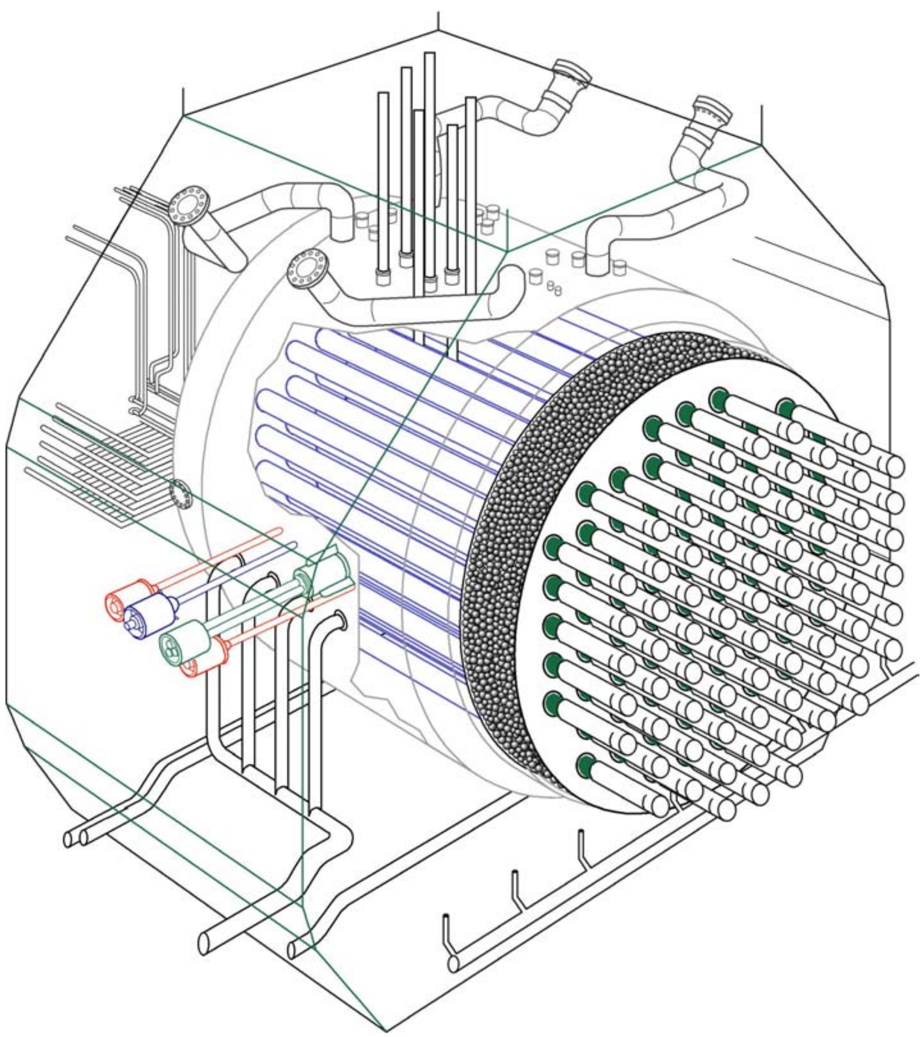


Table of Contents

1 Objectives

1.1 Course Overview

CANDU Fundamentals is designed as an introductory course in the operation of a CANDU reactor. Starting with the basics of atomic theory the course explains the construction of the reactor, its major systems and enough reactor physics so that the participant will understand the control and operating practices in a CANDU plant. Emphasis is placed on nuclear safety and the systems that minimize the risk from fission products in the reactor core.

1.2 Atomic Structure

- Name the fundamental atomic particles.
- State the mass and electric charge if the fundamental particles.
- Describe an atom as pictured by the Bohr model.
- Recognize, interpret, and use the $^{A}_{Z}X$ notation for atoms (nuclides).
- Describe what the term isotope means.
- Recognize and use the names of the hydrogen isotopes.

1.3 Radioactivity – Spontaneous Nuclear Processes

- Write a typical equation for the production of each type of radiation: α, β, and γ.
- List the properties of each type of radiation: α, β, and γ.
- Discuss how each type of radiation interacts with matterα, β, and γ.
- State how to shield against alphas and betas.
- State how to shield against γ rays and calculate γ ray shielding in terms of ½-value layers.

1.4 Nuclear Stability and Instability

- Discuss the stability of nuclides in terms of neutron proton ratios and forces in the nucleus.

- From a plot of n against p, state the emission a given nuclide is likely to undergo.

- Given a chart or table of nuclides, list all members of the decay chain of a given radioactive nuclide.

1.5 Activity

- Define the common units of activity: Becquerel, Curie.

- Define the half-life and discuss activity in terms of half-lives.

- Solve simple activity and half-life calculations.

1.6 Neutrons and Neutron Interactions

- Write equations describing each of the following neutron interactions: transmutation, radiative capture, photoneutron reaction with deuterium.

- Describe elastic and inelastic scattering of neutrons.

1.7 Fission

- State where the energy released by fission comes from (mass to energy conversion).

- Write a typical fission reaction.

- State how much energy a fission releases.

- State how the major portion of the energy from fission is carried away.

- Define each of the following: thermal neutrons, fast neutrons, prompt neutrons, delayed neutrons.

- Describe what is meant by neutron cross-section and neutron flux.

- Explain a self-sustaining chain reaction.

1.8 Fuel, Moderator, and Reactor Arrangement

- Explain the purpose of a moderator.

- Sketch the basic arrangement of fuel and moderator in a CANDU reactor.
- Explain why the arrangement of fuel and moderator in a CANDU reactor is used.
- State the basic differences between fresh and equilibrium fuel.
- Compare the moderating properties of heavy water, light water, and graphite.

1.9 Nuclear Safety

- Describe the makeup of background radiation, and state how much nuclear stations contribute to it.
- State what is meant by Defence in Depth.
- Describe the five parts of the Defence in Depth model.
- List the five barriers that protect the public from fission products.
- Explain how the principle of Control, Cool and Contain guides reactor operation.
- Explain how the following concepts contribute to the reliability and/or availability of systems or equipment: redundancy, independence, diversity, periodic testing, fail-safe operation, operational surveillance, preventive maintenance, and predictive maintenance.
- State the purpose of the following documents: Safety Report, Station Operating Licence, Operating Policies and Principles (OP&Ps), Operating Procedures, and Certificates of Approval.
- State the possible consequences of an OP&P violation.
- Describe how the authorization of some positions supports nuclear safety.

1.10 Nuclear Power Reactors

- State the type of fuel, coolant and moderator used in a CANDU reactor.

- Give the advantages and disadvantages of a nuclear power plant compared to a fossil fuelled plant for each of the following: economics, energy production flexibility, environmental considerations.

1.11 CANDU Reactor Construction

- State the function of each of the following reactor components: pressure tube, calandria rupture disc, calandria tube, annulus gas, end fitting, feeder pipe, closure plug, fuelling machine, fuel latches, biological and thermal shield, end shield, shield plug, shield tank, reactor vault.

- Given a diagram of a fuel channel or an end fitting in a reactor face, label the following components: end fitting, channel closure plug, feeder coupling, annulus bellows, journal bearings, fuel bundle, pressure tube, calandria tube, calandria-side tube sheet, liner tube, shield plug, end shield, fuelling machine-side tube sheet.

- State the purposes of the annulus gas system.

- Describe three advantages of the CANDU pressure tube design compared to a pressure vessel design.

1.12 Moderator and Moderator System

- Define the term isotopic and state the importance of maintaining the isotopic as high as possible.

- State the purpose of a D_2O upgrader.

- State the three main radioactive isotopes produced in the moderator.

- Describe the radiation hazard associated with each isotope.

- State the two major reasons why the rate of tritium production in the moderator is greater than in the heat transport coolant.

- State two purposes of the moderator circulation system.

- State the three main moderator heat sources.

- State why a backup cooling arrangement is needed.

- Given a diagram of a moderator, circulating system label the following major system components: pumps, heat exchangers, moderator temperature control valves.

- Describe how moderator temperature is controlled.

1.13 Moderator Cover Gas System & Moderator Auxiliary Systems

- Given a diagram of the main moderator system, show where the purification system is connected.

- State the three major functions of a moderator cover gas system.

- State two additional functions for the moderator cover gas system for a reactor with a dump tank.

- Given a diagram of a moderator cover gas system label the following: compressors (two functions), recombination unit, heat exchanger, recombination unit inlet heater, flame arrestors.

- State the function of each of the following components of the moderator cover gas system: compressors (two functions), recombination unit, heat exchanger, recombination unit inlet heater, flame arrestors.

- State the role of gas chromatography in the cover gas system.

- State why the following gases are sometimes added to the cover gas system: oxygen, helium.

- State the function of each of the following moderator auxiliary systems: purification system (two functions), liquid poison addition system, D_2O collection system, auxiliary cooling.

- State the importance of maintaining the chemical purity of the moderator D_2O as high as possible.

- Given a diagram of the moderator purification system, label the following major system components and state their function: ion exchange columns (IX columns), filters, strainers, purification coolers.

- Name the two neutron absorbing poisons added to moderator D_2O.

- List four points from which moderator D_2O is routinely collected.

1.14 Heat Transport System

- State the two purposes of the heat transport coolant.

- Discuss the nuclear and conventional hazards that may be present around heat transport system equipment in the plant.

- Given a diagram of a typical heat transport system (HTS), label the major components: HTS Circulation pumps, fuel channel, reactor inlet header, reactor outlet header, feeders (feeder pipes), boilers/steam generators.

- Explain why bi-directional flow is used in HTS operation.

- State why HTS pumps are placed after the boilers in the main HTS circulating system.

- State the purpose of the shutdown cooling system.

- Explain the importance of circulation by natural convection to HTS operation.

1.15 Heat Transport Auxiliary Systems

- State the two major purposes of the heat transport pressure and inventory control system.

- Given a diagram of a pressure and inventory control system, label the diagram showing the following components: pressurizer, pressurizer steam bleed valves, pressurizer heaters, feed valves, bleed valves, feed pumps (pressurizing pumps), bleed condenser, bleed cooler, D_2O storage tank.

- State the function of each of the following components: pressurizer, pressurizer steam bleed valves, pressurizer heaters, feed valves, bleed valves, feed pumps (pressurizing pumps), bleed condenser, bleed cooler, D_2O storage tank.

- State the function of each of the following HTS auxiliary systems: Pressure relief system, Purification system, HTS D_2O

collection system, HTS D_2O recovery system, Fuelling machine D_2O supply.

- State the importance of chemical control in the HTS.
- State the function and basic principle of operation of the HTS pump gland and gland seal supply system.
- State why HTS D_2O is not added to moderator and why moderator D_2O is not added to HTS D_2O.
- State two reasons why there is more leakage from the heat transport system than from the moderator system.
- List four typical heat transport system leakage collection points.

1.16 Reactor Fuel

- Define the term failed (defective) fuel.
- State how normal CANLUB fuel is used and why it has this name.
- Given a diagram of a fuel bundle, label the following components: fuel element, fuel sheath, fuel pellet, end plate, bearing pad.
- State seven characteristics (two nuclear and five non-nuclear) that the fissile fuel material should have.
- State four characteristics (one nuclear and three non-nuclear) that the fuel sheathing material should have.
- Give three ways the CANLUB graphite layer helps prevent fuel defects.
- Explain six precautions taken in the handling of fresh fuel.
- Explain three precautions required when handling spent fuel.
- Describe the general process followed during refuelling of a channel.
- State three in-plant operational consequences of leaving failed fuel in the reactor.

- State and explain two uses of depleted fuel and give the U-235 concentrations for normal and depleted fuel.

1.17 Neutron life cycle

- Sketch the life cycle of a neutron including all possible fates of the neutron.
- Discuss why reactors use reflectors.

1.18 Criticality and Neutron Multiplication

- Define the neutron multiplication constant (k).
- Define reactivity (Δk) and state its common units.
- Discuss what is meant by sub-critical, critical, and super-critical in terms of the values of k and Δk and state whether power is increasing, decreasing, or remaining constant.
- State and explain how the reactor can be critical at any power level.
- Given a method of criticality control, discuss how it affects the neutron cycle.

1.19 Changes In Reactor Power With Time

- Define Reactor Period.
- Explain why and how delayed neutrons affect changes in reactor power.
- Explain why power does not drop to zero in a sub-critical core.

1.20 Xenon: A Fission Product Poison

- Explain why xenon is the most important fission product poison.
- Explain how xenon is produced in, and how it is removed from the reactor.
- Sketch xenon concentration as a function of time for a shutdown or trip from full power.
- Discuss the phrase xenon poison out.

1.21 Reactivity Effects of Temperature Changes

- Define the following terms: temperature coefficient of reactivity, void reactivity, power coefficient
- Explain why and how reactivity changes when the temperature of the fuel changes.
- Explain why a negative temperature coefficient is desirable.

1.22 Neutron Flux Control

- Explain why a flat flux distribution is desirable.
- Explain how each of the methods used in CANDU reactors flattens the flux.
- Explain what flux oscillations are and how liquid control zones are used to prevent them.

1.23 Reactivity Mechanisms

- State how core reactivity (long term) is normally maintained.
- State the two general functions of reactivity mechanisms.
- For each of the following reactivity mechanisms, state whether it is used for reactor power regulation or protection: liquid zone control, absorber rods, adjuster rods, shutoff rods, liquid poison addition, liquid poison injection, moderator level, moderator dump.
- Describe the principle of operation of each of the three types of protective shutdown systems found in CANDU reactors.
- Describe how the two-out-of-three trip logic triggers an emergency shutdown system.
- State three advantages of using two-out-of-three trip logic.
- Define the term fail-safe.
- Describe how fail-safe design contributes to the reliability of a safety shutdown system.

1.24 Emergency Coolant Injection & Containment

- Name the four special safety systems designed to protect the public from radiation.

- Explain the purpose of the containment system.
- Explain the purpose of the emergency coolant injection system.
- Given a diagram of the emergency coolant injection system: high pressure water supply, isolation (or injection) valves, recovery sump, recovery pumps, recovery heat exchangers, low pressure water.
- Explain the purpose of each of the following components in an ECI system: high pressure water supply, isolation (or injection) valves, recovery sump, recovery pumps, recovery heat exchangers, low pressure water
- Describe the basic operation of the emergency coolant injection system in response to a loss of coolant accident (LOCA).
- Describe two types of containment systems used for CANDU reactors.
- Given a diagram of a negative pressure containment system, label the following components: the dousing tank, pressure relief valves, a vacuum duct, the pressure relief duct, the containment structure, and the vacuum building.
- Given a diagram of a pressure suppression containment system, label the following components: the dousing tank, a dousing valve, the containment structure.

1.25 Conventional Side

- State the two major functions of the steam and feedwater cycle.
- Describe the condition of the steam (moisture, temperature, pressure) at each of the following locations in the steam path: outlet of high-pressure turbine, outlet of moisture separator, outlet of reheater, and outlet of low-pressure turbine.
- State the function of the turbine-generator set.
- Describe the need for both high and low pressure turbines.
- Explain the advantage of having a vacuum in the condenser during operation.

- Given a diagram, label the following components, show the main steam and water connection between them, and indicate the direction of flow: boiler (steam generator), safety valve, atmospheric steam discharge valve (ASDV), emergency stop valve, governor valve, HP turbine, moisture separator, reheater, LP turbine, condenser, condensate extraction pump, feedheaters, deaerator, and boiler feed pump.

- State the purpose of each of the following components of the steam and feedwater cycle: boiler (steam generator), safety valve, atmospheric steam discharge valve (ASDV), emergency stop valve, governor valve, HP turbine, moisture separator, reheater, LP turbine, condenser, condensate extraction pump, feedheaters, deaerator, and boiler feed pump.

- Explain how an alternative heat sink is provided if the turbine is unavailable for operation.

- Describe the process for preparation of the condensate before it is returned to the boiler.

- Describe the turbine generator lubricating system.

- State the purpose of the turning gear.

- State the reasons and methods for maintaining good chemistry in the steam and feedwater.

1.26 Other Major Systems

- Describe how an AC generator produces electrical energy.

- Explain how heat is generated in and removed from a large AC generator.

- Describe the major hazards associated with the boiler, turbine, generator, steam and feedwater system.

- State the purpose of each of the following: main output transformer, switchyard, unit service transformer, and system service transformer.

- Name the four classes of power used in a CANDU station, and explain the purpose of these classifications.

- Name the power source which supplies power when one or both Class IV power sources fail.
- State the function of the Emergency Power Supply (EPS).
- Given a diagram, label the following light water systems: water treatment, condenser cooling water, common service water, unit low pressure service water, unit high pressure recirculating service water, and closed loop demineralized service water system.
- State the purpose of each of the following light water systems: water treatment, condenser cooling water, common service water, unit low pressure service water, unit high pressure recirculating service water, and closed loop demineralized service water system.
- State the purpose of the Emergency Water System (EWS).
- State the purposes of three separate air systems in a nuclear generating plant.
- State how equipment is identified in the station and on flowsheets.
- Describe how piping systems are coded and why.
- Briefly, describe major features of waste management for liquid and solid waste.
- Describe how D_2O is managed in a CANDU station to minimize losses.
- Describe the purpose and process of the tritium removal facility.

2 ATOMIC STRUCTURE

Nearly 2500 years ago Greek scholars speculated that the substances around us are made of tiny particles called atoms. A limited number of different kinds of atoms in various combinations can construct the whole vast array of nature. John Dalton, a nineteenth century English Chemist, formalized the modern theory of atoms in six basic points:

1. Ordinary matter is composed of particles called atoms.

2. Atoms are far too small to be observed with the naked eye.

3. Different chemical elements are made of atoms with different masses.

4. All atoms of the same chemical element are identical.

5. Atoms combine in simple ratios to form new substances but the atoms themselves remain unchanged.

6. Atoms cannot be divided, created, or destroyed.

With a minor change to point 4 to account for isotopes (see below) the first five points are correct. The twentieth century has demonstrated point 6 to be wrong. Atoms can be split into more fundamental particles: the proton, neutron, and electron.

2.1 Fundamental Particles

2.1.1 Proton

A proton is a very small particle. Its diameter is about one hundred thousandth ($1/100{,}000 = 10^{-5}$) of the diameter of a hydrogen atom. The diameter of a hydrogen atom is about one ten-billionths of a meter, 10^{-10}m.

The proton carries a single unit positive charge (+1e) and has a mass of about one mass unit (1u). The mass unit is very small: $1u=1.66 \times 10^{-27}$ kg. The proton mass is 1.0073 u and most of the time we round this off to 1u.

2.1.2 Neutron

A neutron is a neutral (uncharged) particle the same size as the proton. Its mass is 1.0087 u, about 2½ electron masses heavier than the proton. We generally approximate its mass as 1u.

2.1.3 Electron
An electron is the smallest of the three fundamental particles, with a mass of only 0.000 548 u, about 1/1840 of the mass of a nucleon. (Nucleon is a name for either of the two heavier fundamental particles.)

The electron carries a unit negative charge (-1e). The unit charge is so small it takes a flow of 6.24 trillion charges a second to measure a current of one microampere.

2.2 Structure Of Atoms
We shall now see how to assemble these fundamental particles to make atoms.

Danish physicist Niels Bohr received the Nobel Prize in 1922, for his theory about the structure of atoms. The picture that the Bohr model presents is of a small clump of protons and neutrons (the nucleus) surrounded by electrons in orbit. The atom according to this model, is like a very small solar system, with the electric force between the positive nucleus and the negative electrons playing the role of gravity.

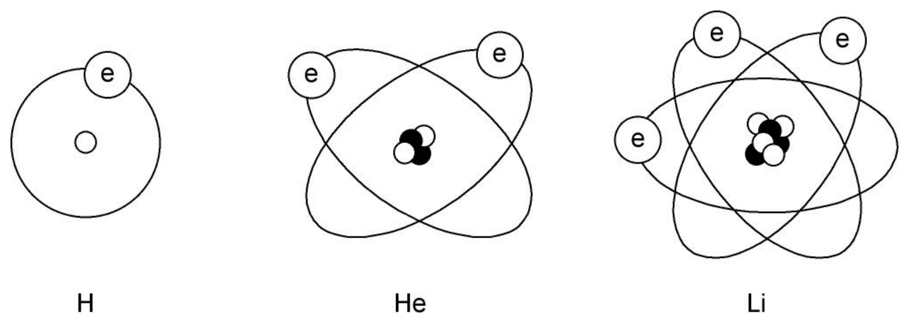


Figure 2.1

Each atom of an element has a characteristic number of protons in its nucleus. A neutral (that is, uncharged) atom has the same number of electrons in orbit as there are protons in the nucleus. Figure 2.1 illustrates the first three elements: hydrogen, helium, and lithium.

There are over 110 chemical elements. Ninety elements exist naturally in the world around us and most of these (81) are made of stable atoms. A few unstable atoms (i.e., radioactive atoms) also occur naturally, and the "man-made" atoms are all unstable.

2.3 Atomic Notation
Each kind of atom can be identified and represented by its chemical symbol, its atomic number (number of protons), and its atomic mass number (equal to the number of nucleons) as follows:

□ $^{A}_{Z}X$ Where: Z = Atomic Number

X = Chemical Symbol

A = Atomic Mass Number

The symbol $^{A}_{Z}X$ represents a neutral atom of chemical element X.

For example, the three elements of Figure 2.1 are:

Hydrogen $^{1}_{1}H$ (1 proton, 1 electron)

Helium $^{4}_{2}He$ (2 protons, 2 neutrons, 2 electrons)

Lithium $^{7}_{3}Li$ (3 protons, 4 neutrons, 3 electrons)

Since the number of protons uniquely determines the chemical symbol, we often write these in a simpler way:
$^{4}_{2}He$ becomes He-4 or helium-4.

2.4 Isotopes

The lithium atom in Figure 2.1 has 3 protons and 4 neutrons in its nucleus. Only 92.5% of naturally occurring lithium atoms are like this. The other 7.5% of lithium atoms have three protons and three neutrons. We call these different kinds of lithium atoms isotopes of lithium. The symbols Li-6 and Li-7 represent them.

Isotopes of an element have the same number of protons in their atoms but varying numbers of neutrons. All isotopes of a given element have similar chemical and physical properties but may show very large variations in nuclear properties (in lighter nuclei the mass varies greatly between isotopes).

Isotopes of the elements hydrogen and uranium are particularly significant in this course. Hydrogen has three isotopes (hydrogen, deuterium, and tritium) shown in Figure 2.2. The first two occur naturally, although deuterium is only 0.015 % abundant (about one atom in every 7000). CANDU reactors use deuterium in the form of heavy water (D_2O) to slow fast neutrons and to carry heat from the fuel. Heavy water production requires an expensive separation process (discussed in another course).

The third isotope, tritium, is produced in CANDU reactors. It is radioactive and can be a serious health hazard.

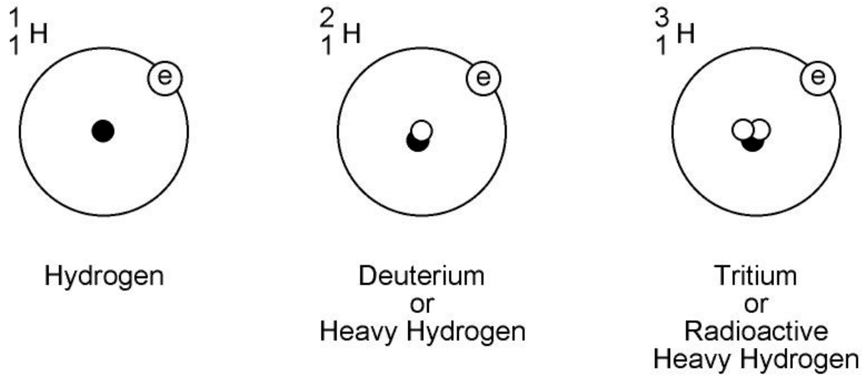


Figure 2.2

Natural uranium used for CANDU fuel has two isotopes, $^{238}_{92}U$ and $^{235}_{92}U$. U-235 is 0.7% abundant and will fission (split, releasing energy) when struck by a low energy (slow speed) neutron. It is said to be fissile. U-235 is the only naturally occurring fissile material. U-238 is 99.3% abundant and is not fissile. Nevertheless, it strongly affects the behaviour of nuclear fuel, as we will see later.

The existence of chemical elements with different atoms could cause confusion. When the numbers of nucleons in the nucleus of an atom are specified (i.e., both Z and A) the word nuclide sometimes replaces the word atom. For example, "The Chart of Nuclides" is a chart setting out the properties of each distinct atomic type (all the isotopes of each element).

2.5 Summary Of Key Ideas

- Atoms are made of protons, neutrons and electrons.
- Protons have a mass of 1 amu and a positive electric charge. Neutrons have a mass of 1 amu and no electrical charge. Electrons have a mass of 1/1840 amu and a negative electrical charge.
- The number of protons and hence the number of electrons in a neutral atom determine the chemical and most of the physical properties of an atom.
- Isotopes are atoms with the same number of protons and different numbers of neutrons in the nucleus.

- Isotopes of an element behave differently in a nuclear reaction.
- ${}^{A}_{Z}X$ notation is the standard for specifying nuclides.

.

2.6 Assignment

1. List the mass, charge, and atomic location of each of the fundamental particles.

2. Define an isotope.

3. Describe the atomic structure of ${}^{3}_{1}H$

4. Sketch the atom ${}^{10}_{5}B$.

5. Define or describe each of the following: atom, element, nuclide, nucleon, atomic number, mass number, and mass unit.

3 Radioactivity - Spontaneous Nuclear Processes

Becquerel was the first to detect radioactivity. In 1896 he was carrying out experiments with fluorescent salts (which contained uranium) and found that his photographic plates had been exposed despite being well wrapped against light. The penetrating "rays" he discovered were later shown to be of three distinct types: alpha particles (α), beta particles (β), and gamma rays (γ).

3.1 Types of Emissions

All nuclides of atomic number greater than 83 are unstable (that is, radioactive) and eventually decay (or disintegrate) by emitting an alpha particle or a beta particle. The new nuclides formed (daughter nuclides) also decay until a stable nuclide of atomic number 83 or less is formed. There are also several naturally occurring radioactive nuclides with mass number less than 83, and many artificial radioactive nuclides have been discovered.

3.1.1 Alpha Emissions

The alpha particle is emitted, typically, from a heavy nuclide such as U-238. This is expressed as:

$$^{238}_{92}U \rightarrow {}^{234}_{90}Th + \alpha$$

Examination of the alpha particle shows it is a helium-4 nucleus so it is sometimes written:

$$^{238}_{92}U \rightarrow {}^{234}_{90}Th + {}^{4}_{2}\alpha$$

(parent) $\rightarrow$ (daughter) + (α)

or $$^{A}_{Z}X \rightarrow {}^{A-4}_{Z-2}X + \alpha$$

These equations represent a parent nucleus emitting a fast moving α particle (helium-4 nucleus), producing a new daughter nucleus.

The alpha particle does not have any electrons (remember it is a helium nucleus) and therefore has a charge of +2e, (usually given simply as +2). The mass of the alpha particle is 4.0015 u and its speed when first emitted is typically a few percent of the speed of light.

3.1.2 Beta Emissions

Beta particles are emitted by neutron-rich nuclides, i.e., a nuclide with too many neutrons. This is a typical example:

$$ {}^{90}_{38}Sr \rightarrow {}^{90}_{39}Y + \beta^{-} $$

You may put the mass number and charge number onto the symbol if desired, giving:

$$ {}^{90}_{38}Sr \rightarrow {}^{90}_{39}Y + {}^{0}_{-1}\beta $$

or $$ {}^{A}_{Z}X \rightarrow {}^{A}_{Z+1}X + \beta^{-} $$

As noted from the above expressions the daughter nuclide from beta decay appears one position higher in the periodic table of elements. A neutron in the nucleus has changed into a proton so the atomic number goes up one, but the atomic mass number remains the same.

The beta particle is a very fast moving electron originating in a nucleon inside the nucleus. It has the same mass as any electron, 0.000548 u, and the same charge, -1. The speeds of beta particles range from about 90 to 99% of the speed of light.

3.1.3 Gamma Emissions

An alpha or beta emission usually leaves the disrupted daughter nucleus in an excited state. Excited states are not the same as unstable nuclides. An excited nucleus has excess energy. Both stable and unstable nuclides can be in an excited state. The kind of de-excitation could be emission of a suitable particle (α, β, neutron, or proton) but in most cases, the de-excitation takes place by the emission of one or more gamma photons. The name photon emphasizes that gamma radiation has particle-like properties. A typical example is written:

$$ {}^{60}_{27}\mathrm{Co} \rightarrow {}^{60}_{28}\mathrm{Ni}^{*} + \beta^{-} \qquad (\beta \text{ emission}) $$

Then $$ {}^{60}_{28}\mathrm{Ni}^{*} \rightarrow {}^{60}_{28}\mathrm{Ni} + \gamma \qquad (\gamma \text{ emission}) $$

Cobalt-60 emits a beta particle, leaving the daughter nickel-60 nucleus in an excited state (indicated by the asterisk). Almost immediately, the excited nickel-60 emits γ-rays until it is de-excited. The duration of the excited state is very short, usually much less than 10^{-9}s so we usually write the beta and gamma decays as though they are a single event.

$$ {}^{60}_{27}Co \rightarrow {}^{60}_{28}Ni + \beta + \gamma $$

The generalized gamma decay can be written:

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

As you can see, there is no change in Z or A. The gamma ray has no charge and no mass (it is pure energy) and cannot affect the atomic number or mass number of the nuclide.

Gamma rays are electromagnetic radiation like light rays, radio waves, and x-rays. Changes in charge configuration generate electromagnetic radiation. Different kinds of electromagnetic radiation are distinguished by their photon energy. A gamma photon has more energy than most x-ray photons, which in turn have more energy than ultra-violet photons and so on, down to the longest wave length radio waves. Figure 3.1 shows the electromagnetic spectrum. Low frequencies coincide with low photon energies, long waves, and wave like properties. High-energy gamma rays are more particle-like in their interactions. The speed of all electromagnetic radiation is $c = 3 \times 10^8$ m/s.

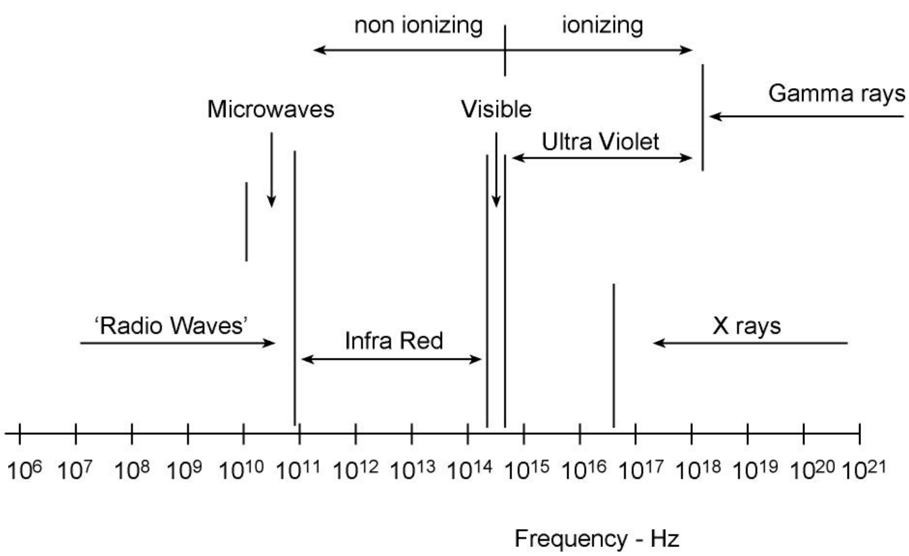


Figure 3.1 – Electro-Magnetic Spectrum

3.2 Interaction of Radiation with Matter

Alpha and beta particles are classed as ionizing particles. This is because they carry electric charge, which causes the atoms they approach to separate into ions. Each separation creates an ion-pair. Gamma rays are said to be indirectly ionizing, as described later in this section. Table 3.1 summarizes the properties of these different kinds of radiation.

3.2.1 Interactions of Alpha Particles

Alpha particles with their charge of +2 and their mass of 4 u create intense ionization. In dry air the alpha generates about 50 000 ion-pairs per centimeter of its path, giving up about 34 eV per pair produced. A 4 MeV alpha dissipates its energy in about 2.5 cm of travel. It slows, stops, and becomes a normal helium atom by adopting two electrons from its surroundings. Near the end of its path, it transfers some energy to neighbouring atoms by atomic excitation.

In liquids or solids, the number of ion-pairs generated per centimeter is much greater so the distance the alpha travels is much less. Typically, the range (straight-line distance) of an alpha crosses the same mass of material in different materials. The alpha particle range in solid materials is generally less than 0.1 mm, about the thickness of a sheet of paper.

3.2.2 Interactions of Beta Particles

Beta particles have a charge of -1, a mass of 0.000 548 u, and travel very fast (90-99% c). They cause less intense ionization than alpha particles, typically 100-300 ion-pairs per centimeter of path in dry air. Because of their small mass the beta particles are deflected easily and do not travel in a straight line. In dry air, their total length of path is typically 20 m. Their actual range might less than half of this. Beta particles are more penetrating than alphas: they will penetrate a sheet of paper. Generally, 1mm or so of a dense material is sufficient to stop them.

Rapid slowing or quick changes in direction cause beta particles to emit X-rays. This process usually accounts for only a few percent of the beta particle energy loss, with most of the energy lost by ionization. This unusual radiation has a suitably unusual name, bremsstrahlung radiation, from the German word for “braking”.

3.2.3 Gamma Ray Interactions with Atoms

Gamma rays do not interact with matter in the same way as alpha and beta particles. They have no charge and no mass and do not lose energy steadily in small, scattered amounts. Instead, they give it away in larger chunks in direct interactions. Three reactions between gamma rays and atoms follow.

The Photoelectric Effect.

This gamma ray interaction can take place for gamma rays of low energy. An incident gamma ray reacts with an electron in an atomic orbit. The gamma photon gives all of its energy to the orbiting electron and ceases to exist. The electron is ejected from the atom and behaves like a beta particle. The ejected electron is called a photoelectron.

In many materials, the photoelectric effect is not important for photon energies above 0.1 MeV.

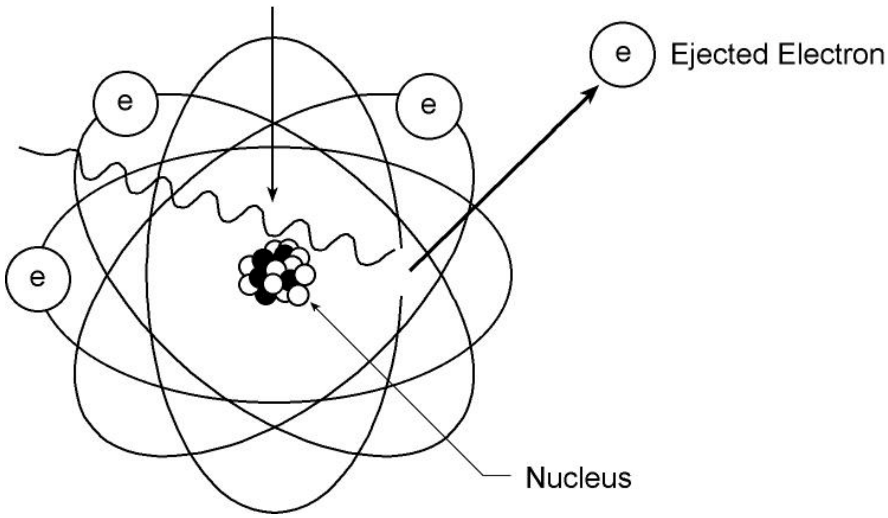


Figure 3.2
Photoelectric Effect

The Compton Effect
This gamma ray interaction is most important for gamma photons with energies between 0.1 and 10 MeV. The incident gamma ray is "scattered" by hitting an electron. The electron receives some of the gamma ray energy and is ejected from the atom. The Compton electron is usually much more energetic than a photoelectron. It causes ionization just as a beta particle does.

The scattered gamma ray is really a different gamma ray, as the original photon is absorbed and a new one emitted at a lower energy. After a series of such interactions, a low-energy gamma ray is produced, which is then absorbed by the photoelectric effect.

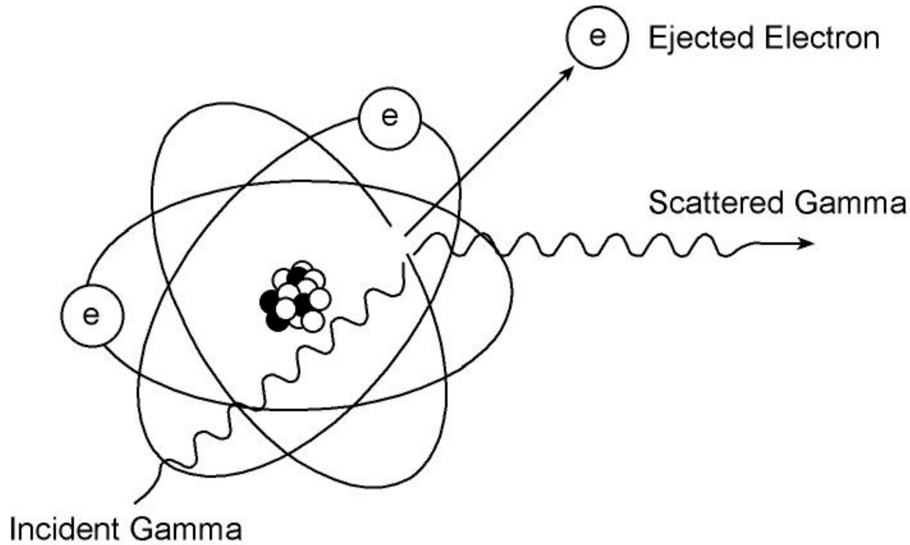


Figure 3.3
Compton Effect

Pair Production
This gamma ray interaction always occurs near an atomic nucleus that recoils. The gamma ray gives its energy to the creation of an electron-positron pair. (A positron is a positively charged electron!) The minimum gamma photon energy that can do this is 1.02 MeV (the energy equivalent of 2 electron masses). The process most often happens for high-energy gamma rays.

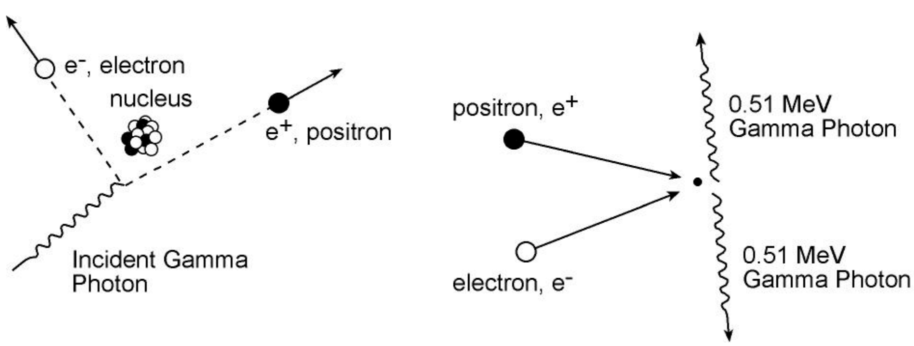


Figure 3.4
Pair Production

The positive and negative electrons created both cause ionization but their fates differ. Once it has slowed, the positron meets with another atomic-electron and they "mutually annihilate". Both cease to exist but two gamma rays of 0.511 MeV each are created.

The 0.511 MeV gamma rays go on and cause one of the other possible gamma ray interactions. The electron eventually settles down with some accommodating atom and become a normal atomic electron.

3.3 Direct and Indirect Ionization

Alphas and betas cause direct ionization. Each ion-pair created takes a small amount of energy and slows the alpha or beta a little bit. Eventually the particle stops. Alphas of a given energy all travel the same straight-line distance (range) in a given material. Similarly, betas of a given energy all have about the same range in a given material. By contrast, the gamma rays do not have a range. They may interact immediately or travel a very long distance between interactions. The gamma ray energy is transferred in large chunks and is deposited in the material by indirect (i.e., secondary) ionizations near each of the interaction sites. A small fraction of the gamma rays may penetrate quite thick materials and emerge on the other side with no loss of energy.

Table 3.1

Type of Radiation	Approximate Mass (AMU)	Charge	Energy Range (MeV)	Remarks
α	4	+2	4 to 8	Very short range, highly ionizing
β	0.0005	-1	0.5 to 3.5	Short Range
γ	0	0	Up to 10 (most below 3)	Long Range

3.4 Shielding

It is easy to shield against alphas or betas; we simply need material of thickness equal to or greater than their range. Shielding materials for betas should not stop them too quickly or the stopping process causes bremsstrahlung radiation (X-ray radiation). This then needs shielding.

Shielding against gamma rays and x-rays is not so easy. No matter how thick the shielding, some of the rays will still penetrate. For any particular photon energy, we can always find the amount of material that cuts the intensity in half. We call this the half value layer (HVL). Two half value layers reduce the intensity to ¼ of the original, and a third reduces it to $^1/_8$.

As an example, for typical gamma rays from fission products about 15 cm of water is a half value layer. In the irradiated fuel bays, water is maintained at least 4.5 m (30 HVLs) depth above the fuel. That means that the absorption in the water reduces γ ray intensity reaching the surface of the bay by a factor of 2^{30}. That is, it halves it 30 times. In round numbers $1/2^{30}$ is a reduction of 10^{-9} or one billionth of the original intensity. (You should check these numbers on your calculator.)

Materials containing heavy atoms shield gamma rays most effectively. Lead is often used where space for shielding is limited. Where lighter, less expensive materials (e.g., concrete, or water) are used, greater thickness is needed.

3.5 Summary of Key Ideas

- The three major particles emitted by spontaneous radioactive decay are alpha, beta and gamma.

- Alpha particles are doubly charged helium nuclei, which move slowly when they are emitted. They are emitted from large nuclei such as U-235, U-238 or Thorium.

- Beta particles are electrons. At the time the are emitted they are generally traveling at a speed greater than 90% of the speed of light. They are emitted from a nucleus with too many neutrons. A neutron in the nucleus changes to a proton and a beta particle is emitted.

- Gamma usually accompanies alpha or beta decay. They are photons of electromagnetic energy that travel at the speed of light.

- Alpha and beta particles are directly ionizing radiations. They leave a trail of ionized atoms in their wake.

- Gamma rays are indirectly ionizing radiation, and interact with atoms to generate ions. The three gamma interactions are Compton effect, photoelectric effect and pair production.

- Beta and alpha can be shielded by placing material between the source of the radiation between the source and a person.

- Gamma is the most difficult to shield. The effectiveness of a material in shielding gamma is referred to as a half value layer; the thickness of material required to reduce the gamma energy by one-half.

3.6 Assignment

1. Using ${}^{A}_{Z}X$ notation, write equations for alpha, beta and gamma decay.
2. Briefly, describe how alpha, beta, and gamma deposit their energy in matter.
3. List the masses and charges for α and β particles.
4. What is ionization?
5. Why is it said that γ rays do not cause direct ionization?
6. Describe methods used to shield against α or β particles.
7. What type of material makes good gamma ray shielding?
8. For a material of half value thickness of 6 cm, shielding 1 MeV gamma rays, calculate the thickness needed to reduce the intensity by 1 000.

4 Nuclear Stability And Instability

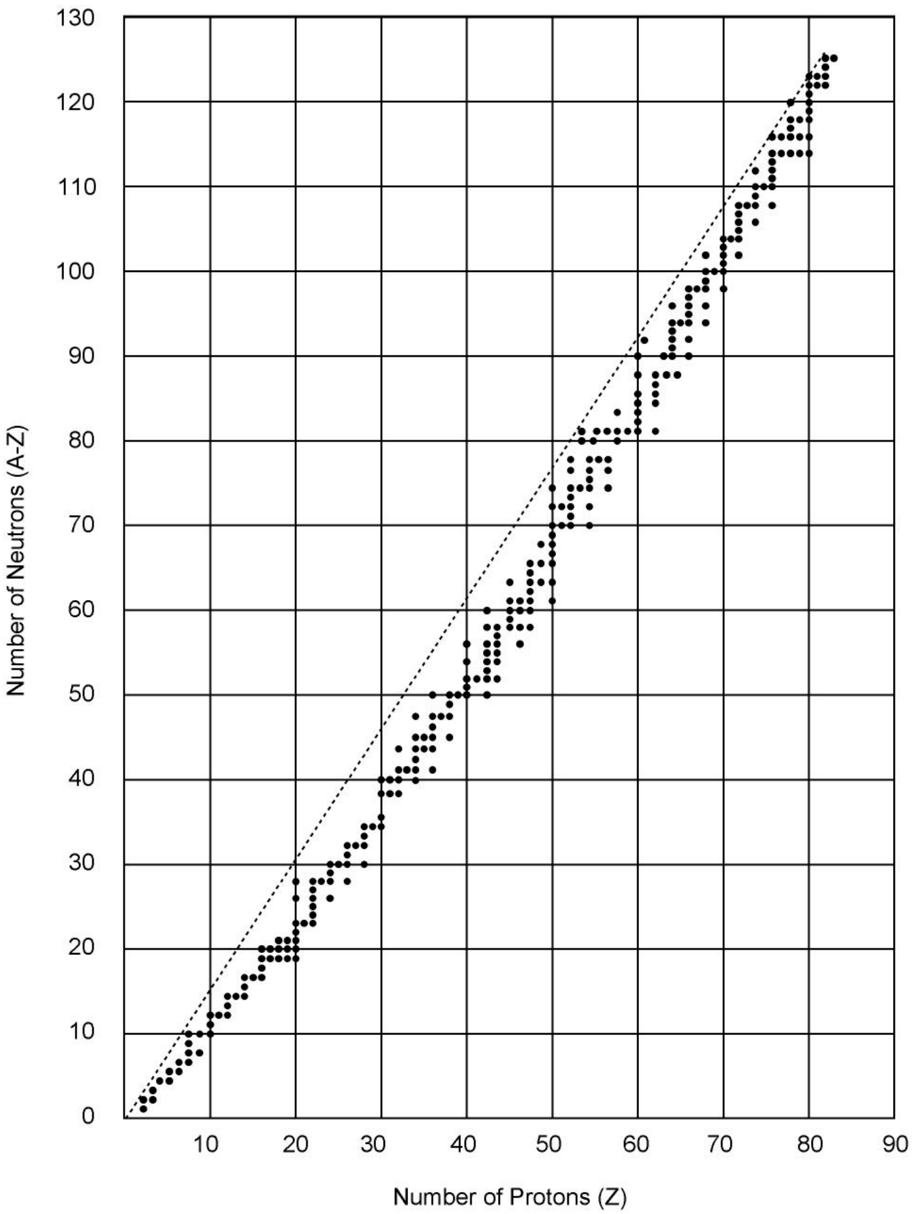


Figure 4.1

Plot of N vs. Z

Each black dot in Figure 4.1 represents a stable nuclide. Where more than one dot appears for a particular atomic number, those dots represent stable isotopes. For example if we look at atomic number one, we will see two dots. These represent hydrogen (H-1) and deuterium (H-2). For atomic number 8, there are three dots representing O-16, O-17 and O-18.

Looking at the general shape of the dot distribution, we can draw the simple conclusion that for light nuclides the number of neutrons is nearly the same as the number of protons. There are some exceptions, such as H-1, which has only one proton. For intermediate mass nuclides, the neutron to proton ratio is higher, about 1.3. (For Rh-103, n:p = 1.29.) For the heavy nuclides the ratio goes up to about 1.5, e.g., for gold, Au-197, n:p = $\frac{118}{79}$ = 1.5.

In general, we can say that if the neutron-proton ratio is outside of this range the nuclide is unstable. We find for instance that two protons cannot combine to form a nucleus without the aid of neutrons. If we look at the nucleus of He-3, we see that it contains two protons and one neutron. The neutron helps "dilute" the electric force, which tends to push the protons apart. In a sense, the neutron "glues" the protons together. Excess neutrons permit the short-range attractive force between adjacent nucleons to overcome the long-range repulsive electric forces among the protons. In He-4, 2 neutrons and 2 protons give a nucleus with the nucleons very tightly bound together.

Adding more neutrons does not always increase stability. There is no evidence that He-5 exists, and He-6 (which has been observed) has less than a 1 s half-life. In general, extra neutrons aid stability but too few or too many neutrons cause instability. Unfortunately we cannot specify it much closer than that.

For example, look at Cu-64, (Z = 29). You will not find a dot for it on the graph because it is unstable. Yet, if we do plot it we find it sits right between Cu-63 and Cu-65, both of which are stable. Having a n:p ratio in the right range is important for stability but some unstable nuclides also have n:p ratios in the right range. We are safe in saying that everything away from the stability line is unstable; the neutron proton ratio must be "right" for a stable nuclide. Most, but not all nuclides with n:p ratios along the stability line are stable.

We have already seen that all of the very heavy nuclides (Z > 83), are unstable. If they have a suitable n:p ratio (and thus do not beta decay) they will α decay because of the large repulsive electric force. If extra neutrons are present to dilute this electric force, α decay may be prevented but beta decay occurs because of the high n:p ratio.

4.1 Neutron Rich Nuclides

For lighter nuclides a relatively easy way to get a "wrong" n:p ratio is to add a neutron to a stable nuclide (by neutron absorption). For example, adding a neutron to the nucleus of O-18 gives O-19, which is unstable. Conversion of a stable nucleus into and unstable one is called activation. Neutron absorption does not always cause activation. For example, adding a neutron to H-1 produces the stable H-2 nuclide.

Another way to get nuclides that are neutron rich is to split (fission) a heavy nuclide into two intermediate mass nuclides. These fission products are almost certain to be unstable. (Occasionally an uneven split leaves one stable fission product.) Consider splitting a nucleus of U-235. Its n:p ratio is about 1.55. The two fission products are also likely to have n:p ratios near 1.55, too high for nuclides near the middle of the mass range. The dashed line in Figure 4.1 shows the position of all nuclides with a neutron to proton ratio of about 1.5. Only the heaviest stable nuclides lie squarely on this line.

Suppose for example that two fission products are ${}^{95}_{36}Kr$ and ${}^{139}_{56}Ba$.

If we plot these on the graph, we find that they are quite far from the stability zone. They will decay or disintegrate by emitting a particle. Will the particles be alphas or betas? Well we can make a guess.

Perhaps ${}^{95}_{36}Kr$ is an alpha emitter.

$${}^{95}_{36}Kr \rightarrow {}^{91}_{34}X + {}^{4}_{2}\alpha \text{ ?}$$

Where would this new nuclide plot? Is it more stable, less stable or about the same? You should suspect that its stability is about the same and alpha decay is unlikely. So let's second guess (it's usually more reliable) and try a beta decay.

$${}^{95}_{36}Kr \rightarrow {}^{95}_{37}X + {}^{0}_{-1}\beta \text{ ?}$$

How does this one look on the graph? Yes, it looks better. We assume therefore that beta emissions are likely. Experiments confirm this. Because most fission products have too many neutrons, they decay by beta decay. A few unstable fission products release a neutron immediately following beta decay. (These are called delayed neutrons.) No fission products emit alpha particles.

4.2 Interchangeable Nucleons

For beta emissions, notice that the number of nucleons remains constant although the nuclides have changed. Inside the nucleus, a neutron has changed into a proton.

$${}^{95}_{36}Kr \rightarrow {}^{95}_{37}Rb + \beta^{-}$$

$$\left.\begin{matrix} 36\ protons \\ +59\ neutrons \end{matrix}\right\} \rightarrow \left\{\begin{matrix} 37\ protons \\ +58\ neutrons \end{matrix}\right. + \beta^{-}$$

The reverse process can also happen in some nuclides. A proton can turn into a neutron, either by emitting a positive electron or by capturing an orbiting electron, as described next.

4.3 Neutron Deficient Nuclides

There are many neutron deficient nuclides, (nuclides that plot beneath the curve in Figure 4.1), but we are not likely to meet with them in our reactor technology. What type of emissions should we expect from them? Try some guesses as we did in the last section. Are they alpha emitters? Are they beta emitters? Satisfy yourself that neither of these is reasonable.

These neutron deficient nuclides decay by either positron emission or electron capture or both. Positron? What is that? The positron is a positively charged electron. It is "exactly" like an electron, except for its positive charge. We saw in the last module that an electron and positron can annihilate one another, emitting a pair of gamma rays.

Putting positron emission into the usual symbolic form yeilds:

$$\square{}^{A}_{Z}X \rightarrow {}^{A}_{Z-1}X + {}^{0}_{+1}\beta$$

If you plot ${}^{A}_{Z-1}X$ on Figure 4.1, you will see that it is closer to the stability region than ${}^{A}_{Z}X$.

For electron capture (also called K-capture because the electron comes from the K-shell orbit) we write:

$${}^{A}_{Z}X + {}^{0}_{-1}e \rightarrow {}^{A}_{Z-1}X$$

The nuclide produced is the same one produced by positron emission. In each case, a proton inside the nucleus has turned into a neutron.

4.4 Heavy Nuclides

For nuclides above atomic number 83 we find there are several possible decays. Most of the naturally occurring heavy nuclides emit alpha particles. Some emit beta particles, and a few may also undergo spontaneous fission. Some synthesized heavy nuclides are also positron emitters (rare) or undergo electron capture (more likely).

4.5 Summary of Key Ideas.

- The n:p ration or a nuclei is a predictor of its stability against beta decay

- For light nuclei the n:p ratio is in the neighborhood of 1:1 for O-16, C-12, B-5. The ratio increases to about 1.5 for the heaviest nuclei: U-235, Au-197

- Fission products usually have a n:p ration that is too high for the mass of the fission product

- If nuclei are bombarded with neutrons, they absorb some of the neutrons and will tend to become neutron rich.

- Sometime a delayed neutron will be released following a beta decay.

- Neutron deficient nuclei decay by positron (anti-electron) emission or electron capture.

- Heavy nuclei can decay by numerous means: alpha, beta, spontaneous fission.

4.6 Assignment

1. Why do all nuclides except H-1 have neutrons in their structure?

2. Guess which type of particle the following nuclides emit. Use the graph of Figure 4.1 and then check your answers using a chart of the nuclides or Table of Isotopes (e.g. in the Science Data Book).

3. Sr-90, Br-87, Xe-135, I-135, I-131, Sm-149, Co-60, B-10, N-16, U-238, Pu-239, Cu-64, Mn-56, H-3, Cs-137.

4. What happens if the neutron to proton ratio of a nuclide is too high or too low?

5. For the natural heavy nuclide U-238, write down the stages in its decay to become a stable nuclide. Use a chart of the nuclides. You can repeat the exercise with U-235 as the starting nuclide

5 Activity and Half-Life

The activity of a radioactive material is the number of nuclei that decay per unit time, often expressed as "disintegrations per second". In the SI system, this unit is called the becquerel (Bq). The becquerel is defined as one radioactive disintegration (decay) per second. Another unit, widely used, is the curie (Ci). One curie is now defined as 3.7×10^{10} Bq. Originally, it was the activity of 1 g of radium-226. The discovery of radium won Marie Curie the first of two Nobel prizes.

5.1 The Law of Radioactive Decay

A pure radioactive substance decays at a fixed fractional rate. That is, in each second a constant fraction of the total amount present decays. Consequently, the actual number of atoms decaying per unit time is proportional to the amount of the substance.

Consider a particular sample of a radionuclide. The continual decay decreases the quantity of the sample and so the activity decreases. This process continues until the radioactive material is gone. Figure 5.1 plots the quantity of the sample Q against time T.

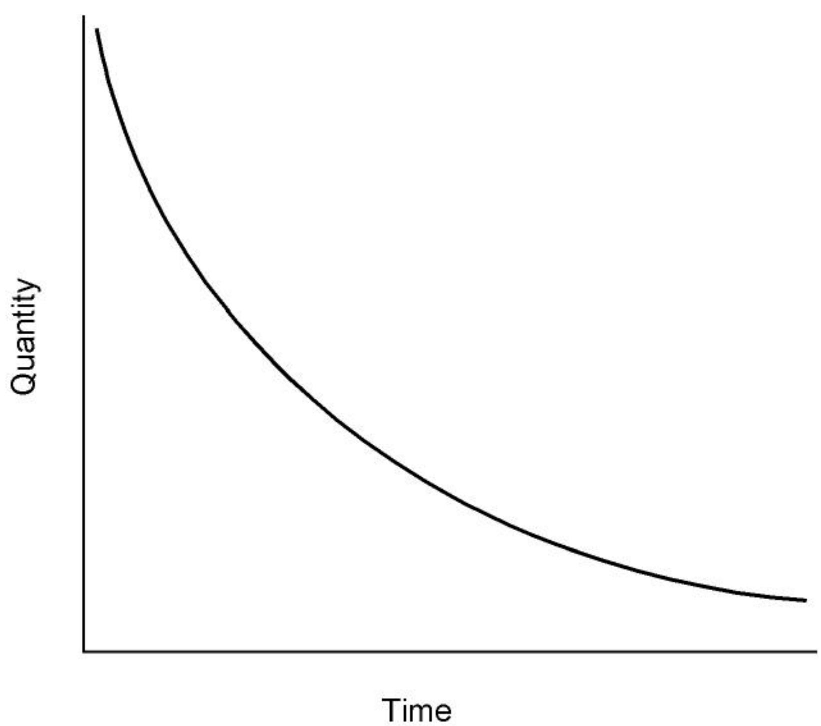


Figure 5.1
Quantity v Time for a Radionuclide

We have already said that activity is proportional to the quantity of the radioactive substance so we can also plot the activity against time (Figure 5.2).

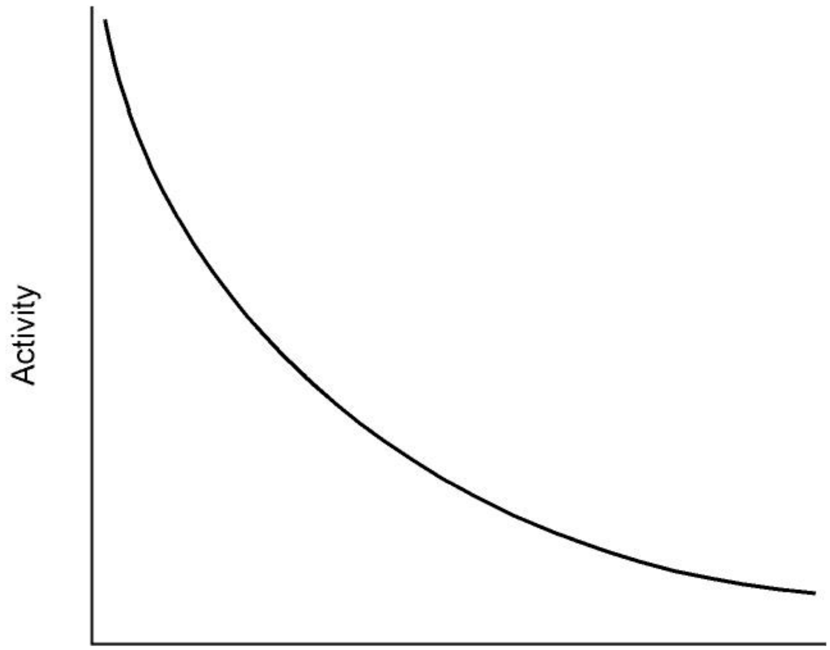


Figure 5.2
Activity v Time for a Radionuclide

These two graphs (Figures 5.1 and 5.2) are mathematically identical and only differ in having different vertical scales. In practice, we prefer to use the second form because activity is the quantity usually measured and activity is usually what interests us most. By contrast we seldom can measure, and often don't care about the actual quantity of the radioactive substance. For example, the activity of the moderator quoted in curies per kilogram gives a clear indication of the radiation hazard, but tells us nothing directly about the amount of tritium in the moderator.

5.2 Half-life

If we plot graphs of activity vs. time for different radioactive materials, we find they have different rates of decay (Figure 5.3). To distinguish between the different rates, we use half-life (Figure 5.4). The half-life ($T_{1/2}$) is the time interval for the activity of a specimen to fall to half of its original value.

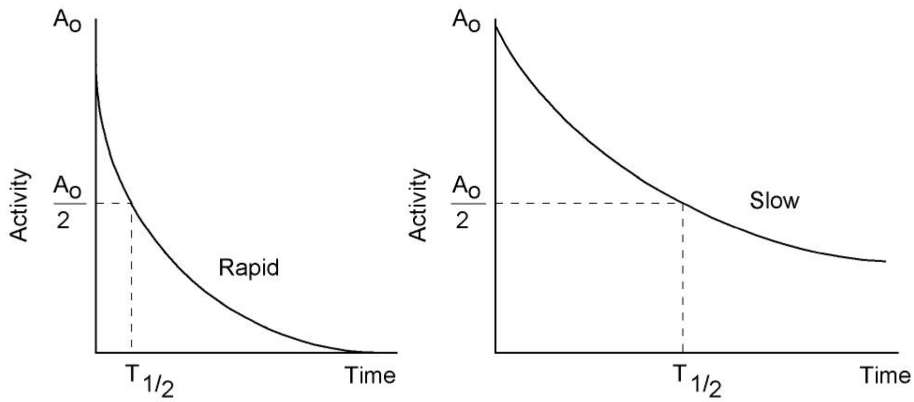


Figure 5.3 A and B -
Activity Plots of Two Different Radioactive Materials

The time interval between activity A_0 and activity ½ A_0 is, by definition, one half-life. For an exponential decay curve (which these are), it does not matter where we start with A_0. For any starting point on the curve, the time for the activity to decrease to ½ the starting value is the same (Figure 5.4).

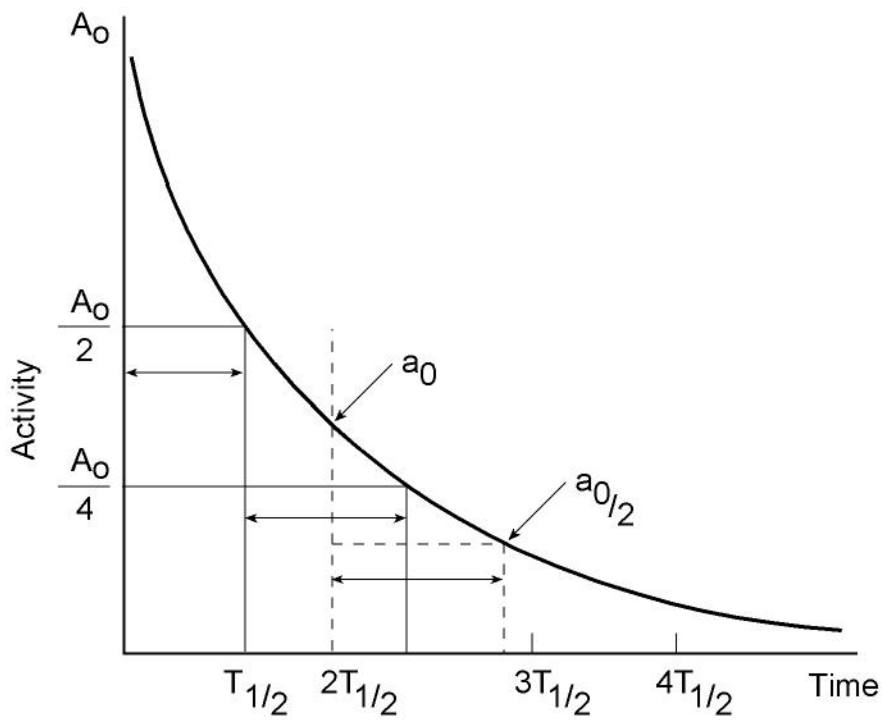


Figure 5.4 –
Activity and Half-life

The time from A_0 to ½ A_0 is the same as from ½ A_0 to ¼ A_0. It also takes the same time from a_0 to ½ a_0. This leads to the formula $At=A_0$ (1/2) n where n is the number of half-lives, i.e., n = t/$T_{½}$ where t is the elapsed time. In this relationship, we usually take n to be an integer but it need not be.

Another form of the equation $At=A_0\ (1/2)^n$ is $A_0/A_t = 2$ n

Before reviewing the following examples, try the end-of-chapter-exercises. Many people find these calculations easier to do than to read about.

Examples

1. Suppose a radioactive substance has an activity of 6144 Bq. How many half-lives will it take for the activity to fall to 6 Bq?

   $$\frac{A_0}{A_t} = 2^n$$

   Substituting the values, $\frac{6144}{6} = 2^n$, or, $1024 = 2^n$

   $\therefore \quad n = 10$

   Answer:

   It takes 10 half-lives for the activity to fall from 6144 to 6 Bq.

2. What will the activity be 6 half-lives later for a radioactive substance which has an activity now of 192 Bq?

   $$A^t = A_0(1/2)^n$$

   $A_t = 192(½)^6 \qquad$ *i.e.* $(192\ x\ ½\ x\ ½ \times ½ \times ½ \times ½ \times ½)$

   or $= \frac{192}{2^6} = \frac{192}{64}$, so $A_t = 3$ Bq

   Answer:

   After 6 half-lives an original activity of 192 Bq falls to 3 Bq.

3. If the half-life in the first example is 25 minutes, what is the time t?

$t = n\,T_{½}$

$=10$ half-lives x 25 minutes per half-life

$= 250$ m or 4h 10 m

5.3 Range of Half-lives

Half-lives vary from very short (fractions of a second) to very long (billions of years). From an operational point of view, these values are important in comparison to reactor life, operational times, outage times, fuel life, etc.

For example, fresh CANDU fuel contains natural uranium. The half-life of U-238 is 4.5 billion years and for U-235 is 700 million years. Although both of these are decaying by α emission, we never notice any change in their activity over the lifetime of the reactor. Fresh fuel will be the same no matter how long we keep it. By contrast, the half-life of N-16 (produced by activation in the reactor core) is only 7s and the activity changes faster than most of us can calculate.

Irradiated fuel contains isotopes of uranium and neptunium that decay to make fissile plutonium. The beta decays of U-239 and Np-239 have half-lives of 23m and 2.3d respectively. They convert into fissile Pu-239 on a quite short time scale. The Pu-239 decays by α decay with a half-life of 25 thousand years so its quantity does not change due to α decay in the 1 to 2 years the fuel is in the reactor.

Activity (rate of decay) depends on half-life, so it too has a wide range of values. The becquerel is a very small unit of activity and is usually seen with a metric prefix. For example: kilo becquerel (k Bq = 10^3 Bq), mega becquerel (M Bq = 10^6 Bq), giga becquerel (G Bq = 10^9 Bq) or even, sometimes, tera becquerel (T Bq = 10^{12} Bq). The curie, on the other hand, is a large unit. It is customary to see it as milli curie (m Ci = 10^{-3} Ci), micro curie (μ Ci = 10^{-6} Ci), pico curie (p Ci = 10^{-12} Ci), or, occasionally, nano curie (n Ci = 10^{-9} Ci). As an example, contamination with about 8×10^{-9} g of radioactive I-131 (about 3.7×10^{13} atoms of I-131) results in an activity of about 1 m Ci = 37 M Bq.

5.4 Summary of Key Ideas

- Activity is the rate at which a radioactive material is decaying.

- Activity is measured in d.p.s., Bq or Ci.
- Nuclei decay exponentially and each has a characteristic half-life.
- Half-lives vary over a tremendous range, from tiny fractions of a second to billions of years.

5.5 Assignment

1. Give the relationship between disintegrations per second and the becquerel.

2. Name a widely used base unit for activity other than the becquerel.

3. Fe-59 has a half-life of 45 days. If a sample has an activity of 1000 decays per second, what is its activity after one year?

4. The activity of a radioactive specimen is 2×10^7 Bq. After 20 days, the activity is 2×10^4 Bq. What is the half-life of this specimen? (Calculate to the nearest whole number of half-lives.)

6 Neutrons and Neutron Interactions

A nuclear reactor will not operate without neutrons. Neutrons induce the fission reaction, which produces the heat in CANDU reactors, and fission creates more neutrons. The neutrons produced also engage in other reactions. It is important to know about these neutron interactions.

6.1 Neutron Production

Most of the neutrons in a CANDU reactor come directly from fission. About ½% of the neutrons in a reactor at power are emitted as part of fission product decay. These two important types of neutrons, known as prompt and delayed neutrons respectively, are discussed in the next section. The only other important neutron source in a CANDU is the photoneutron reaction.

6.1.1 The Photoneutron Reaction

An energetic gamma ray may interact with a deuterium nucleus, removing a neutron. The deuterium nucleus becomes a normal hydrogen nucleus and the neutron is free to move around.

$$\gamma + {}^{2}_{1}H \rightarrow {}^{1}_{1}H + {}^{1}_{0}n$$

The minimum gamma ray energy for this reaction, 2.22 MeV, is known as the threshold energy. In this example, a gamma ray of energy of 2.22 MeV is absorbed and the total mass increases by 0.00239 u. Looking ahead for a moment, module 6 describes how energy can be converted to mass, and mass to energy. The photoneutron reaction is an example of conversion of energy to mass, as was electron-positron pair production, introduced in module 3. The threshold energy can be equated with the mass difference measured between the initial deuterium atom and the reaction products (the hydrogen atom and neutron) to give the useful mass to energy conversion factor: 1 u = 931.5 MeV.

6.2 Neutron Interactions

This section introduces five reactions that can occur when a neutron interacts with a nucleus. In the first two, known as scattering reactions, a neutron emerges from the reaction. In the remaining reactions, known as absorption reactions, the neutron is absorbed into the nucleus and something different emerges.

6.3 Elastic Scattering—(n, n)

Elastic scattering resembles a billiard ball collision. A neutron collides with a nucleus, transfers some energy to it, and bounces off in a different direction. (Sometimes it absorbs the neutron and then re-emits it, conserving kinetic energy.) The fraction of its initial energy lost depends on whether it hits the target nucleus dead-on or at an angle—exactly like the cue ball striking a ball on the billiard table. The target nucleus gains the energy lost by the neutron, and then moves at an increased speed.

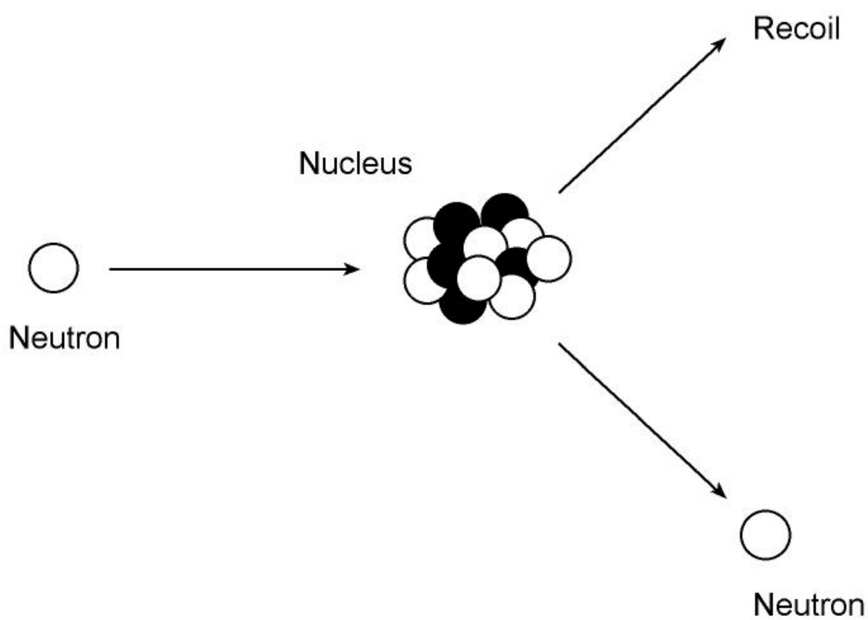


Figure 6.1
Elastic Scattering

Light nuclei are the most effective for slowing neutrons. A neutron colliding with a heavy nucleus rebounds with little loss of speed and transfers very little energy—rather like firing the cue ball at a cannon ball. On the other hand, neutrons will not be scattered by the light electron clouds surrounding the nucleus, but will travel straight on—much like baseballs through a fog.

6.4 Inelastic Scattering—(n, nγ)

A neutron may strike a nucleus and be temporarily absorbed, forming a compound nucleus. This will be in an excited state. It may de-excite by emitting another neutron of lower energy, together with a gamma photon, which takes the remaining energy. This process is called inelastic scattering. It generally happens only when high-energy neutrons interact with heavy nuclei and has little practical importance for reactor operation.

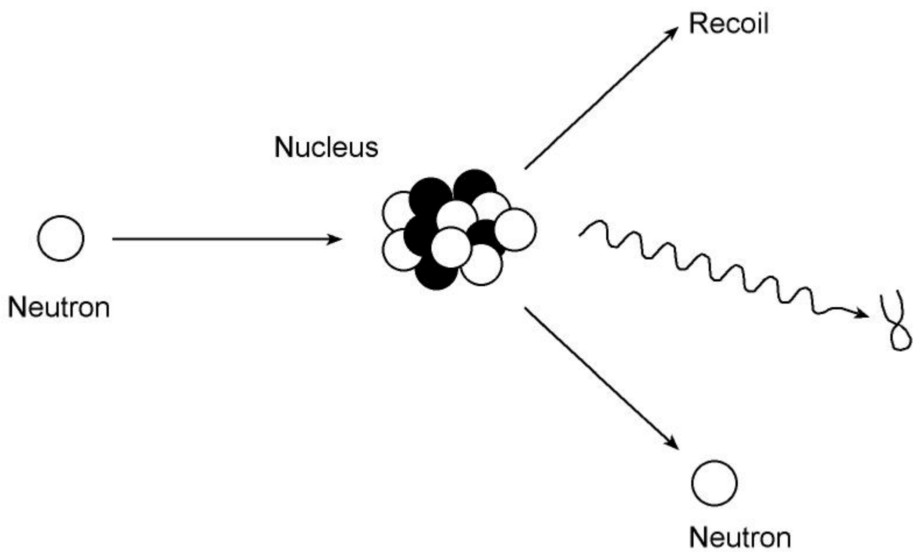


Figure 6.2
Inelastic Scattering

6.5 Transmutation—(n, p), (n, α)

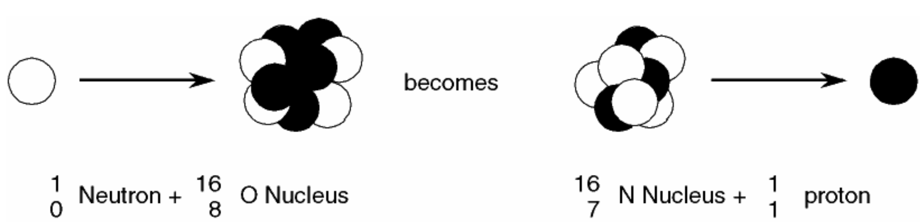
A nucleus may absorb a neutron forming a compound nucleus, which then de-energizes by emitting a charged particle, either a proton or an alpha particle. This produces a nucleus of a different element. Such a reaction is called a transmutation.

Transmutation is the transformation of one element into another by a nuclear reaction.

Examples:

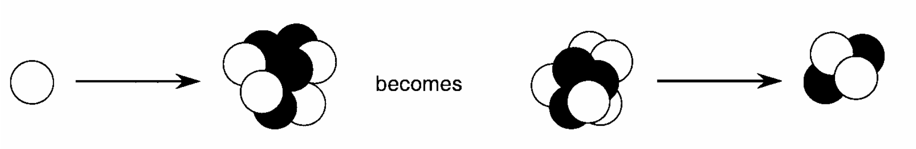
6.5.1 Neutron-Proton Reaction (n, p)

Oxygen-16 captures a neutron and emits a proton to form nitrogen-16:



The product, nitrogen-16, is radioactive with a half-life of 7.1 seconds so this example is an activation reaction. N-16 is a beta emitter, but more important, it also emits very penetrating, high-energy gamma rays.

6.5.2 Neutron-Alpha Reaction (n, α)
Neutrons captured by boron-10 cause the following reaction:

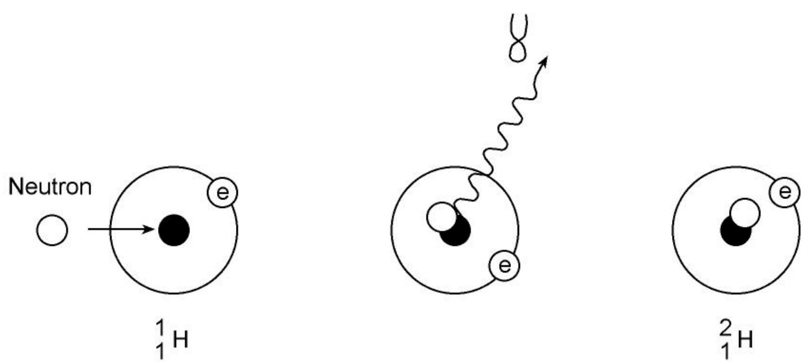


$^{1}_{0}$Neutron + $^{10}_{5}$B nucleus $^{7}_{3}$Li nucleus + α

6.6 Radiative Capture—(n, γ)

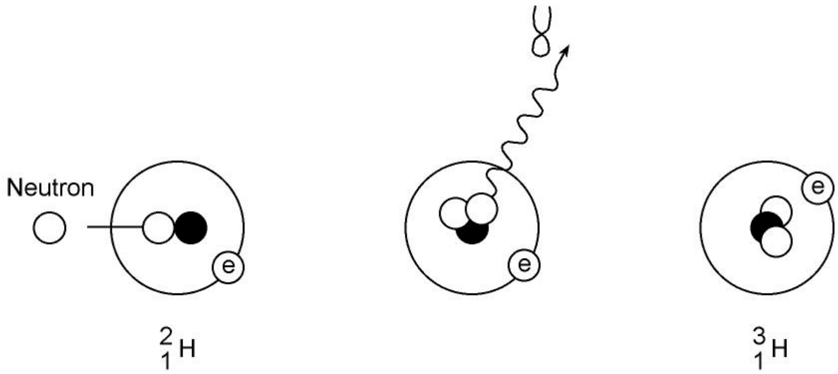
This is the most common nuclear reaction. The compound nucleus formed emits only a gamma photon. In other words, the product nucleus is an isotope of the same element as the original nucleus. Its mass number increases by one.

Examples
The simplest radiative capture occurs when hydrogen absorbs a neutron to produce deuterium (heavy Hydrogen);



The deuterium formed is a stable nuclide. However, many radiative capture products are radioactive and are beta-gamma emitters.

Deuterium itself undergoes a radiative capture reaction to form tritium;



The tritium isotope is unstable and is a major radiation hazard in CANDU reactors.

Stable cobalt-59 undergoes radiative capture to form highly radioactive Co-60:

$$^{59}_{27}Co \; + \; ^{1}_{0}n \;\rightarrow\; ^{60}_{27}Co \; + \; \gamma$$

Cobalt-60 has a long half-life (5¼ years) and emits very penetrating gamma radiation when it decays, making it a serious hazard among activated corrosion products. Normal steel usually contains a small amount of cobalt, but the concentration in reactor grade materials is limited to reduce the radiation hazard.

Cobalt-60 is an isotope commonly used in radiation treatment of cancer.

6.7 Fission

This most important reaction is the subject of the next lesson.

The five neutron reactions introduced in this section are of two types: scattering reactions (the first two), and absorption reactions (the others). The following examples illustrate a convenient shorthand notation for these reactions.

$^{1}_{0}n \; + \; ^{2}_{1}H \;\rightarrow\; ^{2}_{1}H \; + \; ^{1}_{0}n$ can be written $\quad ^{2}_{1}H\,(n,n)^{2}_{1}H$

$^{1}_{0}n + {}^{238}_{92}U \;\rightarrow\; ^{238}_{92}U + {}^{1}_{0}n + \gamma$ can be written $\quad ^{238}_{92}U\,(n,n\gamma)^{238}_{92}U$

$^{1}_{0}n \; + \; ^{16}_{8}O \;\rightarrow\; ^{16}_{7}N \; + ^{1}_{1}p$ can be written $\quad ^{16}_{8}O\;(n,p)\;^{16}_{7}N$

$^{1}_{0}n \; + \; ^{10}_{5}B \;\rightarrow\; ^{7}_{3}Li \; + \; ^{4}_{2}\alpha$ can be written $\quad ^{10}_{5}B\;(n,\alpha)\;^{7}_{3}Li$

${}^{1}_{0}n + {}^{59}_{27}Co \rightarrow {}^{60}_{27}Co + \gamma$	can be written	${}^{59}_{27}Co\,(n,\gamma)\,{}^{60}_{27}Co$
${}^{1}_{0}n + {}^{2}_{1}H \rightarrow {}^{3}_{1}H + \gamma$	can be written	$D(n,\gamma)T$
${}^{1}_{0}n + {}^{1}_{1}H \rightarrow {}^{2}_{1}H + \gamma$	can be written	${}^{1}_{1}H(n,\gamma)\,{}^{2}_{1}H$
$\gamma + {}^{2}_{1}H \rightarrow {}^{1}_{1}H + {}^{1}_{0}n$	can be written	${}^{2}_{1}H\;(\gamma,\,n)\;{}^{1}_{1}H$

6.8 Summary of Key Ideas

- Important sources of neutrons in the reactor are fission, delayed neutrons and photoneutrons.
- Photoneutrons are released from nuclei when a high energy gamma interacts with a nucleus and releases a neutron.
- The most important photoneutron reaction in a CANDU reactor is photoneutron released from a deuterium nucleus by a gamma ray with an energy greater than 2.2 MeV.
- Neutrons may be scattered by elastic and inelastic scattering.
- Neutrons also cause transmutation reactions. A transmutation occurs when a neutron is absorbed by a nucleus and some other particle is emitted.
- (n, γ), (n, p), (n, α) are among the common transmutation reactions.

6.9 Assignment

1. Write the equation for the photo-neutron reaction with H-2.

2. Describe elastic and inelastic scattering of neutrons.

3. Identify the following reactions:

   a) $^{1}_{0}n + ^{40}_{18}Ar \rightarrow ^{41}_{18}Ar + \gamma$

   b) $^{1}_{0}n + ^{16}_{8}O \rightarrow ^{16}_{7}N + ^{1}_{1}p$

4. List the examples of neutron reactions in this chapter that are also activations.

5. Complete the following equations and name the reaction. (The atomic numbers are 6 for carbon, 7 for nitrogen, and 8 for oxygen.)

   a) C-13 (n, γ) ?

   b) O-17 (n, p) ?

   c) O-17 (n, α) ?

   d) N-14 (n, p) ?

6. Complete the following table:

Name of Reaction	**Reaction type (n,x)**	**Target Nucleus**	**Product Nucleus**
?	(n, γ)	$^{2}_{1}H$	**?**
Transmutation	(n, p)	**?**	$^{16}_{7}N$
Radiative Capture	**?**	$^{18}_{8}O$	**?**
?	(n, **?**)	$^{12}_{6}C$	$^{13}_{6}C$

7 Fission

In 1939 Hahn and Strassman, while bombarding U-235 nuclei with neutrons, discovered that sometimes U-235 splits into two nuclei of medium mass. There are two important results:

1. Energy is produced.

2. More neutrons are released.

The process was named fission, which can be defined as the splitting of a heavy nucleus into two lighter nuclei. Immediately after hearing about this discovery, Lise Meitner and Otto Frisch recognized that neutrons produced by fission might be made to cause fission, one fission after another, releasing enormous energy.

7.1 Energy Released by Fission

Before the twentieth century, mass and energy seemed to be separate unrelated quantities, each governed by its own conservation law.

a) Conservation of Mass states that mass cannot be created or destroyed.

b) Conservation of Energy states that energy cannot be created or destroyed.

   Burning carbon illustrates both laws. A carbon atom reacts chemically with a molecule of oxygen according to the equation:

$$C + O_2 \rightarrow CO_2$$

   Each reaction releases almost 5 eV of energy. If a quantity of carbon is burned and all the carbon dioxide (CO_2) gas is collected and weighed its weight equals the combined weight of the carbon (C) and the oxygen (O_2). Mass is conserved.

   The 5 eV of heat energy released comes from chemical potential energy, converted to heat in burning. One form of energy is changed to another, but no energy is created or disappears. The amount of energy does not change, so energy is conserved.

The two conservation laws can still be applied to most processes, but fission and other nuclear reactions defy these separate laws.

Destruction of mass creates energy, and destruction of energy creates mass. Einstein predicted conversion between mass and energy in 1905. He uncovered contradictions in classical physics and showed that mass and energy are related by the formula:

$$E = mc^2$$

where: E = energy (joules)
m = change in mass (kilograms)
c = speed of light (approximately 3 x 10^8 m/s)

This relationship is likely true in general, but chemical reactions do not show it. The following example shows why. Consider the complete combustion of one kilogram of coal:

Energy from complete combustion of 1 kg of coal
= 3.36 x 10^7 joules, which, in other units, is almost 10 kWh.
(3.6 × 106 J = 1 kWh)

From, $E = mc^2$

$$\text{m (converted)} = \frac{E}{c^2} = \frac{3.36 \times 10^7\, J}{\left(3 \times 10^8\, m/s\right)^2} = 3.7 \text{ x } 10\text{-}10 \text{ kg}$$

This is a very small fraction of the initial 1 kg and impossibly small to measure.

A similar calculation shows that burning one atom of carbon converts only a few billionths of a mass unit to energy.

Energy from converting one C atom to CO_2 = 5 eV

Using the mass to energy conversion factor 931.5 MeV = 1 u yields the mass converted to energy as

5×10^{-6} MeV/ 931.5 (MeV/u) = 0.000 000 005 4 u

This is about 1/10 000 of an electron mass and technology is not yet able to measure such a small mass loss.

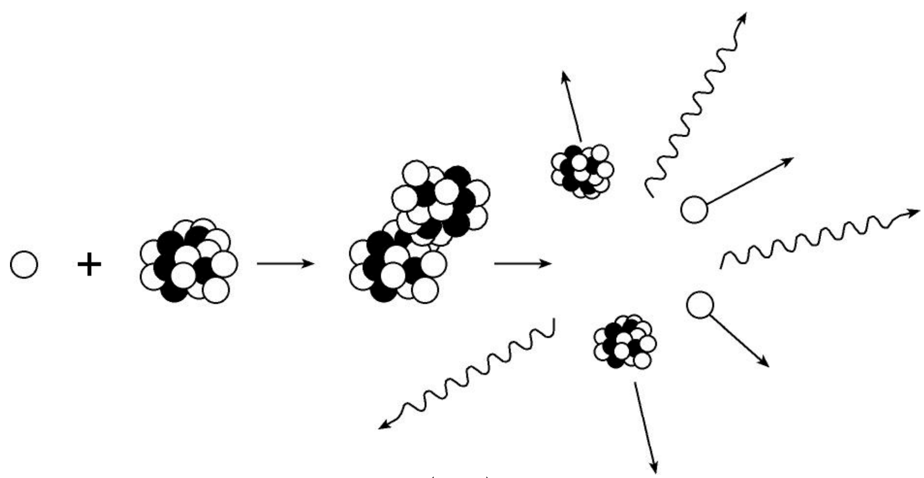
For comparison, consider the complete fissioning of 1 gram of U-235.

Energy from fissioning of 1 gram of U-235 = 8.2 x 10^{10} joules. This is very nearly 1 MWd (megawatt-day) of heat energy. (24 000 kWh = 1 MWd)

$$m\ (\text{Converted}) = \frac{E}{c^2} = \frac{8.2\times10^{10}\,J}{\left(3\times10^{8}\,{}^{m}\!/\!{}_{s}\right)^2} = 9 \times 10^{-4}\ \text{g}$$

The actual mass converted to energy to generate 1 MWd is almost 1 mg, nearly 0.1% of the original mass, and can be measured.
The example shows that the complete fissioning of all the atoms in one gram of U-235 would produce 1 MWd of thermal energy. A single fission does not create much energy but a gram of natural uranium contains 1.8 x 10^{19} U-235 atoms, and a CANDU reactor can fission about ¾ of these. (Compare fission of 1 g of U-235 with burning 1 kg of coal).

We now look at the fission of a single U-235 atom in more detail. A neutron enters the U-235 nucleus to form a highly excited compound nucleus, U-236, which in turn fissions. The following figure shows a typical fission:



$${}^{1}_{0}n + {}^{235}_{92}U \rightarrow \left({}^{236}_{92}U\right)^{*} \rightarrow {}^{95}_{37}Rb + {}^{139}_{55}Cs + 2\,{}^{1}_{0}n + \gamma$$

The mass of the reactants	236.05 u
The mass of the products	235.865 u
Mass converted (subtract)	0.19 u

The transformation of 0.19 u of mass releases nearly 180 MeV of energy immediately. Adding the energy released later by the radioactive decay of the fission products brings the total to about 200 MeV.

7.2 Fission Fragments

The general equation for the fissioning of uranium is:

$$ {}^{1}_{0}n + {}^{235}_{92}U \rightarrow \left({}^{236}_{92}U\right)^* \rightarrow 2\ F.F. \rightarrow 2\ F.P. + 2.5\ {}^{1}_{0}n + \gamma $$

The compound nucleus, after capturing the thermal neutron, has a huge excess of energy. The asterisk indicates an excited state of U-236. The equation shows the immediate break-up (fission) of the compound nucleus into two fission fragments (F.F.).

The initial fission fragments leave the fission site with velocities around 9 x 10^6 m/s (that is, 32 million kilometres per hour). Most of the energy from fission (≈ 84%) is in the form of kinetic energy of the fission fragments.

The fragments are highly positively charged and stop quickly, depositing their energy in a very short distance (5 x 10^{-4} cm). Ionization transfers most of the energy to the surrounding fuel. Excitation of nearby atoms and some direct collisions of the fragments with atomic nuclei also transfer some energy. The fuel gets hot as the motions of its atoms and molecules increase by these interactions.

The initial fragments are highly excited and unstable. They decay almost instantly into longer lived, unstable nuclei known as fission products, (F.P), illustrated in the equation above. The fission neutrons and prompt gamma rays shown in the above fission reaction are emitted in this process.

Fission products have a neutron/proton ratio similar to the U-235 nucleus (n:p = 143/92 = 1.55), yet are much lighter nuclei. The neutron/proton ratio required for stability is smaller, near 1.3 for the light fragment with A ≈ 95 and near 1.4 for the heavy fragment, with A ≈ 140. Fission products are neutron rich and consequently decay by beta-gamma emission.

The mass of the fission products falls within the narrow range shown in figure 7.1. A typical fission produces a heavy fragment and a lighter one that carries the largest proportion of the kinetic energy.

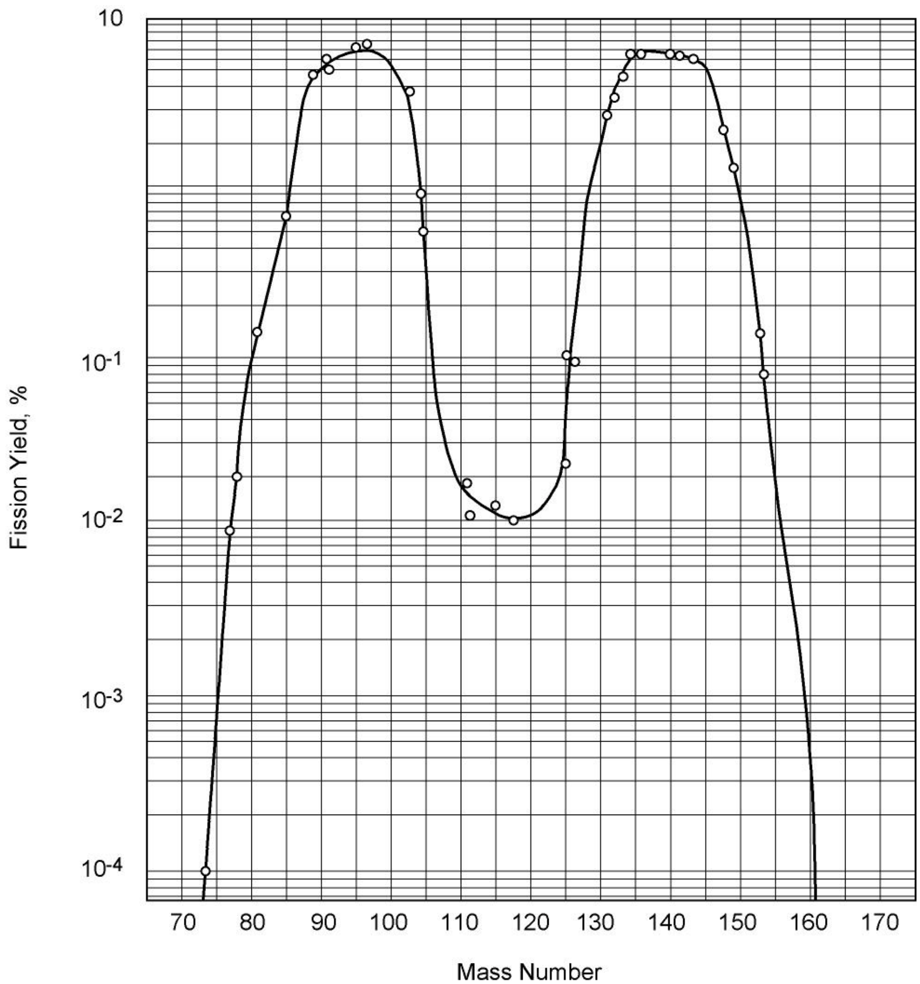


Figure 7.1
Yield of Fission Products

7.3 Chain Reaction

A typical fission produces between 0 and 5 neutrons. The average is approximately 2.5 neutrons per U-235 fission. Under the right circumstances, these neutrons will produce further fissions. Figure 7.2 shows that if one neutron makes two, two give four, four give eight, and so on. This results in over one thousand fissions in ten generations.

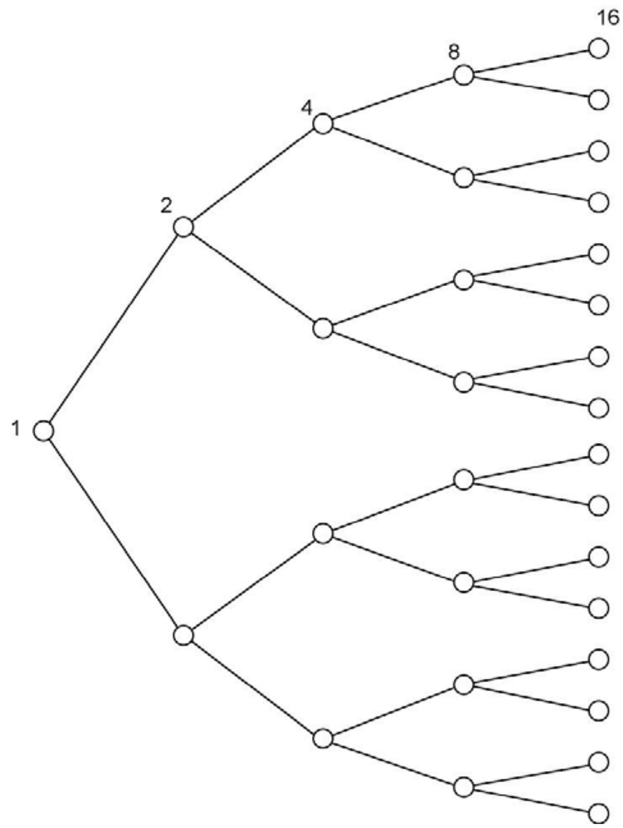


Figure 7.2
The Chain Reaction

his type of multiplication is unsuitable for a power reactor where steady power production is required. For a power reactor, we want each fission to cause just one other fission; so 1.5 neutrons must meet some fate other than causing fission. This special condition, where each fission causes one more, is called a "self-sustaining chain reaction" and will be discussed in detail in a later module.

7.4 Neutrons

7.4.1 Prompt and Delayed Neutrons

Most (99.35%) of the neutrons from the fission of U-235 are born at the time of the fission (10^{-14} seconds after the neutron is absorbed). A small number of the fission products emit a neutron while decaying. These decaying fission products yield the remaining 0.65% of the neutrons.

The neutrons born "instantly" at the time of fission are called prompt neutrons. The average lifetime before neutron emission from the fission products is 13 seconds. These neutrons are called delayed neutrons and will be seen to be indispensable to reactor control.

7.4.2 Neutron Energy

Neutrons from fission have relatively high energies near 2 MeV. High-energy neutrons travel at speeds a few percent of the speed of light and are called fast neutrons. They slow by undergoing elastic and inelastic collisions with surrounding nuclei until they reach energy equilibrium with their surroundings.

Once slowed, the neutrons diffuse through the core, jostled by surrounding molecules. (In subsequent collisions with neighbouring molecules the neutron is just as likely to pick up a bit of energy as to lose some). Such neutrons, in thermal equilibrium with their surroundings, are called thermal neutrons. A thermal neutron has energy of 0.025 eV at 20°C. Thermal neutrons are also slow neutrons.

7.4.3 Neutron Flux

Thermal neutrons are much more likely to interact with nuclei than are fast neutrons. The effect of the thermal neutrons at any point in the reactor depends on both the number of neutrons and their speeds. The quantity that relates these properties is the neutron flux, represented by the Greek letter ϕ (phi). Neutron flux measures the number of neutrons crossing a volume each second (moving in random directions). In this course, neutron flux can be thought of as a function of neutron population, that is, higher flux means more neutron “visits” to potential targets. Figure 7.3 shows the thermal neutron flux in a Bruce reactor. Flux distributions will be discussed later in more detail.

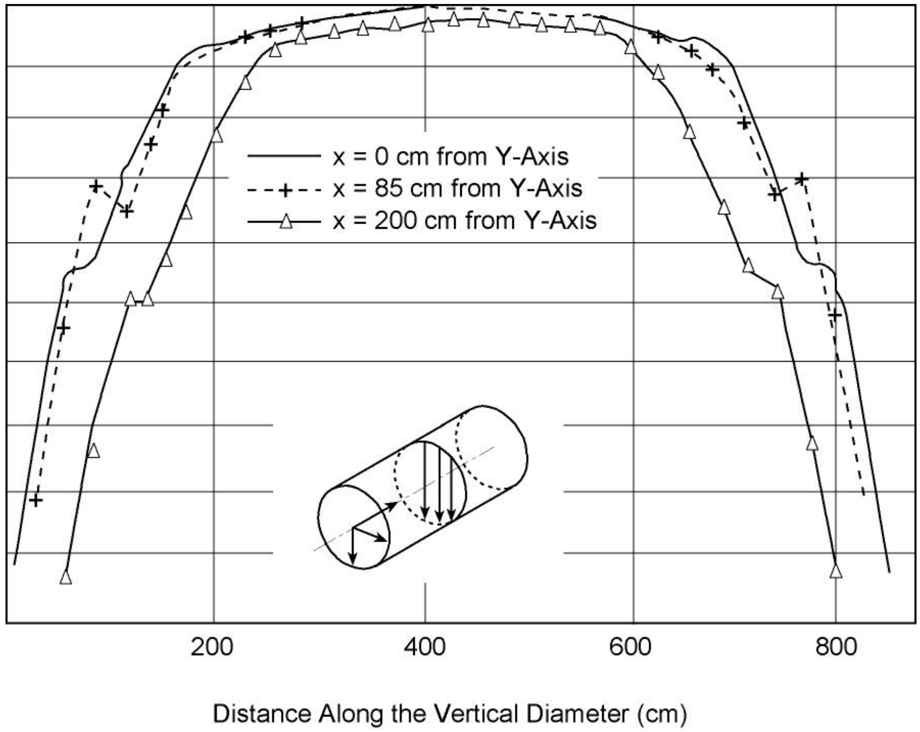


Figure 7.3
Neutron Flux

7.5 Neutron Cross-Sections

Module five examined two types of neutron reaction: scattering and absorption. Different nuclei have different probabilities of reacting with a neutron. A given target nucleus, struck by a neutron, has a different likelihood of undergoing these different reactions.

Neutron cross-section represents the probability that a reaction occurs when neutrons bombard a target nucleus. The Greek letter sigma (σ) denotes the cross-section. A nucleus has different cross sections for different reactions; subscripts denote the type of cross-section. For example, σ_a is the absorption cross-section and σ_f is the fission cross-section. The unit for cross-section is a barn (1 barn = 10^{-24} cm^2). To a neutron, an area of 10^{-24} cm^2 appears "as easy to hit as the broad side of a barn".

You can think of the neutron cross-section, with the dimension of area, as the effective target area of the nucleus for an incoming neutron, although the cross-section has no simple relationship with the actual geometry of the nucleus. The size of the cross section depends on:

1. The composition of isotopes in the target.

2. The energy of the incoming neutron.

The next two sections cover these two effects.

7.5.1 Effect of Composition

Uranium-235 has a fission cross-section of 580 barns for thermal neutrons, but makes up only 0.7% of natural uranium. The other 99.3% is U-238, which has a zero fission cross-section for thermal neutrons. Thus, the fission cross-section of natural uranium (used in CANDU fuel) is:

$$\sigma_f^{Nat.U} = 0.993 \times 0 + 0.007 \times 580b \approx 4 \text{ barns}$$

Enriched fuel with U-235 (typical for a light water reactor) has a fission cross-section:

$$\sigma_f^{2\%\,enriched} = 0.98 \times 0 + 0.02 \times 580b \approx 11.6 \text{ barns}$$

As you can see, enrichment increases the fission cross-section of the fuel. Fission is a more probable fate for a neutron entering enriched fuel, almost 3x as likely as for CANDU fuel.

One hundred tonnes of uranium fuel (typical for a large reactor) contains about 700 kg of U-235 if the fuel is natural uranium, 2 tonnes of U-235 if it is 2% enriched. Enrichment allows the fission process in light water reactors to compete effectively with neutron absorption by light water. CANDU accommodates a lower fission probability with a reactor design that is neutron efficient, that is, wastes few neutrons. However, we are getting ahead of ourselves. The next section describes how this is done.

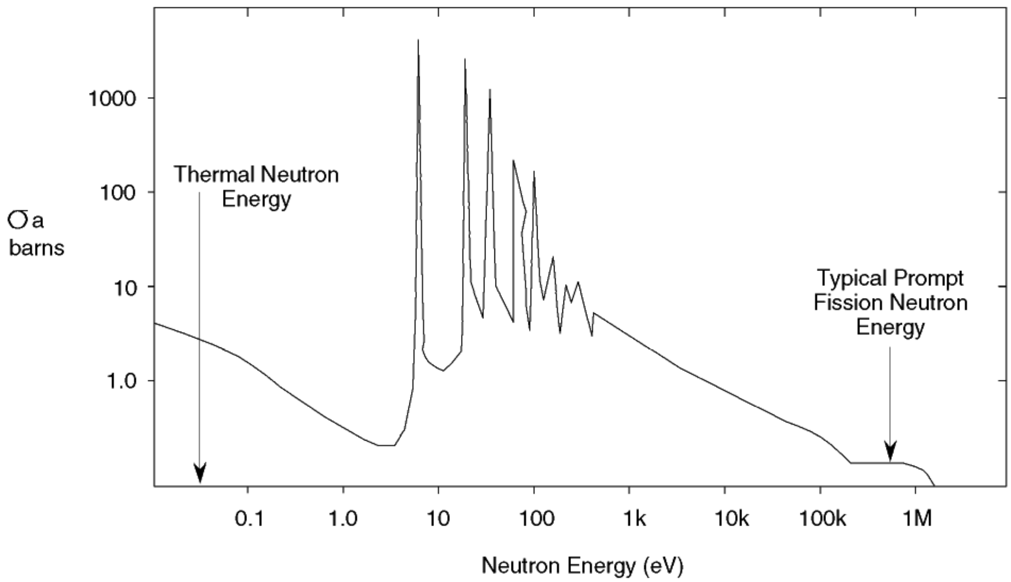


Figure 7.4 - Variation of the Absorption Cross Section of U-238 with Neutron Energy

7.5.2 Effect of Neutron Energy

For absorption reactions, cross-section decreases overall as neutron energy increases. For example, the fission cross-section for U-235 is 580 barns for thermal neutrons and only 2 barns for fast (2 MeV) neutron. This means fission is more probable (made 290 times more likely) if the neutrons are thermalized.

Figure 7.4 shows the absorption cross-section for U-238. For neutron energies near the thermal neutron energy, the cross-section falls off smoothly as neutron energy increases. This is typical of absorption cross-sections for most nuclei. As thermal neutrons travel at faster speed the apparent size of the target decreases, but the frequency of neutron “visits” increases. For many materials (but not all), these effects offset each other so the amount of absorption in the material is not much affected by thermal neutron speed.

The peaks in the energy range of ≈ 10 eV to ≈ 1 keV of figure 7.4 are called Resonance Absorption Peaks. The highest peak is over 6 000 barns. The peaks represent the only time absorption in U-238 is significant. U-238 is almost certain to absorb neutrons that enter the fuel in this energy range. Most nuclei have resonances, but the U-238 resonances are particularly important to us because there is so much U-238 in a CANDU core.

7.6 Summary of Key Ideas

- Fission of a nucleus results in releasing energy and more neutrons.
- The energy released comes from the conversion of mass following Einstein's famous formula $E = mc^2$.
- A single fission results in about 200 MeV of energy.
- 85% of the energy shows up as kinetic energy of the fission products.
- The rest of the energy is divided between gammas emitted at the time of the fission and kinetic energy of the neutrons.
- More energy is released after the fission when the fission products decay.
- A chain reaction occurs when the neutrons released by one fission cause fissions in other nuclei.
- Prompt neutrons appear at the time of fission.
- Delayed neutrons appear after the fission when certain the fission products decay.
- Almost all nuclides will absorb neutrons. The probability that a neutron is absorbed is called the neutron cross section. The unit of cross section is the barn.

7.7 ASSIGNMENT

1. Explain where the energy released by fission comes from.

2. Write the general fission reaction for ${}^{235}_{92}U$.

3. State how much energy a fission releases and how the majority of this energy shows up.

4. Explain a self-sustaining chain reaction.

5. Define:
   a) thermal neutron
   b) prompt neutron
   c) delayed neutron

6. Define neutron cross-section and state its units.

7. How does the probability of fission in U-235 vary with neutron energy?

8. Why are delayed neutrons important, although they contribute only a small fraction of the neutrons in the reactor core?

8 Fuel, Moderator, and Reactor Management

An aggregate of material that will just give a self-sustaining chain reaction is called a critical mass. If we had a small pile of pure U-235 and initiated fission, many neutrons would escape before they could cause further fissions. The chain reaction would die away. For a larger pile, fewer neutrons escape before causing fission and for some particular size, the pile supports a self-sustaining chain reaction. A pile of that size contains the critical mass of U-235.

It is an important feature of natural uranium (0.7% U-235, 99.3% U-238) that it cannot be made critical by itself, no matter how large a pile is assembled. The resonance peaks in U-238 absorb so many of the neutrons that too few neutrons remain to sustain a chain reaction.

To obtain a self-sustaining chain reaction with natural uranium the neutrons must be slowed to thermal energy (to increase the probability of fission) away from the fuel (to limit U-238 absorption). The neutrons move in random directions, so we cannot directly control this. However, by concentrating the fuel in channels separated by an effective moderator, fewer neutrons are exposed to resonance capture and there are high odds of thermalization. By using this process, most of the neutrons are:

1. Away from the U-238 while slowing through the energy range of the U-238 resonances, and

2. Far more likely, once thermalized, to be absorbed by U-235 and cause fission.

8.1 Moderator

CANDU reactors, like most other power reactors in the world, are thermal reactors. That is, they use thermal neutrons to cause fissions. For the reactor to operate, the fast neutrons released during fission must be slowed to thermal energies before they will cause another fission.

The function of the moderator is to slow the fission neutrons without absorbing them. To perform this function effectively a moderator must:

1. Thermalize the neutrons in as few collisions as possible over a short distance,

2. Not absorb too many of the neutrons.

Fast neutrons lose their energy mainly by elastic collisions with other nuclei. Elastic scattering with light nuclei is more effective than elastic scattering with heavy nuclei. It takes an average of 18 collisions to thermalize a neutron in pure hydrogen but 2172 collisions to thermalize the same neutron in U-238. Thus, only light nuclei are suitable as moderators.

The second point is low absorption. Boron-10 could thermalize a neutron in ~90 collisions but, with an absorption cross-section of 3840 barns it would absorb the neutrons it thermalized.

Because of these nuclear considerations and other engineering and economic considerations, only three moderators are suitable for thermal reactors*: light water (H_20), heavy water (D_2O) and graphite (C). Table 8.1 summarizes the properties of each.

Moderator	Average Number of Collisions to Thermalize	σ_s (barns)	σ_a (barns)
H_2O	20	103	0.664
D_2O	36	13.6	0.0010
C	115	4.8	0.0034

Table 8.1

Clearly, light water thermalizes a neutron faster than either heavy water or graphite (higher scattering cross-section coupled with fewer collisions to thermalization). However, light water's absorption cross-section is 664 times that of heavy water and 195 times that of graphite. Due to light water's neutron absorption, it is impossible to obtain a self-sustaining chain reaction with natural uranium fuel and a light water moderator. Light water moderated reactors must use 2 to 3% enriched fuel (uranium in which the percentage of U-235 has been increased from 0.7% to 2 or 3%).

Most reactor designs, including the CANDU, use UO_2 rather than uranium metal for fuel. Ceramic fuel (UO_2) has excellent corrosion resistance and is very stable in a radiation environment, making it a good choice for reactor fuel. However, it is impossible to obtain a critical mass of unenriched UO_2 with a graphite moderator. Heavy water is the only possible moderator for a reactor using natural uranium UO_2 fuel.

8.2 Fresh Fuel and Equilibrium Fuelling

When a reactor is first fuelled, the fuel is called fresh fuel. This initial fuel load is good for about 6 months. After this we remove and replace a few fuel bundles each day, a state called equilibrium fuelling.

Radical changes occur in the composition of the fuel between the fresh and equilibrium conditions. Three significant changes are the depletion of the U-235, mostly by fission, the build-up of fission products, and the build-up of Pu-239 (a fissile fuel) by the following nuclear processes:

$$ {}^{1}_{0}n + {}^{238}_{92}U \rightarrow {}^{239}_{92}U + \gamma \qquad \text{(radiative capture)} $$

$$ {}^{239}_{92}U \rightarrow {}^{239}_{93}Np + \beta + \gamma \qquad \text{(beta decay, } T_{1/2} = 23.5 \text{ m)} $$

$$ {}^{239}_{93}Np \rightarrow {}^{239}_{94}Pu + \beta + \gamma \qquad \text{(beta decay, } T_{1/2} = 2.35 \text{ d)} $$

Fresh fuel contains 0.7% U-235 and no Pu-239. By the time fuel is removed from the reactor the U-235 is depleted to near 0.2% and there is a similar amount of Pu-239. Plutonium fission provides a significant portion of the power produced by a CANDU reactor.

Each atom that fissions produces two new atoms so the fission products build up to a concentration over 1%. The content of U-238 in the fuel changes very little.

8.3 Reactor Arrangement

Figures 8.1 and 8.2 show the axial and radial arrangement of the fuel in the moderator. This arrangement permits most of the fast neutrons from fission to leave the fuel and enter the moderator before significant resonance absorption occurs. The moderator thermalizes the neutrons away from the U-238 in the fuel and most neutrons re-enter the fuel as thermal neutrons. This arrangement accomplishes two goals:

1. slowing the neutrons to thermal energy, which significantly increases the fission cross-section,

2. minimizing neutron captures by:

   a) Keeping the U-238 away from most of the neutrons while they are passing through the resonance energy range.

   b) Returning thermal neutrons to the fuel quickly to reduce absorption in the moderator.

The channel spacing shown in Figure 8.2 is very important and is carefully chosen for CANDU reactors. Any significant increase or decrease in this spacing decreases the probability of sustaining a chain reaction.

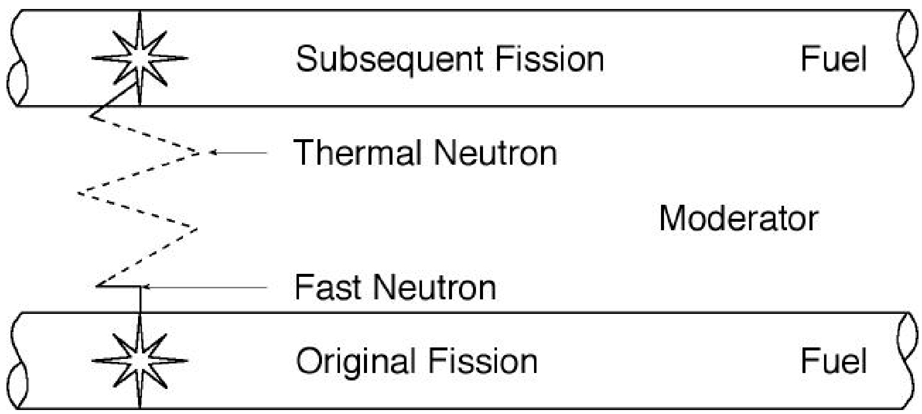


Figure 8.1
Axial Reactor Arrangement

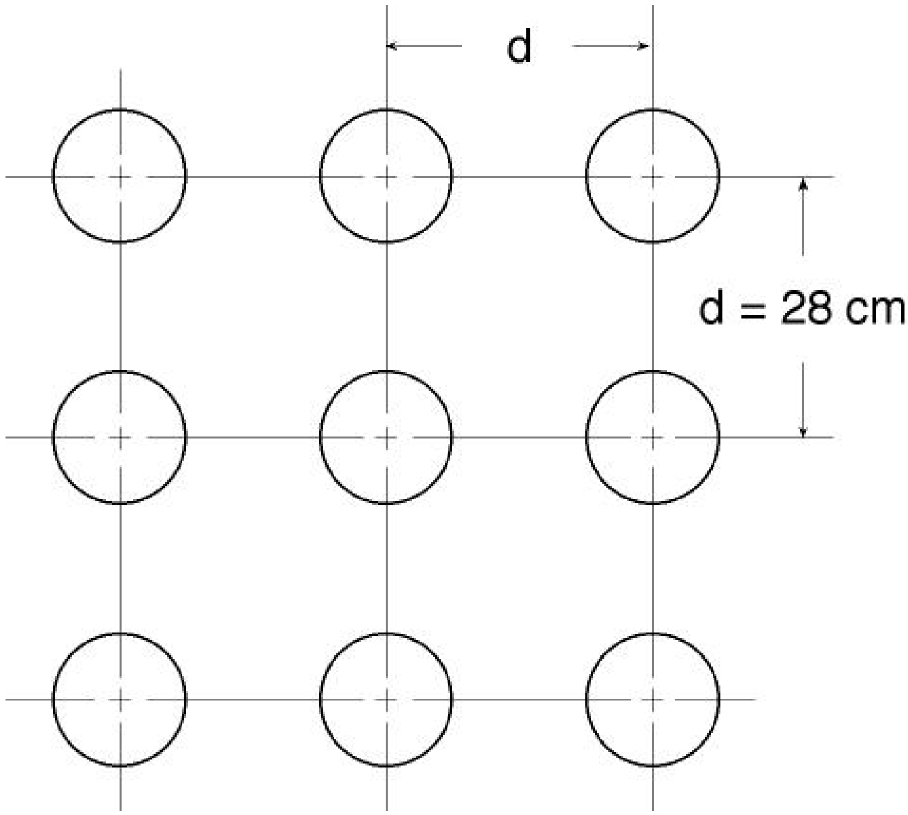


Figure 8.2
Radial Fuel Arrangement

An important safety feature of CANDU fuel is that it can constitute a critical mass only in heavy water in an arrangement similar to the one used. There is no chance of a criticality accident in the handling or storage of CANDU fuel provided storage and handling of D_2O and fuel occur in physically separated locations. New fuel is stored in a configuration that does not support criticality, in an area with good drainage, separated from any D_2O

8.4 Summary of Key Ideas

- Sufficient mass of U-235 would become critical on its own.
- No amount of natural Uranium can achieve a critical mass because of absorption in U-238.
- Neutrons must be away from the fuel when they are slowed so they do not undergo resonance capture by U-238.

- Materials suitable for moderators are H_2O, D_2O and carbon. D_2O is the best moderator, it has good slowing down properties and a very low absorption cross section.

- During the initial stages of irradiation, Pu-239 builds up in the fuel. Pu-239, like U-235 is fissile and produces significant amounts of power in the reactor.

- In a CANDU reactor the fuel is placed in channels about 4 meters long (in the reactor).

- The channels are arranged in a square lattice spaced about 28 cm from center to center.

8.5 Assignment

1. Describe the arrangement of the fuel and moderator in a CANDU reactor.

2. Explain why the arrangement described in question one is important.

3. Explain why heavy water (D_2O) is a better choice as a moderator than light water (H_2O). Can you suggest any disadvantages?

4. Describe the differences between fresh and equilibrium fuel.

5. How do you expect your answer to question one would be affected for:

   a. a light water moderated reactor with enriched UO_2 fuel,

   b. a graphite moderated reactor with enriched UO_2 fuel.

6. Suggest reasons why the fuel in the spent fuel bay will not go critical.

9 Nuclear Safety

9.1 Introduction

There is no question that there is a real risk to the public from the radioactive fission products produced in nuclear power plant fuel. If plants are to be allowed to operate this risk must be extremely low. Managing the risk (keeping it low) is referred to as nuclear safety. Nuclear safety is the way the plant is designed, built, operated and maintained. It is the hardware, work processes, the administrative processes surrounding a nuclear plant. This section provides a very brief overview of the philosophy and practices in nuclear plant operation.

9.2 Units of Radiation Exposure

Large amounts of radiation present an extreme health hazard to exposed people. However, there is debate on the effects of low levels of radiation. In fact, we are exposed to low levels of radiation each and every day. Before we discuss these sources, we need to define the units of radiation exposure. The initial measurements of radiation were energy absorbed by a unit mass of living tissue. The unit was named Roentgen after one of the early experimenters with radioactive material. The units used for Roentgen is the rad.

The damage done to a humans is not only a function of the energy absorbed but also how localized the damage is. One joule of gamma energy is less damaging than one joule of energy delivered by an alpha source. The energy from the alpha source will be deposited in a local area of the body; the energy from the gamma will be distributed over a wider volume and will be less damaging. A second unit was developed to take into account not only the absorbed energy but also the type of radiation that caused the damage. This unit is the Roentgen-equivalent-man or rem. Rem is the standard for measuring radiation dose.

Another catch was the implementation of SI units. Nothing went unscathed and rads were replaced by grays, and rem by sieverts. The relationship is that grays are 100 times larger than rads and sieverts are 100 times larger than rem. It is all very confusing since rem were already too large to be practical so they were replaced with a larger even more useless unit. Practically, radiation doses in a plant are measured in millirem. If the official SI units are used the practical measurement are in microsieverts.

Dose is one of the major considerations in the basic design and construction of the plant. In order to get a license, a utility has to have a sound plant design and demonstrate that, during a year, the dose received by someone living at the station fence will not receive a significant dose. In addition, it has to be shown that the probability of a failure that results in the maximum allowable dose to the person at the fence is very low. Everything that impacts on a release of radiation or the probability of a release of radiation is nuclear safety.

9.3 Radiation

The following introduces the concept of normal background radiation in our environment. Understanding background radiation is essential to a rational discussion of radiation hazards.

Background radiation is made up of a number of natural and man-made sources. Naturally, occurring radiation comes from radioactive elements, which have existed in the earth since its creation. All naturally occurring elements above 82 are radioactive as well as a few isotopes of elements with lower atomic numbers. In addition, the earth is subject to constant bombardment by cosmic radiation, which creates certain radionuclides in the atmosphere, such as tritium (hydrogen 3 -H^3) and carbon 14 (C^{14}). Man-made radiation results from nuclear weapons testing programs, the use of medical techniques which involve ionizing radiation, certain consumer items like luminous watches, television receivers, and video display terminals, and, of particular interest to us, nuclear power generation.

A by-product of the nuclear generation of electricity is large quantities of radioactive material contained within the reactors. Problems can occur if this radiation escapes into the environment. The largest source of the radioactivity and hence the greatest potential acute hazard is irradiated fuel; fuel in the reactor which contains radioactive elements from the fission process. How we prevent this acute hazard from becoming chronic is dealt with in the next section. Some chronic hazards are:

- the relatively large volume of low and medium level radioactive plant wastes such as process equipment, personal protective equipment, and clean-up materials, and
- low level radioactivity (mostly tritium) emitted from the plant on a more or less continuous basis as a normal consequence of operation.

Figure 9.1 gives the expected annual radiation dose to a member of the public from each of the sources listed above. Please be wary of these average numbers; they are averages over many varying circumstances. However, they do give some feel for the meaning of the numbers.

Sources of Radiation	Millirem/Year
Natural Radiation (Cosmic Rays, Potassium-40, Building Materials, radon, etc.)	202
Medical Exposures	110
Nuclear Weapons Test Fallout	2
Occupational	4
Consumer Products	2
Nuclear Power	0.3

Figure 9.1
Annual Individual Exposure to Background Radiation in Ontario (Average for the Ontario Population)

As can be seen, the contribution of nuclear power to the radiation exposure of members of the public is extremely low. Exposure of Nuclear Energy Workers is more significant. The exposure limit set by the Canadian Nuclear Safety Commission (CNSC) for Nuclear Energy Workers is 5000 millirem/year. (For non-nuclear energy workers, the limit is 500 millirem/year.) This figure represents the amount of exposure a person can sustain year after year with no measurable effect. For comparison with actual exposures in and around our nuclear plants, see Figure 9.2.

Proximity to Station	Annual Radiation Dose above Background	Relative Value
Station Operating Staff	400 millirem/year	5000 millirem/year CNSC limit
Station Office Staff	20 millirem/year	The same as spending 4 months in Denver, Colorado.[1]
Person at Station Exclusion Fence	0.3 millirem/year	The same as a round-trip by air between Toronto and Vancouver.
Residential Area 1 kilometre from Station	0.1 millirem/year	The same as radon exposure from living 2 months in a brick building.

Figure 9.2
Typical Radiation Exposures in and Around Our Plants

Note that the typical dose for operating staff (including maintenance staff) in station stations is 400 millirem/year, considerably less than the legal limit. As the distance from the reactor increases, radiation normally falls off dramatically until at the Station Exclusion Fence, the typical potential exposure is 0.3 millirem/year.

In conclusion, the incremental increase from nuclear power over background radiation likely to be received by members of the public is very low. Medical risks at these low dosages cannot be accurately determined. In the worst case it is believed that the effect of radiation is directly proportional to the dose (e.g., half the dose, half the effect). However, there is evidence that this assumption overstates the actual risk. For example, people living in Denver, Colorado, where natural levels of radiation are higher, actually exhibit a lower incidence of cancer than those in other areas of the United States. This is not to suggest that low levels of radiation are beneficial, but merely to illustrate that the effects of low-level radiation are far from clear.

---

[1] **Background radiation increases at higher altitudes due to reduced shielding from the atmosphere against cosmic radiation.**

Even at the typical exposure of 400 millirem/year for operating staff, employees receive much less than the minimum dose considered dangerous to health. Even so, the Nuclear Business is always seeking ways of doing work to lower radiation exposure.

9.4 Conventional Safety

Nuclear generating stations have extensive conventional safety programs. It is not possible for people to work safely in one environment and not in another. In order to promote a safe work environment the utilities promotes safety in all aspects of the job and even off the job.

The first nuclear plants set safety targets that were twice as stringent as those in more conventional industrial plants. The safety targets have gotten better and better over the years so that now the targets are zero lost time accidents, not 1 lost time accident.

9.5 ALARA - A Philosophy towards Hazards

The radiation hazards associated with our facilities are very real. Control and containment methods are designed to prevent any harm to the public. The principle observed is to "play it safe" by reducing hazard levels As Low As Reasonably Achievable (ALARA). The practical application of ALARA has reduced the radiation dose received by those who work in the operating part of the plant by about a factor of 10 over the 40-50 years of plant operation in Canada.

9.6 Self-Checking

Most incidents have a number of causes but one of the causes is usually some type of human error. There is a major attempt by the industry to limit human errors. One way to keep incidents, from human error, low is through the technique of self-checking. This technique is designed to reduce the number of inappropriate actions that could lead to an incident by helping staff to focus consciously on the details of the task they are performing. A deliberate review of both an intended action and the expected response will often make it possible to identify a potential problem before it becomes real. Not all of our actions have the potential to cause serious problems.

9.7 Reactor Safety

There is an acute hazard posed by the radioactive materials contained within nuclear stations. In order to minimize the potential threat from these materials, a number of principles have been developed and incorporated into the design and operation of nuclear generating stations. Collectively, these principles are known as Reactor Safety. The golden rule of Reactor Safety can be stated as:

THERE IS A MINIMUM RISK TO THE PUBLIC AND THE ENVIRONMENT FROM REACTOR FUEL, PROVIDED THAT AT ALL TIMES:

- THE REACTOR POWER IS CONTROLLED,
- THE FUEL IS COOLED, AND
- THE RADIOACTIVITY IS CONTAINED.

This rule is often shortened to CONTROL, COOL, AND CONTAIN. This section is intended as a brief introduction to some of the key concepts of Reactor Safety. It will examine basic reliability concepts, the Defence in Depth model and the role of station documentation.

9.8 Defence In Depth

There are different ways of achieving the golden rule (CONTROL, COOL AND CONTAIN). Many of these have been incorporated into an important concept known as Defence in Depth. This underlies the whole process of design, construction, commissioning, and operation of a CANDU reactor. One way of presenting this concept is the five-part model illustrated in Figure 9.3.

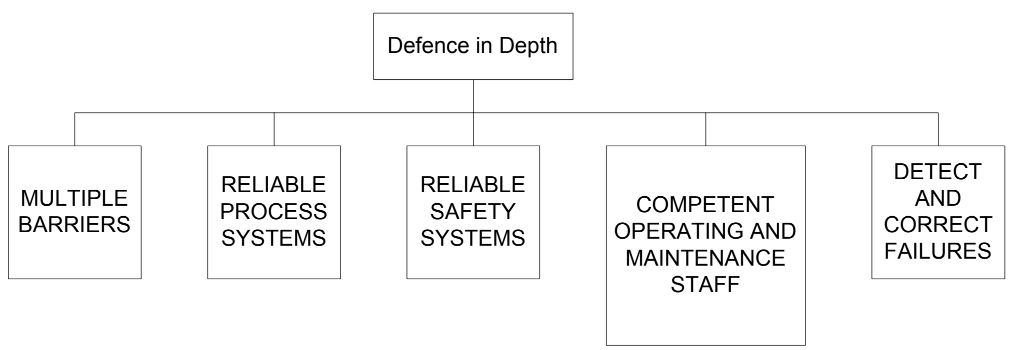


Figure 9.3
Defence in Depth Model

The Defence in Depth concept assumes the following:

1. Nuclear station design will have some flaws,
2. Equipment will occasionally fail, and
3. Operating personnel will occasionally make mistakes.

The key is to ensure sufficient depth of defense that flaws, failures and mistakes can be accommodated without increasing the risk or consequences of an accident. If we look at each of the major blocks of the model in turn, we can see how this is accomplished.

9.8.1 Reliable Process Systems

Process systems are those systems performing a continuous function in the normal operation of the plant. For example, the primary heat transport system is a process system that is continuously active in the removal of heat from the fuel. The reactor regulating system is a process system that is continuously active in the normal control of reactor power. Reliable process systems ensure that heat is produced and electricity generated while maintaining control, cooling and containing.

9.8.2 Reliable Safety Systems

Safety systems are poised systems that operate only to compensate for the failure of process systems. They can do this by shutting down the reactor to regain control (shutdown systems), by providing additional cooling to the fuel (emergency coolant injection system), and by containing radioactivity, which has escaped from the fuel (containment system). Reliability in this context means that in the rare event these systems are called upon to act, they will be available to perform their intended function.

9.8.3 Multiple Barriers

The multiple barrier approach that has been built into station design is intended to prevent or impede the release of radioactivity from the fuel to the public. There are five passive barriers (refer to Figure 9.4) continuously available:

1. The uranium fuel is molded into ceramic fuel pellets which have a high melting point and lock in most of the fission products,

2. The fuel sheath which is made of high integrity welded metal (zircaloy) and contains the ceramic fuel,

3. The heat transport system which is constructed of high strength pressure tubes, piping and vessels and contains the fuel bundles,

4. The containment system which provides a relatively leak tight envelope maintained slightly below atmospheric pressure. This partial vacuum encourages

air to leak in instead of out thereby helping to prevent release of radioactivity that escapes from the heat transport system, and

5. The exclusion zone of at least one kilometre radius around the reactor that ensures any radioactive releases from the station are well diluted by the time they reach the boundary.

For radioactivity to reach the public from the fuel, it would have to breach each of the five barriers in succession. This provides a significant degree of protection to the public.

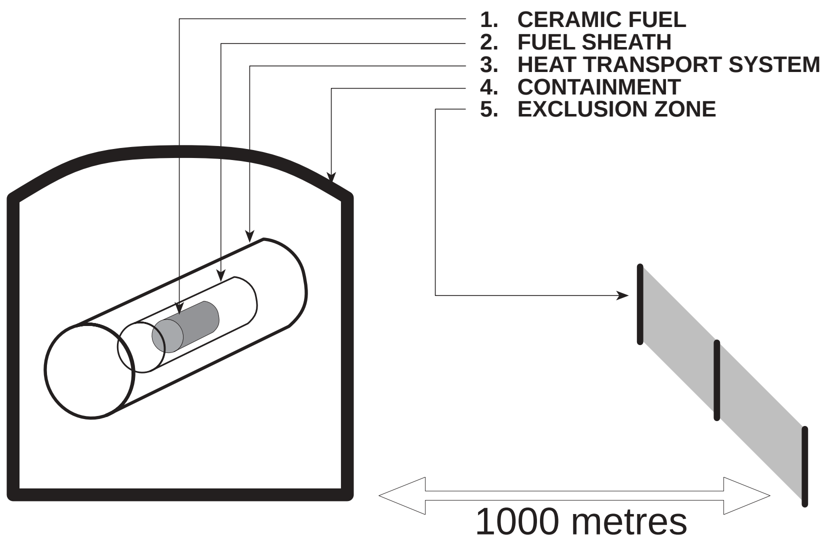


Figure 9.4
Physical barriers

9.8.4 Competent Operating and Maintenance Staff

The safety systems are designed to operate automatically and the five passive barriers are always in place, but the Defence in Depth concept does not allow reliance on equipment and systems to prevent accidents. It is important that operating and maintenance staff are knowledgeable about system conditions, alert for any evidence that systems or equipment may be on the verge of failure, and act promptly to prevent or minimize the consequences of such failures. To achieve a high level of competence, the qualification criteria for each job family are clearly defined. Considerable effort goes into performance-based training of staff to meet those criteria and maintain their qualification.

9.8.5 Detect and Correct Failures

Adequate detection and correction of failures requires not just competent staff but also processes and procedures for the staff to carry out in a systematic fashion. For example, a routine testing program for safety systems helps meet the availability targets. An operational surveillance program in conjunction with a planned preventive maintenance program helps to ensure that equipment and systems are monitored, inspected and repaired before they fail. Failures, when they do occur, are thoroughly investigated and solutions applied through a rigorous change approval process. Elaborate work control processes exist, allowing the quick reporting, prioritizing and repair of deficiencies.

9.9 Basic Reliability Concepts

Reliability is concerned with the overall operation of nuclear generating stations. In addition to trained and motivated staff, overall station reliability is a function of the reliability of systems and equipment. Reliability is critical piece of the Defence in Depth model and is therefore critical to the safe operation of our reactors. The following material introduces basic reliability concepts as they relate to CANDU equipment and systems.

9.9.1 Definitions

Reliability is defined as the probability that a device will work adequately for the period intended under the operating conditions encountered.

Reliability is a probability with a numerical value ranging from 0 (totally unreliable) to 1 (always operates for the time intended). If a pump is judged to have a reliability of 0.99 for its first year of operation (based on historical data for this type of pump), this means that for 1000 hours of operation the pump will be unavailable for no more than 10 hours.

Reliability is concerned with whether an operating component in a process system is likely to fail. When dealing with poised systems, the concern is whether a system or component will be available when called upon to operate. A process system is a system that operates when the plant is producing power. A poised system is one that is sitting waiting to operate in the event of specific events. In your car the engine cooling system is a process system, the air bag is a poised system.

Availability is related to reliability but is defined as the fraction of time that a device is available to work if called upon to do so.

Availability has a value of from 0 (never available) to 1 (always available) and is generally expressed as years per year or hours per year. The value, which is more frequently encountered, however, is unavailability. For example, if a poised system has an unavailability target of 10-3 years/year, this means that it will be unavailable for no more than 8 hours during the year (1 year = 8760 hours and 8/8760 is approximately 10-3).

9.9.2 Concepts

High reliability and availability can be achieved by attention to a number of reliability principles during design and operation of a station.

Redundancy

If only one component exists to perform a certain function, when it fails, the system fails. This problem can be reduced by installing additional components, so that if one fails, there is another to do the job. In other words, higher reliability can be attained by providing a backup (or redundant) component. It is important to understand that this redundancy is provided primarily to ensure reliable operation, not to allow more convenient maintenance. Taking redundant equipment out of service for maintenance will lower the reliability of the system.

We can look at the space shuttle program to provide us with an example. The computer control system in each shuttle contains more than one computer. Redundancy is provided by running the same software control program on more than one computer. If one computer fails, another is immediately available to assume control.

Independence

Independence is the physical separation of systems or components so that a fault in one system will not affect the others. Using the space shuttle, an example of independence is separate power supplies for each of the computers. This way failure of the power supply to a computer does not at the same time disable the other computers.

Diversity
Diversity is an attempt to ensure that there is more than one way of doing a job. Again using the space shuttle, diversity is provided by running entirely different software control programs on different computers to achieve the same purpose. The software is even created by a different design team. This ensures that a bug in one piece of software is not duplicated in the other so that one mistake cannot disable more than one computer.

Periodic Testing
When a component in a process system fails the effects are immediately apparent. Failure of a poised system, on the other hand, is not readily apparent and can only be determined by testing. Since it is not possible to determine at what point the failure occurred, the unavailability is considered to be half the time since the system was last tested (plus however long it takes to make the repairs). It follows that unavailability can be kept low by more frequent testing. The frequency of testing must, however, be balanced against:

- Wear and tear on the system and components caused by testing,
- Unavailability due to removing components from service for the duration of the test,
- The risk (by human error) of leaving the system in a degraded state after a test, and
- The danger of activating the system during the testing process.

Fail Safe Operation
A system or component is called fail safe if after failing it leaves the remainder of the system in a safer state. For example, train locomotives are equipped with a deadman brake. It must be depressed by the engineer to allow the locomotive to move. If the engineer falls over dead, his foot will come off the brake and the locomotive will come to a halt.

Operational Surveillance
Operational surveillance is a process of continual monitoring and trending of process parameters and equipment with the intent of spotting potential problems before they become real problems. Thus, corrective action can be taken before a major problem occurs. An example is vibration monitoring of rotating equipment. If unusual vibrations are detected, the equipment can be stopped and repaired before the vibration causes serious damage.

Preventive Maintenance
Reliability data on different types of equipment offers a means of estimating when failures are likely to occur. By planning replacement or maintenance before any appreciable deterioration occurs that can contribute to the predicted failure, it is possible to reduce the number of unscheduled outages and consequent loss of production. This sometimes has the appearance of throwing away good equipment, but the reliability statistics indicate that the equipment is likely to fail shortly and probably inconveniently (remember Murphy's Law).

Predictive Maintenance
The best form of preventive maintenance is predictive maintenance, which is based on equipment condition. Maintenance or replacement is only done when diagnostic test results (such as vibration monitoring) indicate equipment degradation.

9.10 Documentation

Operation of a nuclear station is governed by a licence issued by the federal nuclear regulator, the Canadian Nuclear Safety Commission (CNSC). To support the application for a licence, the station designers prepare a Safety Report that describes the physical plant and how it supports protection of the public, the environment and the employees. The safety report also analyzes how well the plant will cope with a number of accident scenarios specified by the CNSC. The safety report is updated every three years. When granted, the Station Operating Licence is the contract between the utility and the CNSC and defines the general boundaries within which the station will be operated.

Within the licence, one of the clauses dictates that operation of the station will be governed by a set of Operating Policies and Principles (OP&P). The OP&Ps ensure safe station operation by defining limits on station operation. These limits are either stated qualitatively or spelled out with quantitative values. The OP&Ps embody good operating practices based on established reactor safety principles. For example, the OP&Ps define the requirement for a maintenance program, periodic testing, and reactor power limits. Violation of an

OP&P would place the plant in a state which has not been analyzed in the Safety Report, and which might therefore be unsafe. To operate in such a state could impair the capability of the plant to respond properly to accident conditions.

Subordinate to the OP&Ps, station Operating Procedures, which include operating manuals and maintenance manuals, define the precise details of station operation and maintenance. These procedures are rigorously prepared, verified and approved.

To provide some assurance that station operation remains within the bounds specified by the OP&Ps while enabling improvements to be made, each station has a Change Control Process in place to ensure that all planned deviations in plant operation or design are properly analyzed and approved. On a day-to-day basis, the Work Authorization Process serves a similar purpose by enabling the control room staff to monitor work to ensure that it will not step outside the bounds of the OP&Ps. It also serves to protect workers doing the job.

The operating licence is not the only contract a utility has with regulatory agencies. A provincial government agency grants Certificates of Approval that govern operation of non-nuclear facilities at plants such as the water treatment plants, or limit the temperature differential between the cooling water inlet and outlet at a generating station. These are contracts between the utility and the regulator (in Ontario this is the Ministry of the Environment) concerning conventional processes within the plant.

9.11 Nuclear Station Radioactive Emissions

With respect to the Nuclear Business' obligation to control radioactive emissions from the nuclear stations, performance has consistently met the standards imposed by the CNSC. Typically, utilities control discharges to less than 1% of the regulatory limits set by the CNSC. Utilities monitor airborne emissions for tritium, iodine, noble gases, and particulates, and waterborne emissions for tritium and gross radioactivity.

The small amount of radioactivity released from plants is either to atmosphere or to the lake. Within the plant, a release to atmosphere is reduced significantly by dilution in the surrounding air. For the public, further dilution is provided by the exclusion zone (the last of the five barriers) around the stations. In any event, the amount and type of radioactivity that is released is carefully controlled and monitored. Typically, a utility’s internal emission target is 1% of the allowable emission for the radioactive substance.

In order to ensure that operating targets are achieved and maintained, an extensive program of environmental monitoring has been established. Samples are taken from fixed positions around nuclear sites at regular intervals. Measurements are taken of both the air and water at sampling sites. Samples are also taken of lake sediments, fish, fruit and vegetables. In addition to the monitoring carried out by a utility, both federal and provincial regulatory agencies carry out independent sampling as well.

9.12 The Role of Licenced Positions within the Nuclear Station

Nuclear stations are operated within the framework of the station operating licence granted by the CNSC. The CNSC is responsible for verifying that operation is carried out within the terms of the licence. Those terms require the following positions within the station organization to be approved or licensed by the CNSC:

- Shift Manager,
- Authorized Nuclear Operator.

To obtain authorization from the CNSC for these positions, individuals undergo a rigorous training and examination process. The process is monitored and audited by the CNSC. In fact, the CNSC monitors the training of all employees; the program for control room staff is the most extensive. Approval of some department managers usually requires previous experience as a Shift Manager and requires a formal interview with the CNSC. These requirements are outlined in the operating license.

The Shift Manager is the senior position on shift. This position has the ultimate responsibility for managing the station (both operation and maintenance) to ensure that the Station License, OP&Ps and other high level procedures are not violated. In the large multi-unit stations, there is a third licensed position, the Control Room Shift Supervisor. The Control Room Shift Supervisor reports to the Shift Manager and has the responsibility for supervising control room operations of all units. This includes monitoring operations and maintenance carried out in the station to ensure that they comply with the OP&Ps.

The Authorized Nuclear Operator (ANO) carries out operations according to approved procedures. The ANO exercises direct control over one unit in the station. A unit is a reactor, associated turbines, generator and all of the associated support systems. The ANO is responsible for carrying out panel operations, directing field operations and maintenance on the unit. The ANO authorizes most work in the unit, but sometimes the approval of the shift manager is required as well. Work on special safety systems or reactivity devices (devices for controlling reactor power) would typically require this type of approval. This is a procedural barrier to prevent OP&P violations. The OP&Ps and operating procedures indicate where higher level of approval is needed.

The important feature of each of the above positions is that, in addition to normal supervisory responsibilities, they are directly licensed by the CNSC to provide assurance that station operation is carried out within the limits of the Operating Licence. As such, they are legally required to be intimately involved in the work carried out by most work groups within the station through the work approval process.

9.13 Assignment

1. Explain the reactor safety philosophy that is encompassed by the words control, cool and contain.
2. Explain the purpose of self-checking.
3. Define self-checking.
4. What are the three basic assumptions upon which the Defence in Depth concept is based?
5. Identify the five parts to the Defence in Depth model and briefly describe the intent of each.
6. List in order the five major barriers designed to prevent the release of fission products from the fuel to the environment.
7. Explain the difference between reliable and available.
8. CANDU stations have two different types of automatic shutdown system. What principle does this illustrate?
9. Provide an example of system independence.
10. How does redundancy contribute to higher reliability for a system?
11. How does the frequency of testing affect unavailability of a poised system?
12. State at least two reasons why frequency of testing of safety systems must be limited.
13. A heat exchanger requires cooling flow at all times. A valve upstream of the heat exchanger regulates cooling flow. What would be the fail-safe position of that valve?
14. How does the Safety Report support the Operating Licence?
15. Apart from the legal consequence, what is the danger of violating a limit defined in the OP&Ps?
16. What roles do the Operating Licence and Certificates of Approval play?

17. What are the four airborne and two waterborne emissions monitored at nuclear stations?

18. Why does the CNSC directly license several supervisory positions within the station.

19. How does the Control Room Shift Supervisor monitor work going on in the station to ensure that the operating license is being observed?

10 Nuclear Power Reactors

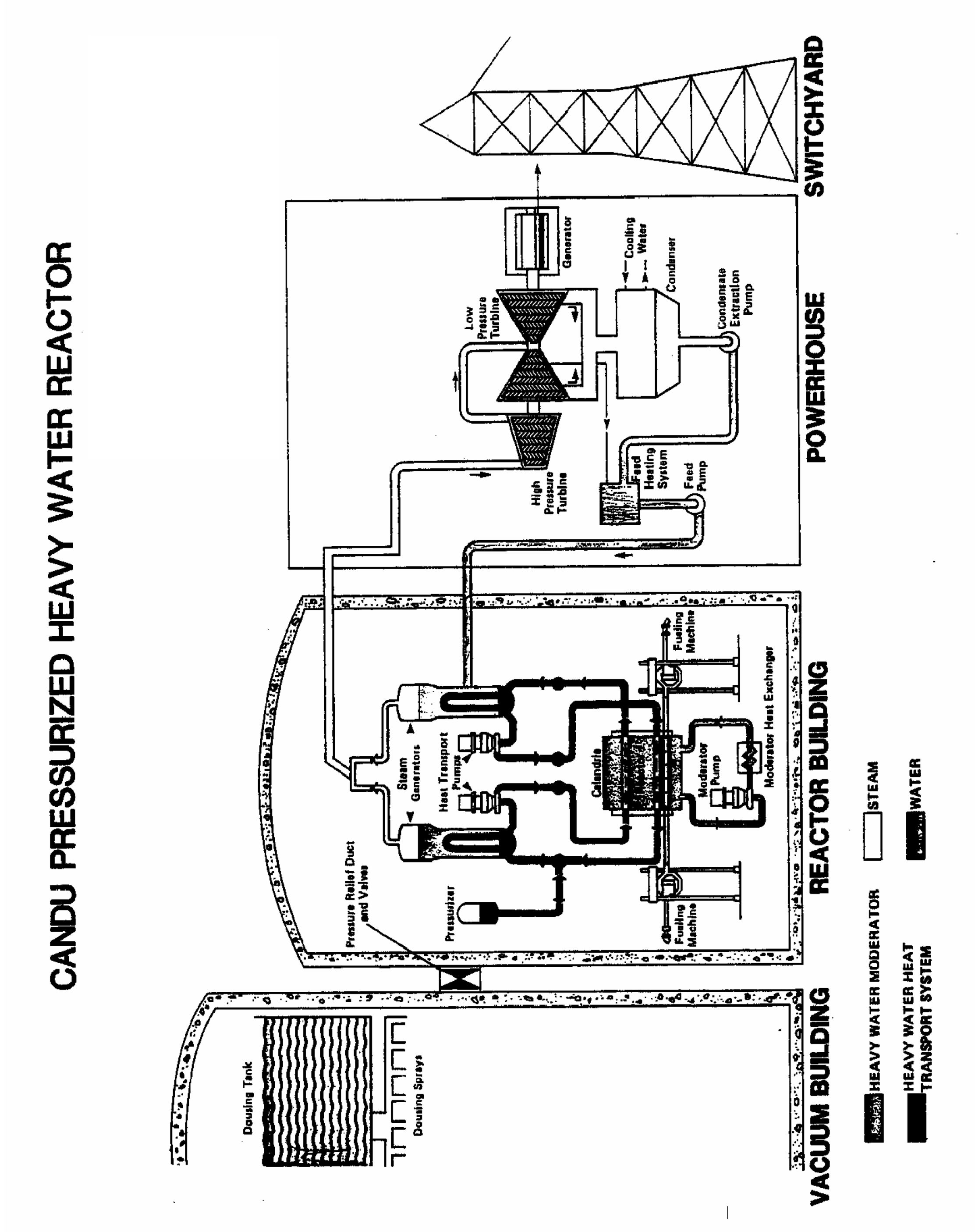


Figure 10.1

10.1 What is a Nuclear Power Station?

The purpose of a power station is to generate electricity safely reliably and economically.

Figure 10.1 is the schematic of a single nuclear generating unit. Often several units share equipment. Each nuclear power station in Ontario includes four CANDU reactors. Quebec and New Brunswick have built single unit stations. The next sections of the course are mostly about the systems and equipment shown inside the reactor building of Figure 10.1

An obvious difference between a nuclear power station and a fossil fuelled plant is the heat source. In the CANDU reactor, heavy water coolant is pumped over hot uranium dioxide fuel and becomes hot. It then flows into a boiler where it gives this heat to ordinary water, converting it to steam. In a conventional plant, heat to make steam comes from burning coal or oil. In each case the steam drives a turbine that turns a generator.

The name CANDU comes from CANada Deuterium Uranium.

Heavy water is deuterium oxide, D_2O. Deuterium is a heavy form of hydrogen, which is found naturally at a concentration of about one deuterium atom for every 7000 hydrogen atoms.

The part of the nuclear reactor that produces heat is called the reactor core. It includes the fuel, the coolant and the moderator. The nuclear fuel can get hot only when the moderator surrounds it. A distinctive feature of CANDU reactors is the heavy water moderator.

10.2 Hazards

The fuel generates heat and intense radiation when it is in the reactor. It is safe to handle before it is used, but it contains deadly quantities of radioactive matter after it has been in the core for a short time. These radioactive materials continue to produce radiation for a long time.

The thermal power output from the core of a CANDU at full power ranges between 1700 and 3000 MW_{th}. If systems that remove the heat fail, the heat will damage the fuel and reactor components. This could allow lethal radioactive material to escape from the reactor core. Damage results when thermal expansion and high pressure cause stress and distortion of reactor parts. Also, materials may become weak or melt at high temperature.

Considering this obvious hazard, why do we have nuclear reactors? The next section of this module gives some reasons. The rest of the course shows how safe, reliable and economical nuclear power production is possible.

10.3 Summary Of The Key Ideas

- The purpose of a power station is to generate electricity safely, economically and reliably.

- The core of a CANDU reactor contains natural uranium dioxide fuel, heavy water coolant and heavy water moderator.

- The fuel in a reactor generates heat to make steam, but produces radiation and radioactive materials in the process.

- Heavy water has unique nuclear properties that allow CANDU reactors to use natural (unenriched) uranium in the fuel.

- 1 MW (megawatt) = 1000 kW (kilowatts). Power units measure the role energy is produced or delivered. Electric energy is sold in watt or kilowatt hours (kWh). For example, a rate of energy production (power) of) MW delivers 1 kWh of energy every 3.6 seconds. You will also see the symbols MWth (megawatts thermal) to indicate heat energy production wad MW_e (megawatts electrical) for electrical generation.

10.4 Production Methods Compared

10.4.1 Alternatives

We have become used to reliable and relatively inexpensive electric power. As a result, we do not always use electric energy appropriately or efficiently. Effective conservation can reduce the demand for electric power. This can delay or even eliminate the need for some new power stations.

Small power sources (for example, windmills in remote locations and dams on smaller rivers) also can fill certain niches. Most utilities forecast the need for additional large power stations. Until someone finds a better technology, the choice is between nuclear power and fossil fuelled plants.

10.4.2 Economics

The inventors of nuclear power expected it to be a much cheaper energy source than energy from fossil fuels (coal, gas or oil). The economic advantage of electricity from nuclear power compared to fossil fuel comes from the cost of the uranium fuel. For comparison, in

1990, nuclear fuel costs in Ontario were about 10% of the fuel cost for an equivalent sized fossil plant. A large coal fired plant uses about 20,000 tonnes of coal a day. The equivalent nuclear plant uses about 20 fuel bundles, each with less than 20 kg of uranium.

People who predicted cheap nuclear power overlooked costs that offset fuel savings. Nuclear plants are expensive to finance, build and maintain. Capital cost is at least three times that of an equivalent fossil plant Also, nuclear plants take 3 or 4 years longer to construct, commission and bring into service.

Predicting the cost of both conventional and nuclear power production is difficult. Financing for construction must be in place years before revenue from power sales begin. The cost of nuclear power, with its high front-end cost, depends strongly on interest rates. Fuel cost is a much more important part of the price of electricity from fossil fuels.

From initial construction to in-service presently takes 8-10 years for a nuclear plant. The designer is expecting that a modular construction technique will allow the newest design of small reactor to be built in half that time.

Cost estimates for each kind of power generation depends on how prices change during the life of the plant. Fossil generated power becomes expensive if fuel costs increase sharply after the plant is built Cost increases after the plant starts operating do not affect the cost of nuclear power as much. Borrowing money to build a nuclear plant is a good strategy when costs are expected to increase, especially if interest rates are low. When fuel costs are stable or expected to fall or the cost of borrowing money is high, coal may be a better choice.

Cost comparisons affect how we run our power plants. Electricity demand varies, so plants do not all run continuously at full power. In a fossil fuelled plant, power reductions save money on fuel. Most of the costs of nuclear power continue, whether or not the plant is producing energy. As a result, the economic way to run a nuclear plant is to keep it operating at full power.

We call continuous operation at full power base load operation. When demand changes, the fossil fuelled plants adjust their output. We call this load following or demand power operation. Some nuclear units also adjust their output to the load. Large power maneuvers are not possible because of operating problems caused by nuclear processes in the fuel. This will be discussed more in future sections of this course.

Frequent power changes increase the risk of damaging the nuclear fuel because of thermal stress.

When a nuclear unit is down for repairs, expensive electricity from a fossil fuelled plant replaces the lost production. Meanwhile, nuclear costs remain high during the shut down. Equipment failures, unplanned maintenance and operating errors cause expensive outages in nuclear plants. It is important to the utilities to prevent unplanned outages and when they do occur to limit the duration.

Experience with CANDU reactors shows nuclear power is much less expensive than fossil fuel power when nuclear plants are kept running 80% or more of the time. New CANDUs routinely do this well or better.

Good performance over the life of a reactor will result in a significant cost advantage of nuclear over fossil fuel. Poor operation can make nuclear power very expensive.

A capacity factor of 60% is sometimes given as a crude estimate of the break-even point between nuclear and fossil fuelled plants.

The advantage of uranium over fossil fuel will likely increase as world oil and gas supplies dwindle and concerns about greenhouse gases grow. The supply of uranium is finite too. Canada is fortunate to have supplies for itself and for a large export market for at least 50 to 60 years. It is possible, but not now economic, to use thorium as the nuclear fuel in a CANDU reactor. When uranium is used up, thorium could extend fuel supplies for 100 years or more.

10.4.3 Environmental Effects

The effects of nuclear and fossil fuelled power plants on the environment are very different. Radiation is an inevitable disadvantage of nuclear power production. Both heat and radiation are produced when the process of nuclear fission converts uranium, which is slightly radioactive, into highly radioactive fission products. Fission products in the fuel emit radiation long after the fission process has stopped. Activation produces some radioactive substances in materials exposed to radiation from the core.

These factors affect the construction of nuclear plants and how work is performed in them. Radioactive releases to the public, however, are extremely small. Releases are carefully monitored. The legal limits on release of radioactive materials are lower than the radiation level from sources in the natural environment. Typical releases during operation are a fraction of allowable releases. We know there are small emissions, but sampling near CANDU plants does not show radiation levels higher than normal background.

What alarms people is the chance of an accidental release of large amounts of radioactivity. This course describes equipment that helps prevent such accidents. The following principle of safe operation is oversimplified, but is essentially correct.

If the fuel is kept wet it will not fail and release radioactive material.

The following sections will point out features of the reactor design intended to make sure the fuel stays wet. We examine reactor safety, which depends on people as well as equipment, in other courses.

Activation is the name given to any interaction of radiation with matter that converts over time matter into a radioactive substance.

Conventional plants also produce wastes. A large coal fired unit without scrubbers generates over 15,000 tonnes of carbon dioxide ($C0_2$), 200 tonnes of acid gas (sulphur dioxide and oxides of nitrogen), several tonnes of fly ash and 500 tonnes of ash each day. These wastes are much less harmful than nuclear radiation, but they are not completely harmless. The above includes, for example, more than a tonne of poisons such as arsenic and mercury.

Scrubbers reduce releases from modern conventional plants, but the release of some acid gases and toxic materials is inevitable. Scrubbers cannot remove carbon dioxide. CO_2, while not directly harmful to health, adds to global warming.

With or without scrubbers, a conventional plant produces a large quantity of solid waste that requires disposal. A large nuclear unit, in contrast, uses fewer than 20 nuclear fuel bundles a day. That is less than a half tonne of waste. At present, water pools at the plant site hold highly radioactive used fuel. Concrete silos can safely store older fuel.

Long-term storage of nuclear waste is another problem. Radioactivity from nuclear decay decreases over time. The radioactivity of different materials decreases at different rates. In 100 years or so, the radioactivity of the used fuel comes from a few very long-lived radioactive substances, mainly plutonium and the remaining, unused uranium.

Because of these long-lived isotopes, the used fuel must be isolated from the environment for a very long time. Designing a secure storage method has been difficult. The political problem of setting up a disposal site also will be difficult.

In summary, used nuclear fuel contains extremely dangerous radioactive materials. This material is kept localized, with a very small chance of release. Waste from fossil fuelled plants is less harmful, but is not kept isolated from the environment. For example, the smokestack spreads substances that may harm health or the environment over a large area.

All power plants discard heat into the surroundings. Nuclear plants are not as efficient as modern conventional plants and produce more waste heat. In Canada, plants are sited on large bodies of cold water. The effect of this heat on the surroundings is not believed to be particularly harmful.

10.5 Summary of Key Ideas

- Nuclear power is less expensive than fossil fuel power if the nuclear plant is operated reliably at high power for long periods. The cost advantage comes from the low fuelling cost.

- Nuclear power production is less flexible than power production from a conventional plant. There are two reasons. Costs in a nuclear plant continue even when it is not producing power. Technical reasons limit the depth and frequency of power maneuvers.

- The consequences of radiation release from a nuclear plant are very serious. Extreme care is taken to run the plants safely. The hazard from a fossil fuelled plant is much less, but some harm to the environment is inevitable.

- Conventional plants produce huge volumes of relatively harmless waste. Nuclear plants produce very small quantities of highly hazardous waste, and this material remains hazardous for a very long time.

- Both conventional and nuclear plants produce large quantities of waste heat, but this waste heat is not believed to cause serious problems.

10.6 The Energy Flow Path

The introduction stated that an obvious difference between fossil fuelled and nuclear power plants is the source of heat. You will see another difference if you compare the size of the steam equipment.

A good conventional plant produces hot, high pressure (dry) steam. A CANDU reactor delivers steam at lower temperature and pressure (that is, saturated almost wet, steam). This requires larger volumes of steam flow to transfer the same amount of energy. As a result, the steam piping and the steam turbine are physically larger than similar equipment in an equivalent fossil fuelled plant

To increase steam temperature and pressure in the CANDU, the design would have to use thicker material. The tubing that carries the coolant past the fuel interferes with the fission process by absorbing neutrons that cause fission. It cannot be thicker. Also, the fuel would need to be redesigned to operate at a higher temperature.

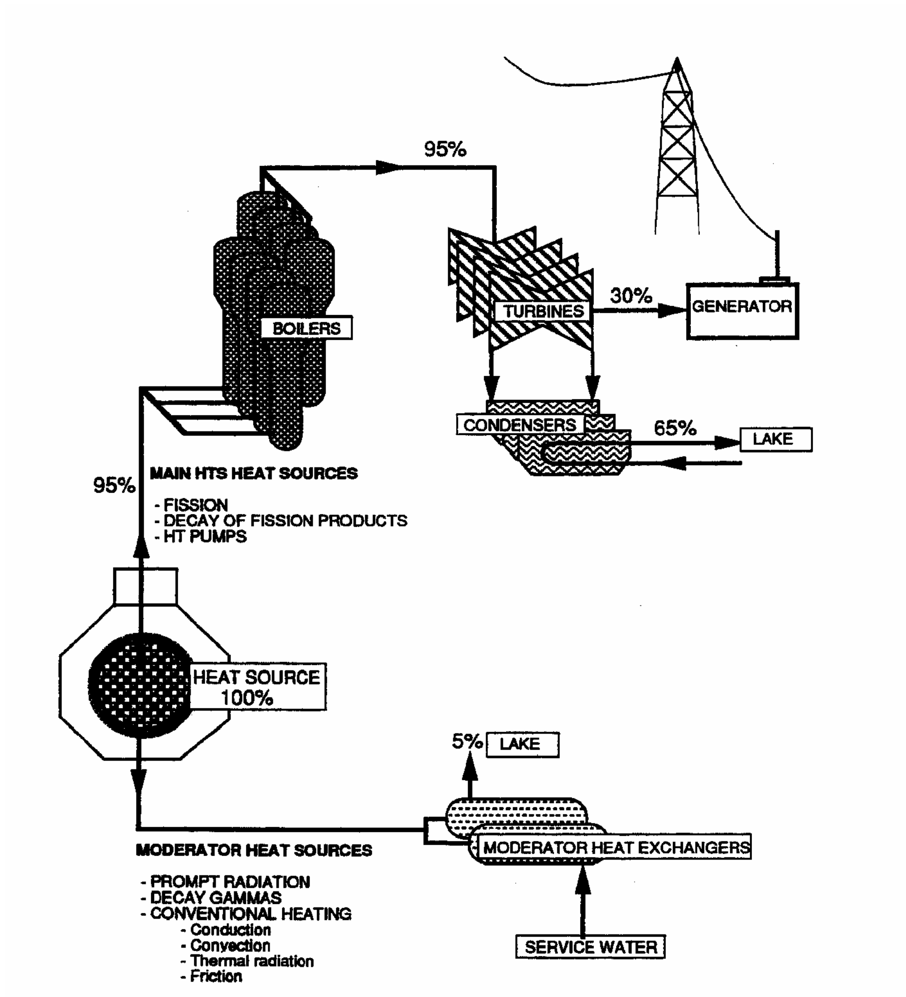


Figure 10.2
Major Energy Flow in a CANDU Power Station

Using saturated steam affects the thermal efficiency of the nuclear plant.

Equipment that turns heat energy into mechanical energy is never very efficient. In round numbers, a conventional plant throws away about 60% of the heat energy generated by burning oil or coal. A nuclear plant wastes as much as 70%. (Your automobile engine is even less efficient!)

You can see the source of this inefficiency for a steam turbine. The turbine, to operate, must have high pressure at one end and low pressure at the other. The condenser condenses the exhaust steam with cold water to maintain the pressure difference. Heat removed by the water is wasted.

The efficiency of the energy transfer depends almost completely on the temperature difference between the hot steam and the cold condensing water. With cooler steam, a higher percentage of the heat is wasted.

We now describe the energy flow path more carefully, using typical numerical values. This is useful background for understanding the purpose and operation of systems described later.

The CANDU operating license limits the total heat output from the fuel at full power. The smallest units have a license limit of almost 1700 MWth. The limit for the largest units is over 2800 MWth. The megawatt is a measurement of power, the rate of producing and transferring energy. A unit producing 2800 MWth must get rid of 2800 megajoules of energy each second. To see where this energy goes, refer to Figure 10.2 as you read the following.

The steam condensers reject about 65% of the total heat energy produced. This waste heat passes to the lake.

The turbine converts about 30% of the total heat energy produced into mechanical energy. The generator converts the mechanical energy to electric energy. The electrical grid distributes most of this energy to customers. The station uses some of this electricity to operate equipment.

About 5% of the total energy produced shows up as heat in the moderator system. This percentage includes a small amount of heat that escapes into the shielding around the core. Heat exchangers cool the equipment and transfer the heat to the lake.

In principle, the steam could be exhausted to the atmosphere. This would be even less efficient.

The various energies have names. The next few paragraphs define these and illustrate them with design numbers.

The reactor thermal power is the net heat transferred from the fuel to the coolant. A large unit might deliver 2700 MWth to the boilers. Steam pressure developed in the boilers turns the turbine that drives the electric generator.

The gross electrical power is the electric power produced by the generator. Usually about 30% of the reactor thermal power is converted to electricity; this is the thermal efficiency of the energy conversion process.

The unit thermal efficiency is defined as the ratio of the unit gross electrical power to the reactor thermal power.

The electric power used by equipment in the plant is called the station service power. Station service power takes 5% or so of the generator output. The rest of the electric power is delivered to the grid. It is called the unit net electrical power. In our example the power produced by a large unit operating at its maximum design capacity is as follows:

reactor thermal power	2700 MWth
gross electrical power	837 MWe
station service power	47 MWe
net electrical power	790 MWe
unit thermal efficiency	837/2700= 31%

At 80% of full power, the net electrical production would be 632 MWe.

This number leaves out the heat lost to the moderator. For this example, moderator heat is about 120 MWth, so total heat from the fuel is about 2820 MWth

Some station documents quote overall net efficiency. This number compares net electrical power to total power from the fuel. In the example it is 790/2820= 28%.

10.7 Summary Of Key Ideas

- There is a license limit on the total thermal power from the fuel.
- The reactor thermal power, which makes steam, is less than the total heat from the fuel. The moderator system removes about 5% of the heat.
- The turbine/generator set converts about 30% of the reactor thermal power to electricity. This is the gross electrical power.
- The ratio of gross electrical power to reactor thermal power is the unit thermal efficiency.
- The station equipment uses about 5% of the gross electrical power (the station service power). The net electrical power that remains is sold to customers to generate revenue.

10.8 Assignment

1. The reactor core includes fuel, coolant and moderator. What are these for and what are they made of in a CANDU reactor?

2. Give the most important advantage and the most important disadvantage (in your opinion) of nuclear power production compared to conventional power production. Briefly explain your selection.

3. Briefly outline how you think improved energy conservation would affect: (There is no right answer to this question).

    a) The economics of power production from nuclear and from fossil fuels;

    b) The environment

4. What is meant by:

    a) Net Electrical Power?

    b) Reactor Thermal Power?

    c) Thermal Efficiency?

11 Candu Reactor Construction

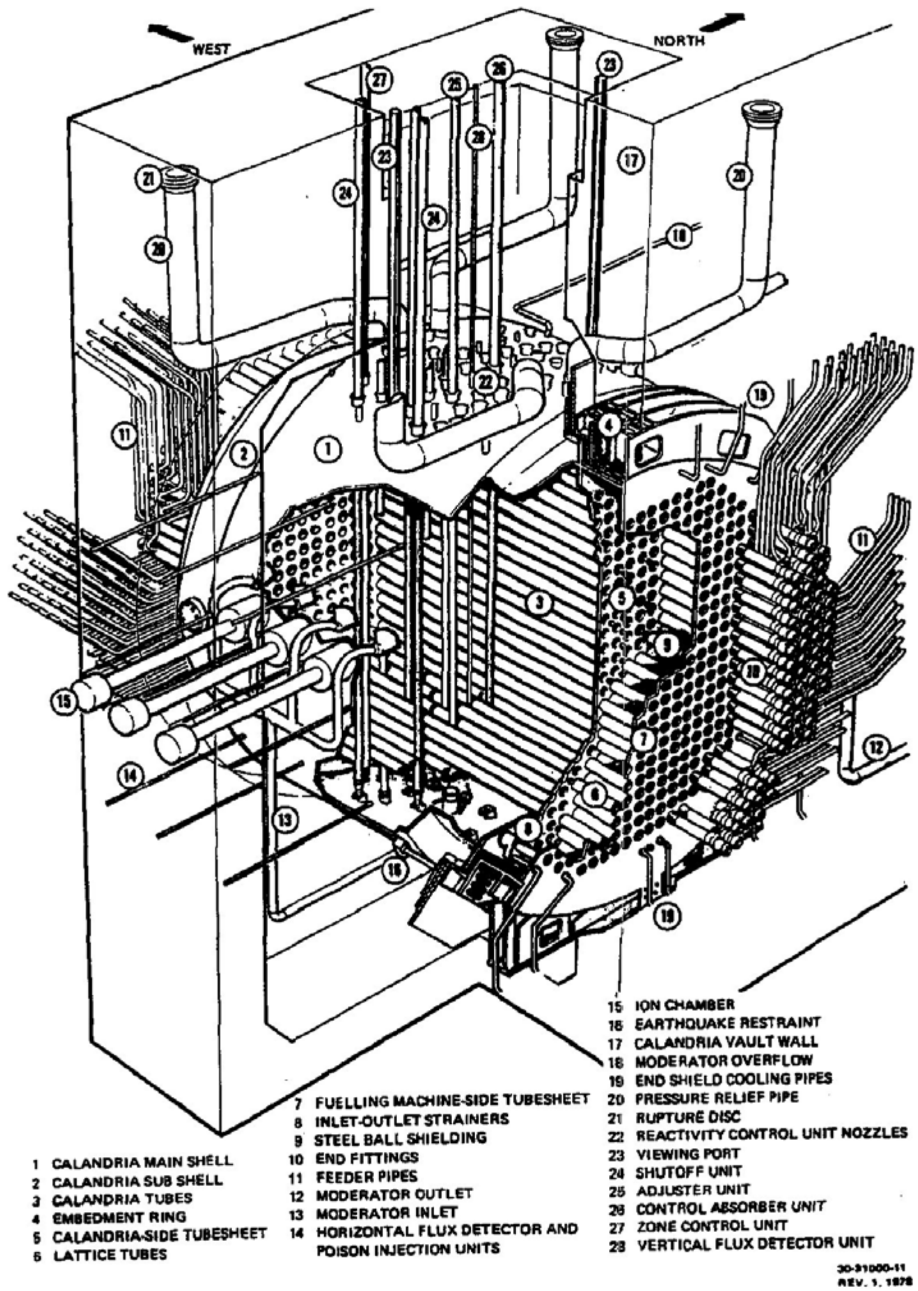


Figure 11.1
CANDU Reactor Assembly

11.1 INTRODUCTION

This module outlines some choices made by the designers of the CANDU reactor, and shows how the design came about. You need to understand the reason for particular design features so you can remember what the components are for, or locate them on a drawing.

Figure 11.1 is a detailed drawing of the Pickering B reactor. Simplified drawings on the next few pages illustrate important CANDU features. As you read this module, look back at Figure 11.1 from time to time to check the arrangement of the reactor components. Before we describe the design, we will review a few key reactor physics ideas.

The splitting (fission) of atoms in the reactor fuel produces heat. Each watt of heat takes about 30 billion fissions each second. Vast numbers of thermal neutrons bombard the fuel atoms and cause these fissions. Fissile atoms are atoms that can be split by thermal neutrons.

Each of the 30 billion fissions yields 2 or 3 fast neutrons as well as heat. The reactor designer arranges the fuel so that fast neutrons escape from it. The neutrons must slow down in the moderator (not in the fuel), and only then return to the fuel. If the moderator or structure absorbs too many neutrons in the process, the chain reaction fizzles out.

11.2 Key CANDU Components

The key parts of a reactor are the fuel, moderator and coolant. A CANDU reactor uses natural uranium dioxide fuel with heavy water moderator and heavy water coolant. Figure 11.2 shows their arrangement. Bundles of thin fuel elements allow easy escape of fast neutrons from the fuel. The fuel channels are about 30 cm apart. This distance allows the neutrons to lose most of their energy before they find their way back to the fuel.

Thermal neutrons move slowly. A reactor designed to operate using thermal neutrons is called a thermonuclear reactor. This name is often shortened to thermal reactor.

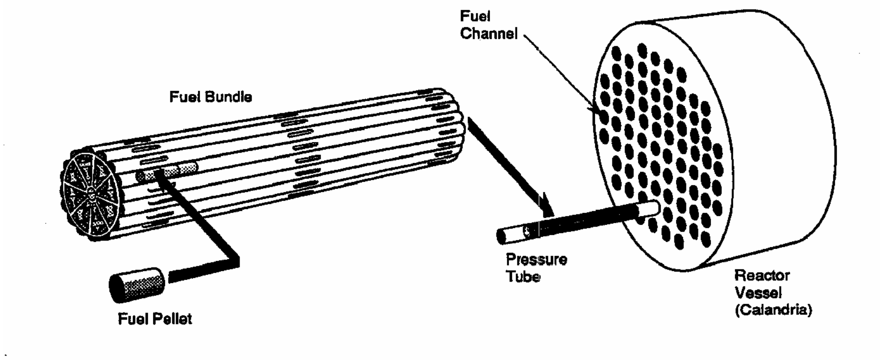


Figure 11.2
The Arrangement of Fuel, Coolant and Moderator in a CANDU Reactor

11.2.1 Fuel

Most commercial reactors, including CANDU, use uranium dioxide fuel. CANDU is unique in using natural uranium. Natural uranium is 99.3% U-238, which is not fissile, and 0.7% fissile U-235. Most reactors use enriched fuel with 2% to 4% U-235. The chance of a collision between a neutron and a fissile atom is greater in enriched fuel, so a chain reaction continues with fewer neutrons to support it.

Canadian designers produced a reactor design that did not waste neutrons, and so could use natural uranium fuel. Such a reactor is fuel efficient, and the fuel is relatively inexpensive. Also, after World War II, Canada was one of the few countries that knew how to build a nuclear bomb. By choosing a reactor design that did not need uranium enrichment, Canadian politicians showed there were no plans to make bombs.

Bombs require plutonium or highly enriched uranium. The world's first commercial reactor, Calder Hall in the U.K., made military grade plutonium as well as electricity. A compact, enriched uranium, submarine reactor was scaled up for the first U.S. power plant. CANDU was designed from the outset for commercial use only.

11.2.2 Coolant

In a light water moderated reactor, the moderator also serves as the coolant. Liquid water in contact with hot fuel becomes very hot. It must remain liquid or lose its ability to slow neutrons. The reactor must operate at high pressure to prevent the hot liquid from turning to steam. This method does not work well for a heavy water moderated reactor.

A lot of heavy water is needed to slow neutrons down. This makes a heavy water moderated reactor large. A large pressure vessel is difficult to build and very expensive. Designers of the first CANDU could not find a Canadian manufacturer who could make such a large pressure vessel.

A pressure tube reactor design solved this problem. This design separates the moderator and coolant. Look at Figure 11.2 again. Pressure tubes running horizontally through the reactor contain the fuel. High-pressure heavy water coolant passes through the pressure tube and over the fuel.

11.2.3 Moderator

The concentration of U-235 in natural uranium is low, so the number of neutrons bombarding the fuel must be high. Designing a reactor around the chosen fuel requires exceptional care to reduce neutron losses.

Canadian scientists knew that uranium dioxide fuel, using natural uranium required a D_2O moderator. Any other moderator would absorb too many neutrons. The CANadian Deuterium Uranium (CANDU) reactor was born.

A large tank with hundreds of passageways (channels) through it contains the moderator. This complicated tank is called the calandria. It is about 6 m long and 7 m across.

The calandria is not a pressure vessel. The moderator is cooled so it will not boil and pressurize the structure. Calandria rupture discs protect the calandria from overpressure. Figure 11.1 shows these discs.

Heavy water absorbs few neutrons, but is not as effective as light water in slowing them down. For the same power output, a heavy water reactor is larger than a light water moderated reactor.

Canadians, doing war research, had become experts on heavy water.

11.3 The Reactor Core Structure

A pressure tube design, separating moderator and coolant, required detailed design work to fit the parts together. You will need to refer to figures 11.3, 11.4 and 11.5 as you read the following description.

Figures 11.3 and 11.4 show the moderator and coolant separated by two tubes with a donut shaped space (annulus) between them. The calandria tubes, about six meters long, are the walls of the channels through the calandria. The pressure tubes, placed inside the calandria tubes, each contain 12 or 13 fuel bundles. The gas in the space between the tubes, the annulus gas, insulates the cool moderator from the hot heat transport system.

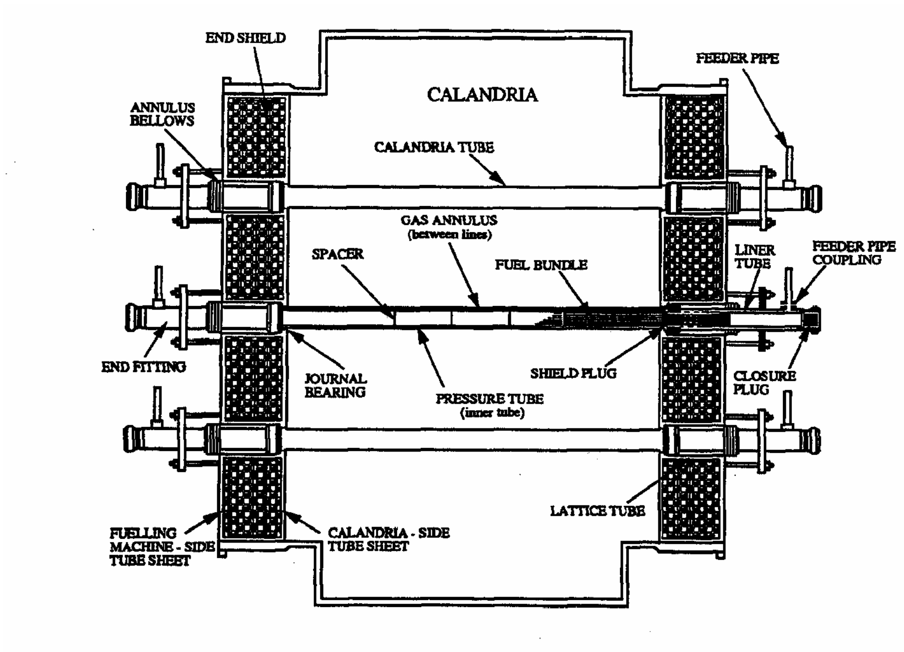


Figure 11.3
Reactor Core Schematic

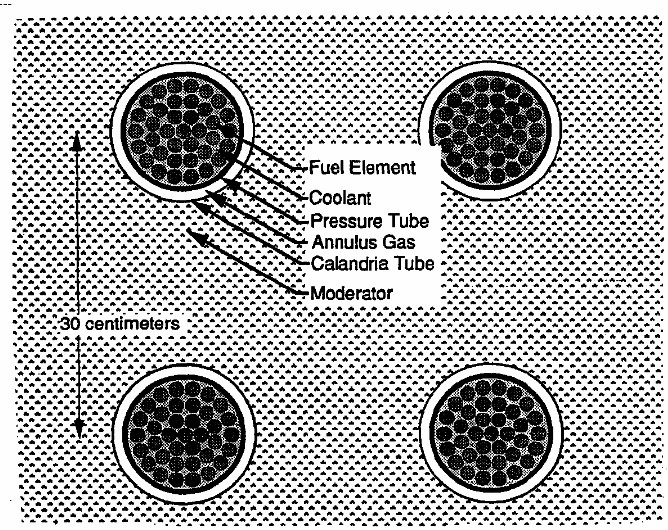


Figure 11.4
The CANDU Lattice

The walls of the calandria tubes and pressure tubes absorb few neutrons passing through them. These tubes are made of a zirconium alloy, a metal that absorbs fewer neutrons than other metals. It is used where neutron absorption must be low. The tubes in the reactor core are rather transparent to neutrons, which see the fuel surrounded by D_2O.

Figure 11.5 shows several connections. The calandria tubes are attached to the calandria-side tube sheet (the flat inside face of the calandria). A mechanical rolled joint connects the zirconium alloy tube to the stainless steel tube sheet. Similarly, a rolled joint attaches each end of a pressure tube to a stainless steel end fitting. The end fittings support the pressure tubes and allow connections to them. The rolled joints are the only direct connections to the zirconium alloy tubes.

High pressure D_2O coolant flows to and from the fuel in the pressure tubes through feeder pipes. The end fittings have feeder couplings for attaching the feeders. A removable closure plug closes each end fitting. There is a pressure tight metal disc seal fitted to each closure plug.

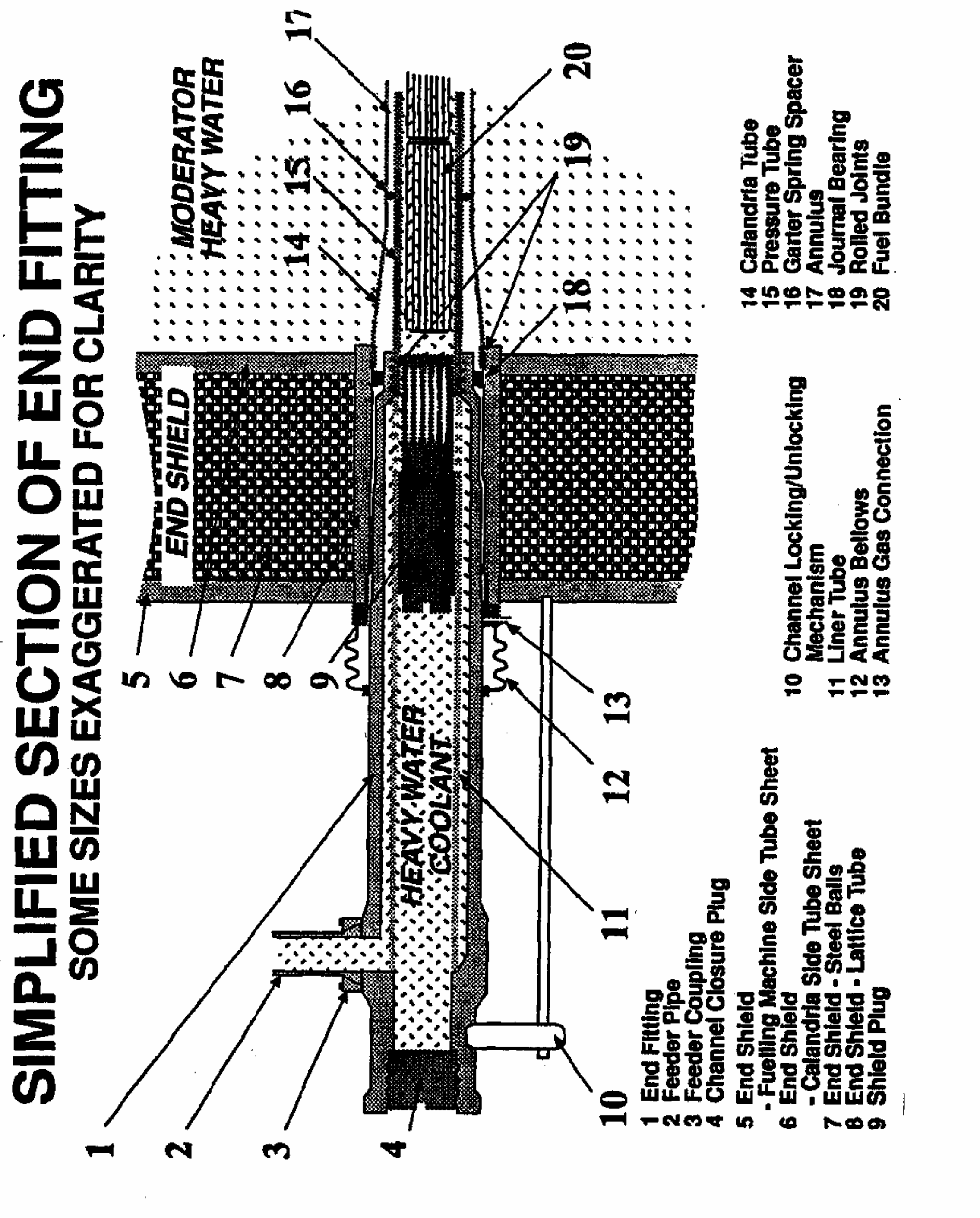


Figure 11.5
A Typical End Fitting

The end fittings allow a pressure tight connection with the remotely controlled fuelling machines. These machines insert and remove fuel during reactor operation. A fuel latch in the channel prevents the string of bundles shifting, unless an attached fuelling machine releases the latch.

The end fittings support the pressure tube at its ends. Garter spring spacers along the pressure tube keep it from sagging into contact with the calandria tubes.

Each end fitting rests on a journal bearing on which it can slide.

Normally one end of the fuel channel is clamped and cannot move, while the other end is free to move. This movement allows for thermal expansion and contraction of the pressure tube. It also accommodates pressure tube creep.

Figures 11.3 and 11.5 show a flexible seal, the annulus bellows, between the end fitting and the reactor face. This seal is flexible to allow movement of the pressure tube relative to the reactor face, as just described.

Carbon dioxide circulates through the gas annulus, entering and leaving by tubing attached to the bellows. This gas provides insulation between the moderator and pressure tubes. The annulus gas system circulates the insulating gas, allowing for control of conditions between the tubes. Detection of moisture in the gas may warn of a tube leak.

The annulus gas system has changed since the first design. At one time the annular spaces between the calandria tubes and pressure tubes were open to the air. This insulates the moderator from the pressure tube, but has a couple of disadvantages. Argon in the air becomes activated, creating a radiation hazard. Moisture and excessive oxygen in the air corrode the zirconium alloy tubes.

11.4 Summary of Key Ideas

- CANDU fuel is made from uranium dioxide using natural uranium. Remotely controlled fuelling machines insert and remove fuel bundles through end fittings on each pressure tube.

- The fuelling machine makes a leak tight seal with the end fitting. When the fuelling machine is not attached to the reactor, the fuel channel is sealed by a closure plug inserted into the end fitting.

- Heavy water coolant circulates through the pressure tubes, over the fuel. It enters and leaves by feeder pipes connected by feeder couplings to the end fitting.

- The fuel string in the pressure tube is held in place by a fuel latch in the end fitting.

- Heavy water moderator surrounds the pressure tubes containing the fuel. A carbon dioxide annulus gas insulation separates the hot pressure tube from the cool moderator.

- The annulus gas system circulates the annulus gas, providing controllable conditions for the tubes and making it possible to detect tube leaks.

- The calandria holds the moderator. It is a large horizontal cylinder, closed on the flat ends by calandria tube sheets. Each tube sheet has hundreds of openings, nearly 30 cm apart on a square lattice. Calandria tubes seal the openings in the tube sheets. They run from end to end, making open channels through the tank.

- The calandria is not a pressure vessel. It is protected from over pressure by calandria rupture disks.

- Zirconium alloy calandria tubes and pressure tubes reduce neutron absorption. Mechanical rolled joints connect the zirconium alloy tubes to stainless steel reactor components:

    a) the calandria tubes are connected to the calandria tube sheet.

    b) the pressure tubes are connected to end fittings.

11.5 Advantages And Disadvantages

Natural uranium dioxide fuel requires a heavy water moderator. A heavy water moderated reactor is fuel-efficient but large. A pressure tube design was chosen to avoid building a large pressure vessel. This design has advantages in the following three areas:

a) Low Fuelling Cost (discussed in the previous section),

b) On Power Fuelling,

c) Flexibility of Reactor Monitoring and Control.

11.5.1 On Power Fuelling

The ability to insert and remove fuel with the reactor running gives several advantages not available to batch fuelled reactors.

Power production figures can be kept high because there is no lengthy shutdown for refuelling. This, more than any other factor, accounts for the high production from newer CANDU reactors compared to all other types.

Defective fuel (fuel from which fission products can escape) can be removed as soon as it is discovered. This helps lower the radiation dose to station staff.

Detailed fuel management is possible. Fuelling can shape the power distribution across the core. Fuel burnup can be optimized.

The fuelling workload is distributed throughout the year instead of conflicting with a busy maintenance schedule during a shutdown.

Enriched uranium reactors discard fuel with U-235 concentrations higher than fresh CANDU fuel.

11.5.2 Flexibility of Reactor Monitoring and Control

The calandria houses a variety of monitoring instruments and control devices. The design and operation of these devices is simpler because they do not operate in a hot, high pressure environment. For example, neutron absorbing rods used for emergency shut down do not have to be inserted into a high pressure core.

Individual channels can be monitored for temperature and for radiation levels.

There are also some disadvantages of the pressurized heavy water reactor.

a) Heavy water is expensive.

b) The core is large and complex.

c) Heavy water absorbs few neutrons, but when it does, it produces radioactive tritium. Tritium is a significant radiation hazard, often contributing more than half of a CANDU station's radiation dose.

11.6 Summary of Key Ideas

- CANDU reactors are fuel-efficient and use relatively inexpensive fuel.

- CANDU reactors have typically had higher lifetime production figures than any other type of reactor. This is mainly because on-power fuelling allows for long running time without shutting down.

- The fuelling workload is easier to manage when it is distributed throughout the year; another advantage of on-power fuelling.

- Detailed fuel management is possible using on-power fuelling. Defective fuel can be located and removed. Fuel burnup can be optimized. The reactor power distribution can be adjusted.

- A low-pressure moderator, with individual pressure tubes, allows for flexible monitoring and control of the reactor. This is because individual channels can be monitored and because instruments and control devices in the calandria do not need to operate in a hot, high-pressure environment.

11.7 Shielding

Radiation affects work near the reactor. Some equipment is accessible while the reactor is running. You can approach other equipment only if the reactor is shut down. Equipment that cannot be approached operates by remote control.

Radiation shielding protects the operating staff from intense neutron and gamma radiation. A thick radiation shield that provides protection for normal work near the operating reactor is called a biological shield. Shielding which provides adequate protection only for a shut down reactor is called shutdown shielding. Shielding that absorbs heat is called thermal shielding.

11.7.1 End Shields

No work is allowed at the faces (flat ends) of a running reactor. An end shield provides shutdown shielding only. Thickness of shielding at the reactor face is about one meter. Space is limited by the need to fuel the reactor. Figures 11.1 and 11.5 show the end shield.

The inner and outer walls of the end shield are, respectively, the calandria tube sheet and the fuelling machine side tube sheet. The zirconium alloy calandria tubes stop at the calandria tube sheet. The channels through the end shield are enclosed by steel lattice tubes. The lattice tubes are part of the end shield, as the calandria tubes are part of the calandria.

The end shields, as well as providing shutdown shielding, support the journal bearings. The journal bearings rest on the lattice tubes and, in turn, support the end fittings. Recall that the end fittings support the pressure tubes that hold the fuel. The small contact region of the journal bearing limits heat transfer from the hot fuel channel to the end shield.

11.7.2 The Reactor Face

Steel balls fill the end shields. Figures 11.1 and 11.5 show the balls between the tube sheets. Ordinary water cools these balls. (The Pickering A shield has water-cooled steel slabs instead of balls). Cooling removes heat deposited by neutrons and gamma rays. The small area of contact also conducts some heat from the hot fuel channels into the end shields.

Removable panels of thermal insulation are placed over the end shields. The end fittings stick through the insulation for fuelling machine access.

The end shields would be useless if radiation could stream from the ends of the fuel channels. The D_2O in the end fitting does not provide enough shielding. Inside the fuel channel, between the closure plug and the fuel is a stainless steel shield plug that completes the radiation shielding by plugging the holes in the end shield. The length of the shield plug is about the same as the thickness of the end shield.

The shield plug also affects the coolant flow. Inside the end fitting is a liner tube. Flow from the feeder coupling to the pressure tube is around the outside of the liner tube. Holes in the liner tube allow the coolant into the fuel channel. The shield plug shape directs this flow smoothly into the channel.

11.7.3 Radial Shielding

The end shields are similar for all CANDU reactors. Radial shielding is different from reactor to reactor.

Figure 11.6 shows the arrangement at Bruce and Darlington. A steel, light water filled shield tank surrounds the calandria. This tank, like the end shields, provides shutdown shielding from nuclear radiation. It also acts as a thermal shield absorbing heat from thermal radiation.

The reactor vault, the room that encloses the reactor, has thick concrete walls. This, with the shield tank and end shield, provides full biological shielding on all sides of the reactor.

Pickering A was built without a surrounding water filled tank. Thick concrete vault walls provide both biological and thermal shielding.

Cooling pipes, embedded in the walls, protect the concrete from overheating.

The extension at the top of the tank provides biological shielding on top of the reactor.

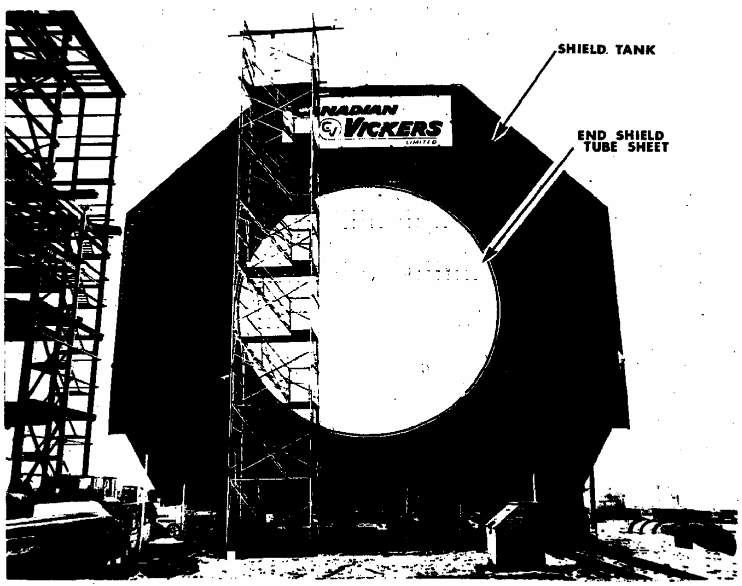


Figure 11.6
Calandria with Shield Tank (Bruce A)

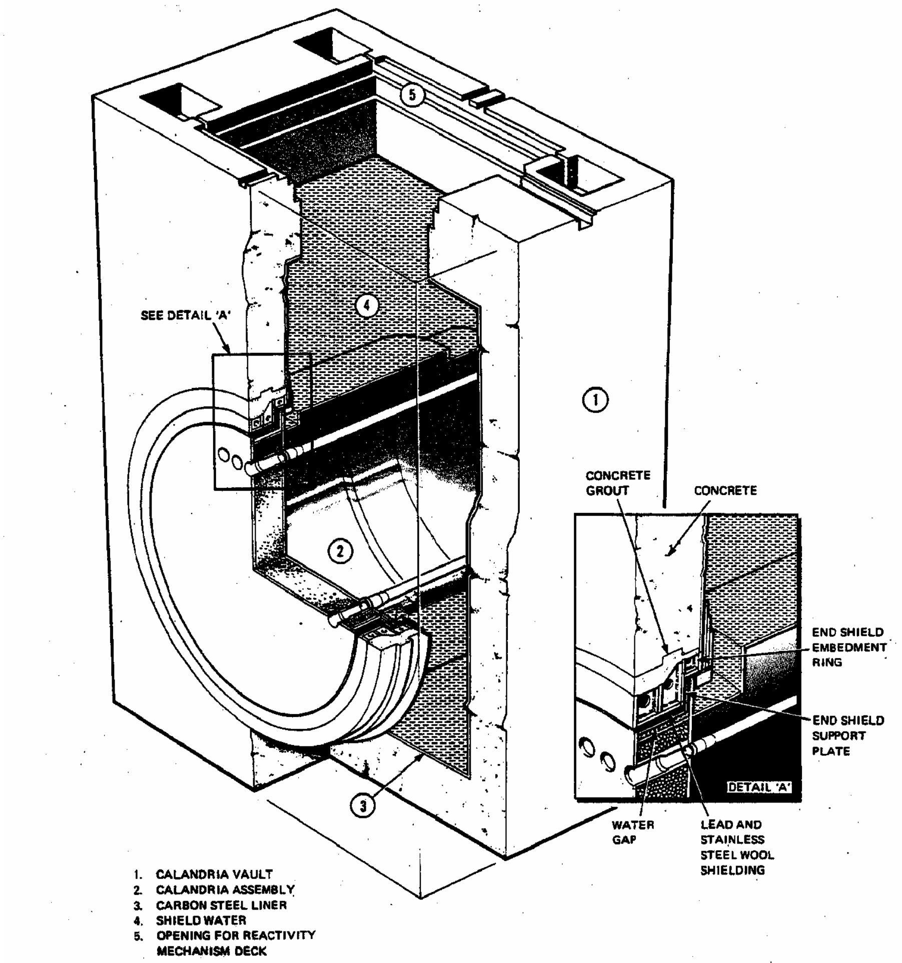


Figure 11.7
Pickering B Calandria Vault

Figure 11.7 shows the Pickering B and CANDU 600 design. The calandria vault (sometimes also called the reactor vault) supports the calandria. This vault is a steel lined concrete tank filled with light water. It combines thermal and biological radial shielding.

In this design, fuelling machine vaults at each end of the reactor house the fuelling machines. The biological shield at the reactor face is the combined shielding of the end shield and the walls of fuelling machine vault.

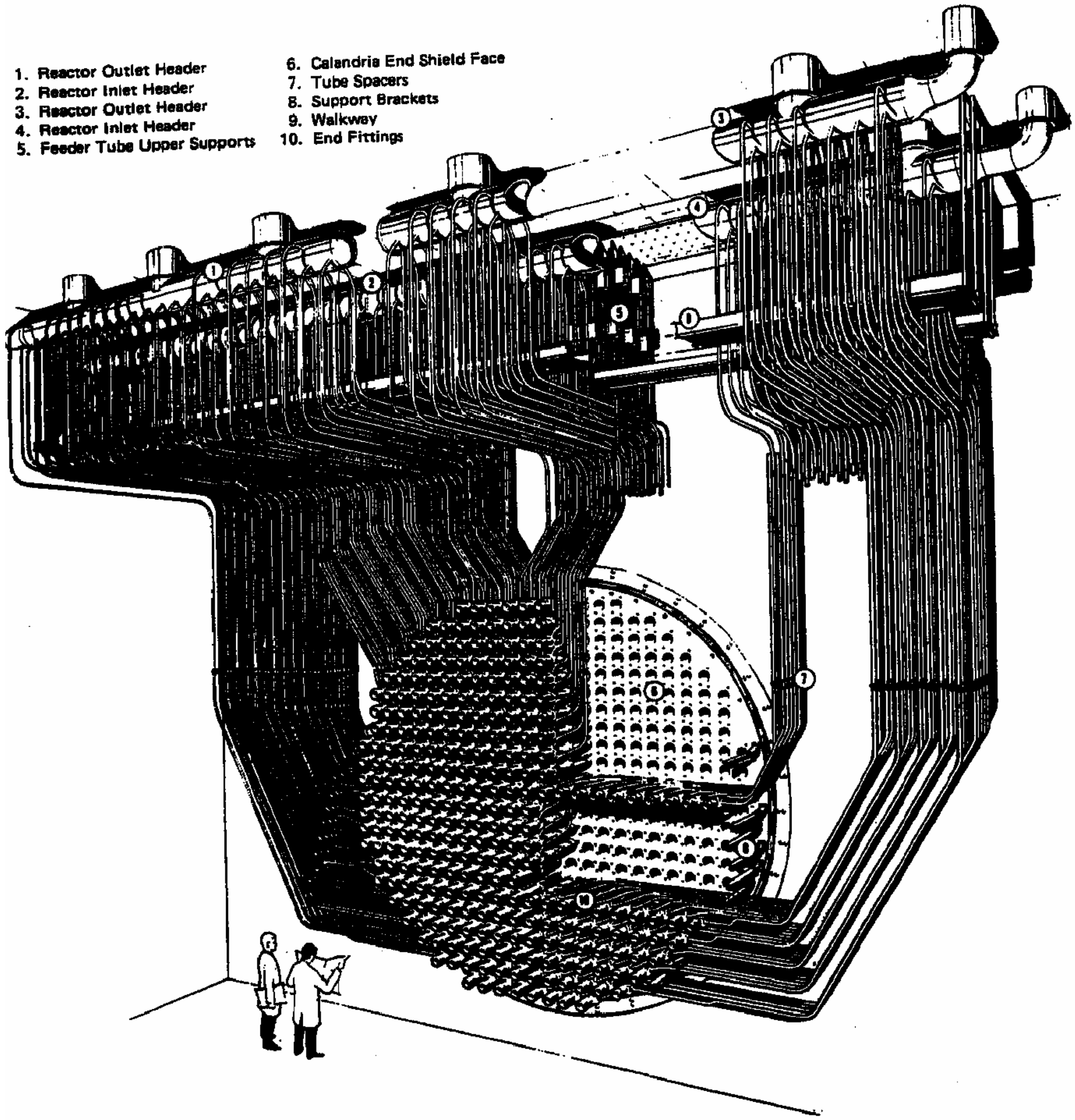


Figure 11.8
Feeders and Headers

11.8 Summary Of The Key Ideas

- All CANDU reactors have end shields. These, with the shield plugs in the fuel channels, allow work at the face of the shut down reactor.

- Some CANDU reactors have shield tanks. These allow work inside the reactor vault of a shut down reactor.

- Some CANDU reactors have a water-filled calandria vault. These allow work around the reactor (but not at the face) whether or not the reactor is shut down.

- The end shields, shield tank and calandria vault together provide thermal shielding and radiation shielding.

- All CANDU reactors provide full biological shielding by thick concrete walls. Each reactor design arranges this shielding differently. Normal work goes on outside the biological shielding.

11.9 Assignment

1. Describe the advantages of a pressure tube reactor compared to a pressure vessel type.

2. What is the purpose of the end shields?

3. Explain why an annulus gas system is needed.

4. Distinguish between a shield plug and a closure plug.

5. What is the purpose of:
   a) an end fitting?
   b) a calandria rupture disc?
   c) fuelling machines?
   d) a fuel latch

6. Describe the shape, size and structure of the calandria.

7. Where is the fuel in a CANDU core and how is it cooled?

Optional Exercise

8. Use coloured pencils to highlight major components in Figure 11.1. Identify as many of the components listed in the objectives as you can find. Indicate the location of the components that are not shown. (Try to locate components without looking at the labels below the figure, but check before you highlight them.)

12 Moderator And Moderator System

12.1 Introduction

Nuclear fuel produces heat by fission. In the fission process, fissile atoms split after absorbing slow neutrons. This releases fast neutrons and generates heat. Fast neutrons are not very good at causing fissions. The moderator slows the fast neutrons so they will cause more fissions. It must do this without absorbing too many, or the chain reaction will stop.

Most neutrons, on collision with a heavy water molecule, hit and bounce. These collisions transfer energy from fast moving neutrons to the heavy water. This slows the neutrons and heats the heavy water. The moderator system cools the heavy water by circulating it through heat exchangers. The last section of this module describes the circulation system.

Moderator water contains mostly heavy water with a very small amount of ordinary water. Fewer than 2% of the fission neutrons are absorbed in the moderator. The small H_2O impurity captures about half of these. The next section explains the importance of keeping the light water impurity small, and describes how this is done.

Occasionally a deuterium nucleus in D_2O captures a neutron and becomes tritium. Tritium is a radiation hazard. Neutrons may also interact with the oxygen nuclei, producing other radiation hazards. This module discusses these hazards.

The water molecules are broken into fragments by the energetic collisions. The next module describes the effects of this on moderator operation.

12.2 D_2O Isotopic

A sample of moderator water is typically about 99.8% by weight D_2O and 0.2% D_2O. We say it has an isotopic of 99.8%. The exact definition is:

$$\text{Sample } D_2O \text{ Isotopic} = \frac{\text{Mass of } D_2O \text{ In Sample}}{\text{Mass } D_2O + \text{Mass } H_2O \text{ In Sample}} \times 100\%$$

The number of neutrons absorbed is sensitive to changes in isotopic. An impurity of less than 0.2% light water will absorb half the neutrons absorbed by the moderator.

If the isotopic drops a few tenths of one per cent, extra fuel can offset the neutron losses. This increases fuel costs. A decrease in isotopic from 99.8% to 99.7% increases fuel costs between a half million dollars and one million dollars a year. Similarly, upgrading the moderator by 0.1% saves this amount of money.

If the isotopic drops below 99.5%, the reactor may stop running. A D_20 upgrader at the station upgrades low isotopic heavy water to 99.9% or higher. The isotopic of the upgrader product depends on the amount and isotopic of the heavy water it must process, and the time available.

Careless handling of heavy water is expensive. Your job may involve transferring heavy water, by adjusting valves or emptying drums. If so, always think twice. A wrong transfer can mix low and high isotopic. The result could be extra upgrading and fuel costs, or, expensive heavy water could go down the drain. Even worse, you might cause a power outage.

12.3 Summary Of The Key Ideas

- The heavy water isotopic is the mass % of D_2O. Moderator isotopic is usually near 99.8%. The other 0.2% is ordinary water.

- The isotopic requirement for the coolant is less rigid. The coolant is exposed to far fewer thermal neutron collisions, so its isotopic has less effect on neutron absorption.

- A small decrease in isotopic increases the fuel cost. A large drop in the isotopic causes the reactor to shut down.

- A D_2O upgrader maintains high isotopic. Mistakes that downgrade heavy water are costly.

12.4 Radiation Hazards

Significant numbers of neutrons are present in the core when reactor power is a few percent of full power or higher. These neutrons interact with D_2O to produce radioactive nitrogen-16, oxygen-19 and tritium.

Tritium (H-3) forms when deuterium absorbs a neutron. Neutrons interact with naturally occurring oxygen isotopes to produce N-16 and O-19. These isotopes affect work you do in the plant.

N-16 and O-19 emit very high energy gamma rays along with energetic beta particles. The beta particles do not penetrate pipe walls, but the penetrating gamma rays are a hazard around equipment containing these isotopes.

N-16 and O-19 have short half-lives. These isotopes decay to harmless levels a few minutes after a reactor shut down, allowing access to the equipment. Radiation from these isotopes returns to unsafe levels just seconds after restarting.

Systems that handle water circulating from the reactor core can be approached only with the reactor shutdown. Very heavy shielding around this equipment may allow work on nearby equipment with the reactor running. D_2O that leaks from a reactor at power exposes anyone nearby to beta and gamma radiation and tritium.

A few systems require attention when the reactor is at power. When required, a delay tank between the core and the equipment delays the flow of D_2O. This gives time for N-16 and O-19 to decay before they reach accessible areas.

The naturally occurring oxygen isotopes are O-16 (99.76%). O-18 (0.2%) and O-17 (0.04%). O-19 comes from neutron absorption in an O-18 nucleus. N-16 comes from an (n, p) reaction with O-16. (Some N-17 is produced by an (n, p) reaction with O-17 wad some C-14 is produced by an (n, p) reaction with O-17).

O-19 has a half-life of about 27 seconds. N-16 has a half-life of 7.1 seconds.

Tritium (H-3) has a half-life of 12.3 years. Its concentration builds gradually in moderator and heat transport D_2O. It hardly decreases on a shut down.

Tritium emits a low energy beta particle and no gamma ray. Normal radiation instruments cannot detect tritium. A person with Radiation Protection Training (RPT) qualifications can check most workplace radiation hazards

The low energy beta particle from tritium is not an external radiation hazard. Tritium, nevertheless, is a serious internal radiation hazard. Tritiated water vapour enters the body through the lungs and through the skin. It then disperses to all parts of the body, just as normal water does. Body tissues and organs have no dead layer of skin to protect them.

Moderator water has the highest tritium concentration in the plant Tritium escapes when the system is open for maintenance. Little escapes otherwise because the calandria is not pressurized, and there are few leakage points. In older stations, a combination of high tritium concentration and small leaks can, nevertheless, produce a significant tritium hazard from the moderator system.

Tritium also builds up in the heat transport system heavy water. This water is hot and pressurized, and the system has many more possible leak points. During normal operation the coolant contributes more tritium to the station than does the moderator water.

Neutrons produce less tritium in the coolant than in the moderator because:

a) Moderator water spends most of its time exposed to neutrons in the reactor core. The heat transport heavy water spends less than 5% of its time passing through the core.

b) The concentration of thermal neutrons is higher in the moderator than in the coolant.

Station staff wear positive pressure plastic suits with supplied breathing air to work in atmospheres containing tritium. These suits are required even when a small leak or spill of tritiated D_2O occurs. Full protection is needed any time maintenance crews open a D_2O system.

In the future it will be possible to control the tritium problem. In 1990, a tritium removal facility started operating at the Darlington site. It is designed to remove 99.5% of the tritium in the heavy water it processes. This water with a very low tritium concentration is put back into an operating reactor moderator diluting the tritium in the moderator The moderator in an operating reactor typically contains 14 ci/l if tritium.

12.5 Summary Of The Key Ideas

- Neutron interactions in the moderator produce the radioactive isotopes H-3 (tritium), N-16 and O-19.

- The radiation hazard from tritium is independent of reactor power. The tritium beta particles do internal biological damage.

- N-16 and O-19 produce intense gamma radiation, limiting access to equipment in a running reactor. They disappear after a shutdown.

- Heavy water may leak in normal operation or when a system is open for maintenance. This produces dangerous levels of tritium in the atmosphere. Protective equipment allows work where there is tritium.

- Moderator D_2O has the highest tritium content in the plant No other system exposes the D_2O to so many neutrons. Bombardment of the moderator is continuous and the thermal neutron concentration is higher there than anywhere else.

12.6 The Main Moderator System

The main moderator system, sometimes called the moderator circulating system, has one main purpose. It maintains a constant moderator temperature in the calandria. The moderator temperature is, typically, 60ºC to 80ºC.

The circulation system also supplies D_2O flow to several auxiliary systems.

If the tritium concentration is very low it may be possible to do short jobs wearing an appropriate air filter, or an air supply but no plastic suit.

12.6.1 Moderator Heat Sources

Previously it was pointed out that the overall efficiency of a plant is about 30%. The steam condensers reject about 65% of the heat produced and the moderator heat exchangers discard about 5%. Even 5% of the total is a large amount of energy. If moderator heat removal stops, the moderator in a reactor at full power starts boiling in just a few minutes. Even a shut down reactor causes significant moderator heating.

At full reactor power, the sources of moderator heat are:

a) The prompt radiation from fission (neutrons and gamma rays) produce 70% to 80% of the heat in the moderator. The neutrons typically contribute more than half of this.

The moderator absorbs energy slowing the fast neutrons. The prompt gamma rays that accompany fission deposit energy in the moderator water and in the shielding. They also deposit energy in structures, (for example, the calandria tubes) which the moderator cools.

This heat source disappears when the fission process stops.

b) Gamma rays from fission product decay and from decay of activation products in reactor components produce 15% to 25% of the heat in the moderator. This heat decreases slowly after a reactor shut down.

c) Conventional heating (conduction, convection, thermal radiation and friction) accounts for about 3% to 5% of moderator heating.

The annulus gas does not insulate the hot pressure tube perfectly.

Conduction and convection transfer some heat through the annulus gas and heat radiation transfers some heat across it. The moderator pumps, when running, also produce heat by fluid friction.

The operator can reduce the conventional heating of the moderator after a reactor shut down by cooling the heat transport system.

Each neutron loses about 2 MeV of energy and each fission creates, on average, 2.5 neutrons. This works out to about 2.5% of the heat produced by one fission.

Heat may be conducted through the calandria tube sheet in or out of the moderator, depending on the temperature difference between the moderator and the end shield.

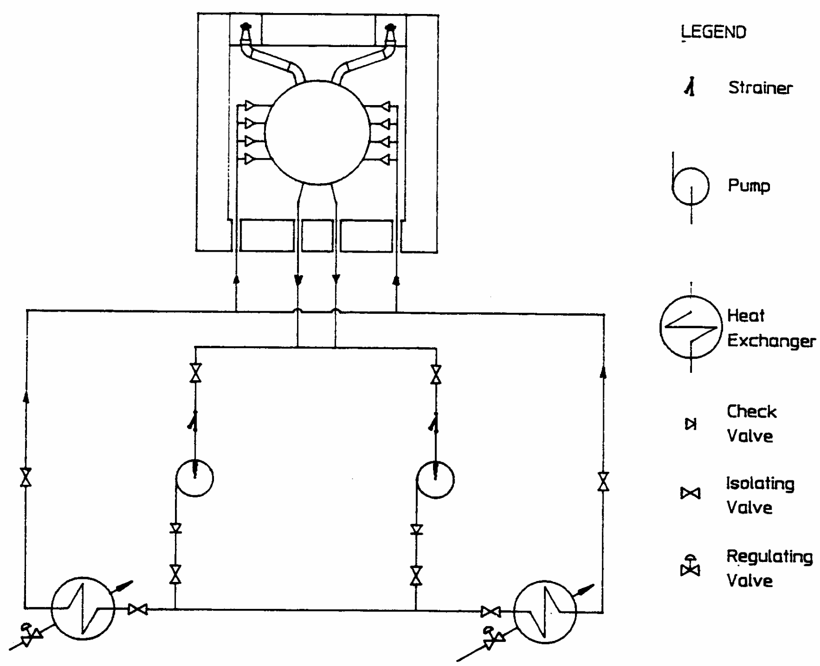


Figure 12.1
Main Moderator Circulating System (Typical)

Safe plant operation requires moderator heat removal to be available always. A review of the heat sources shows there is a lot of heat, even with the reactor shut down. All plants provide some sort of backup cooling arrangement to remove heat if the normal equipment is not available.

The backup cooling equipment must be large enough to handle the reduced heat output after a shutdown. In some plants there is a smaller auxiliary system with pumps, piping and heat exchangers. Other plants provide standby pumps or small, backup motors for the main pumps.

12.7 The Moderator Circulation System

Figure 12.1 shows the arrangement of the main moderator circulation system.

Moderator pumps take suction from outlets at the bottom of the calandria. They return cooled D_2O through inlets on the sides of the calandria. Heat exchangers downstream from the pumps transfer heat to the service water system, a light water cooling system.

The flow of service water through the heat exchangers controls temperature. When the moderator outlet temperature is high, the moderator temperature control valves in the service water piping open further. This increases the flow of cooling water, removing more heat. As temperature drops, the valves close in, reducing heat removal.

Two calandria outlets feed the pumps through a common pipe, the suction header. A balance header connects the pump discharge outlets. These connections provide common suction and discharge conditions for each pump, helping balance the flow through the heat exchangers.

The piping arrangement also allows heat removal to continue after isolation of a single pump or heat exchanger for maintenance. Notice the check valves (to prevent reverse flow through a failed pump) and the isolation valves (for maintenance).

Heat exchangers and pumps are equipped with drain valves to allow maintenance. In some plants the service water temperature can drop below 4ºC, the freezing point of heavy water. Damage can occur to a heat exchanger when it is isolated for maintenance if the cold cooling water is not drained.

Some stations have two banks of smaller pumps.

12.8 Summary of Key Ideas

- The main moderator system provides moderator temperature control. It also provides flow through moderator auxiliary circuits.

- Heat enters the moderator from three sources: prompt and delayed nuclear radiation and conventional heat transfer. Moderator heat is typically 5% or so of the gross heat produced by the reactor.

- Prompt radiation from fission produces about 75% of moderator heating. Over half of this results from the moderator doing its job: slowing down fast neutrons. Absorption of prompt gamma radiation accounts for the rest.

- Heating by decay gamma radiation comes mostly from decay of fission products in the fuel.

- Conventional heating of the moderator is less than 5% of the total moderator heat. It comes mainly from the hot pressure tubes.

- The main moderator circulating system removes heat from the moderator. Pumps draw hot D_2O from the bottom of the calandria and return it through heat exchangers that remove heat.

- Because there is a lot of heat produced in the moderator, even with the reactor shut down, heat removal from the moderator must continue at all times. Auxiliary equipment is available for heat removal if the normal equipment is unavailable.

- Temperature control valves vary service water flow through the heat exchangers to control moderator temperature. This adjusts cooling flow to match heat production.

12.9 Assignment

1. Define isotopic, and give a typical value for moderator isotopic.

2. What is the problem if the moderator isotopic decreases by:

   a) 0.1%?

   b) 0.5%?

3. What is done with low isotopic heavy water?

4. List the three main radiation hazards associated with the moderator system. Describe how each affects maintenance work.

5. Explain the following statements:

   a) Tritium production is higher in the moderator than in the coolant.

   b) Tritium from the heat transport coolant contributes more to the station staff radiation dose than tritium from the moderator.

6. What purpose does the moderator circulation system provide in addition to removing heat from the moderator water?

7. What are the main moderator heat sources in a shut down reactor?

8. a) What is the biggest moderator heat source if the reactor is at 50% full power?

   c) What happens to the moderator temperature control valve if reactor power increases?

9. Why is it necessary to have backup cooling for the main moderator system?

10. Label the two main pumps, the two heat exchangers and both temperature control valves in Figure 12.1.

13 Moderator Auxiliary Systems

13.1 Introduction

The moderator must slow neutrons and not absorb them. We previously examined moderator properties related to this function. This module describes several systems that support moderator system operation.

The last section described the heating caused by fast neutrons slowing down. Temperature changes cause thermal expansion and contraction of the heavy water. Expansion in a moderator system full of liquid could raise the calandria pressure, which is not a pressure vessel. A gas cushion, the moderator cover gas, protects the calandria structure from high pressure. This module describes the moderator cover gas system.

Fast neutrons split the water molecules that slow them down. Gamma rays also fragment D_2O. The gases D_2 and O_2 form from the fragments. The cover gas system removes these gases from the calandria so they will not collect and explode. Equipment in the cover gas system combines the gas molecules, making them into heavy water again.

The cover gas system keeps the pressure a little higher than the surrounding atmosphere. This prevents air from leaking into the system and contaminating it. The moderator purification system removes chemical contaminants and corrosion products from the moderator water.

Air is 78% nitrogen (N_2), 21% oxygen (O_2) and 1% argon (Ar) with traces of other things. O_2 is corrosive. N_2 makes the moderator acidic and Ar becomes radioactive.

Because the system is above atmospheric pressure, leaks that occur are outward. The D_20 collection system collects and returns D_2O that leaks from moderator equipment

It has already been stressed how important high isotopic is to keep absorption low. Reactor regulation sometimes needs changes in neutron absorption. The liquid poison addition system adds neutron-absorbing chemicals and the moderator purification system removes them.

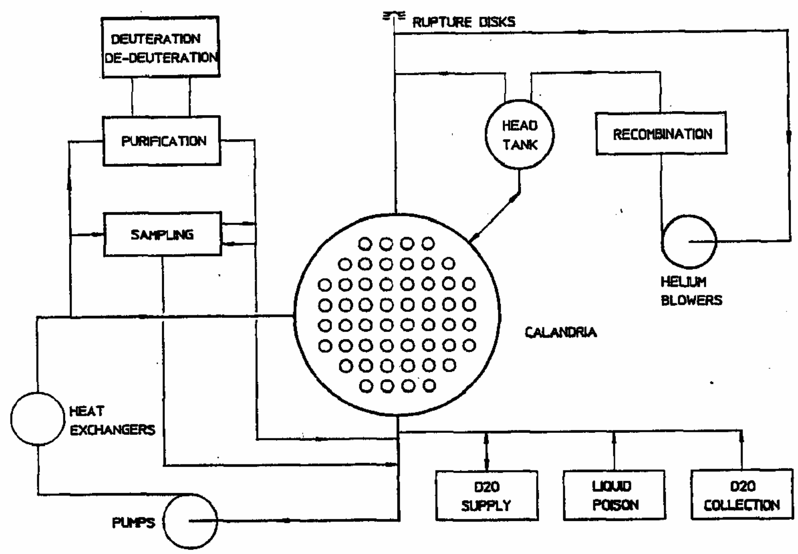


Figure 13.1
Moderator and Auxiliary Systems

Figure 13.1 shows the connections between the auxiliary systems and the main system. Notice how moderator pump pressure provides flow through the purification system.

Holes in heat exchanger tubes cause D_2O and tritium leaks into the service water. Finding and fixing leaks costs less than downgrading moderator water with service water.

Figure 13.1 is not a realistic drawing. Systems differ from station to station. For example, some stations connect the purification system downstream from the heat exchangers, taking advantage of the cool D_2O they provide. Other stations connect the purification system at the pump discharge to get better flow, and then provide a separate purification system heat exchanger.

13.2 Summary Of The Key Ideas

In a moderator system full of heavy water, that is, without a cover gas, thermal expansion could raise the pressure and damage the calandria.

Cover gas above atmospheric pressure allows heavy water and helium to leak out. It stops air from leaking in to contaminate the system.

Fast neutrons and gamma rays split water molecules. This produces hydrogen D_2 and oxygen gases that could collect and explode.

Reactor control uses neutron absorbing chemicals added to and removed from the moderator water as needed.

The main moderator pumps supply purification flow.

13.3 The Moderator Cover Gas System

13.3.1 Purposes

Figure 13.2 and the upper right hand corner of Figure 13.1 show the cover gas system. Figure 13.3 shows the layout for a system with a dump tank.

Dropping the moderator out of the reactor core stops the fission chain reaction.

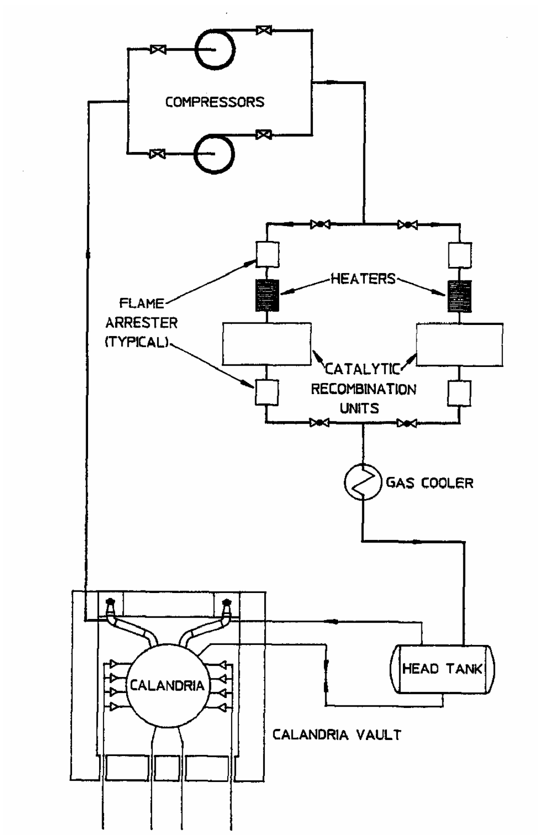


Figure 13.2
Moderator Cover Gas System for a Reactor Without Moderator Dump

The moderator cover gas system fulfils several functions.

The cover gas system controls pressure in the calandria.

The cover gas keeps a cushion above the moderator. This allows for expansion and contraction of the heavy water. It also absorbs transient high pressure caused by the liquid injection shutdown system.

Recall that rupture disks also protect the calandria from high pressure. Figure 13.1 shows these.

The cover gas provides a non-corrosive, non-radioactive atmosphere in parts of the system not filled with water. The cover gas is helium.

It is chemically inert, does not break down when radiation bombards it and neutrons cannot activate it.

Helium is, for a gas, a good heat conductor. It cools components not cooled by moderator water. The cover gas system removes D_2O and O_2 gases from the calandria.

Gamma rays and fast neutrons fragment water molecules, a process called radiolysis.

D_2O (RADIATION INDUCED)$\rightarrow D_2 + O_2$

The mixture of hydrogen and oxygen becomes an explosion hazard as the gases collect. Oxygen corrodes system components.

13.3.2 Description

Figures 13.2 and 13.3 show several common features. We will describe these first, and then discuss additional functions for the moderator cover gas system for reactors with moderator dump.

The helium compressors keep the pressure in the gas space above the calandria at about 110 kPa(a). The pressure is high enough to keep air out at the lowest pressure point of the system, the compressor suction.

The compressors also circulate the cover gas through the recombination unit.

Do not confuse the liquid poison injection system, an automatic shutdown system, with the liquid poison manual addition system described in this module.

13.3.3 The Recombination Unit

The recombination unit takes the radiolysis products, deuterium and oxygen gases, and combines them to make D_2O. The recombiner uses a catalyst that promotes a controlled chemical reaction at low levels of the reactants. This keeps the concentration of D_2 and O_2 in the system low enough that they cannot explode.

The chemical reaction between hydrogen and oxygen produces heat. This keeps the catalyst hot and dry. A small heat exchanger cools the hot gas.

When the system is not used, the catalyst becomes wet and will not work. On restarting, the heater at the inlet to the recombiner warms the cover gas, drying the catalyst.

Flame arrestors on the inlet and outlet prevent propagation of flames that might arise in the recombination unit.

Small lines can take a cover gas sample upstream or downstream from the recombination unit. The sample passes through a gas chromatograph.

The gas chromatograph finds the concentrations of O_2, D_2 and N_2. (N_2 shows there is air leaking into the system). If the deuterium concentration is high, the operator must act and shut down the reactor.

Oxygen is corrosive and some of it will combine with other elements in the system. There may not be enough oxygen in the cover gas to combine with the D_2. Oxygen addition lines introduce oxygen to the recombination units, if required.

Some helium will leak in normal operation. High-pressure cylinders supply makeup helium through a helium addition line connected to piping into the head tank.

Normally the concentration of reactants will be low and there will be no flame. Operating conditions could increase the concentration unexpectedly, however.

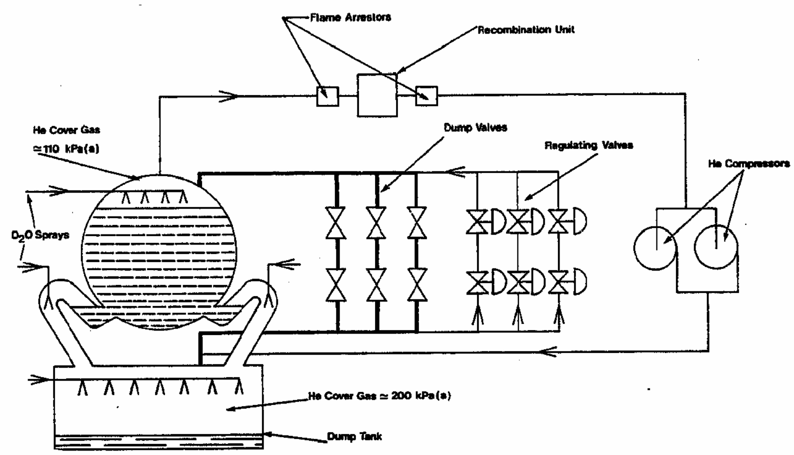


Figure 13.3
Cover Gas System for a Reactor with Dump Tank

At Pickering A, the cover gas system is part of the moderator dump emergency shutdown system.

A reactor malfunction may require a rapid shut down. With moderator dump, the cover gas system dumps the moderator from the calandria, shutting down the reactor. Future sections describe reactor protection by shutdown systems.

Large liquid ring compressors apply a pressure of about 200 kPa(a) to the dump tank. The regulating valves adjust the pressure in the gas space at the top of the calandria. A differential pressure of about 100 kPa supports the water in the calandria. The gooseneck design of the dump ports allows the dump tank pressure to prevent the water from draining into the dump tank.

Pickering A uses 2 x 100% (one operating and one standby) compressors. They take their seal water from the moderator, as light water would downgrade the D_2O.

Six dump valves trigger a reactor shut down. These valves, closed during normal operation, open quickly in an emergency. This equalizes the pressure in the calandria and dump tank. The moderator falls through the dump ports into the dump tank, shutting down the reactor.

At Pickering A, the cover gas system helps regulate reactor power.

The cover gas system at Pickering A maintains and regulates the moderator level in the calandria. Lowering the level increases neutron leakage from the core, and this reduces power.

Six regulating valves control moderator level by adjusting pressure in the dump tank. The height of heavy water that can be supported depends on the dump tank pressure. To lower moderator level, the valves go to a more open position. This lowers pressure in the dump tank. To raise moderator level the valves go to a more closed position.

13.4 Summary Of The Key Ideas

- The cover gas acts as a cushion, absorbing high pressure that could damage the calandria. Helium compressors keep the pressure slightly above atmospheric pressure.

- The helium cover gas is an inert atmosphere, neither radioactive nor corrosive, in parts of the system not filled with water.

- The cover gas removes O_2 and D_2 from the calandria and carries them to the recombination unit. The compressors circulate the cover gas.

- The recombination unit safely converts O_2 and D_2 to D_2O. The recombination usually produces a lot of heat. Flame arrestors prevent propagation of flames that might arise in the recombination unit. A heat exchanger cools the hot gases as they leave the heat recombiner.

- Dampness can prevent the catalyst from working. If the recombiner catalyst becomes damp through lack of use, an inlet heater warms the gases, drying the catalyst.

- A head tank helps maintain a full calandria in the system without dump. The cover gas fills ducts above the calandria. Figure 13.2 shows head tank and ducts.

- A gas chromatograph samples the cover gas for O_2, D_2 and D_2O. If there is not enough O_2 to combine with the available D_2, oxygen can be added through oxygen addition lines.

- Helium leaks from the system, high-pressure He cylinders replace it.

- At Pickering A, the cover gas system is part of the reactor control systems. This includes power regulation and emergency shut down.

13.5 Moderator Purification System

13.5.1 Purpose

Figure 13.4 shows the purification system. The purification system has two tasks. First, it must remove insoluble and soluble corrosion products and other impurities from the moderator water. Secondly it must remove unwanted neutron absorbing chemicals (called poisons), used for reactor regulation.

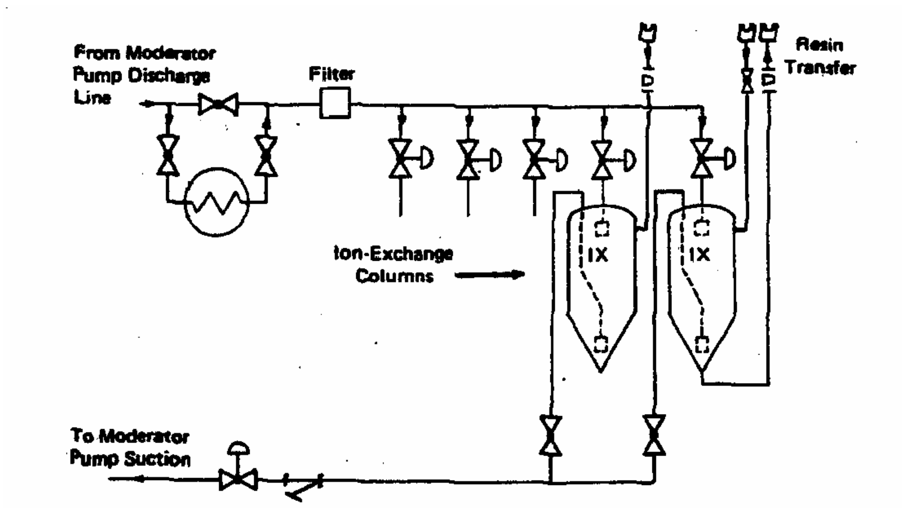


Figure 13.4
Moderator Purification System

Typical impurities include pump-lubricating oil and small particles produced by wear.

The purification system must keep the moderator water very clean for several reasons. Impurities cause corrosion and erosion damage. Neutron activation converts some impurities into radiation hazards. Impurities increase D_2 and O_2 gas releases from the moderator water.

13.5.2 Description

The ion exchange (IX) columns hold chemical resins that remove the soluble impurities. There are several IX columns in parallel. Purification flow increases if two or more columns operate simultaneously. The operator can valve in a fresh column to replace one that is used up. A filter removes insoluble particles. The filters precede the IX columns so the IX resin does not become clogged. Downstream from the IX columns, the strainer keeps resin from entering the moderator system.

The purification cooler in Figure 13.4 drops the moderator water temperature before cleaning. High temperature damages the resin, causing it to release trapped impurities back into the system. At some stations, this cooling is done by the main circulation system heat exchangers.

The chemical resins and filters concentrate activated impurities from the moderator. Heavy shielding protects workers from this equipment

Replacement of filters or resins requires exceptional care because of the radiation hazard.

13.6 Summary of Key Ideas

- The moderator purification system has two uses. It cleans the moderator water and helps regulate the amount of neutron poison.

- High chemical purity of moderator water curtails corrosion, diminishes the hazards of activated substances and reduces releases of D_2 and O_2.

- The moderator circulation pumps provide purification flow.

- Before purification, the moderator water is cooled. In some stations, the main circulation system heat exchanger provide this function. In other stations, there is a separate purification cooler.

- A filter removes particles from the moderator water and ion exchange columns clean it. A strainer prevents resins from escaping into the main system.

13.7 Other Moderator Auxiliaries

13.7.1 Moderator Liquid Poison Addition System

Boron and gadolinium are strong neutron absorbers used for reactor regulation. The liquid poison addition system adds soluble compounds of these elements to the moderator water. Figure 13.5 shows a typical system.

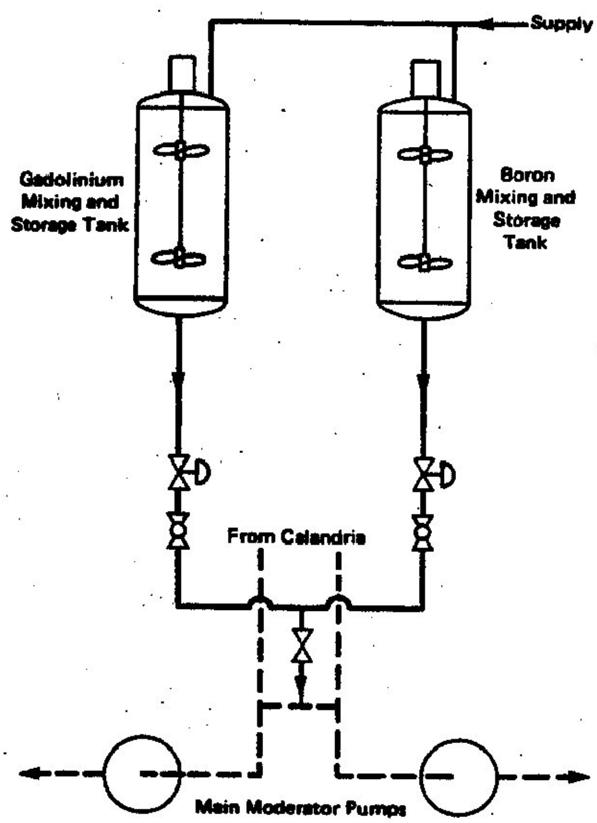


Figure 13.5
Moderator Liquid Poison Addition System

A poison addition pump introduces the selected solution at the moderator pump suction, the lowest pressure point in the main circuit. The control room operator regulates the pump and addition valves. Poison is added for coarse control of the reactor.

Some stations have a gravity feed instead of a pump.

Addition is not directly into the moderator so the effect is not immediate. After addition it stopped, poison continues to enter the system from holdup, in the pipes.

13.7.2 Moderator D_2O Collection System

The moderator D_2O collection system collects moderator water from known leak points. These typically include pump seals, gaskets, and packing around valve stems.

The collection system also accepts drainage from vent lines, from heat exchangers and from pumps. Maintenance workers drain and vent a pump or heat exchanger before opening it for maintenance.

The collection system consists of piping from the collection points, drained by gravity into a collection tank. Sight glasses or flow gauges reveal the leakage rate from various sites. A sample station is used to check isotopic. A pump returns high isotopic, clean water to the main system. Downgraded or dirty D_2O goes for upgrading or cleaning as needed.

A closed collection system lessens the loss of expensive D_2O. It also reduces the release of tritium to the plant atmosphere and diminishes downgrading, as it blocks mixing with atmospheric H_2O.

There is a figure in the HTS sections that shows a similar system that collects heat transport D_2O leakage. There are two separate collection systems to prevent mixing of heat transport coolant and moderator water.

13.7.3 Auxiliary Cooling Systems

Some stations use moderator water to cool certain equipment. A brief description of these auxiliary cooling facilities follows.

13.7.4 Reactivity Mechanism Rod Cooling

Rods to control the reactor need to be cooled. These rods include s adjuster rods, control absorber rods, and shut off rods. Their function will be discussed later. Gamma rays and neutrons heat these rods even when they are out of the core. Some stations cool the out of core rods with a spray of moderator water. Others circulate cover gas past these devices to cool them.

13.7.5 Calandria, Dump Port and Dump Tank Spray Cooling (Pickering A only)

Figure 13.3 shows calandria spray nozzles that cool the exposed calandria, calandria tubes and guide tubes. The figure shows similar sprays for the dump port and dump tank. The sprays operate continuously, both when the moderator is in the calandria and when it is in the dump tank.

During reactor operation, absorption of neutrons and gamma rays heats these components. Without cooling, thermal stress could distort the equipment Spray cooling continues during shutdown, to remove heat produced by decay gamma radiation.

13.8 Summary of Key Ideas

- Some pieces of mechanical equipment (gaskets, seals, valve stems) inevitably leak. The moderator D_2O collection system routes the leakage to a collection tank, keeping it out of contact with the surrounding atmosphere.

- The D_2O collection system also collects drainage from pumps and heat exchangers before they are opened for maintenance.

- The reactor power control scheme includes manually controlled addition of boron or gadolinium to the moderator by the poison addition system.

- Some CANDUs use moderator water to cool out-of-core reactivity devices and calandria components. Other CANDUs use the cover gas to cool them.

13.9 Assignment

1. Why is helium used as a moderator cover gas?

2. Why does a leak in the moderator main system or cover gas system produce a radiation hazard?

3. What is the purpose of:

   a) a recombination unit?

   b) the flame arrestors?

   c) the gas chromatograph?

4. Why is the recombination unit inlet heater not normally needed?

5. Give two uses of the moderator purification system and describe how the purification system equipment carries out these functions.

6. What problems arise if the moderator water is not kept clean?

7. a) What harmful effect does oxygen have on moderator system parts?

   b) Why is oxygen sometimes added to the cover gas? Why is helium added?

8. What are the sources of heat in the reactivity mechanism rods, the calandria tubes and the calandria shell?

9. Describe how the poison addition system is used.

10. Explain why a moderator D_2O collection system is needed and why a closed pipework system is used.

11. What is the role of the strainers in the purification system?

14 Heat Transport System (HTS)

14.1 Introduction

Pressure tubes containing fuel pass through the calandria. Large pumps move heavy water coolant through these fuel channels, removing heat from the fuel. The coolant carries the heat from the core to the boilers, where it makes steam. Coolant is the main link in this heat removal chain. This module describes the heat transport system, including fuel channels, pumps and boilers. The next module describes several heat transport auxiliary systems.

The heavy water coolant removes heat from the fuel and transfers it to the boilers. In normal operation, this is a single task, but it is really two separate functions.

a) The heavy water coolant transfers heat from the fuel to the boilers. This is an essential step leading to steam production and power generation.

b) The heavy water coolant removes heat from the fuel. This task is extremely important, whether or not the reactor is making steam. Keeping the fuel wet protects the fuel. Without adequate cooling, the fuel will fail, releasing hazardous radioactive materials.

Radiation hazards in the heat transport system include the tritium,N-16 and O-19 hazards previously described for the moderator.

When the reactor is running, neutrons make N-16 and O-19. Their penetrating radiation prevents access to equipment containing circulated coolant when the reactor is running. They disappear shortly after a shutdown.

The tritium radiation hazard is always present in the coolant. Tritium releases are more common from the coolant than from moderator water because the coolant is under pressure. High pressure makes small leaks worse. A system opened for maintenance is also a tritium hazard.

Defective fuel releases a range of hazardous radioactive matter into the coolant. Some, for example iodine 131, are vapours that produce a radiation hazard around open equipment. Others, for example cobalt 60, plate out on system piping. These emit penetrating gamma radiation that persists when the reactor is not running.

The heat transport system presents two conventional hazards not seen in the moderator system. These are high pressure and hightemperature.

14.2 Summary Of Key Ideas

- The coolant is the main link in the heat removal chain.
- The coolant transfers heat to the boilers to make steam.
- The coolant protects the fuel by cooling it. This prevents massive fuel failures and radioactive releases.
- N-16, O-19 and tritium are radiation hazards common to coolant and moderator water. The coolant also may contain fission products from defective fuel.
- N-16 and O-19 are hazards that prevent access to HTS equipment with the reactor at power. Hazards from tritium and fission products persist after a reactor shutdown.
- High temperature and high pressure are conventional hazards associated with HTS equipment.

14.3 The Main Heat Transport System

Figure 14.1 shows a typical heat transport system layout The main circulation pumps take cooled D_2O from the boilers and pump it to a reactor inlet header. The header distributes the coolant through feeder pipes to the individual fuel channels.

Plate out is highest on cooler surfaces, such as the boiler outlets, and higher radiation levels are measured in these locations.

Hot coolant leaves each channel through an outlet feeder. The outlet header collects the hot coolant from these feeders and directs it to the boilers (steam generators). The hot coolant gives up its heat through the boiler tube walls. Figure 14.2, a sketch of a typical boiler, shows the boiler tubes.

This completes half the circuit shown in figure 14.2. The coolant continues from the boiler outlet to a second pump. Another inlet header, feeders and fuel channels take the coolant back to the first boiler. The complete pattern resembles a figure eight.

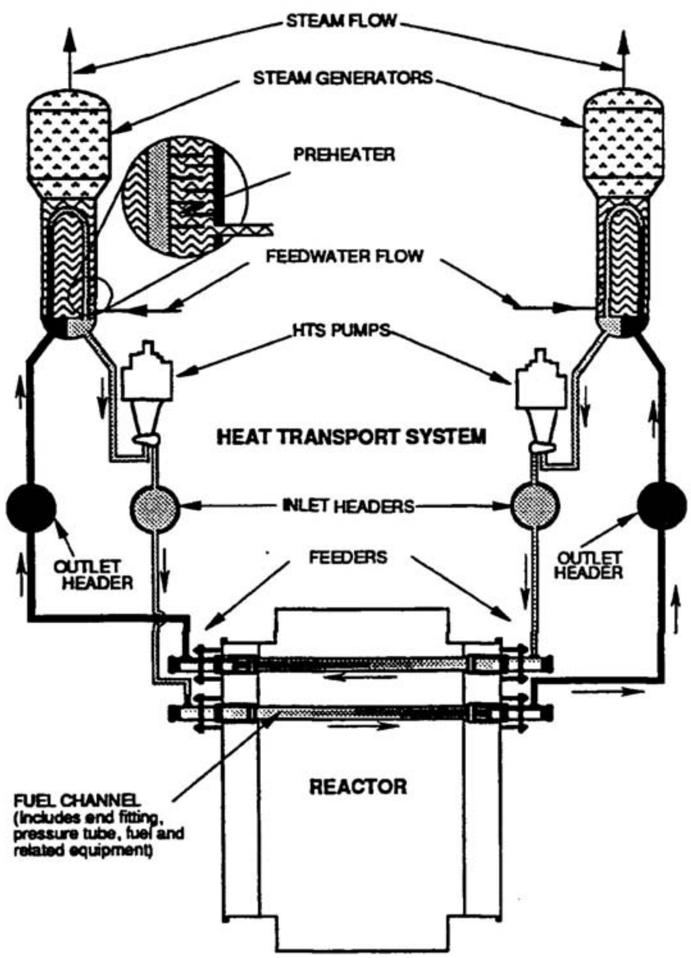


Figure 14.1
Heat Transport System

The figure eight places inlets and outlets at each end of the core. Coolant flows in opposite directions through adjacent channels. Bidirectional coolant flow keeps the temperature of the two reactor faces the same. This decreases thermal stress in the end shields, calandria and calandria tubes. If one reactor face had outlets only, it would operate about 40°C hotter than the opposite face, with inlets only.

The boiler produces steam at about 250°C for the turbine. The coolant enters the boiler somewhat hotter than this, roughly 300°C or so. Its temperature drops about 40°C as it passes through the boiler. It regains the higher temperature as it passes through the reactor core.

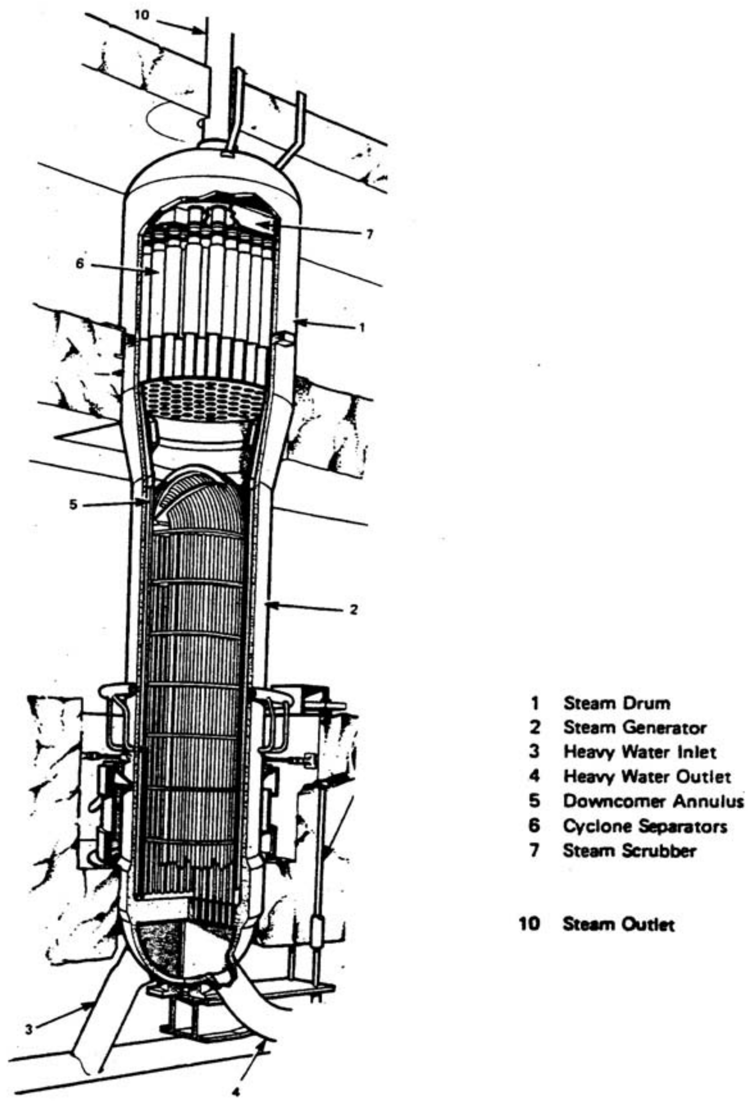


Figure 14-2
Typical Boiler

The location of the main pumps is the coolest point in the circuit at the boiler outlet. The pump location gives the largest possible margin against cavitation.

Briefly, cavitation is caused by localized boiling at the lowest pressure (highest flow) points in the fluid stream. This is immediately followed by rapid condensation of the steam bubbles on the high-pressure side of the pump impeller. Excessive HTS pump cavitation reduces the flow of coolant and may cause damaging pump vibrations. If fluid conditions deteriorate, the pump could fill with steam, stopping cooling flow to the fuel.

To prevent D_2O at 310°C from boiling, the pressure must be about 10 MPa (that is, about 100 atmospheres). The main circulating pumps do not produce this pressure; they supply coolant flow. They generate enough pressure to overcome fluid friction in the fuel channels and boiler tubes. The next module explains how the high pressure is produced and controlled.

14.4 Summary Of Key Ideas

- Pumps at the boiler outlets move coolant over fuel in the pressure tubes.
- The system has a symmetrical figure eight arrangement with boilers, pumps and headers at each end of the reactor.
- Feeder pipes take hot coolant from each channel and pass it to an outlet header. The header collects the hot coolant and supplies it to a boiler.
- The inlet headers take coolant from the pumps and distribute it to individual fuel channels through feeders.
- The pump location gives a high margin of safety against pump cavitation.
- Bidirectional flow maintains uniform temperatures across the core, reducing thermal stress on components.

14.5 Other Features Of The Hts Layout

Before you continue with this section, locate the boilers, main pumps and feeders in Figure 14.3. The figure shows the feeders at one end of the reactor only. In the diagram, the HTS pumps have double discharge and the boilers have dual inlets. That makes it easy for you to distinguish inlet headers from outlet headers in the diagram.

CANDU heat transport system designs are not all the same. The following description is typical of many newer CANDUs.

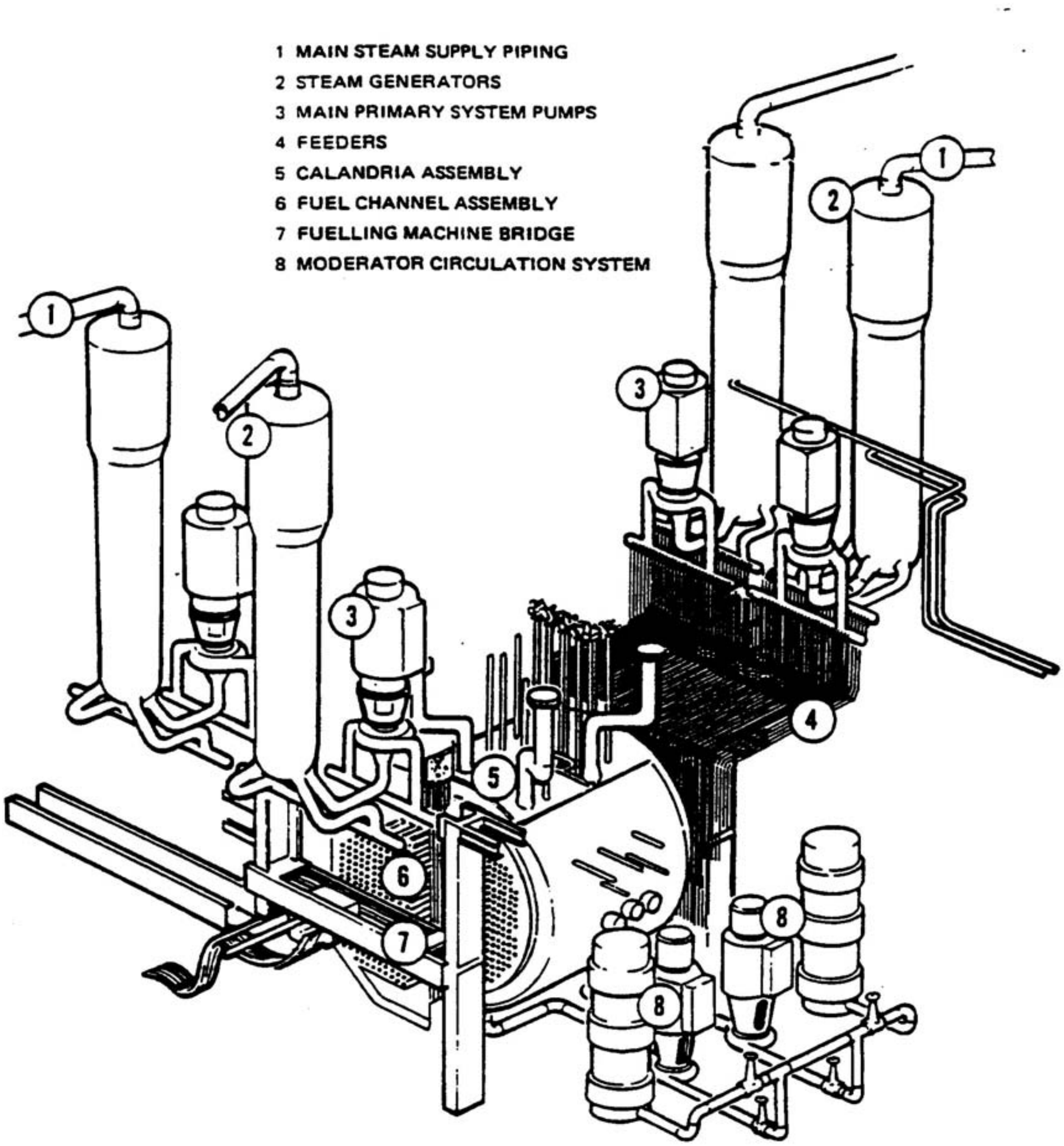


Figure 14.3
Reactor Core, Headers, Pumps and Boilers

Figure 14.3 is a view of a heat transport system with two figure eight loops. (Some plants have a single loop). Each loop supplies coolant to half the core. There are two large pumps and two large boilers in each loop. If one pump in a loop fails, it may be possible to continue operation at reduced power with a single pump.

Such large boilers and pumps were not always available. Older stations use larger numbers of smaller boilers and pumps. This sometimes includes standby pumps or boilers. Designs that are more recent do not include standby equipment in the main HTS. This saves money.

Fuel cooling must continue always, even with the reactor shut down. Without cooling, heat produced by decay of fission products in the fuel can fail the fuel, releasing fission products. A shutdown cooling system cools the fuel when the main pumps or boilers are either unavailable or not required. The size of the substitute pumps and alternate heat exchangers is adequate to remove decay heat. The actual layout differs from plant to plan.t

A feature shared by all CANDUs is the elevation of the headers and boilers above the reactor core. This permits fuel cooling by natural convection if the main pumps and shutdown cooling are both unavailable. For example, loss of electrical power could leave natural convection as the only way to cool the fuel.

Natural convection, also called thermosyphoning, does not produce high flow. It cannot remove full power heat, but is adequate to remove decay heat.

Natural convection occurs when cool D_2O in the boiler tubes falls by gravity into the core. It displaces the hot, less dense D_2O surrounding the fuel. The boilers cool the fluid pushed into them from the fuel channels, and it becomes more dense. The liquid that falls into the core expands on heating and becomes less dense. Thermosyphoning continues as long as the boilers continue to remove heat.

The elevation of the headers above the reactor and the position of the boilers above the headers has another advantage. Operating staff can drain the coolant to the level of the headers, provided it is cool and not under pressure. This drains the boilers and pumps for maintenance. The shutdown cooling system cools the fuel when the coolant does not reach the level of the main pumps. (Terminology at the Bruce site calls this system the maintenance cooling system).

There are no valves for isolating equipment in the main circuit. This eliminates the cost of the valves, reduces heavy water leakage and cuts down on radiation dose to plant staff. Reduced heavy water leakage helps reduce tritium leakage, and exposure of staff who do valve maintenance is reduced.

Early designs, uncertain of reliability, used standby pumps and boilers. Isolating valves allowed equipment maintenance.

14.6 Summary Of The Key Ideas

- A shutdown cooling system removes decay heat when other methods of cooling the fuel are not available. This is typically the case during pump or boiler maintenance.

- The elevated position of the boilers causes circulation by natural convection when the main pumps and shutdown cooling are not available. Natural convection (thermosyphoning) is adequate to remove decay heat from the fuel. It is essential when other methods of cooling the fuel are lost.

- The elevation of boilers and pumps above the reactor core eases their maintenance. Isolation valves are not needed.

14.7 Assignment

1. How are the two purposes of the coolant different from one another and how are they the same?

2. What hazards are present in the HTS that are not hazards in the moderator system?

3. Why are heat transport pumps located with their suction at the boiler outlet?

4. Why does the heat transport system layout have pumps and boilers at each end with bidirectional flow through the core?

5. What is thermosyphoning and when would it be needed?

6. What feature of the HTS layout causes thermosyphoning?

7. Why can the shutdown cooling system have pumps and heat exchangers that are smaller than the main pumps and boilers?

15 Heat Transport Auxiliary Systems

15.1 Introduction

Liquid coolant in the heat transport system (HTS) is a link in the heat removal chain. It protects the fuel by keeping it wet, and it transfers heat from the fuel to the boilers. The auxiliary systems described in the next few sections help the coolant do its job.

High pressure keeps the coolant from turning to steam. The pressure and inventory control system produces and controls coolant pressure in the main HTS circuit. High pressure feed pumps initially produce the high pressure. The pressure and inventory control system must accommodate coolant volume changes caused by thermal expansion and contraction (swell and shrink) of the coolant during operation.

The pressure may increase faster than the control system can counteract. Pressure relief valves prevent pressure from putting piping at risk of rupturing. Loss of coolant pressure and fuel failures would follow a large pipe break.

The coolant purification system protects piping and pump seals from corrosion and erosion that make ruptures more likely. It also removes fission products so that coolant, when it does escape, is less hazardous.

The gland seal system supplies cool, clean HTS D_2O to cool and lubricate the pump seals. This helps prevent seal failure, which would result in pump failure and in the loss of D_2O from the system.

There are many actual and potential heat transport system leaks. The fuelling machine, when connected, becomes part of the heat transport system. This creates additional places where D_2O could escape.

The pressure and inventory control system uses a supply of D_2O from a storage tank to keep the system full. A collection system returns leaked D_2O from leakage points such as pump seals. A recovery system recovers D_2O from unanticipated small breaks.

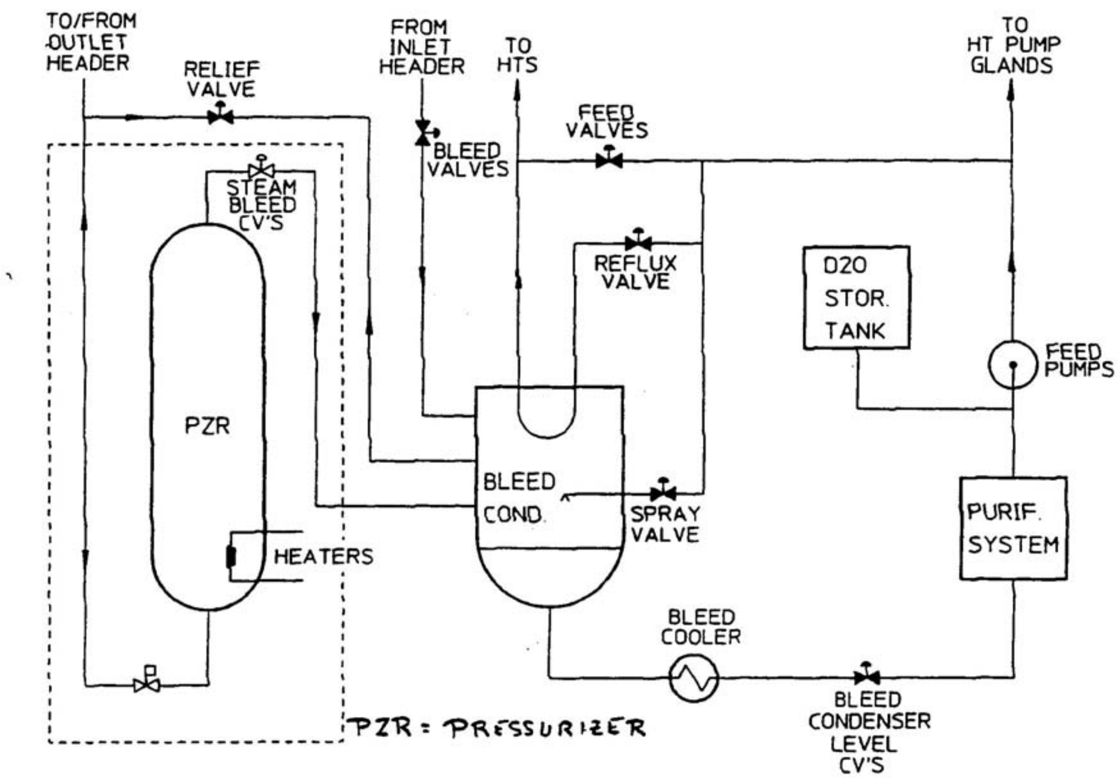


Figure 15.1
Pressure Control Systems

15.2 Summary of Key Ideas

- The words shrink and swell describe coolant thermal expansion and contraction.

- High pressure keeps the hot coolant in the liquid state. If the coolant pressure is too high, piping and pump seals risk rupture.

- HTS auxiliaries help maintain the quantity and quality of the coolant.

15.3 Pressure And Inventory Control

The pressure and inventory control system produces and controls the heat transport system coolant pressure. High pressure feed pumps initially produce the high pressure. The system then holds the pressure at the required value. In other words, the system supplies high-pressure inventory when required, and controls the pressure at the setpoint.

Low pressure and high pressure both cause problems. With low pressure, hot fuel bundle surfaces become covered with steam. This limits heat removal. Extremely high pressure can rupture piping, causing steam filled fuel channels when the pressure drops. Either way, fuel cooling suffers. Fuel bundles will likely fail, releasing radioactive fission products. If poor cooling persists, the fuel and fuel sheath could melt.

Figure 15.1 shows the pressure and inventory control systems found in most CANDU stations. Not all stations have a pressurizer, shown inside dashed lines in the figure. Some details of the CANDU 600 layout are different

We will explain figure 15.1 by describing each major piece of equipment. First we describe pressure control in a system with a pressurizer. Next, we describe pressure control with the pressurizer isolated, and then a system with no pressurizer. The next section describes the bleed condenser, bleed cooler, storage tank and the pressure relief system.

15.3.1 Pressure Control with a Pressurizer

The pressurizer is a large high-pressure tank, partly filled with liquid D_2O and partly filled with steam. One of its purposes is to maintain the heat transport system pressure. A large pipe connects the bottom of the pressurizer directly to the reactor outlet headers at one end of the reactor. Compressed D_2O steam above the liquid in the pressurizer holds the HTS pressure at the pressure setpoint.

For pressure control, there must be a steam space in the pressurizer, and nowhere else in the HTS. This condition exists when the saturated fluids in the pressurizer are hotter than the coolant elsewhere in the system.

When coolant in the main circuit heats and swells, it forces its way into the pressurizer, compressing the steam. The steam acts as a cushion, absorbing some of the pressure increase. The compressed, saturated steam, begins to condense. The steam bleed valves open as the pressure rises, lowering the pressurizer pressure to the setpoint.

When coolant in the main circuit cools and shrinks, the compressed steam in the pressurizer expands. This pushes liquid into the main circuit, limiting the pressure drop. The saturated liquid in the pressurizer begins to boil as the pressure drops. Heaters near the bottom of the tank turn on, and make more steam, raising the pressurizer pressure to the setpoint.

A pressurizer responds immediately to pressure changes. The steam space, by expanding or contracting, begins correcting a pressure error as it happens. The heaters or steam bleed valves then step in to correct pressurizer pressure and hold the system at the demanded pressure.

15.3.2 Inventory Control with a Pressurizer

A second purpose of the pressurizer is to accommodate volume changes of hot coolant. During power maneuvers, the pressurizer accommodates shrink and swell. Inventory control keeps the total amount of coolant in the main system plus pressurizer nearly constant. At high power, the reactor outlet temperature increases. The coolant swells and the liquid level in the pressurizer is high. At lower power, the coolant shrinks and the pressurizer level is lower.

1. Saturated liquid is at the boiling point for the given pressure. It is in equilibrium with saturated steam at the condensation temperature.

2. Five to ten metric tonnes of coolant move in or out of the pressurizer during power manoeuvres between 50% and 100% power.

Inventory control also keeps the pressurizer from completely filling with liquid, or from draining. Pressure control requires a steam space, and there must be enough liquid to cover the heaters.

Figure 15.1 shows coolant feed and bleed valves. When the pressurizer level is too high, excess inventory bleeds from the main circuit through the bleed valves. When the pressurizer level is too low, the feed valves open, allowing the high pressure feed pumps to add D_2O to the main circuit.

15.3.3 Operation with the Pressurizer Isolated

When the reactor is shut down, the coolant temperature may be held near the operating temperature or its temperature may be cooled to below 100°C. During cooldown or warmup, the volume of coolant shrink or swell is too large for the pressurizer. The heat transport system then operates with the pressurizer isolated and the HTS D_2O storage tank accommodates the large volume changes. The coolant may swell 60 m^3 during warmup of a large CANDU.

Figure 15.1 shows the large isolating valve in the connecting pipe between the pressurizer and the main circuit. With the pressurizer out of the picture, the inventory control system becomes the pressure control system. With no steam space anywhere in the HTS, small inventory changes have a large effect on the pressure.

The output pressure of the pressurizing pumps, labeled feed pumps in Figure 15.1, is higher than the HTS pressure. With the feed valve open, the feed pumps transfer inventory from the storage tank into the main system. If system pressure is high, the bleed valve opens to reduce pressure. Inventory from the circuit goes into the bleed condenser, reducing system pressure. This is known as feed and bleed pressure control.

15.3.4 Operation Without a Pressurizer

Some CANDU reactors do not have a pressurizer. Liquid coolant completely fills these reactors. Figure 15.1, with the equipment inside the dashed lines removed, shows this system. Its operation is similar to the operation just described with the pressurizer isolated, except that it is used both when the reactor is shut down and when it is at power.

A feed and bleed system controls pressure by adjusting inventory.

Pressurizing pumps increase the pressure when it drops, and the bleed condenser accepts excess inventory when the pressure is high. The feed and bleed valves control the movement of coolant in and out of the main system.

The feed and bleed system does not respond quickly to swell or shrink. Instruments must sense the pressure error and a control signal must then adjust the control valves.

Stations without a pressurizer limit the coolant shrink and swell when the reactor is running. To limit coolant shrink and swell, the coolant average temperature is kept nearly constant For example, when a power increase causes the reactor outlet temperature to increase, the temperature at the reactor inlet is lowered (by adjusting the boiler pressure).

15.4 Summary Of Key Ideas

- The pressure and inventory control system supplies coolant to the main circuit at the pressure required, and controls the system pressure.

- A CANDU reactor without a pressurizer controls pressure with a feed and bleed pressure control system, (pressurizing feed pumps with feed and bleed valves).
- A CANDU reactor with a pressurizer uses the feed and bleed system for inventory control. The feed and bleed system provides pressure control with the pressurizer isolated during warmup or cooldown.
- The steam cushion in the upper part of the pressurizer controls the heat transport system pressure. It expands to correct low pressure when the coolant in the main circuit shrinks. It absorbs pressure increases when coolant in the main circuit swells.
- Pressurizer heaters and steam bleed valves control the pressurizer pressure. As pressure rises, the steam bleed valves open to lower it. As pressure falls, the heaters come on to raise it.

15.5 Other Equipment In The Pressure And Inventory Control System

Now we will describe the role of the other equipment shown in Figure 15.1.

The bleed condenser reduces the temperature and pressure of coolant leaving the main system. D_2O enters the condenser and flashes to steam. Reflux flow cools the steam, which condenses and collects in the condenser. If reflux flow cannot condense all the steam coming into the bleed condenser, the pressure will rise. Spray cooling then comes on to help bring the pressure down.

The D_2O condensate flows through the bleed cooler. This drops the D_2O temperature below 60°C. Hot D_2O will damage the ion exchange resins and feed pump seals. The cool, clean D_2O is stored in the low-pressure storage tank, or sent to the feed pumps for return to the main system.

Normally the feed and bleed valves are slightly open to supply continuous coolant flow to the purification system. The bleed condenser level control valves control the flow of liquid from the bleed condenser. The bleed condenser pressure pushes the D2O through the system to the feed pumps, or up into the storage tank. The storage tank at the pump suction has three functions.

a. It is large enough to accommodate coolant swell and shrink. The storage tank in a reactor without a pressurizer handles all inventory changes. For a reactor with pressurizer the tank handles inventory during warmup and cooldown, but not during normal operation.

b. It has a reserve supply of D_2O to make up for small coolant leaks.

c. Its elevation provides pressure at the suction of the feed

The HTS coolant pressure is near 10 MPa, with a temperature near 300°C. Cooling keeps the bleed condenser temperature near 200°C. with a corresponding pressure of about 2 MPa.

The CANDU 600 coolant purification flow is separate from the feed/bleed loop. There is almost no feed or bleed flow in normal operation

15.6 The Pressure Relief System

The pressure relief system protects the heat transport system piping from mechanical overpressure. Overpressure could occur, for example, if the feed valves failed open, allowing the full discharge pressure from the feed pumps to pressurize the system. Relief valves that discharge coolant directly from the reactor outlet header into the bleed condenser provide the protection. The names pressure relief valves and liquid relief valves are both used.

15.7 Summary Of Key Ideas

- The bleed condenser collects, cools and lowers the pressure of coolant discharged from the main circuit. Coolant may enter through the steam bleed valves, the bleed valves or the liquid relief valves.

- The bleed cooler lowers the coolant temperature to protect the purification resins and feed pump seals. The low-pressure storage tank also requires cool D_2O.

- The feed pumps have a very high discharge pressure. Depending on the valve settings, this may pressurize the main circuit, increase the pressurizer liquid level or provide flow through the system, including purification flow.

- The feed valves supply inventory to the main system as required. In a system without a pressurizer, this raises the pressure. When a pressurizer is present, increased feed raises its level.

- The bleed valves drain inventory from the main system as required. In a system without a pressurizer this decreases the pressure. When a pressurizer is present, increased bleed lowers its level.

- The feed and bleed valves, when both open, supply purification flow.

- The storage tank accommodates coolant shrink and swell that the pressurizer cannot handle. It has a reserve supply to offset small leaks. Its elevation prevents feed pump cavitation.

- The pressure relief system protects the HTS from mechanical overpressure.

15.8 Other HTS Auxiliaries

15.8.1 The Purification System

Impurities in the system may arise from wear, corrosion and erosion. The fuel or fuelling machines also may introduce impurities. Radioactive impurities also come from fuel defects.

The purification system has two tasks. It must keep the coolant clean and it must control the coolant pH at a high value. Good chemical control is important for several reasons:

a) Protection from corrosion. Hot D_2O is corrosive. The corrosion of heat transport system components is reduced when the coolant pH is kept high.

b) Protection from particulate damage. Particulates erode material, deposit on equipment and clog instrument lines. Abrasion by particulates can damage pump seals. Activated deposits are a maintenance hazard.

c) Removal of radioactive materials. Coolant may contain fission products from failed fuel and activated corrosion products. This increases radiation fields around HTS equipment, where some soluble materials plate out. Leaks are particularly hazardous when the coolant contains radioactive material. Regulations require high purification flow to remove radioactivity after a fuel bundle failure.

The layouts of the coolant and moderator purification systems are similar. Filters precede the ion exchange (IX) columns and keep the particulates from clogging up the resins. Strainers downstream of the IX columns prevent resin from entering the heat transport system.

The coolant purification system has fewer IX columns than the moderator purification system. It does not need reserve cleanup columns for removing reactivity control poisons. The ion exchange resins are also different from those used to clean moderator water. The coolant purification system resin, in addition to removing impurities, keeps the coolant pH high.

15.9 Summary Of Key Ideas

- The purification system keeps the coolant clean and maintains a high coolant pH.
- Good chemical control requires high pH for the coolant to minimize HTS corrosion.
- Particulates must be filtered from the coolant to limit erosion and abrasion damage. Activated particulates deposit on equipment and make it radioactive.
- The purification system removes fission products and activated corrosion products that make the coolant radioactive. These materials can escape through leaks, or when the equipment is opened for maintenance.

15.9.1 HTS Pump Gland Seal

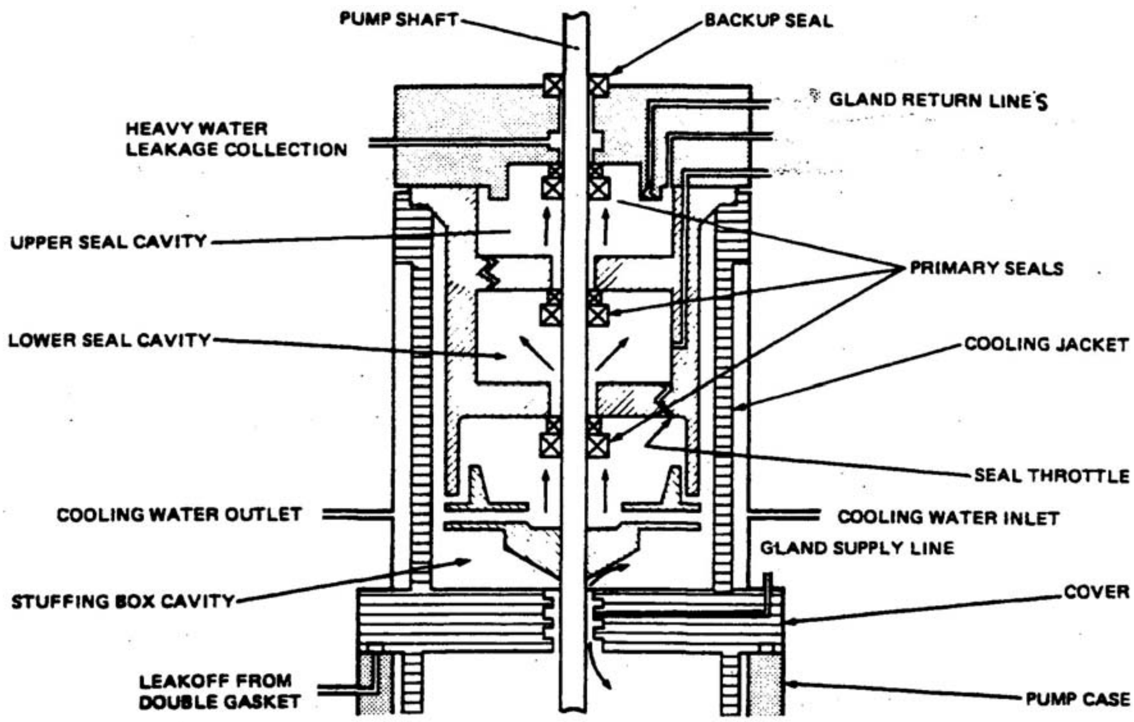


Figure 15.2
Typical Heat Transport Pump Gland Seal

The HTS pump gland and gland seal supply system protect the pump seals and bearings from damage. They also prevent leakage of radioactive coolant through the pump seals.

Figure 15.2 shows a typical pump gland seal. The pump impeller, if shown, would be at the bottom of the diagram. The bottom of the shaft is in hot coolant at almost 10 MPa pressure.

The coolant contains impurities that may be radioactive. The sealing arrangement prevents leakage of the coolant along the shaft. The hot coolant and the impurities it contains could damage pump seals and bearings. Failure of seals could release hot radioactive coolant to the reactor building atmosphere.

The gland supply line takes high pressure, cool, clean, filtered D_2O from the feed pumps. The discharge pressure of these pumps is higher than the pressure in the main heat transport system. The high pressure, clean D_2O keeps the hot coolant out of the seal cavities.

Some of this D_2O flows along the shaft into the main HTS pump, preventing escape of the hot coolant. The rest of the gland supply flows into the seal cavities. The pressure drops from one cavity to the next. The D_2O follows the gland return line to the suction of the feed pumps. A small amount of the water leaks past the primary seals, cooling and lubricating the shaft and seal. D_2O that leaks past the final primary seal passes into the heat transport D_2O collection system.

15.10 Summary Of Key Ideas

- Cool, clean, filtered D_2O from the HTS pressurizing pump discharge supplies the pump glands. The high-pressure gland supply forces its way along the pump shaft, preventing hot, dirty, radioactive coolant from escaping. This prevents damage to pump seals and leakage of radioactive coolant.

15.10.1 The Heat Transport D_2O Collection System

The heat transport D_2O collection system uses a network of pipes to collect coolant that leaks from known leak points. Figure 15.3 is a line diagram of the heat transport collection system. The enclosed system lessens D_2O losses, limits the escape of tritium and diminishes downgrading. Contact between D_2O vapour and atmospheric H_2O would cause downgrading.

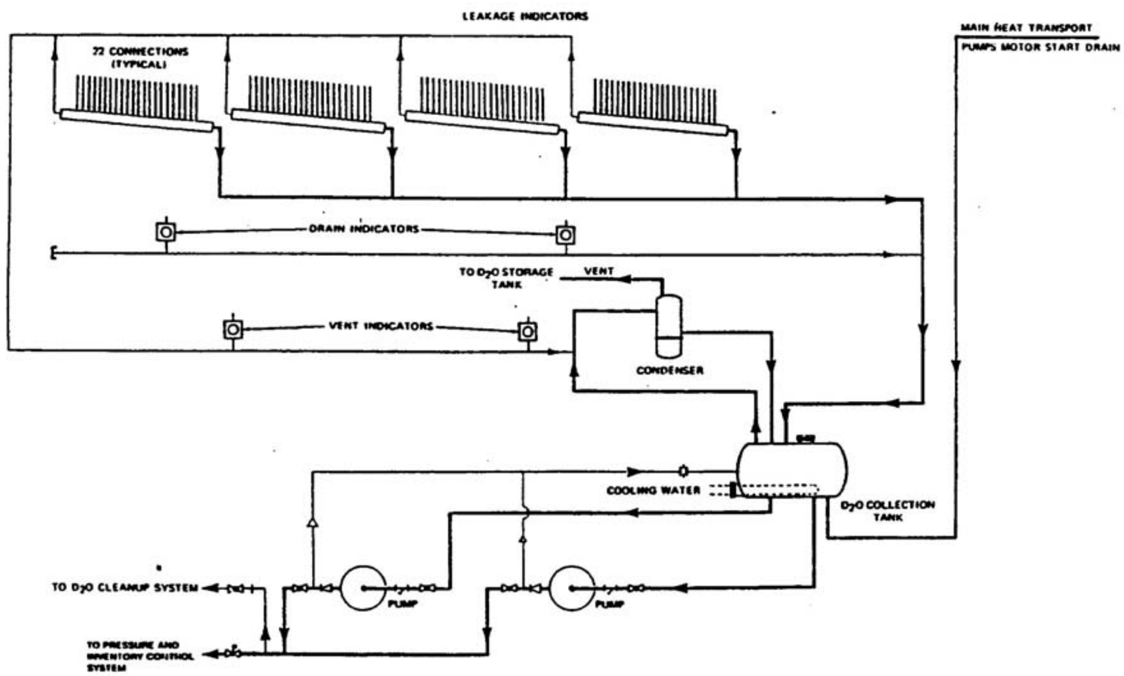


Figure 15.3
HTS D_2O Collection System

The heat transport and moderator collection systems are similar in construction and purpose, but completely separate. Moderator D_2O is never added to the heat transport system because tritium levels in moderator water are too high. Differences in isotopic prevent addition of coolant to the moderator system. Coolant isotopic is usually 98% to 99%, much lower than the 99.8% typical of moderator isotopic. Adding heat transport D_2O to the moderator system would downgrade the moderator water. Downgrading increases fuel costs.

The heat transport collection system collects D_2O from many collection points. There is more leakage from the HTS than from the moderator system. High coolant pressure makes the leak rate from many locations higher, and there are more leak points. A picture of the moderator collection system, comparable to Figure 15.3, would show fewer collection lines.

Coolant that leaks past the main HTS pump seals is the largest flow of D_2O into the collection tank. Another source of D_2O collection is from the inter gasket cavities of heat exchangers and other equipment. Inter gasket cavities are spaces between double gaskets used to seal the mechanical connections between the equipment and the heat transport system. Other collection lines come from valves. There is leakage along the valve stems as well as valve gaskets.

Finally, pumps, heat exchangers and other equipment have D_2O draining and venting lines used for equipment maintenance. The collection system is a convenient place for collecting this drainage.

15.11 Summary Of Key Ideas

- Some D_2O leaks are inevitable. Leakage through pump seals helps cool and lubricate the seals. There is leakage around valve stems and from gaskets on valves, heat exchangers and other equipment.

- The heat transport D_2O collection system takes coolant from these known leakage points and returns it to the HTS. This system also collects D_2O from drain and vent lines provided for equipment maintenance.

- The leak rate from the HTS is much higher than from the moderator system. The HTS has more leakage points and coolant pressure is much higher than the pressure of moderator heavy water.

- Moderator water, with its high tritium content, is never added to the heat transport system. The coolant, with low tritium content but high leak rate, already causes a large part of the station tritium hazard.

- Coolant isotopic is usually lower than moderator isotopic. Coolant is never added to moderator water because it would downgrade the moderator water.

15.11.1 The Heat Transport D_2O Recovery System

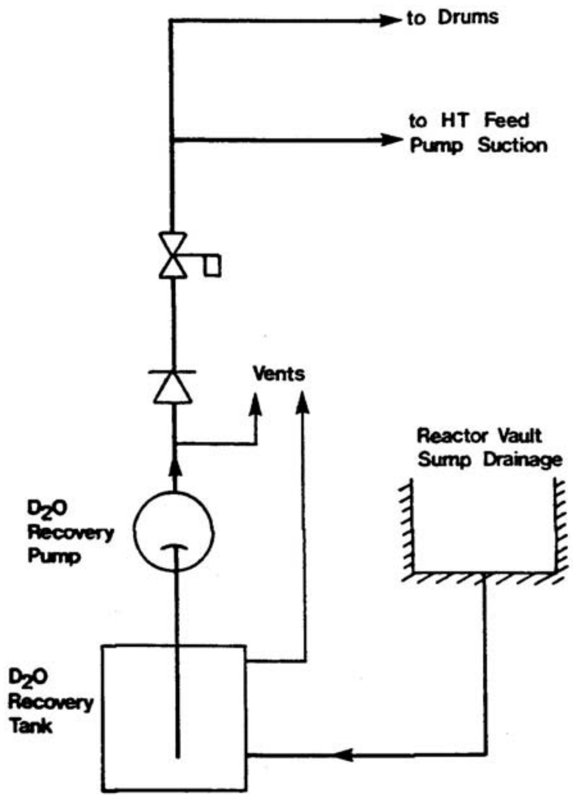


Figure 15.4
Heat Transport D_20 Recovery System

The D_2O recovery system collects and returns coolant lost from unanticipated small and moderate leaks. If the coolant is not returned, the supply of D_2O might run out. Light water injection would then be needed to keep the fuel wet, and this would downgrade the coolant.

Figure 15.4 is a line diagram of the recovery system. Leaking coolant drains to a sump. The sump drains to a recovery tank. A recovery pump transfers the coolant to the feed pumps for return to the heat transport system.

15.12 Summary Of Key Ideas

The HTS D_2O recovery system recovers coolant from moderate leaks and returns it to the heat transport system. This allows D_2O to cool the fuel. If there is not enough D_2O, H_2O cooling is required. This causes expensive downgrading of the coolant.

15.13 Fuelling Machine D_20 Supply

The fuelling machine, when it is attached to a fuel channel, is part of the heat transport system pressure boundary. The fuelling machine has its own D_2O supply to cool the used fuel in the fuelling machine.

Some fuelling machines inject their cool D_2O into the channel during fuelling. This lessens the transfer of radioactive impurities from the heat transport system into the fuelling machine.

15.14 Summary Of Key Ideas

- The fuelling machines, when locked to channels during fuelling, become part of the high pressure HTS boundary.

- Used fuel in the fuelling machine must be cooled. This coolant may mix with HTS coolant during fuelling.

15.15 Assignment

1. State the functions of the following components for a feed/bleed pressure control system and for a system with pressurizer.

   a) feed and bleed valves,

   b) feed pumps,

   c) storage tank,

   d) bleed condenser and bleed cooler.

2. State how the pressurizer controls system pressure.

3. Compare the functions of the pressurizer steam bleed valves and the HTS pressure relief valves.

4. The moderator and heat transport purification systems both clean D_2O and control its pH. Each has one specific task that is not required of the other. What are these two tasks?

5. Compare the D_2O collection and D_2O recovery system functions.

6. Describe how the pump gland seal works. Why is it important to filter particulates from the gland seal D_2O supply?

7. Give and explain differences between the heat transport D_2O collection system and the moderator D_2O collection system in the following areas:

   a) Quantity of heavy water collected,

   b) Isotopic of the heavy water,

   c) Radioactivity of the heavy water.

8. Why would moderator water not be used to supply the fuelling machine?

16 Reactor Fuel

16.1 Introduction

This module shows how fuel materials are selected and assembled to make a fuel bundle that is safe and economic. It also introduces some fuel performance and operating features.

Economic fuel operation requires a bundle to generate heat energy continuously for a long time. It does this routinely in the hostile reactor environment. Good fuel design allows energy extraction without fission product releases during normal plant operation.

Good fuel design also should prevent or limit fission product releases during accidents. No fuel design can prevent all releases in all circumstances. Fuel design and performance features combine with safe operating practices to prevent releases.

Sometimes a fuel bundle fails during normal operation. Failed fuel is fuel that releases fission products through a defect into the heat transport system. Defects vary from small holes in a poorly made bundle to large splits.

Figure 16.1 shows the design chosen for a CANDU fuel bundle. This bundle, used in all routine fuelling of CANDU reactors, is called CANLUB fuel. Many years of successful operating experience prove the safe and economic use of this fuel in normal operation.

To date, no serious accident has subjected CANDU fuel to extreme stress. Tests and analysis suggest it will contain most fission products effectively in these situations too.

The bundle in figure 16.1 is mostly uranium dioxide (UO_2), sheathed and held together by zircaloy. Zircaloy 4, the specific alloy used for this purpose, is 98% zirconium and 1.7% tin. The UO_2 is in the form of high-density pellets. Each fuel element is a thin zircaloy tube, the fuel sheath, filled with fuel pellets. A thin lubricating layer of graphite between the pellet and the sheath is what gives the fuel the proprietary name CANLUB.

Welded end caps seal the ends of each tube. End plates hold the elements in the required arrangement. Spacers brazed to the fuel elements keep them properly spaced. Bearing pads brazed to the outer elements keep the bundle centered in the fuel channel

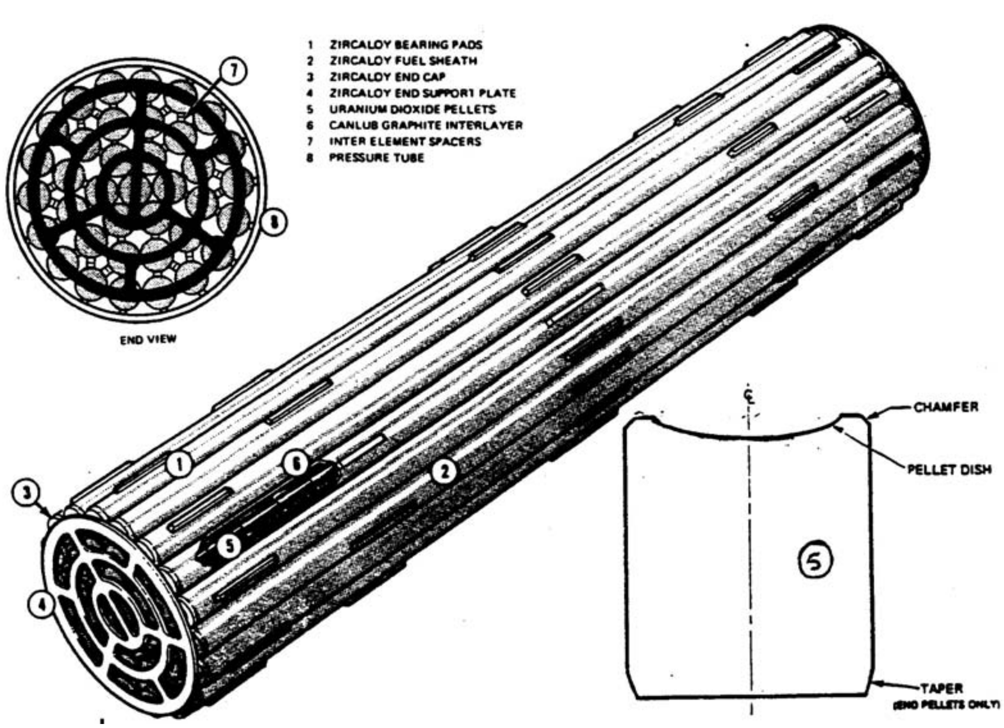


Figure 16.1
37-Element CANLUB Fuel Bundle & Fuel Pellet

The37 element bundle shown in figure 16.1 is used in the majority of CANDUs. Pickering bundles have 28,thicker fuel elements. The overall bundle dimensions are the same and both bundles contain about the same amount of uranium.

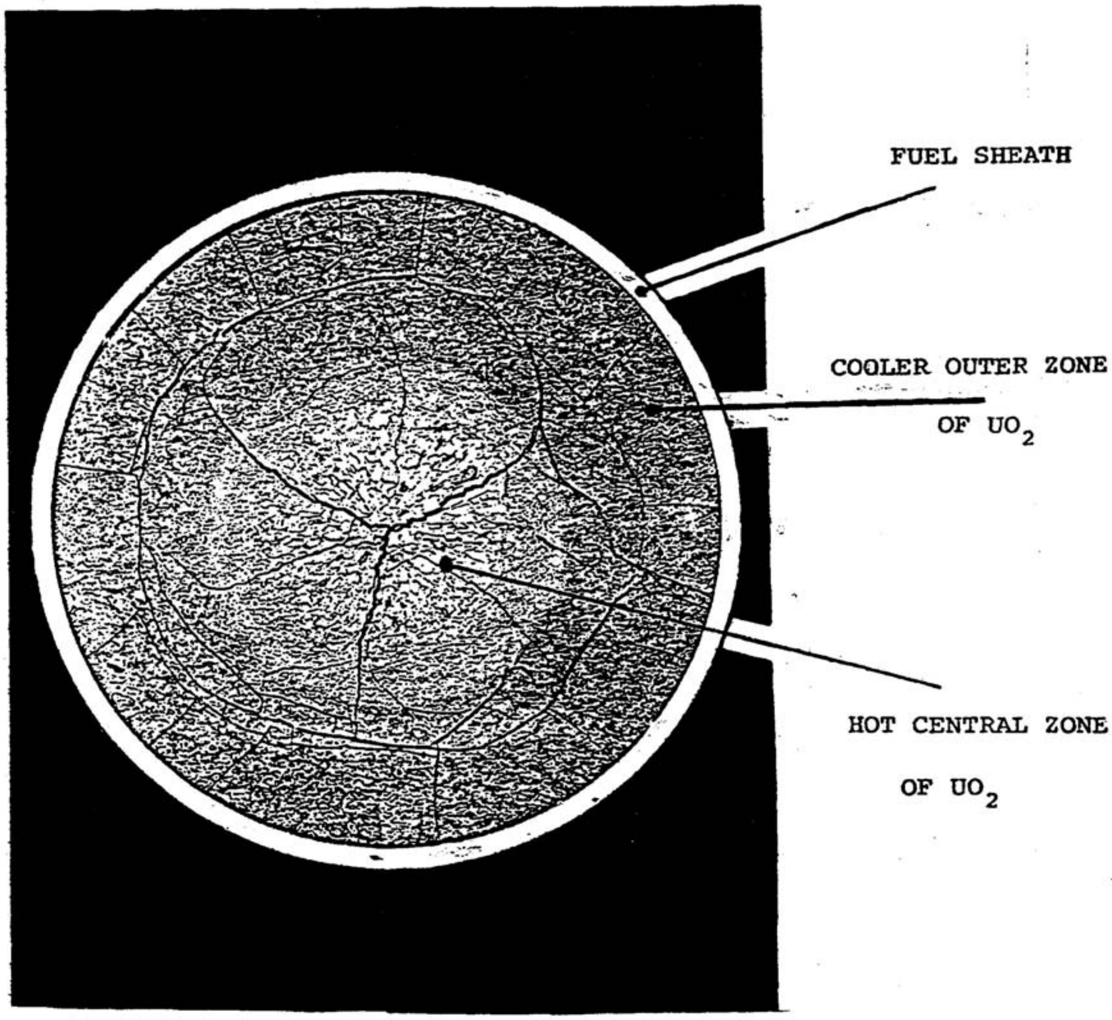


Figure 16.2
Cross Section through a CANDU Fuel Element

16.2 Summary Of Key Ideas

- Failed fuel is fuel that releases radioactive material. Good fuel design and construction produce fuel bundles that operate safely and economically.

- Safe operation implies that fuel does not release radioactive fission products during normal use or during accidents or upsets. Design and operating features combine to limit or prevent releases.

- Economic operation requires each bundle to supply its share of thermal power over a long period. Economics also requires safe reactor operation.

- Fuel elements are assembled to make fuel bundles. The elements are natural UO_2 sheathed in sealed zircaloy tubes. Zircaloy end plates, spacers and bearing pads keep the elements spaced properly.

- CANLUB fuel has a thin layer of graphite between the fuel pellet and the sheath.

16.3 Material and Fabrication

Uranium metal, uranium alloys and uranium compounds have been suggested or tried as reactor fuel. The combination of uranium dioxide (UO_2) for fissile material and zircaloy for sheathing and structural parts is the most common choice for commercial power reactors world wide.

16.3.1 The Fissile Material

There are several characteristics the fissile material should have:

High Fissile Content

High density uranium dioxide pellets contain only half as much uranium as the same volume of pure uranium metal. Natural uranium (0.7% U-235) nevertheless provides enough fissile U-235 to make a critical reactor using natural uranium dioxide fuel and a heavy water moderator.

Other moderators require fuel enrichment of UO_2 to between 2% and 5% U-235. Natural uranium metal can be used with a less expensive graphite moderator, but uranium metal is a troublesome fuel in other ways. This will be seen as we compare its properties with UO_2.

Separation and removal of U-238 increases the concentration of U-235 from the 0.7% found in nature. A CANDU would produce more energy per gram of U-235 using enriched fuel with enrichment between 1% and 2%.

Better use of uranium might not be worth the cost of enrichment. Even without enrichment, a CANDU reactor uses 15% less uranium than an equivalent light water reactor.

Efficient Heat Transfer
Poor thermal conductivity is the main disadvantage of uranium dioxide. Because of it, the fuel pellet interior is much hotter than the exterior. The hottest elements in the core have central temperatures near 1800 ºC for a 37 element bundle and 2200ºC for a 28 element one.

The manufacturer makes a dish shaped end in each fuel pellet to allow for thermal expansion. The hot interior of the pellet expands more than the exterior. The dish shape, shown in figure 16.1, also provides space for fission product gases to collect.

Hot uranium dioxide is not brittle. Shading in figure 16.2 marks the inner and outer regions of the pellet. During operation there are no cracks in the soft inner region. Thermal stress causes permanent radial cracks in the brittle outer region. These cracks may strain the surrounding sheath, but are not likely to cause fuel failures in normal operation.

Fission Product Containment
Uranium dioxide holds 95% of the fission products within its structure. Most of the fission products released come from the hot inner part of the pellet. These migrate to cracks in the cooler part of the pellet and escape into the spaces between the pellets. Some of this free inventory escapes if the fuel element ruptures.

The pellet is the first barrier to the release of fission products to the public. Even a 5% release is not acceptable. For example, in the Chernobyl reactor explosion, between 5% and 10% of the fission products in the core escaped to the atmosphere.

Chemical Compatibility with the Surroundings
Uranium dioxide resists corrosion better than most materials. In contrast, hot uranium metal is very corrosive in hot water. Commercial reactors with metal fuel use gas cooling and a graphite moderator. It does not react with the sheathing material. If the sheathing fails, UO_2 reacts weakly with water. Corrosion is not a problem in normal operation.

Uranium dioxide reacts with oxygen. This could be a problem after a reactor accident if, for example, air contacts the fuel pellets. Breakdown of the UO_2 would result in fission product release.

High Melting Temperature
Uranium dioxide melts at a very high temperature, higher than 2700ºC. This partly offsets the disadvantage of poor thermal conductivity. Even though the central temperature is very high, there is a large margin to melting in normal operation.

Under accident conditions the surface of the bundle may become covered with steam. This causes poor heat transfer and uranium dioxide in the centre of the pellet may melt. Two mechanisms can cause fuel element failure under these conditions: thermal expansion of molten UO_2 stresses the sheath; hot, molten UO_2 may contact the sheath and melt a hole in it. It is important to keep the fuel wet because water cooling will usually prevent fuel failure.

Stability in the Reactor Core
Uranium dioxide is stable under wide temperature variations and intense neutron and gamma radiation. Some materials behave badly when irradiated.

The main disadvantage of uranium metal, for example, is that it changes shape and size during exposure in the core. Uranium metal was used in the world's first commercial power reactor, the British Calder Hall reactor. The fuel was made from natural uranium metal castings, clad with a magnesium alloy. Fuel was routinely removed before it had time to distort. This limits time in the reactor and makes uranium metal fuel uneconomic.

Ease of Fabrication
Uranium dioxide is a chemically inert black powder. A punch and die operation forms compacted UO_2 pellets. These pellets are too big and not dense enough or strong enough for reactor fuel. Sintering the pellets at high temperature in a hydrogen atmosphere reduces their volume by 25%. This makes them into hard, dense ceramic pellets. Grinding them to size polishes the surface. This improves the thermal contact with the sheath.

The expensive difficulties that can be faced in making fuel are illustrated by uranium metal fuel. Uranium metal can be machined, however, the turnings spontaneously combust in air. Fine machinings and dust are hazardous, both because uranium metal is chemically toxic and because it is radioactive.

16.3.2 The Sheath Material
The fuel sheath is the second barrier to the release of fission products. Zircaloy satisfies the requirements for a good fuel sheathing material.

Low Neutron Absorption
Zircaloy absorbs very few neutrons. High neutron absorption rules out most other materials. For example, steel has good structural properties but absorbs too many neutrons. More fuel is needed if neutron absorption is too high.

Mechanical Strength
Zircaloy has good structural rigidity and ductility under operating conditions. Aluminum satisfies the other requirements for a good sheathing material. This less expensive choice has inadequate high temperature strength.

When the fuel temperature changes, the fuel and sheathing do not expand or contract at the same rate. Friction between fuel and sheath can stretch and weaken the sheath. In CANLUB fuel, a thin layer of graphite between the pellet and sheath reduces friction and sheath strain.

Zircaloy loses strength at high temperature. It melts above 1800ºC, but begins to weaken above 1000ºC. At low temperature, below about 1500ºC, zircaloy becomes brittle. This is especially true for irradiated bundles. Fuel should not be moved in a cold core.

Adequate Thermal Conductivity
The zircaloy sheath has good thermal conductivity, but the gap between pellet and sheath hampers heat removal. In normal operation the heat transport system pressure forces the sheath against the smoothly ground surfaces of the pellets. The CANLUB graphite layer may improve the thermal contact.

Accident conditions impair heat removal from some fuel elements. When the heat transport pressure is low, gas pressure inside the element may lift the sheath away from the pellet. Only old bundles contain enough fission product gas for this. Accident conditions produce high temperatures that expand the gas and strain the sheath.

Mechanical stress will cause some fuel failures during a large loss of coolant accident. Stress may come from internal gas pressure or thermal expansion of UO_2. In an accident, this mechanical stress combines with sheath weakness at high temperature to cause failures.

Chemical Compatibility with Fuel and Coolant
Zircaloy has good corrosion resistance in water at the normal operating temperature. It is chemically compatible with uranium dioxide, but some fission products attack it The CANLUB graphite layer, by

holding on to some of the more corrosive fission products, helps protect the sheath from chemical attack.

In an accident, high temperature steam quickly oxidizes the zircaloy sheath. Oxidation makes the sheath brittle. Oxidation will cause some fuel failures in a large loss of coolant accident

16.4 Summary Of Key Ideas

- Uranium dioxide (UO_2) is a black ceramic material that is easy to fabricate.
- UO_2 remains intact under intense heat and radiation in the reactor core.
- UO_2 is chemically compatible with fuel sheathing material and with hot coolant.
- The fuel pellet is the first barrier to fission product release. The ceramic matrix of UO_2 holds about 95% of the fission products.
- Natural UO_2 contains about half as much uranium as does pure uranium metal. This is enough to build a heavy water moderated reactor without fuel enrichment.
- The low thermal conductivity of UO_2 results in high fuel temperatures. Low thermal conductivity is acceptable because UO_2 has a very high melting temperature.
- The fuel sheath is the second barrier to fission product release. Zircaloy has the best combination of properties to fill this role.
- Zircaloy has low neutron absorption combined with good mechanical strength and good thermal conductivity.
- Zircaloy resists corrosion in normal operation.
- A thin graphite layer between pellet and sheath reduces destructive chemical and mechanical interactions between the pellet and sheath.
- Zircaloy is brittle at low temperatures. Fuel could crack if moved in a cold core.

- Zircaloy weakens at high temperature and hot steam oxidizes it, making it brittle.

16.5 Fuel Handling

16.5.1 New Fuel Handling

The ceramic UO_2 is brittle and may chip if fuel bundles receive rough handling. These chips or the sharp edges they leave may puncture the sheath.

The new fuel should not introduce contaminants to the heat transport system. These could increase corrosion or cause erosion damage to the fuel or heat transport system. In the turbulent flow of the fuel channel small pieces of debris have punctured fuel sheaths.

To satisfy the preceding requirements, several precautions are taken in handling new fuel:

a) Bundles are kept in their original containers on the shipping pallets until needed for fuelling. Devices (accelerometers) attached to the shipping containers indicate any severe jolts the bundles might have received during handling prior to fuelling.

b) The bundles are unpacked by band and inspected. Bundles are handled horizontally as the pellets rattle and may chip when the bundle is turned end over end.

c) During fuelling, there is a time limit on how long a bundle may sit in the coolant cross flow in the end fitting. A bundle exposed to excessive vibration is not fuelled.

d) Before a bundle is selected for fuelling, its size is carefully checked for dimensional accuracy. If the outer diameter is too small, the coolant flow will make it vibrate in the fuel channel This could chip the uranium dioxide or cause fretting damage to the sheath or pressure tube. If the outer diameter is too large, excessive force is needed to move it. The pressure tube sags slightly and an oversized bundle will bind as it slides.

e) Visible dirt is removed with a clean cloth to make sure contaminants are not introduced to the heat transport system.

f) Bundles are handled with clean cotton gloves to prevent contamination from sweat.

g) New fuel is loaded by hand into the new fuel transfer system. The transfer system then loads it into a fuelling machine through a shielded port.

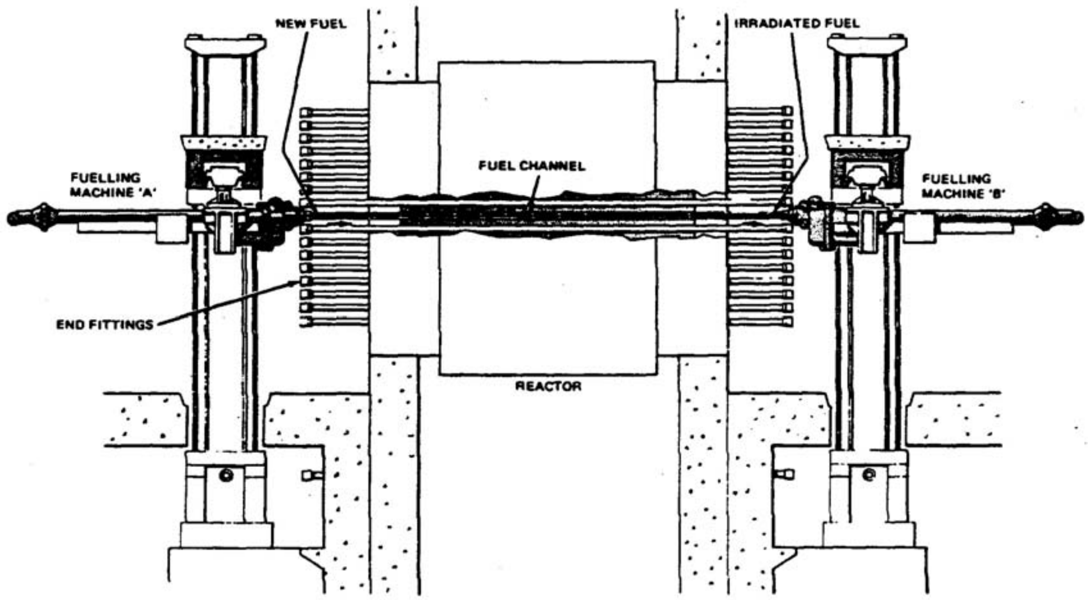


Figure 16.3
The Fuelling Machines in Action

16.5.2 Fuelling

In a CANDU reactor, fuelling is a routine operation. A pair of remotely controlled fuelling machines insert new fuel and remove old fuel while the reactor is running. Figure 16.3 shows the machines aligned with a fuel channel.

The two machines move into alignment at opposite ends of the reactor. Each fuelling machine has a snout that locks on to the end fitting of the channel to be fuelled. When the seal is tight, pressure in the fuelling machines increases to the pressure of the heat transport system.

Behind the fuelling machine snout is a rotating magazine, much like a revolver barrel. A tool in each fuelling machine, the ram assembly, removes the closure plug and the shield plug before fuelling starts. The magazine stores the plugs during fuelling.

Several magazine chambers in one fuelling machine contain pairs of new fuel bundles. The corresponding chambers in the second machine are empty. The ram inserts new bundles two at a time, displacing irradiated bundles into the empty chambers of the second machine.

After fuelling, the ram replaces the shield and closure plugs and the fuelling machines release from the channel.

Either machine can load or receive fuel They insert fuel in the opposite direction in adjacent channels. At some stations, fuelling is in the same direction as coolant flow. In others, fuelling machines insert fuel against the flow.

The fuelling machines usually replace four or eight bundles in a channel on each visit. The fuelling engineer works out details of the fuelling strategy. Fuelling of a channel typically takes two to three hours. Steady operation at full power requires about 100 to 140 bundles (a dozen or so channels) per week.

16.5.3 Handling Spent Fuel

Fission product decay generates heat in spent fuel. The amount of heat is large when the fuel is first removed. Water cooling continues during removal and transfer. Even a few minutes of air exposure may be enough to cause a bundle defect. Cooling continues during storage.

Fuel from the reactor is lethally radioactive. It must be handled remotely and shielded. The fuelling machine discharges spent fuel to remotely controlled equipment. The irradiated fuel transfer system then moves it to the shielded spent fuel storage location. Underwater storage provides both shielding and cooling.

Irradiated fuel is brittle when cool. Handling is reduced by transferring and storing the irradiated bundles on trays.

Defective fuel should be removed from the reactor. Most defects get worse if they are left in. These defects can have several serious consequences: Fission products released into the coolant increase the radiation dose to plant staff. High radiation levels can limit the time workers are allowed to spend in the plant. This increases the cost of routine work at the station.

High concentration of fission products in the coolant increases the risk of release to the public. High iodine 131 concentration requires that the plant be shut down.

Contamination of the coolant and heat transport system piping increases background radiation and makes it hard to detect or locate the next defect.

Once removed, a defective fuel bundle is canned in a sealed water filled container. It, like the intact bundles, is stored under water for cooling and shielding.

The operating license limits the amount of radioactive fission products in the coolant.

16.6 Depleted Fuel and Flux Flattening

In addition to normal CANLUB fuel, depleted fuel is sometimes used for special purposes. Depleted fuel bundles are the same as standard bundles, except they contain about 0.4% or 0.5% U-235, compared to 0.7% for the natural uranium bundles.

During normal, long-term operation, the exposure of the fuel to neutrons varies from place to place in the reactor. The fuelling engineer selects fuel replacement times and places to even out the availability of neutrons across the core. This is known as neutron flux flattening. The oldest fuel, which absorbs many neutrons uselessly, is left longer in areas of the core where the neutron flux tends to be high. Fresh fuel, which gives a higher rate of fission per neutron, is inserted into regions where the neutron flux is lower.

Depleted fuel is used to help flatten flux in a freshly fuelled reactor. In a new or a retubed reactor, all the fuel is new fuel. During the first month or two of operation the uranium content is high. Neutron absorbing fission products take time to build, but the fissile plutonium content increases rapidly at first. The flux flattening effect of high burnup fuel is missing. Typically, one depleted bundle, near the middle of each central core channel, helps flatten the flux.

Depleted bundles are often used to flatten flux when defective fuel is removed from a channel. Replacing fuel ahead of schedule often causes a local hot spot. The fuelling engineer selects locations for depleted bundles. Depleted bundles are placed in the channel to keep the power output similar to before.

Used fuel that is removed to get the defective bundle out is not replaced in the channel. Too much handling of irradiated fuel will likely damage it.

16.7 Summary Of The Key Ideas

- New fuel is handled horizontally with cotton gloves. Visible dirt is removed with a dean cloth. Careful handling reduces fuel pellet chipping and sheath damage and prevents chemical contamination.

- There is a limit on the time a new bundle may sit in coolant cross flow. Bundle vibration in the cross flow could chip the pellets.

- Accurate bundle dimensions are important to prevent damage to the bundle and pressure tube. Bundles that fit loosely in the channel vibrate. Bundles that are too tight require excessive force to move them.

- Identical fuelling machines lock on opposite ends of a fuel channel to refuel it. One machine inserts new fuel and the other accepts spent fuel.

- The fuelling machine includes a snout, a revolving magazine and a ram tool. The snout connects the fuelling machine to the channel. The ram inserts and removes plugs and fuel bundles and stores them in the magazine.

- Spent fuel requires cooling and shielding. Handling is kept to a minimum because the irradiated fuel is brittle. Spent fuel is stored underwater on trays. Defective fuel is placed in sealed canisters for underwater storage.

- Fission products from failed fuel increase the radiation dose to plant staff and makes later defect detection and location difficult. High concentrations could require a shutdown to reduce public risk.

- The fuelling engineer uses depleted fuel, with 0.4% or 0.5% U-235 content instead of 0.7%, to flatten flux. This is necessary in a fresh core where all the fuel is new fuel. It is often required when a channel is refuelled prematurely to remove defective fuel

16.8 Assignment

1. The text lists desirable properties of fuel materials, both for the fissile content of the fuel and for the fuel sheath. For each property listed, state if it is important for economic operation of the fuel or for safety in normal operation.

2. For each of the properties in question 1 that you listed as affecting safe operation, describe how the fuel could behave in an accident.

3. How does the CANLUB graphite layer help prevent fuel defects in the normal operation of CANDU fuel?

4. Give two reasons for using depleted fuel.

5. What are the three operational results of not refueling a channel containing failed fuel?

6. Describe how refueling of a channel is done. Include the removal of old fuel in your description.

7. Compare the handling of fresh and spent fuel.

17 Neutron Life Cycle

A typical neutron, from birth as a prompt fission neutron to absorption in the fuel, survives for about 0.001 s (the neutron lifetime) in a CANDU. During this short lifetime, it travels about 25 cm while slowing and then 30 cm while diffusing, before the fuel absorbs it.

On this journey, it typically scatters 120 or 130 times, 36 times to thermalize. If a neutron is not absorbed in the fuel then one of the earlier interactions absorbs it, or it may escape into the shielding. Figure 17.1 give a pictorial view of the most likely fates of a neutron. The remainder of this module examines each of the possible fates a neutron faces.

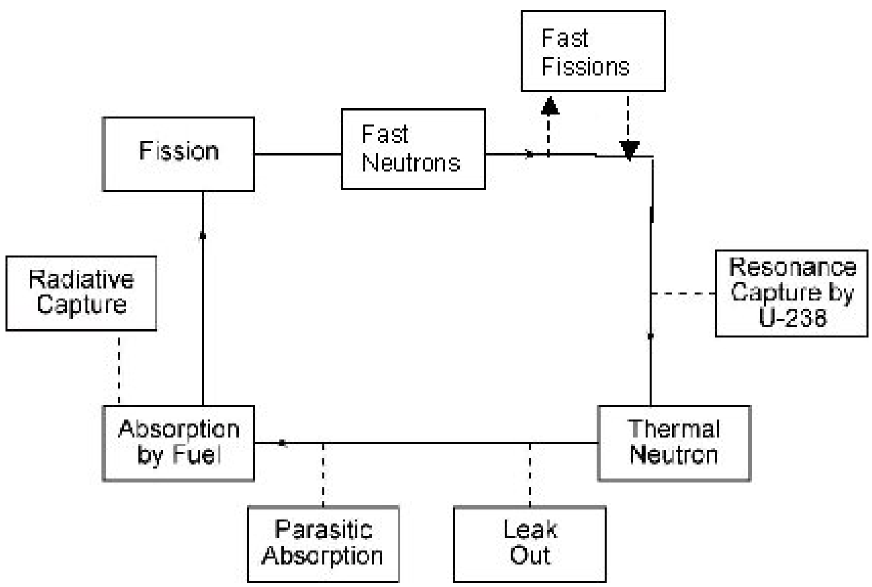


Figure 17.1
Neutron Life Cycle

14.1 Absorption by Equilibrium Fuel

Natural uranium UO_2 fuel, after the reactor has operated for a while, contains U-235, U-238, various isotopes of plutonium, and a variety of fission products. The overall composition of the fuel changes very little with operation because we continually remove old fuel and replace it with fresh.

Over 50% of the thermal neutrons absorbed by equilibrium fuel simply undergo radiative capture. The remaining thermal neutrons (almost 50%) cause fission of U-235 or Pu-239. The net result is that we get about 1.2 fast neutrons per thermal neutron absorbed by the fuel. That is, if 100 thermal neutrons enter the fuel we get 120 fast neutrons in return.

17.1 Fast Fission

An exact accounting of neutron absorption in the fuel includes interactions between fast neutrons (> 2 MeV) and U-238. U-238 cannot fission with a thermal neutron, but U-238 fission can occur with fast neutrons (*fast fission*). This rare process would be completely insignificant but for the fact that our core contains such a large quantity of U-238. Fast fission increases the number of fast neutrons a little from those produced by thermal fission alone.

By far the most significant effect of U-238 in the core is resonance capture. This is important enough to require a new heading.

17.2 Resonance Capture

U-238 has several extremely high absorption peaks in the energy range of about 10 eV to 1 keV, with cross-sections as high as 6,000 barns. Most neutrons that return to the fuel while in this energy range are absorbed.

This is the single largest loss of neutrons in a CANDU; about 10% of the fast neutrons undergo resonance capture while thermalizing.

17.3 Parasitic Absorption

A thermal neutron absorbed by something other than U-235 is unavailable to cause fission. Any of the following can absorb thermal neutrons:

- Fuel sheath
- Coolant, moderator and reflector
- Pressure tubes and calandria tubes
- In-core guide tubes and in-core measuring devices
- Various rods and control zone compartments

In total, the materials on this list absorb about 7.5% of the neutrons, most of them in the moderator and the pressure tubes.

17.4 Leakage

While traveling approximately 40 cm from birth to death a neutron may reach the boundary of the reactor and leak out, never to return.

In a CANDU, leakage accounts for the loss of about 2.5% of the neutrons.

Essentially three things affect leakage: size of the reactor, shape of the reactor, and what happens at the boundary. The designer can adjust these effects, as described below, to reduce leakage into the shielding.

17.5 Size and Shape

Figure 17.2 shows three spherical reactors. If some neutrons travel 50 cm, a neutron born at any location in reactor ‘A' has a possibility of escaping. As we increase the size of the reactor to ‘B', the neutrons born inside the dotted circle normally won't leak before they are captured. By increasing the size again to ‘C' a still smaller percentage of the neutrons can leak out.

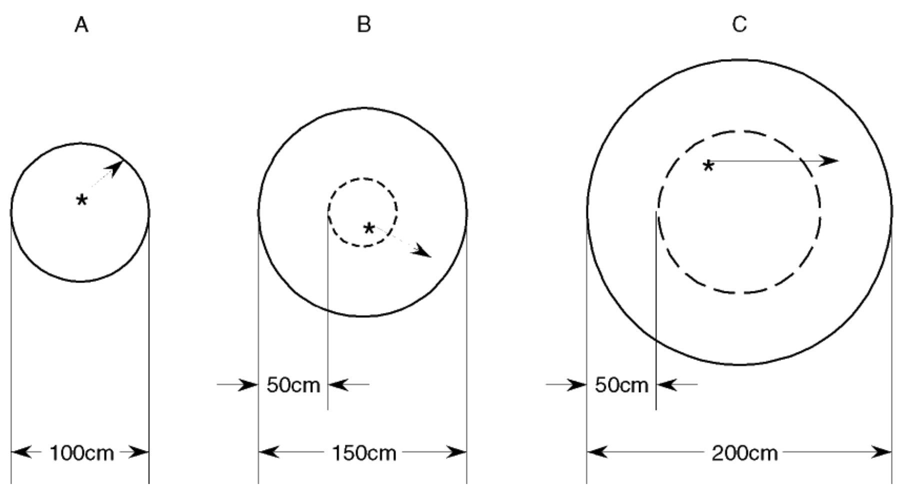


Figure 17.2
Effect of Size

Similar arguments can be made concerning the shape of a reactor. It can be shown that for a given volume of fuel and moderator a sphere always has the smallest leakage. A sphere is not a practical shape from an engineering point of view. Instead, we use a cylindrical reactor core that has the diameter slightly bigger than the length. The actual shape is a compromise between engineering and nuclear considerations.

17.6 Reflectors

The final thing that affects leakage is what happens to a neutron when it reaches the boundary. By surrounding the reactor with material that "bounces" some of the leaking neutrons back into the reactor, the loss due to leakage is reduced. We call this surrounding material a reflector.

An ideal reflector has a high probability of scattering neutrons and a low probability of absorbing them. These properties are shared with the moderator, so the reflector is merely an extension of the moderator as shown in Figure 17.3.

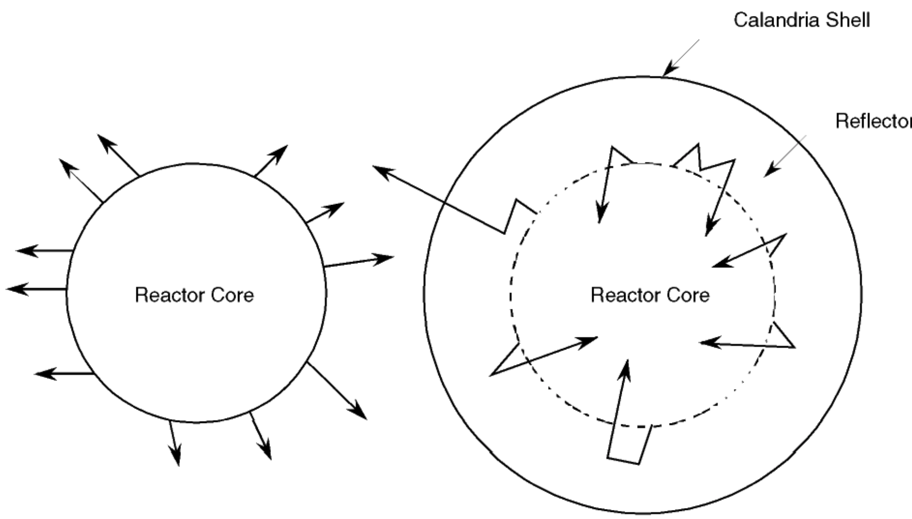


Figure 17.3
Reflector Location

The zone between the fuel region (indicated by the dotted line) and the calandria shell serves as the reflector.

Another effect of the reflector is that it assists with flux flattening. The neutrons that are stopped from leaking add to the flux in a region where flux is naturally low. This *flux flattening* allows bundles near the edge of the core to increase their contribution to the power output, without raising the power from bundles in the high flux region.

17.7 Overall Cycle

Roughly 20% of the neutrons are lost (10% by resonance capture, 7.5 % by parasitic absorption, and 2.5 % by leakage) and do not return to the fuel. About half of those remaining (i.e., 40%) cause fissions, and fission produces 2.5 neutrons, restoring the total to 100%.

Looked at another way, suppose 100 thermal neutrons are absorbed in the fuel. This produces a new generation of 120 fast neutrons, enhanced to 123 by fast fission. A loss of 23 neutrons rounds off to a 20% loss. This leaves 100 thermal neutrons to be absorbed in the fuel again, sustaining continuous energy production in the reactor. Parasitic absorption can be adjusted to hold power steady or to change the number of neutrons in the cycle so power can be increased or decreased.

17.8 Assignment

1. Sketch the neutron life cycle.
2. Discuss each of the possible fates of a neutron.
3. Why do our reactors have reflectors?

18 Criticality and Neutron Multiplication

In the chain reaction illustrated in Figure 18.1, only one neutron is available each time to cause fission. Therefore, the number of fissions occurring per second remains constant.

The power produced depends on the number of fissions per second. If a reactor is producing one watt of power steadily, then 3.1×10^{10} fissions will occur each second. 3.1×10^{10} neutrons are available from these fissions to produce 3.1×10^{10} fissions during the next second, and so on. There is no multiplication of neutrons.

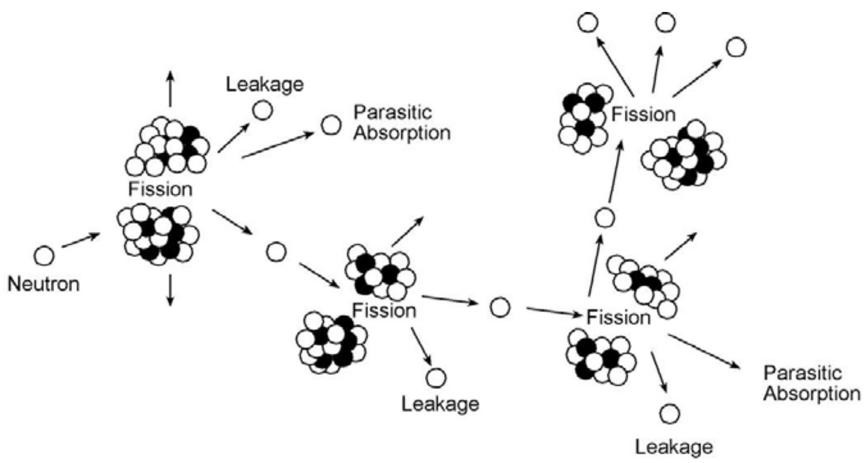


Figure 18.1
A Chain Reaction

When the chain reaction is being maintained steady like this, the power level is steady and the reactor is said to be critical. If the power is increasing or decreasing, the rate of neutron production is not constant.

The neutron multiplication factor, k, based on the neutron cycle introduced in the preceding module, is used to keep track of neutron production.

$$k = \frac{\text{Number of neutrons in a generation}}{\text{Number of neutrons in the preceding generation}}$$

A nuclear reactor can operate with its power steady, increasing, or decreasing. To show how these three different conditions are described by the multiplication factor, let us suppose that we start with 100 neutrons, which is our first generation. Absorption and leakage remove some of these 100 neutrons. Those that remain are available for fission. In a certain time (the generation time), these neutrons cause fission and neutrons of the second generation are produced.

If $k = 1$, there will be 100 neutrons at the beginning of the second generation, 100 at the third, and so on, and fissions continue at the same rate as at the beginning. The power is steady and the reactor is said to be in the critical condition.

Notice from this definition that the reactor may be critical at any power level.

If $k > 1$ (greater than one), say 1.05, the 100 neutrons of the first generation produce $100 \times 1.05 = 105$ neutrons at the beginning of the next generation. This increases again in the third generation and in subsequent generations, leading to a greater number of induced fissions and consequently to a larger neutron population. After 100 generations, for example, the number of neutrons present would be 13150 (100×1.05^{100}). The arithmetic is just like compound interest build-up in a daily interest bank account. A few neutrons can initiate a growing fission chain. The power is increasing and the reactor is said to be super-critical.

In this example, with $k = 1.05$, the power increased 131 times in about one tenth of a second. This is too fast a rate to control and in practice, the multiplication factor is never allowed to become so large.

If $k < 1$ (less than one), 0.95 for instance, the number of neutrons reduces from 100 at the beginning to 95 in the second generation. In this situation, the original 100 neutrons is reduced to one in about 90 generations (100×0.9590). The chain reaction cannot be sustained under this condition. As the neutron population decreases, so does the number of fissions and the power decreases. The reactor is said to be sub-critical.

The term reactivity (Δk) is often used in place of the neutron multiplication factor k. It is defined by the following equation:

$$k = 1 + \Delta k$$

k is always very near to 1 so Δk takes on small positive or negative values. We can say that the reactor is:

critical if	$\Delta k = 0$,
super-critical if	$\Delta k > 0$ (positive reactivity)
sub-critical if	$\Delta k < 0$ (negative reactivity)

Reactivity (Δk) is normally given in units of milli-k, where 1 mk = 10^{-3} k.

Example:

Given $k = 1.004$

$$\Delta k = 1.004 - 1$$

$$= 0.004 \text{ or } 4 \text{ mk}$$

It is important to stress that neither k nor Δk gives any information about the power level in the reactor. They simply tell you whether the current power level is constant, increasing or decreasing.

18.1 Reactivity Control

Reactivity must be controlled for three basic reasons:

1. Maintain the reactor critical and the power level steady,
2. Increase or decrease power at a controlled rate to match the demand,
3. Reduce power quickly in response to an upset.

There must always be excess positive reactivity available in case we need to raise power. Several things influence the excess reactivity such as the burnup of U-235, the production of Pu-239, the production of neutron absorbing fission products, and changes in the temperature of the fuel, coolant, and moderator. Before we look at how to adjust Δk, we will discuss fuel burnup effects that cause slow long-term reactivity changes. The fission product and temperature effects are discussed in separate modules later.

Figure 18.2 illustrates the effect of the burnup of U-235 and the build-up of Pu-239. The graph assumes a freshly fuelled (new) reactor at day zero.

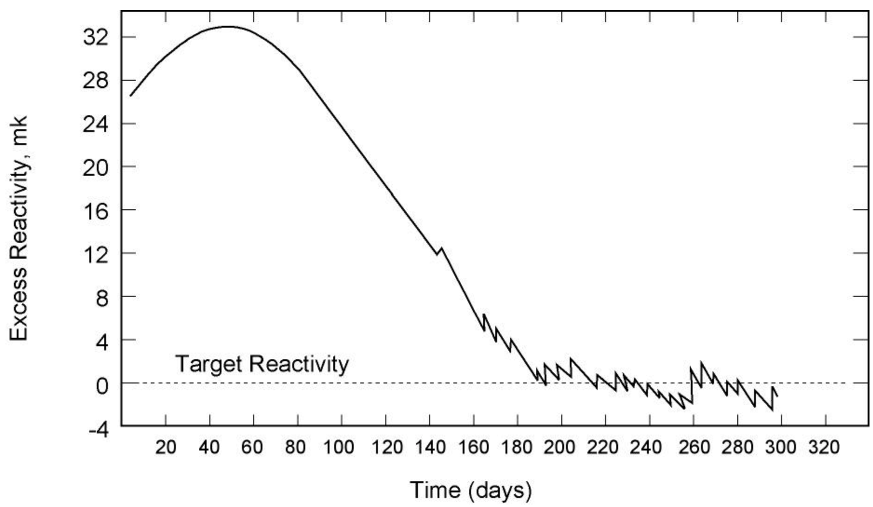


Figure 18.2
Excess Reactivity

As the reactor is operated at power, fissile atoms are consumed causing reactivity to decrease. When the overall reactivity gets close to zero, fissile atoms must be replaced at the rate at which they are consumed (on-power refuelling).

You might not expect the increase in reactivity seen in the first months. It occurs because Pu-239 is initially produced more rapidly than U-235 is consumed. Net production of Pu-239 levels off after one or two months when fission of Pu-239 becomes significant, so that fissile material is not replaced as fast as it burns up and reactivity decreases. The gradual increase of neutron absorbers in the fuel make the reactivity decrease steeper. In operating the reactor, we must adjust the reactivity to compensate for these reactivity changes.

There are three basic methods available to control reactivity:

1. Adjusting the amount of fissile material in the reactor.

2. Adjusting the amount of absorption in the reactor.

3. Adjusting the neutron leakage from the reactor.

18.2 Adjusting Amount of Fissile Material

If more U-235 is inserted into the reactor, U-235 absorbs a greater fraction of the neutrons absorbed by all materials in the core. Thus inserting fissile material is an addition of positive reactivity ($+ \Delta k$). All CANDU reactors accomplish this through on-power fuelling.

18.3 Adjusting the Amount of Absorption

If a neutron absorbing material is introduced into the reactor, it absorbs neutrons that could otherwise have been absorbed by U-235. Thus, insertion of absorbers adds negative reactivity ($- \Delta k$). Practical methods absorb mainly thermal neutrons, changing parasitic absorption. One liquid absorber and three types of solid absorbers are used:

1. Liquid Zone Compartments (used in all CANDU reactors).

2. Adjuster Rods, made of cobalt or stainless steel, (used in all CANDU reactors except Bruce A).

3. Absorber Rods, made of cadmium or stainless steel, (used in all CANDU reactors except Pickering "A").

4. Shutoff Rods, made of cadmium encased in stainless steel, (used in all CANDU reactors).

Light water is used in the liquid zones. A tube is partially filled with light water. Increasing the water level causes more neutrons to be absorbed ($- \Delta k$). Decreasing the level causes fewer neutrons to be absorbed ($+ \Delta k$). Figure 18.3 shows a simplified sketch of a liquid zone control compartment.

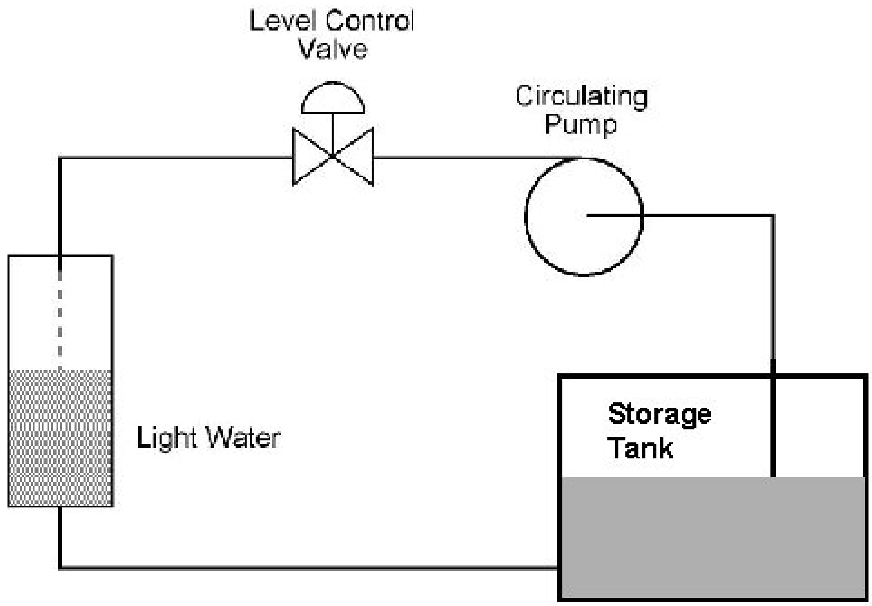


Figure 18.3
Liquid Zone Control

The solid rods are all physically similar. Their names come from the specific purposes for which they are used.

In addition to these absorption devices, parasitic absorption by dissolved neutron absorbers is used in two ways.

1. Neutron absorbers are dissolved in the moderator. The absorbers used, called poisons, are boron and gadolinium. These can be added gradually by the poison addition system or removed by the purification system to adjust Δk. All CANDU reactors use dissolved poisons.

2. All CANDU reactors (except Pickering "A") are able to inject a gadolinium solution rapidly into the core for a fast shutdown.

18.4 Adjusting Neutron Leakage

Lowering the level of the moderator in the calandria increases leakage. If we cause a larger fraction of the neutrons to leak out of the reactor, negative reactivity is inserted (- Δk). Changing the moderator level changes the effectiveness of the reflector. Increasing level reduces leakage and inserts positive reactivity; decreasing level allows more neutrons to leak and inserts negative reactivity.

In addition, the moderator can be dumped rapidly out of the core, stopping the fission process. As level drops, leakage increases and unthermalized neutrons are less likely to be absorbed. They just leak away. Without the moderator, CANDU fuel cannot make a critical mass.

18.5 ASSIGNMENT

1. Define the neutron multiplication constant.

2. Complete the chart below.

	k	Δk	Power
Super-critical			
Critical			
Sub-critical			

3. If $k = 0.997$, find Δk in units of milli-k.

4. List the three basic methods of reactivity control and explain how each works.

5. If the reactor is at a power level of $10^{-4.2}$ F.P. is it critical?

19 Changes In Reactor Power With Time

The two preceding modules discussed how reactivity changes increase or decrease neutron flux and hence, change the thermal power output from the fuel. We saw how the neutron population can change from one generation to the next if the reactor is super-critical or sub-critical.

The rate of change of power is the factor that determines how difficult a reactor is to regulate, or even whether regulation is possible. This lesson considers influences on the rate of change of reactor power.

19.1 Effect of Neutron Lifetime on Changes in Reactor Power

We have seen how neutron density, neutron flux and reactor power increase or decrease over a number of generations. If $k > 1$ an initial power level of P_0 increases to P_0 k in one generation, to $(P_0$ k) x k in two generations, to $P_0 k^3$ in three and after N generations to $P_0 k^N$. The formula for power after N generations, P, is:

$$P = P_0 (k)^N = P_0 (1 + \Delta k)^N \tag{1}$$

This tells us that starting with power P_0, if we insert reactivity Δk the power changes to P after N generations as given in equation (1). The formula gives power in terms of the number of generations that have elapsed, not in terms of time.

The time t required for N generations to elapse is merely:

$$t = \ell \cdot N \quad \text{so} \quad N = \frac{t}{\ell} \tag{2}$$

In this equation ℓ is the average time for one neutron generation. (1) and (2) together calculate the power increase in time t. We shall show later that under normal operating conditions $\ell \approx 0.1$ s for a CANDU reactor.

Example:

Suppose reactor power is steady at 60% FP when Δk = +0.5 mk is inserted (i.e., k = 1.0005). How high will the power go in 100 seconds?

Solution:

From (2)

N = 100 s/0.1 s = 1000 generations

From (1)

P = 60% x (1.0005) 1000 = 60% x 1.65 = 99%

19.2 Reactor Period

To make the arithmetic easier (especially in the days before calculators) we can write equation (1) in a different way:

$$P = P_0 e^{t/\tau} \tag{3}$$

It can be shown that equations (1) and (3) are the same thing.

The constant τ is the reactor period. This equation gives the power conveniently in terms of the elapsed time, t, and reactor period.

In practical terms, to get an idea of how fast power is changing we could talk of the length of time it takes for the power to double, or increase ten-fold, or whatever. Equation (3) makes it natural to use the length of time it takes for the power to change by a factor of e. The power increases by a factor of e, that is, $P = e\,P_0$, when the time duration, t, is equal the reactor period τ. This is our definition of reactor period: the time it takes power to increase by the factor e. (If you were wondering, e = 2.718 281 828 5…)

For small values of reactivity (Δk) encountered in normal operation, equations (1) and (3) give identical results for:

$$\tau = \ell/\Delta k \tag{4}$$

Using this to repeat the example above, τ = 0.1/0.0005 = 200 s

P = 60% x e100/200 = 60% x e0.5 = 60% × 1.65 = 99%

Note that the larger Δk is, the shorter the reactor period becomes, and the faster the power changes will be.

19.3 The Effect of Delayed Neutrons on Power Change

For fission of U-235, 99.35% of the neutrons produced are prompt neutrons, and 0.65% are delayed neutrons emitted by fission products. The time for one generation of prompt neutrons is 0.001 s. The average lifetime of the delayed neutrons is almost 13 seconds. The average lifetime, ℓ, for all the neutrons, prompt and delayed is then:

$$\ell = 0.9935 \times 0.001 \text{ s} + 0.0065 \times 13 \text{ s} = 0.085 \text{ seconds}$$

For simplicity, we usually round off the value of ℓ to 0.1 s, as we did in the earlier example.

Although delayed neutrons represent a small fraction (0.65%) of neutrons generated by fission, they increase the average lifetime of all neutrons from 0.001 s to 0.085 s, that is, by a factor of 85. Formula (4) shows that the period is 85 times longer than it would be for $\ell = 0.001$ s. This reduces the initial rate of power rise by a factor of 85.

In summary, the effect of the delayed neutrons is to make the rate of power changes reasonably slow for small additions of positive reactivity. The delayed neutrons make regulation and protection practical.

19.4 The Effect of Prompt Neutrons Alone, and Prompt Critical

The formulas for reactor period and for power change (1, 2, 3, and 4) accurately predict power changes provided Δk is a small value, typical of reactivity additions used in normal reactor regulation. These formulas do not work at all for large $+\Delta k$ insertions such as would be used to calculate possible upset or accident conditions.

The Chernobyl reactor tragically demonstrated the behaviour of a reactor following the sudden insertion of a large positive reactivity. In that accident power increased from a low level to an estimated 10 000 per cent full power in less than 2 seconds. Why didn't the delayed neutrons limit the rate of power increase? The remainder of this section describes the effect (or non-effect) of delayed neutrons in more detail, to be able to answer this question.

First, consider the role of the delayed neutrons in a constant power reactor (k = 1). Suppose that somehow we could "shut off" the 0.65% of delayed neutrons. Starting with 100 neutrons, after one generation there would be 99.35 (since the delayed neutrons are not showing up). In the second generation, this drops to 98.7 and by the third generation, it has dropped to 98. The power is decreasing as if the reactor is sub-critical.

In fact, the reactor depends on the arrival of the delayed neutrons to "top up" the neutron population and stay critical. When +Δk is added, as long as Δk is very small, the power cannot rise very quickly. The prompt population is still slightly "sub-critical" on its own. (It is important that k × 99.35 be less than 100). The top-up from delayed neutrons increases reactor power, but the increase in the "top-up" must wait until the extra delayed neutrons from the extra fission products at the higher power level begin to show up. This takes several seconds. The slow arrival of the delayed neutrons controls the rate of power increase (decreasing it by a factor of about 85, as we saw earlier).

Now suppose a large +Δk is inserted in the reactor core. The prompt neutrons (multiplied by k) increase enough from generation to generation that the power increases even without the delayed neutrons. The prompt neutron population "takes over" and power rises as though the neutron generation time is $\ell = 0.001$ s, the lifetime of the prompt neutrons alone, and not $\ell = 0.085$ s, the average lifetime we used before. A reactor in this state is said to be prompt critical, that is, critical on prompt neutrons alone. This kind of rapid power increase caused the Chernobyl core to explode.

The rate of power rise of a prompt critical reactor can be illustrated using the earlier example with $\ell = 0.001$ s. We will calculate the power increase in one second instead of one hundred seconds.

With a positive reactivity of 0.5 mk, the reactor period would be given by:

$$T = \frac{\ell}{\Delta k} = \frac{0.001}{0.0005} = 2 \sec onds$$

In one second, the power would increase as given by equation (3), i.e.,

$$P = P^0e^{t/\tau} = P_0e^{1/2} = P0 \text{ x } 1.65$$

For $P_0 = 60\%$ this gives a power rise to almost 100% in 1 s instead of 100 s. (For Chernobyl, the reactivity addition is estimated to have been about 25 mk more than needed to go prompt critical.)

The example shows how rapid the power increases would be, even for reactivity changes less than a mk, if all the neutrons were prompt. Effective reactor regulation is not possible under these circumstances because power changes from small reactivity effects occur too quickly for the regulating system to respond.

Emergency shutdown of the reactor would be an even greater problem. Even very fast protective systems need a second or so to take effect. In this relatively long time interval, severe damage would result from the excessive power levels reached.

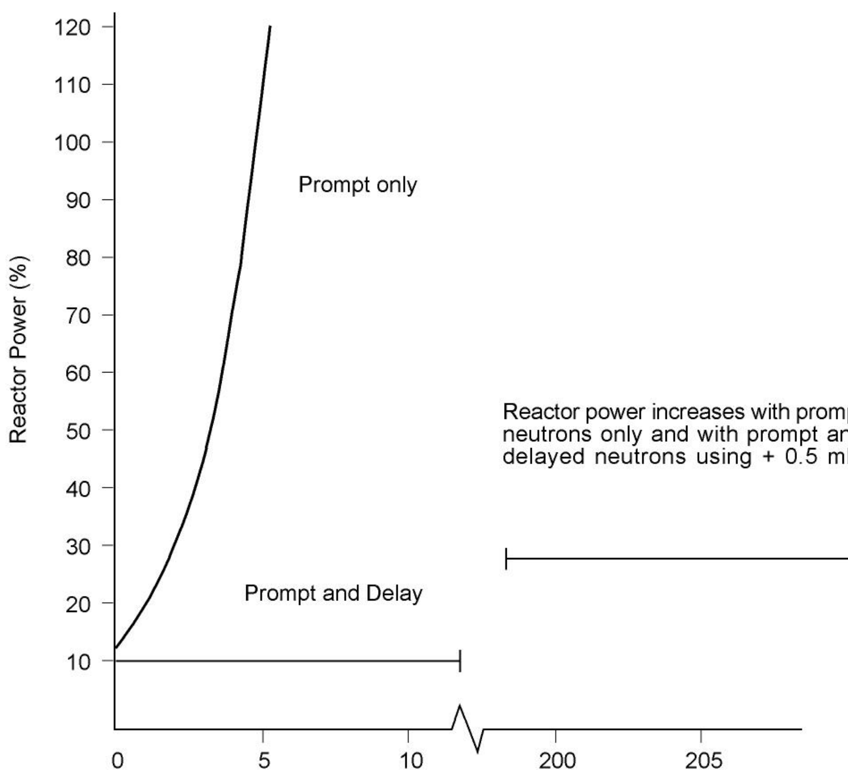


Figure 19.1

Figure 19.1 illustrates the power increase for a reactivity of + 0.5 mk considering only prompt neutrons and considering delayed neutrons.

19.5 Power in the Sub-critical Reactor

When k is made less than one (Δk negative) power decreases from one generation to the next. From what we have said, you would expect power to drop soon to zero. Surprisingly, it does not. First we will describe how a sub-critical reactor behaves and then give the reasons why.

When the reactor is deeply sub-critical, (Δk is large and negative), power is steady at a very low level. If the reactor is not as deeply sub-critical, its power is steady at a higher level. The addition of positive reactivity to a sub-critical reactor (leaving Δk negative) causes a power increase and stabilization at a higher level. Unlike the critical reactor, there is no self-sustaining chain reaction to drive the power higher and higher.

The deeply sub-critical reactor is almost unresponsive; that is, large positive reactivity additions that would be totally unsafe in a critical reactor have almost no effect. A sub-critical reactor that is nearly critical responds much like a critical reactor; that is, even a small reactivity addition can produce a large power rise, with power rising gradually over several minutes. As long as the reactivity addition leaves the core sub-critical, the power stabilizes at a new higher level and does not continue to increase.

This observed behaviour results from the photoneutron reaction. Some fission products emit energetic gamma rays that eject neutrons from deuterium in the heavy water molecules. When the reactor is shut down, that is, when negative Δk is introduced to stop the fission chain reaction, this neutron source cannot be turned off. There are always some photoneutrons.

Photoneutrons may enter the fuel and cause fission. Neutrons from these fissions also cause fissions. (Less than one neutron per fission survives, since k is less than one.) This results in a neutron flux higher than the source neutron flux alone. The core acts like an amplifier for the source flux. This is not a self-sustaining chain reaction; if we could turn off the source neutrons, the flux would go to zero. However, there is always a small, steady trickle of source neutrons in the reactor causing fission.

The observed flux comes from the slowly decreasing photoneutron source (decreasing because the fission products gradually decay over weeks and months) together with some fission neutrons. When the reactor is deeply sub-critical there are mostly source neutrons, not affected by reactivity changes. When the reactor is less sub-critical, (k is larger, but still less than one), more fission neutrons survive and cause more fission, increasing the total neutron population. When the reactor is nearly critical, the bulk of the neutrons are fission neutrons and the core responds much like a critical core. Remember, adding negative reactivity does not make $k = 0$, it simply makes k less than one. As k gets bigger and approaches one, amplification of the source neutrons increases because more fission neutrons survive.

19.6 Assignment

1. Define reactor period.

2. Explain why delayed neutrons are important for reactor control.

3. Describe how, in a sub-critical core, a steady supply of source neutrons produces a flux that is higher than just that of the source

4. Explain why amplification of source neutrons in a sub-critical core is different for different sub-critical values of k.

20 Xenon: A Fission Product Poison

Many fission products absorb neutrons. Most absorption cross-sections are small and are not important in short-term operation. Xenon-135 has a cross-section of approximately 3,000,000 barns, over 4000 times that of U-235. That is, each atom of xenon-135 absorbs as many neutrons as 4000 U-235 atoms. About 6.6% of all fissions produce a nuclide of Xe-135, either directly as a fission product or indirectly as a fission product daughter. Xenon is a major problem in our reactors because of its remarkable neutron absorption and high yield.

20.1 Xenon Production

Xe-135 is produced directly in only 0.3% of all U-235 fissions. The following example is typical:

$$ {}^{1}_{0}n + {}^{235}_{92}U \rightarrow {}^{99}_{38}Sr + {}^{135}_{54}Xe + 2\,{}^{1}_{0}n + \gamma $$

Xenon-135 is mainly produced as a fission product daughter, by iodine decay as follows:

$$ {}^{135}_{53}\mathrm{I} \xrightarrow[T_{1/2}\,=\,6.6\,\mathrm{hr}]{} {}^{135}_{54}\mathrm{Xe} + \beta $$

Iodine-135 is produced in 6.3% of U-235 fissions. Thus, iodine decay accounts for about 95% of total xenon production. (6.3/6.6 = 0.95). Iodine-135 does not absorb neutrons.

20.2 Xenon Loss

Xenon is removed from the reactor by decay as follows:

$$ {}^{135}_{54}\mathrm{Xe} \xrightarrow[T_{1/2}\,=\,9.1\,\mathrm{hr}]{} {}^{135}_{55}\mathrm{Cs} + \beta $$

or by neutron absorption (radiative capture):

$$ {}^{1}_{0}n + {}^{135}_{54}Xe \rightarrow {}^{136}_{54}Xe + \gamma $$

The rate of burnout depends on the neutron flux. For a CANDU at full power, neutron absorption accounts for about 90% of Xe-135 loss, decay for only 10%. Cesium-135 and xenon-136 do not absorb neutrons.

20.3 Equilibrium Xenon Load

There is no xenon in the fuel of a reactor that has been shut down for a long time (or has never been operated). Xenon slowly builds to an equilibrium level after the reactor is started. The equilibrium level depends on the steady state reactor power. Figure 20.1 shows xenon load versus time for various power levels. For CANDU reactors at full power, xenon load builds to about 28 mk of negative reactivity in 35 hours or so.

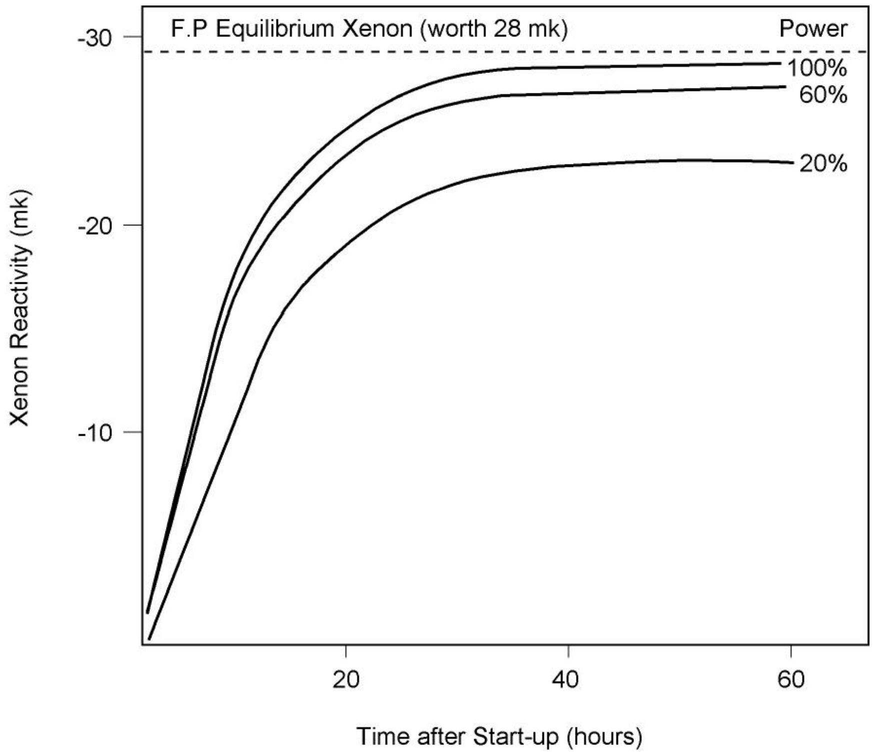


Figure 20.1
Xenon Build-up to Equilibrium

This negative reactivity (-28 mk) is always present in normal steady operation except during the first several hours after start-up. The reactor design includes enough excess positive reactivity to compensate for the normal -28 mk load.

When the normal xenon load is not present, operations must accommodate the excess positive reactivity with neutron absorbing chemicals. Soluble poison (boron or gadolinium) added to the moderator compensate for the missing xenon. As xenon concentration increases, burnout, or ion exchange purification remove the poison.

20.4 Xenon Transients

After operating for around 35 hours, xenon is near its equilibrium level. It then causes problems only if the reactor power is changed. For example, consider what happens to the production and loss of xenon-135 immediately after a reactor trip (or any fast power reduction to 0%).

a) Production:

– from fission (5%) - stops immediately

– from decay of iodine (95%) – continues

Result—on the short term, most of the production continues.

b) Loss:

– by decay (10%) - continues

– by neutron absorption (90%) - stops immediately

Result—on the short term, most of the removal stops.

The consequence of continued production without removal is a dramatic increase in xenon concentration immediately following a trip. Figure 20.2 is a graph of xenon load versus time after a trip from full power.

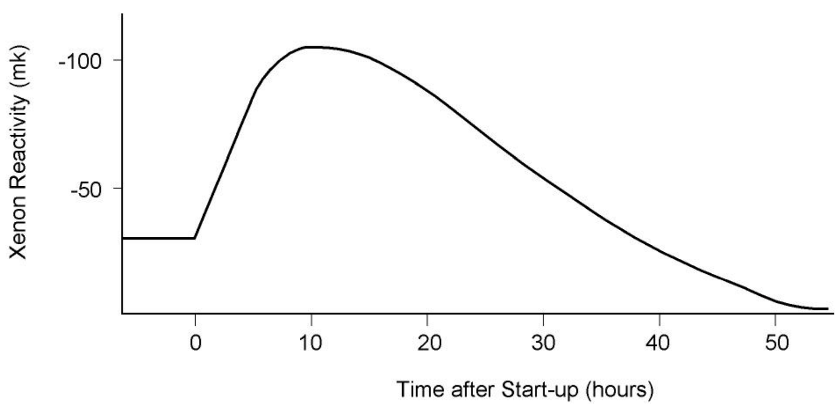


Figure 20.2
Behaviour of Xenon

Notice that negative reactivity from xenon peaks about 10 hours after a full shutdown, at a level much higher than the equilibrium 28 mk load. At the peak of the transient, the decay of Xe-135, which has increased because there is more xenon, matches the production of xenon by iodine decay, which has decreased because there is now less iodine. Iodine decay continually decreases, reducing xenon production. When the peak has passed, xenon decay exceeds production, and the curve gradually falls back towards normal and lower.

Any power reduction causes a transient xenon peak. The smaller the power reduction the smaller the peak, and the earlier it occurs. For example, on a power reduction from 100% to 60%, there is still an initial excess of production over loss, but significant neutron flux remains to burn out the xenon. The peak height and its duration are reduced, and the peak occurs earlier. Figure 20.3 includes a typical reactivity variation for a rapid power reduction (stepback) to 60%.

On a power increase after steady, low-power operation (say from 60% to 100%) the reverse effect occurs. Xenon burns out rapidly while production from iodine decay continues low. Reactivity increases and the control system must insert negative reactivity to compensate. Adding poison to the moderator supplements this as needed.

20.5 Poison Prevent and Poison Override

Withdrawing adjuster rods from the reactor core contributes positive reactivity, up to a maximum of 15 or 20 mk depending on the particular reactor. Excess positive reactivity is required to keep the reactor operating during small xenon transients. As Figure 20.3 shows, the adjusters can accommodate a stepback to 60%.

If the negative reactivity due to xenon exceeds the available adjuster positive reactivity, the reactor goes sub-critical with no way to restart it. We say it is poisoned out. Figure 20.3 shows that on a trip from full power, a reactor that is poisoned out cannot be restarted until 35 hours or so after the trip, when xenon has decayed to near the –28 mk equilibrium level.

Holding reactor power near 60% (or higher) prevents a poison out. It is important to realize that on a turbine trip it may be economically sound to keep the reactor operating and to exhaust steam to a condenser (or the atmosphere). We call this mode of operation poison prevent.

Thirty-five to forty minutes after a trip the negative reactivity from xenon exceeds the positive reactivity of the adjusters. (See figure 20.3 again.) If the reactor is started up during the 35 to 40 minute poison override time and brought up to power before poisoning out, the xenon will be burned up rapidly and a poison out may be prevented.

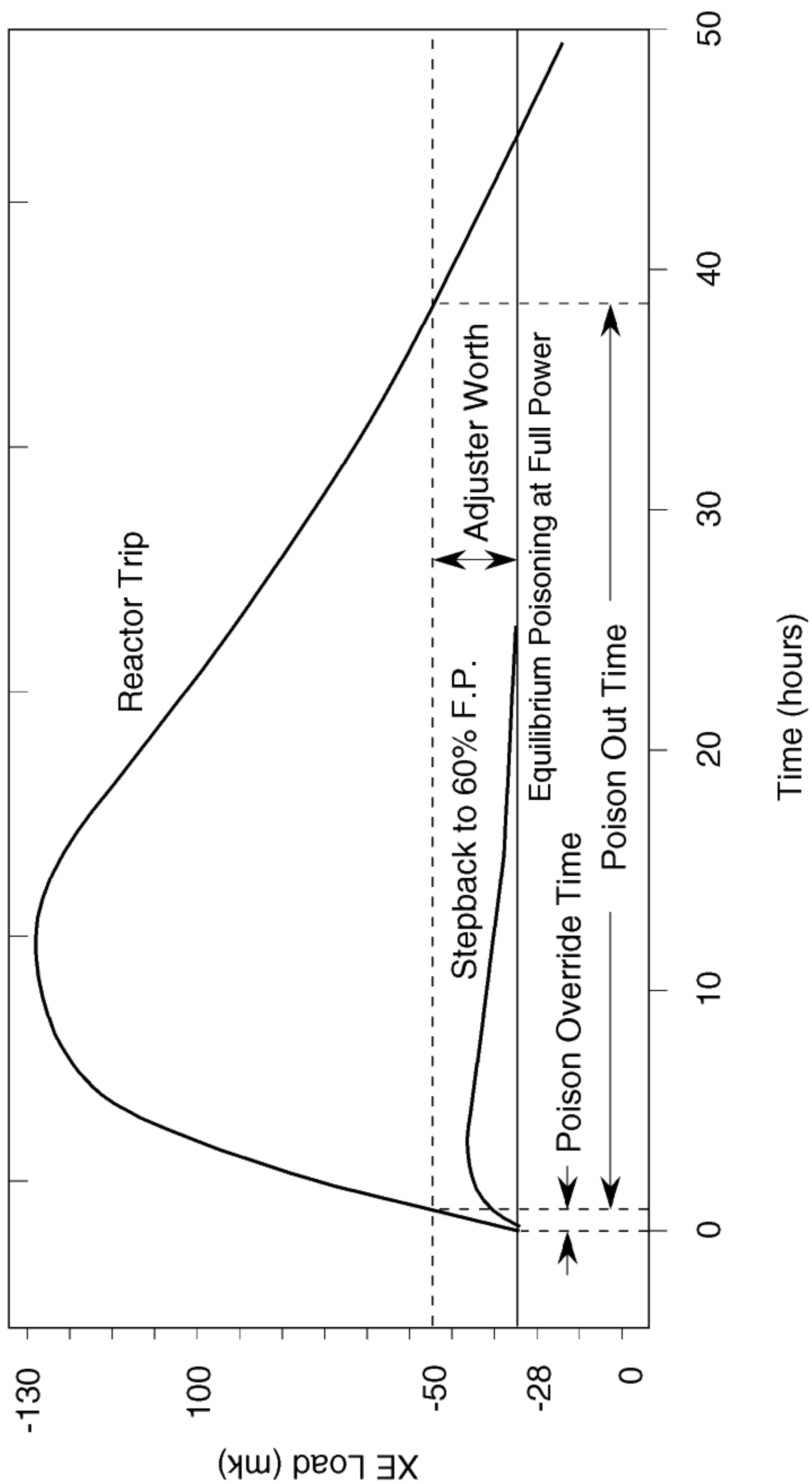


Figure 20.3
Transient Xenon Reactivity

Poison override is possible in principle, and is part of the reactor design, but is usually not practical. Before restarting the reactor following a trip, it is important to find the cause of the trip and eliminate the fault. A number of checks are required before a trip is judged to be spurious (that is, a trip that does not occur in response to an actual failure). The control room staff must make the decision to try to restart in about 20 minutes because the adjusters drive out slowly. Repairs or checks following a trip usually take longer than this. Operating procedures that do not allow the operators to try to "beat the poison out" remove the temptation to take short cuts.

20.6 Other Effects

In a large, high flux reactor, xenon may cause flux increases in one part of the reactor while flux is decreasing elsewhere. Instead of the flat flux shape illustrated previously, transient peaks and valleys occur. This operational problem will be discussed more in future modules.

20.7 Assignment

1. Sketch the behaviour of xenon on a reactor trip from full power.

2. Explain why a xenon poison out occurs.

3. Discuss the production and loss of xenon including the relative magnitude of each term for the following situations:
   (a) on start up,
   (b) on a power decrease from steady full power,
   (c) on a power increase from steady 60% power.

4. What features of xenon-135 and its production make it the most significant fission product in terms of its reactivity effect.

21 Reactivity Effects of Temperature Changes

21.1 The NRX Experiment and Negative Feedback

In 1949, the NRX reactor at the Chalk River Nuclear Laboratory was allowed to "run away" in a controlled test. NRX was a heavy water moderated experimental reactor that used control rods for reactor regulation. The heavy water level was set 3 cm above the height at which the reactor would be critical at low power with the rods withdrawn. The reactor power was allowed to increase "unchecked". (Had there been any surprises it could have been quickly shut down.) The manner in which power changed is not what you might expect based on what you learned previously (See Figure 21.1).

Initially the power increased, but it did not increase indefinitely as you might have expected. As the fuel temperature increased, reactivity decreased and this caused the rate of power rise to slow. Later the reactivity decreased even more as the heavy water got warmer. The total decrease in reactivity was enough, in this demonstration, to make the reactor sub-critical. Consequently, power reached a maximum and began to decrease.

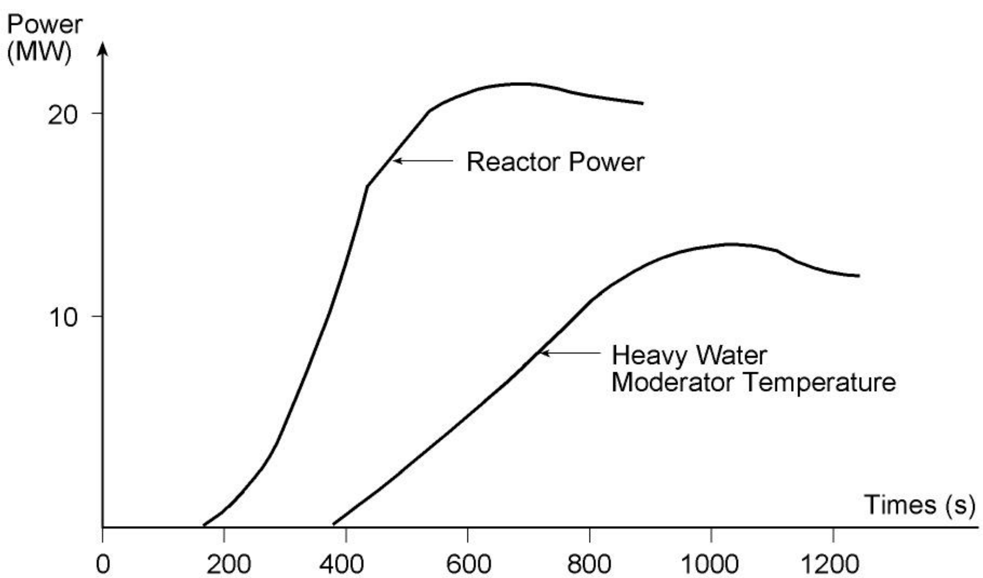


Figure 21.1
The NRX Experiment

The test demonstrated that for small, positive reactivity insertions the NRX reactor was self-regulating. The temperature increases associated with the power increase reduced reactivity, preventing the power from increasing indefinitely.

The power would have continued to rise had the initial reactivity insertion been larger. The point of this example is not to demonstrate that reactor power cannot continuously increase (it well might), but to show that there is a loss in reactivity due to the increase in the temperatures of fuel and heavy water.

Commercial CANDU reactors respond similarly and normal reactor power regulation includes this effect. When a large power increase is required, the regulating system automatically ramps the power in a sequence of small steps. The combination of repeated small reactivity additions and reactivity losses as power rises produces a smooth power increase with very little intervention of the control system needed to limit power rise.

Compare this with what you expect from previous modules, where Δk was assumed not to change. (This is how a reactor behaves when it is super-critical at low power, where the heating effects are very small.) Even a small initial Δk giving a long reactor period eventually leads to a rapid power increase. Power increases faster and faster as time goes on. In the power range that produces significant heat, too fast an increase can lead to equipment failure. The regulating system would need to initiate the power increase by adding a positive Δk and then continually reduce Δk, actively limiting the rate of increase.

The advantage of reactivity loss with power increase is not clear-cut if there is an accidental insertion of a larger amount of positive reactivity. The rate of power rise is a bit slower, which might help. If the accidental insertion is not too big, power might eventually stop rising, but not necessarily below 100%. On the other hand, these effects, if significant, are likely to delay initiation of an automatic trip.

21.2 Temperature Coefficients

The temperature coefficient of reactivity is the change in reactivity per unit temperature increase. It may be positive or negative. It was negative in the example above, because a temperature increase caused a reactivity loss. The units of temperature coefficient are mk/°C.

Temperature changes occur more or less independently in the fuel, the heat transport system, and the moderator. There is a temperature coefficient of reactivity associated with each of these. The “self-regulating” characteristics illustrated above require the overall temperature coefficient of a reactor to be negative, but changes in fuel temperature are the most important in determining the overall coefficient.

The change in reactivity caused by coolant temperature changes is normally very small (and difficult to explain). Moderator temperature changes cause significant reactivity changes, but normal control keeps the moderator temperature constant. Furthermore, moderator heating and cooling are relatively slow (because of the large amount of D_2O), so the effect is not as immediate as the fuel reactivity effect. This course examines only the effects caused by the temperature changes in the fuel.

21.2.1 Fuel Temperature Coefficient of Reactivity

There are two main contributions to the fuel temperature coefficient:

1. Increasing fuel temperature increases resonance capture in U-238.

2. The ratio of fission to absorption in the fuel changes with fuel temperature. (The direction and size of the change depend on whether the fuel is fresh, or there is equilibrium fuelling).

We will look at both effects in turn.

Increased Resonance Absorption

The change in resonance absorption with temperature is always the most important fuel temperature effect.

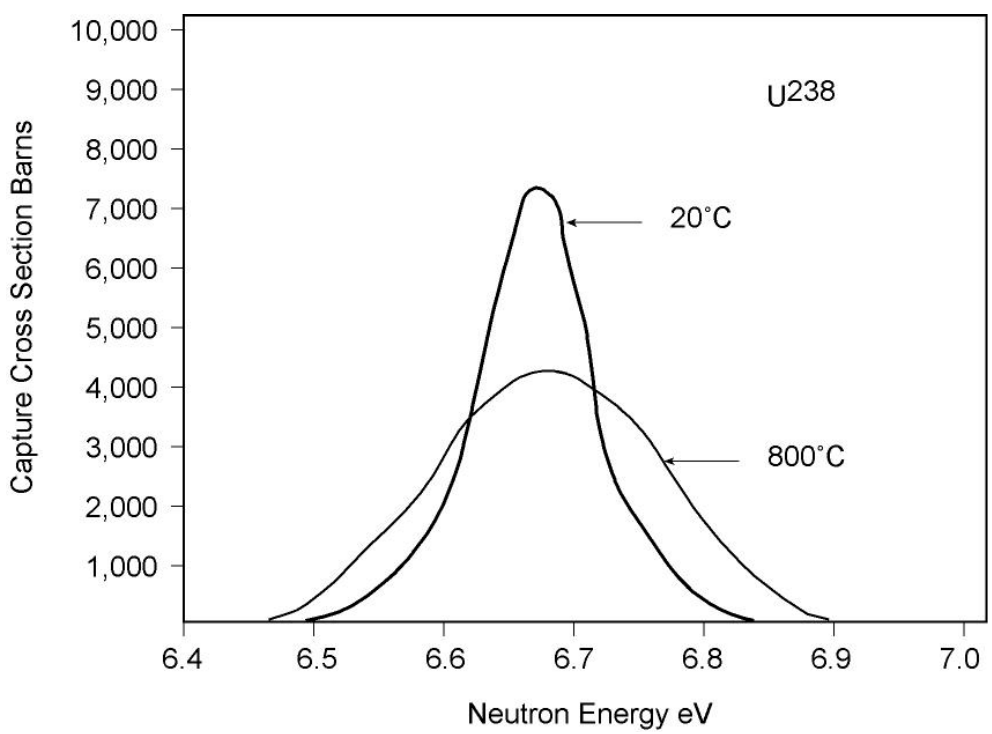


Figure 21.2
Resonance Broadening

Resonance capture accounts for about 10% of neutron loss in a CANDU core. The absorption of resonance energy neutrons in U-238 increases strongly as the fuel temperature increases.

The reason for increased resonance capture is as follows. The width and height of the resonance peaks in the U-238 cross-section depend on the temperature of the U-238. Figure 21.2 shows one particular resonance peak at 20°C and another at 800°C. At the higher temperature, (typical of the effective fuel temperature at high power) the peak is lower, but still high enough to capture almost any neutron in this energy range. At the same time, the resonance spreads over a wider range of neutron energies, exposing more neutrons to capture. Fewer neutrons entering the fuel while thermalizing are able to escape.

Ratio of Fissions to Absorptions
The variation of neutron cross-section with neutron energy has already been introduced. Temperature changes in the thermal neutrons' environment, which includes the hot fuel, are reflected in the thermal neutron energy. Cross-sections for thermal neutrons (apparent target size) get smaller at higher neutron speed, but not all cross sections change in the same way. You should expect the ratio of fission to absorption in the fuel to change with neutron temperature. For fresh fuel, where U-235 is the only fissile nuclide, this ratio goes down as thermal neutrons speed up. For equilibrium fuel, with a significant quantity of Pu-239, the ratio increases.

$$\left(\sigma_f^{fuel} \big/ \sigma_a^{fuel} \right)$$

For fresh fuel, as the thermal neutrons get "hotter", fewer of the neutrons absorbed in the fuel cause fission of the U-235. The opposite effect is observed for fuel with plutonium in it. A non-technical summary of this "hot neutron effect" may help you remember it. U-235 prefers cold neutrons; Pu-239 strongly prefers hot neutrons. Thus, for fresh fuel, increasing thermal neutron temperature decreases reactivity. For equilibrium fuel, reactivity increases with increasing thermal neutron temperature.

Combined Effect
As power rises, the fuel heats up. This broadens the U-238 resonances and heats up the thermal neutrons. The overall change in reactivity is a

combination of both effects. The resonance absorption effect is always larger than the "hot neutron effect".

For fresh fuel, the fuel temperature coefficient is obviously negative, since both effects are negative. A typical value is -0.013 mk/°C. For equilibrium fuel, the strong plutonium effect partly offsets increased resonance capture, reducing the magnitude of the fuel temperature coefficient to about -0.004 mk/°C.

21.2.2 Power Coefficient

Operationally the reactor power is measured and fuel temperature is not. In studying the behaviour of the reactor as power changes, it is handy to define the power coefficient. The power coefficient is the change in reactivity caused by temperature effects when power increases from zero power hot to full power.

A typical value for CANDU reactors is ≈ -5 mk, a decrease in reactivity. The exact value depends on the reactor design and fuel condition. For Bruce B the values are about -9 mk for fresh fuel and -3.5 mk for equilibrium fuel. The reactivity change is almost uniform in the high power range. A power coefficient of –3.5 mk suggests that a 10% increase in power (for example, ramping power from 80% to 90%) results in a loss of about 0.35 mk, equivalent to about 5% change in zone level.

21.2.3 Void Reactivity

If boiling occurs in a coolant channel, steam gradually displaces coolant. The name of this effect is voiding. Partial or total void in a channel affects resonance capture, parasitic absorption, fast fission, and leakage. We leave the explanation of these effects to a more advanced course.

The void reactivity is the change in reactivity for 100% voiding of all coolant channels. It is positive in CANDU reactors. The actual value varies from reactor to reactor but is about +10 mk for the Bruce reactors.

+10 mk is a very large positive reactivity, able to cause an unacceptably fast power rise. However, it is not ordinarily possible for all the coolant to flash rapidly to steam, even on a large pipe break. The safety systems are designed to detect and stop the power rise long before the +10 mk of reactivity is inserted. There are two independent, fast, automatic, safety shutdown systems to make sure a shutdown occurs.

21.3 Summary of Key Ideas

- CANDU fuel has a negative temperature coefficient. As the temperature of the fuel rises, negative reactivity is added to the core.
- The fuel temperature coefficient is comprised primarily from the broadening of U-238 resonance absorption peaks and the change in cross sections of nuclides in the fuel.
- The power coefficient is the total change in reactivity as the reactor power is raised for zero power hot to full power. In a CANDU, this coefficient is negative.
- The void coefficient of reactivity is the reactivity change in the core if the heat transport system voids. In a CANDU reactor this reactivity insertion is positive.

21.4 Assignment

1. Define:

    a) temperature coefficient,

    b) power coefficient,

    c) void reactivity.

2. Discuss the reasons why the fuel temperature coefficient is negative and why the magnitude is lower for equilibrium fuel than it is for fresh fuel.

3. Why is a negative temperature coefficient desirable?

22 Neutron Flux Control

If nothing were done to flatten the flux in our reactors, it would look something like Figure 22.1. The flux would be a maximum in the □efuel of the reactor (where neutrons are moving in from all directions) and decrease toward the boundaries (where neutrons are escaping into the shielding).

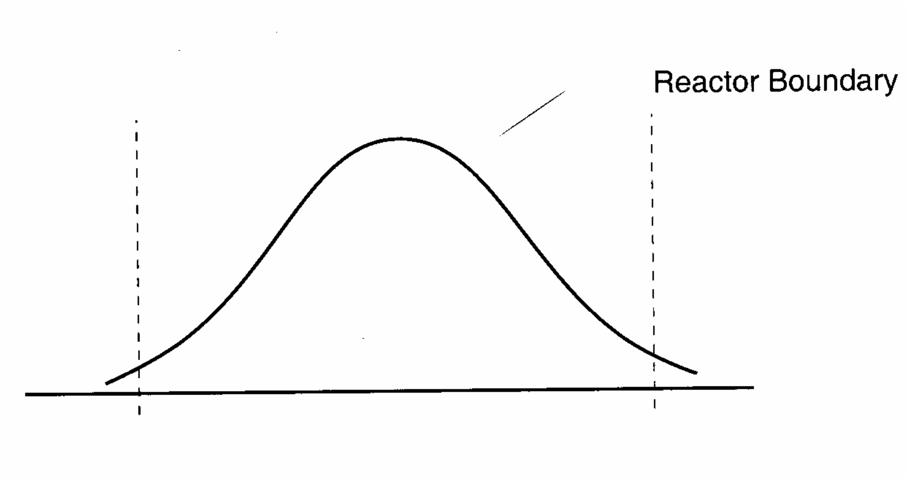


Figure 22.1
Unflattened Flux Distribution

With a distribution like this, the <u>average</u> flux is only about 30% of the maximum flux. The reactor would be producing 30% of the power it could produce if all the bundles were at the same power as a fuel bundle in the center of the reactor, operating at the maximum power it can safely produce.

The ideal flux distribution is perfectly flat ($\phi_{av} = \phi_{max}$). A perfectly flat flux distribution is impossible to achieve, but CANDU reactors have an average flux that is about 60% of the maximum flux. This module tells how to achieve this flux flattening.

Increasing the average flux without increasing the maximum flux has enormous economic benefits without compromising safe operation. For example, without flattened flux, Pickering NGS would be producing only half the power it now produces for roughly the same capital investment.

22.1 Reflectors

Previously the use of reflectors to reduce leakage was discussed, but that is only one advantage. A reflector also helps flatten the flux distribution in the radial direction. Figure 22.2 shows the flux distribution in a reactor without a reflector and with a reflector added. With the same maximum flux, (limited by the maximum allowed power level for the fuel) the reflector increases the average flux by returning neutrons to the low flux region, near the edge of the core.

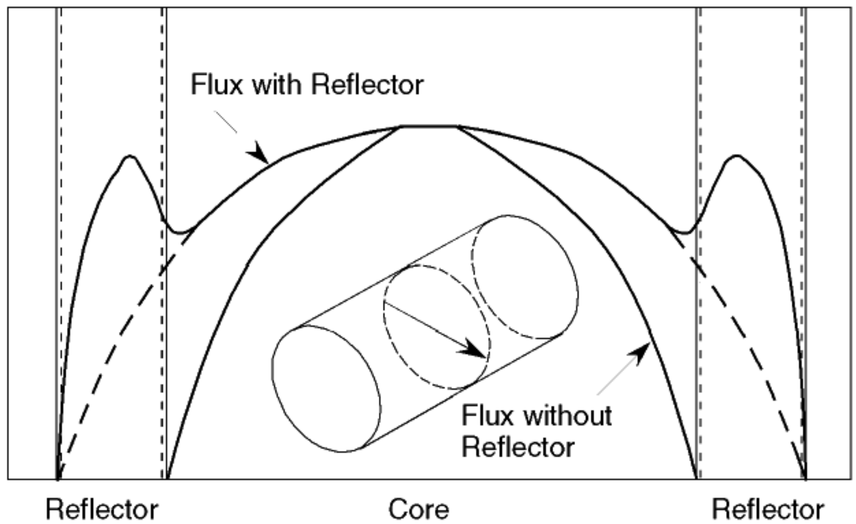


Figure 22.2
Effect of Adding a Reflector

22.2 Bi-Directional Fuelling

Fuelling adjacent fuel channels in opposite directions has a flux flattening effect in the axial direction. Figure 22.3 shows this.

We do not change all the fuel bundles when refueling a channel, so the newer fuel (at the input end of the channel) generates more neutrons than the highly burned up fuel at the exit end. Alternating the direction of fuelling in adjacent channels has the effect of raising flux towards the ends. The amount of flattening depends on the number of bundles fuelled on each visit to the channels. The less the better, from this point of view.

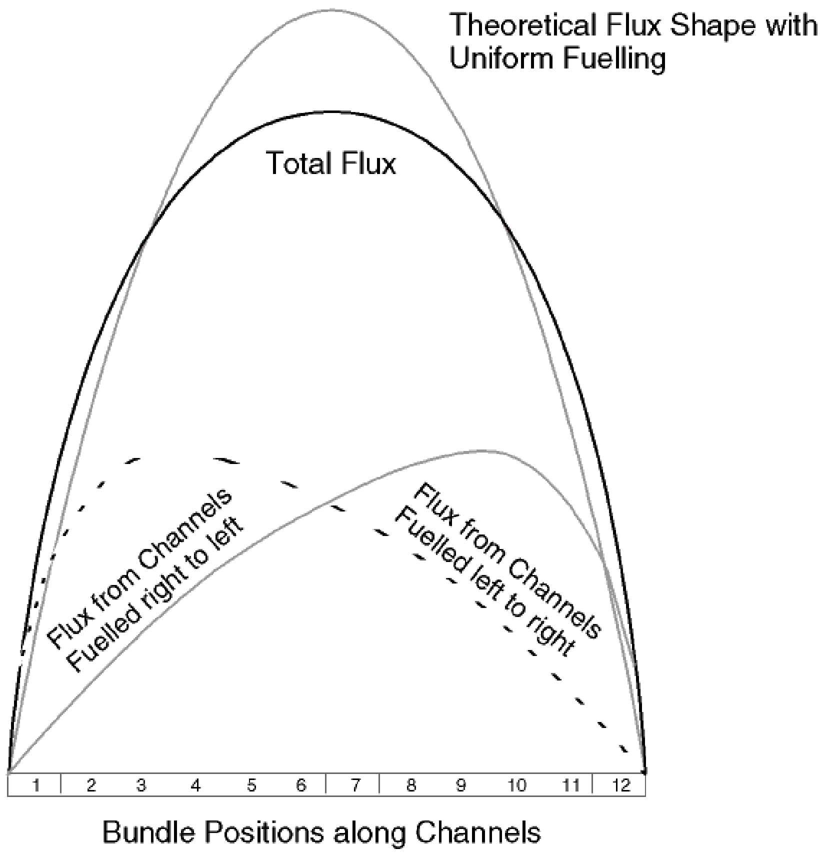


Figure 22.3
Effect of Bi-Directional Fuelling

22.3 Adjuster Rods

The normal position of adjuster rods is fully inserted in the central regions of the core. Thermal neutron absorption depresses or 'adjusts' the flux both radially and axially. Figure 22.4 shows the basic effect adjuster rods have on flux distribution. (Note: Bruce A reactors do not have adjuster rods).

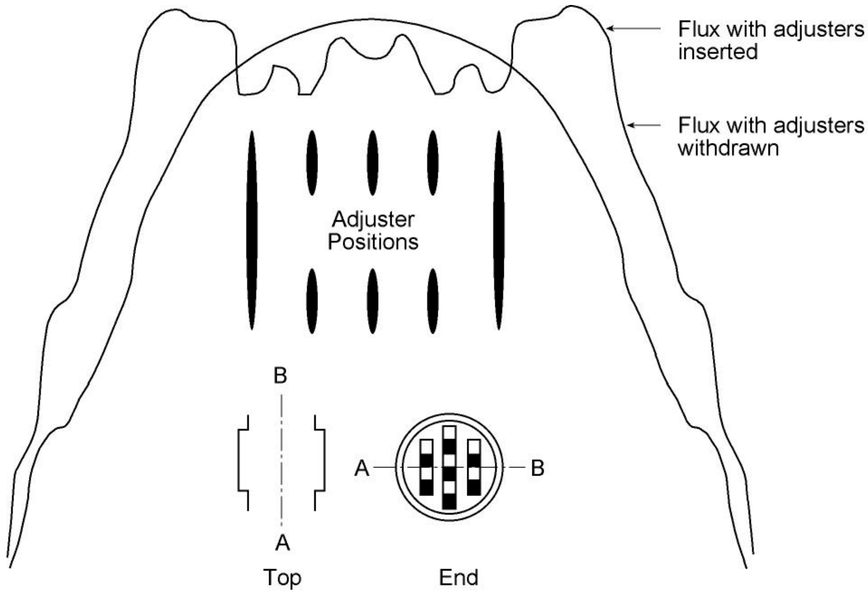


Figure 22.4
Effect of Adjuster Rods

Flux flattening with the use of these rods is quite effective but it does represent a loss in fuel burnup. We accept this because the benefits of increased power production greatly outweigh the higher fuel cost.

A few CANDU reactors use cobalt (Co-59) as the neutron absorbing material in the adjuster rods. The adjuster rods are replaced periodically, and the cobalt-60 is processed and marketed by MDS/NORDION. (Adjuster rods not used for cobalt production are made from mildly absorbing stainless steel.)

22.4 Differential Fuelling

Differential fuelling means that the bundles in the central channels are allowed to reach higher than average burnup while bundles in the outside channels are removed at lower burnup. The central bundles therefore generate relatively fewer neutrons from fission, because they contain fewer fissile nuclei than the outer bundles. Figure 22.5 illustrates this method of flux flattening.

This was the main method of flux flattening chosen for the Bruce A reactors, which formerly used boosters rather than adjusters for xenon override. The extra absorption in the fission products in the high burnup bundles in the central core plays the role of absorption in adjusters. This is more fuel-efficient than using adjusters, but some extra fuelling is required in the outer core to offset high burnup in the inner core and keep the reactor critical.

Daily on-power fuelling is planned by the fuelling engineer to maintain an optimum flat flux shape. This includes using differential fuelling, although reactors with adjusters require less differential fuelling to achieve the same result.

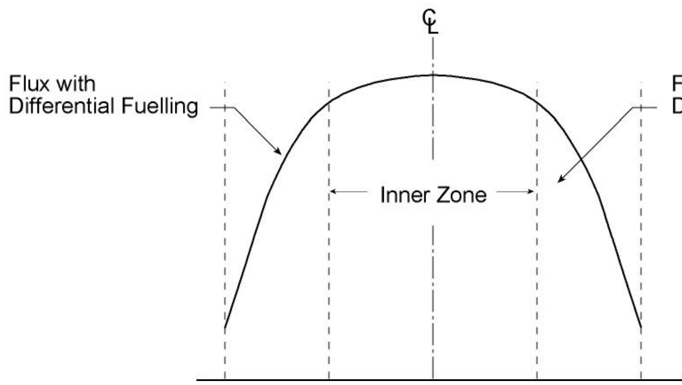


Figure 22.5
Effect of Differential Fuelling

22.5 Flux Oscillations

So far, we have assumed the flux distribution is static. Suppose now that without changing the total power of the reactor, the flux is increased in one region of the reactor. This typically results from refueling a channel. In the region of increased flux, the xenon now burns out more rapidly than it did before the change and its concentration decreases. This decrease in xenon concentration leads to a higher reactivity in this region, which, in turn, leads to another increase in flux. This again leads to increasing local xenon burnup, higher local reactivity, greater flux, and so on.

Meanwhile the control system is trying to hold bulk power constant so the flux away from the "hot spot" is lower than before. In the region of decreased flux, the xenon concentration increases due to reduced burnup while iodine continues to decay. This increased xenon concentration decreases the reactivity in this region, which reduces the flux, in turn increasing the xenon concentration, and so on. The thermal flux, and hence the power, decreases in this region while it increases in the other, the total power of the reactor remaining constant.

These local power excursions do not continue without limit. In the region of increased flux, iodine production increases. The production of xenon from iodine decay gradually increases and ultimately reduces the reactivity there. The flux and power eventually decrease. Similarly, in the region of reduced flux, the accumulated xenon eventually decays, increasing the local reactivity and reversing the flux and power transient in that region.

In this way, the flux and power of a reactor may oscillate between different regions (end to end, side to side, top to bottom) unless action is taken to control them. Calculations show that these xenon-oscillations (also called flux tilts) repeat themselves with a period of 15 to 30 hours.

Since xenon oscillations can occur at constant overall power, they would go unnoticed if we did not monitor the flux distribution at several points in the reactor. Such oscillations represent a hazard to the safe operation of a reactor. They could lead to unacceptably high channel or bundle power.

One purpose of the liquid zone system is to limit such oscillations. Fourteen light water compartments control the power distribution in fourteen zones of the reactor. Each zone has a pair of flux detectors that monitors the zone average power. The Digital Control Computer uses the signals to adjust the light water level in each zone control compartment, keeping power in each zone close to the average.

As an example of how light water zones may be used, look at Figure 22.6. Assume there are only two zones and a flux tilt is developing such that the flux in Zone I is increasing and the flux in Zone II is decreasing. By raising water level in the Zone I control compartment more neutrons are absorbed. Conversely, lowering the level in the Zone II compartment reduces neutron absorption in that zone. Thus, the action of the two zone control compartments returns the flux to a normal flat distribution.

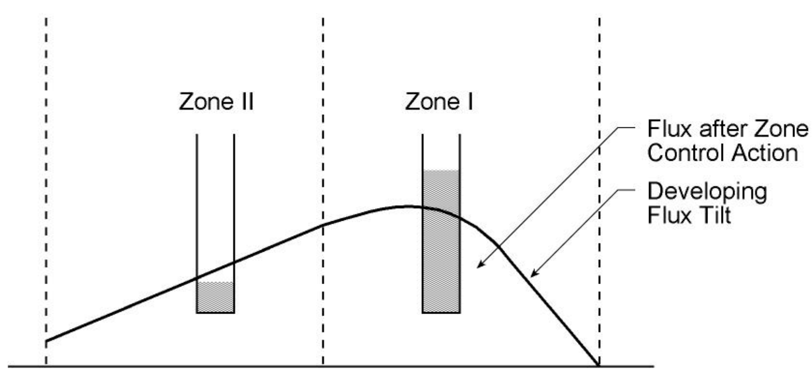


Figure 22.6
Zone Control System

22.6 Summary of Key Ideas

- Flux flattening provides better flux distribution through the core.

- Flux flattening is done in both the radial and axial directions in the core.

- Flux flattening methods used in CANDU reactors consist of reflectors, bi-directional fuelling, adjuster rods, differential fuel burn-up and liquid zones.

- Liquid zones are dynamic and dampen flux oscillations caused by xenon-135.

22.7 Assignment

1. List and briefly describe the five methods of flux flattening used in CANDU reactors.

2. Why is flux flattening desirable?

3. Explain how light water control zones are used to prevent flux oscillations.

23 Reactivity Mechanisms

Fuel fails if it is not kept wet. Fuel will also fail if it generates too much power. Reactivity mechanisms control reactor power. This module describes the reactivity control devices and how they work

Reactivity mechanisms change the value of the neutron multiplication factor in the reactor core. Most affect the parasitic absorption of thermal neutrons in the core. When a device increases the parasitic absorption in a core it is said to be adding negative reactivity. If a devices is decreasing the parasitic absorption it is adding reactivity.

Reactivity mechanisms are devices that increase or decrease neutron losses. Most such devices absorb neutrons. Some devices alter neutron leakage from the core. The reactivity worth of a device is the amount it can change Δk. Changing the core reactivity controls the reactor power output.

Power production uses up fissile atoms in the fuel. This decreases the reactivity of the core. On the short term, an increase in the number of neutrons can offset the loss of fissile atoms. Replacement of spent fuel with fresh fuel maintains the long-term reactivity of the core. Power production can continue for about a week without refueling.

These factors are summarized in Table 23.1. Fuel burnup causes a slow reactivity decrease. The other factors listed change reactivity more quickly. Temperature changes can alter reactivity in seconds or minutes. The effects listed as intermediate can change reactivity over several minutes. These reactivity variations, once started, may continue over hours or days.

Table 23.1 Factors Causing Δk Changes in the Core

Factor	**Time for Δk Change**
Fuel; Burn-up	Long
Refuelling Channel	Intermediate
Xenon Change Following a Power Change	Intermediate
Flux Oscillations	Intermediate
Temperature Change in the Core	Short

The reactivity control devices perform two general functions:

a) Reactor Power Regulation

b) Reactor Protection

Protective reactivity mechanisms, also called shutdown systems, have the single purpose of shutting down the reactor quickly in an emergency. Each unit has two independent shutdown systems. This makes certain that the reactor will shut down if required.

Reactor power regulation devices adjust reactivity to hold the reactor power steady at the demanded power output. The devices also respond to requests for a reactor power change. This could be a power change to match a change in electrical output or it could be a power reduction required because some system is unable to handle the heat the reactor produces.

23.1 Summary Of Key Ideas

- When Δk (reactivity) is positive, reactor power increases. When Δk is negative, reactor power decreases.

- Daily refuelling replaces the reactivity lost by fuel burnup, maintaining long-term reactivity of the core.

- Some reactivity mechanisms shut the reactor down quickly in an emergency. These are for reactor protection.

- Some reactivity mechanisms adjust reactivity in a controlled way to hold the reactor power at the demanded output. These devices can change reactor power. They are for reactor power regulation.

23.2 Reactivity Mechanisms

23.2.1 Fine Reactivity Control

The liquid zone control system provides fine control of k to regulate reactor power. This system holds the power at the demanded setpoint or changes it in a controlled way. It also operates to limit flux oscillations.

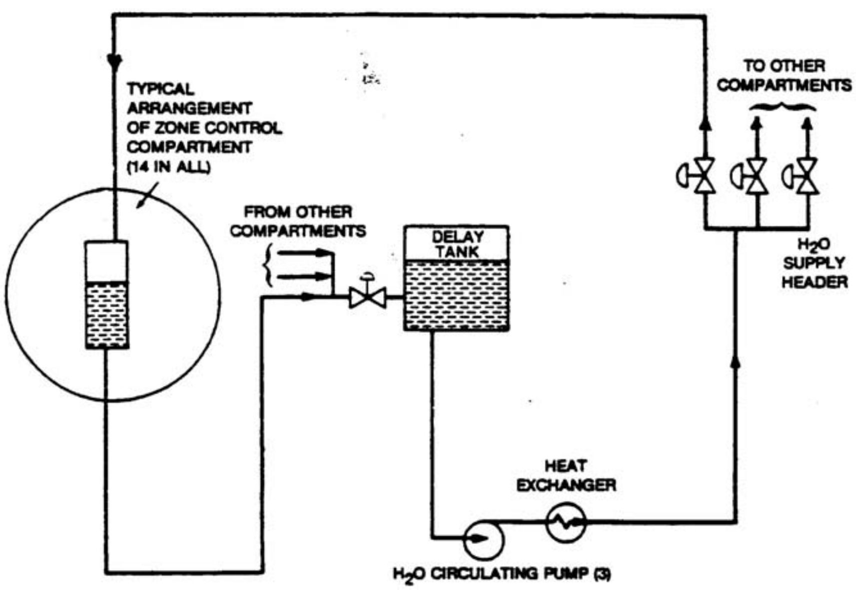


Figure 23.2
Liquid Zone control System (Simplified)

There are fourteen control zones in a CANDU core. Figure 23.2 shows how the liquid zone control system regulates reactivity in one of these regions. A variable level light water compartment sits near the centre of each zone. Signals from the control computer adjust the water flow control valves. This raises or lowers the water level from its nominal half full position. H_2O absorbs neutrons, so raising the water level in a zone compartment decreases reactivity. Reactivity increases when the water level drops.

Bulk reactor power is regulated by adjusting the H_2O level in all fourteen zone compartments together. Independent adjustment of the fourteen zone compartments smoothes out local flux variations. This is necessary to damp out flux oscillations. The Nuclear Theory course describes flux oscillations.

23.2.2 Coarse Reactivity Control

The liquid zone control system continually responds to power measurements and makes small reactivity adjustments. It cannot make large or rapid changes in reactivity. It completely fills or drains as it tries to respond to large reactivity requirements.

Neutron absorbing rods (control absorbers and adjuster rods) are regulating devices that can change reactivity by large amounts. Moderator level adjustments (used at Pickering A only) similarly are able to make large controlled reactivity adjustments.

Control Absorbers

These are neutron-absorbing rods made of cadmium tubes sheathed in steel. Their normal position is out of core. They drive into the core to reduce reactivity. (Reactivity increases as they come back out).

Figure 23.4 shows the control absorber rod equipment A guide tube guides the moving rod between the calandria tubes in the moderator. Cables attached to motor drives on the reactivity mechanism deck above the reactor raise and lower the rods.

These rods reduce reactivity to control power at the demanded value when zone levels cannot do it. They also reduce power gradually to a prearranged low level if certain equipment fails.

On some severe faults, these rods drop quickly into the core. This rapid power reduction is part of the normal power regulation function.

Absorbers are used at all CANDU reactors except Pickering A units.

Moderator Level Control (Pickering A Only)

Moderator level control is part of the regulation system on dump tank units. Figure 23.3 shows the regulating valves and dump valves. Neutron leakage from the core increases as the moderator level drops'. This reduces reactor power. (Reactivity increases as the calandria refills to its normal operating level).

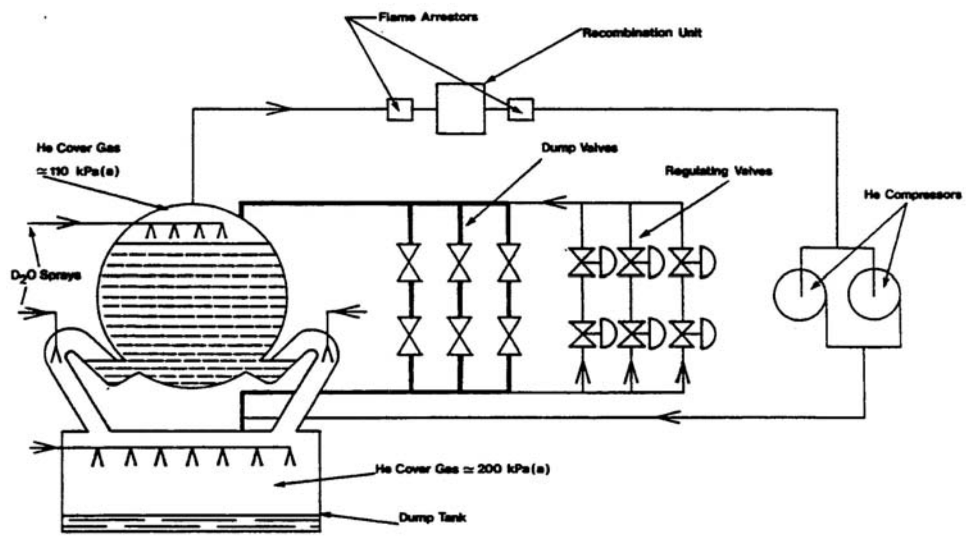


Figure 23.3
Moderator Dump and Level Control at Pickering A

The concepts of moderator level control and moderator dump were introduced previously. Dump tank pressure holds the moderator water in the calandria. The regulating valves adjust the moderator level by controlling the pressure difference between the dump tank and the top of the calandria. This controls the rate of heavy water flow from the calandria into the dump tank. (Pumps, not shown in figure 23.3, return water from the dump tank to the calandria).

Adjuster Rods

Adjusters rods are neutron-absorbing rods similar to the control absorbers in Figure 23.4. They are made of cobalt at some stations and stainless steel elsewhere. Stable cobalt 59 absorbs neutrons and becomes cobalt 60. This radioactive isotope is used to treat certain cancers. The sale of Co-60 partly offsets the cost of lost fuel burnup that results from the absorption of neutrons in the adjusters.

Their normal position is in core. In their normal position, adjusters flatten (that is, adjust) the neutron flux by absorbing neutrons in the central region of the core.

Another function of adjusters is to provide positive reactivity when the reactor regulating system requires it. Withdrawing adjusters from the core removes absorbing material and adds positive reactivity. They were originally designed to allow some ability to prevent poison outages by removing adjusters from the core as the xenon built up.

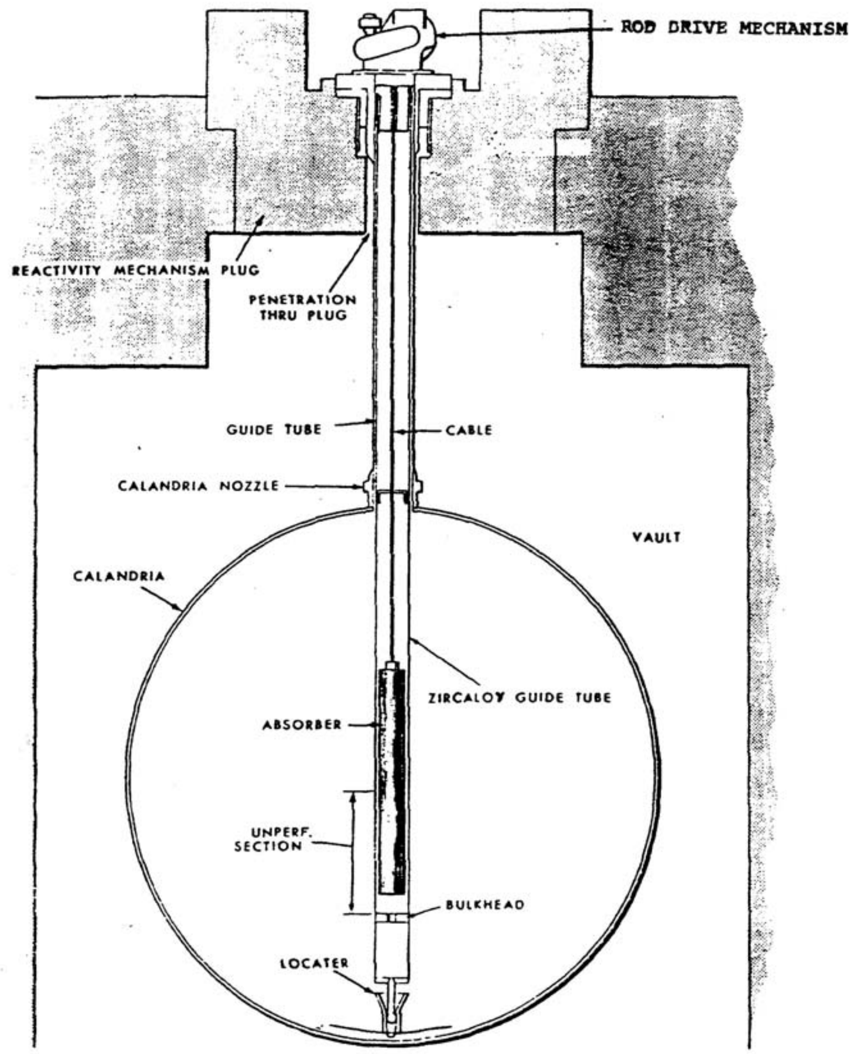


Figure 23.4
Control Absorber, Adjuster and Shutoff Rod Arrangement

A large positive reactivity insertion is required after a large power decrease. Power reductions increase the amount of neutron absorbing xenon in the fuel. The Nuclear Theory course describes these xenon transients. Adding positive reactivity to overcome absorption by xenon is called xenon override.

All CANDU reactors except Bruce A have adjusters. Bruce A was designed to use booster rods for xenon override, but these are no longer used.

23.2.3 Manual and Automatic Reactivity Adjustments

The reactivity device positions in normal reactor operation are: adjusters in the core, calandria full, absorbers out of the core and the zone compartments near half full. The regulating system requests the coarse regulating devices to operate when zone compartment levels are too high or too low. Most of these devices respond automatically.

Operators can prevent unnecessary automatic device movement if they keep the zone levels from deviating too far from middle of their operating range. Operators change liquid zone levels indirectly by manually adding neutron absorber to the moderator with the liquid poison addition system, or removing it with the moderator purification system. For example, when the liquid poison addition system is used to add poison, the regulating system holds reactor power steady by lowering the zone levels.

A combination of poison concentration adjustment and regular fuelling keeps the zone levels in their normal operating range.

23.2.4 Automatic Shutdown Systems

Instruments monitor reactor conditions such as heat transport system pressure, reactor power and coolant flow. Any measurement that shows possible risk of damaging the fuel, or other unsafe operating condition, triggers a reactor shutdown. A shutdown by a protective system is called a reactor trip. A trip occurs automatically whenever a trip parameter (measurement) exceeds its trip set point (safe operating limit). The operator can trip the reactor manually if necessary.

Shutdown system instruments and mechanisms are completely separate from devices used for reactor power regulation. CANDU reactors haves two separate shutdown systems for reactor protection. All units use shut off rods for Shutdown System 1 (SDS#1). All CANDUs except Pickering A use Liquid Poison Injection for Shutdown System 2 (SDS#2). Figure 23.5 shows these two common systems. Pickering A uses moderator dump as a backup to its shutoff rods.

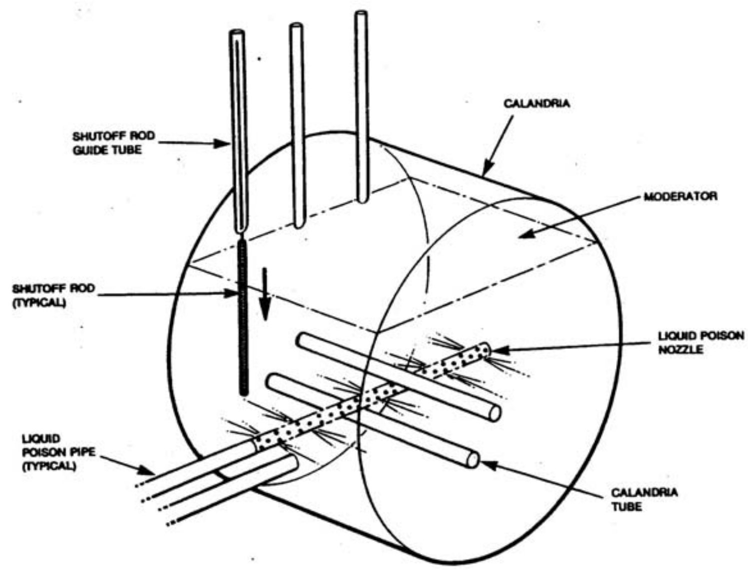


Figure 23.5
Sketch of Shutoff Rods and Liquid Poison Injection

Shutoff Rods
The SDS1 shutoff rods are neutron absorbing rods, nearly identical to the control absorber rods in Figure 23.4. An electromagnetic clutch holds them in their normal, poised, position in the guide tube above the core. A reactor trip signal cuts off the electricity to the clutches. The rods drop into core in about two seconds. Most stations use fast-acting, spring-loaded shutoff rods.

Liquid Poison Injection System
Figure 23.6 shows the liquid poison injection system (SDS#2). SDS#2 automatically injects a large quantity of neutron absorbing poison in a couple of seconds. Do not confuse this protective system with the liquid poison addition system, used for small manual additions of poisons.

A reactor trip signal opens the quick opening valves shown in Figure 23.6. High-pressure helium forces the gadolinium solution from poison tanks into the moderator. The poison enters through horizontal tubes, one tube per tank. Nozzles disperse poison into the calandria along the length of each tube. As a tank discharges, a floating ball rides the liquid surface down the tank and seals the discharge line. This prevents the helium from reaching the calandria and raising its pressure too high.

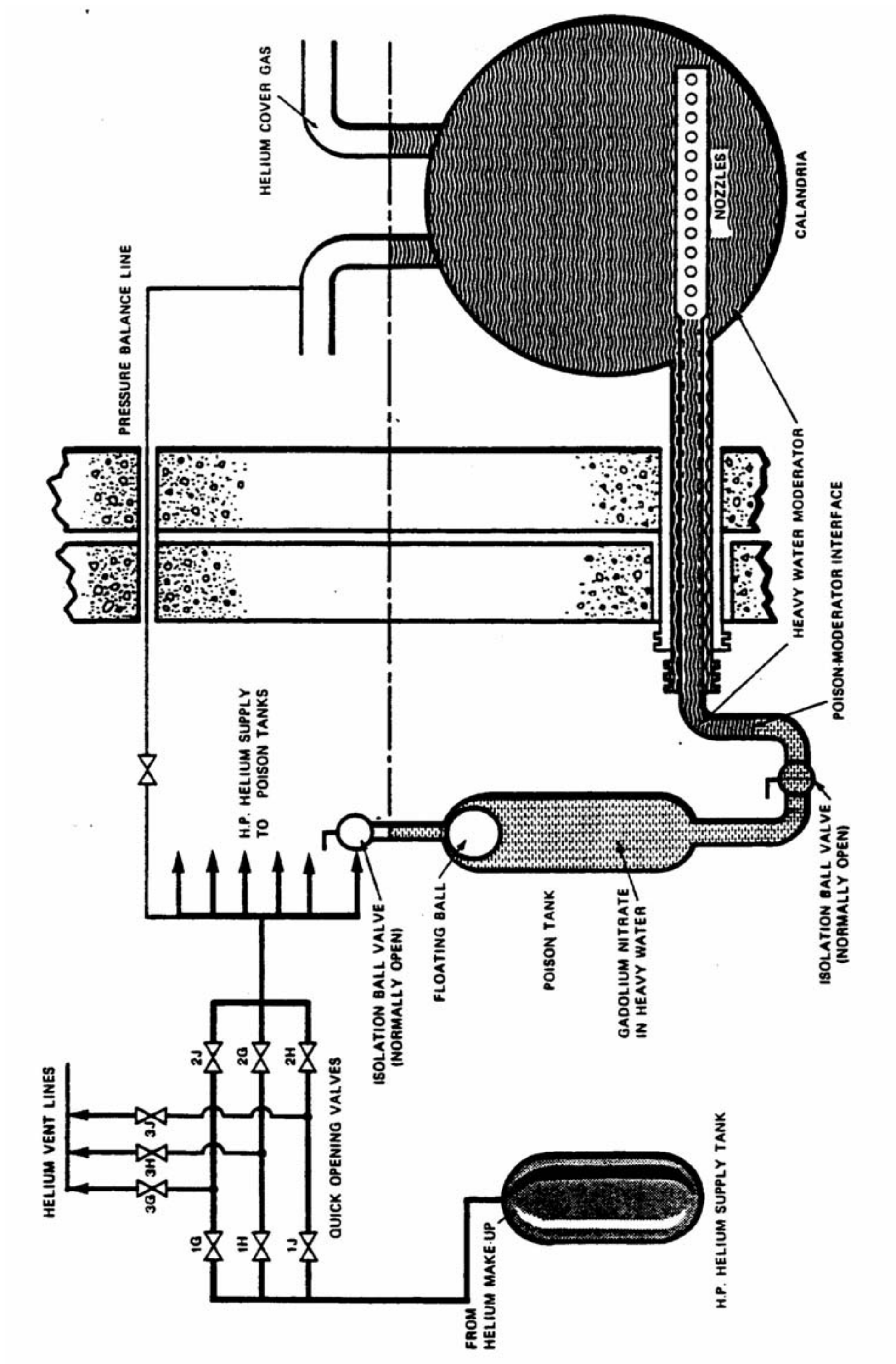


Figure 23.6
The Liquid Poison Injection System

Moderator Dump (Picketing A Only)
Figure 23.3 shows the moderator dump system. introduced in a previous module. On a reactor trip, the large dump valves open. This equalizes the pressure in the calandria and dump tank. Without dump tank pressure to support it, the moderator fails through the dump ports into the dump tank. As the level drops, neutron leakage increases. Without a moderator to slow them, fast neutrons leak away, or are absorbed without causing fissions.

23.3 Reactivity Mechanism Principles of Operation

Table 23.2 summarizes the principles underlying each of the reactivity mechanisms.

Table 23.2 Reactivity Adjustment Operating Principles

Operating Principle	Reactivity Mechanism
Change in the amount of fissile material in the core	Fuelling
Adjust Neutron Leakage from the core	Moderator Level Control
	Moderator Dump
Adjust Neutron Absorption in the core	Liquid Zone Control
	Control Absorbers
	Adjusters
	Shut Off Rods
	Liquid Poison Addition & Moderator Purification
	Liquid Poison Addition

23.4 Summary of Key Ideas

- The liquid zone control system regulates reactor power during normal operation. It regulates bulk power output and helps flatten out local power fluctuations.

- Sometimes effective reactor regulation requires larger or faster reactivity changes than the liquid zone control system provides. Adjuster rods provide reactivity increases. Absorber rods (moderator level at Pickering A) provide reactivity decreases.

- The operator can lower the zone levels indirectly by adding neutron absorbers with the poison addition system. The regulating system, automatically holding reactor power at setpoint, lowers the zone levels to keep the reactivity constant. Similarly, by removing poison from the moderator, the operator can indirectly increase the zone levels.

- Operating staff use fuelling, poison removal (purification) or poison addition to keep the zone levels from becoming too full or too empty.

- Absorbers (or moderator level) operate to hold power at setpoint when the liquid zones are too full to do so. They also can reduce power, in either a gradual reduction or a sudden drop.

- Xenon reactivity effects may be too large for the liquid zone control. Adjusters provide xenon override. Adjusters also have a second purpose. They flatten flux in the central region of the core.

- Shutoff rods, liquid poison injection and moderator dump (at Pickering A) provide rapid reactor shutdown in an emergency.

- Neutron absorbing shutoff rods drop into the core when a trip signal de-energizes the clutches. This is shutdown system one (SDS#l).

- High-pressure helium forces a neutron absorbing solution into the moderator when valves open between the helium tank and the injection tanks. This is shutdown system two (SDS#2).

- At Pickering A, large dump valves open to relieve pressure in the dump tank. The moderator D_2O falls out of the core into the dump tank. Without moderator, neutrons from fission escape from the core, or are absorbed without causing fissions.

23.5 Two-Out-Of-Three Trip System

The shutdown systems must stop the fission process quickly in an emergency. The reactor should shut down when required, but should not shut down unnecessarily. Spurious trips are expensive. After many SDS#l trips and all SDS#2 trips it is impossible to restart the reactor within the next 35 to 40 hours. Apart from the cost of replacement power, the sudden power reduction is hard on equipment.

The shutdown systems are made with very reliable equipment Maintenance programs and frequent testing make certain that the shutdown systems operate correctly.

An important part of the reliability of these systems is the tripping mechanism. Figure 23.7shows the two-out-of-three trip logic. Figures 23.3 (Moderator Dump) and Figure 23.6 (Liquid Injection Shutdown System) both show this valve arrangement The electronic contacts that open the rod clutches to trip SDS#l have a similar arrangement.

Consider the trip system in Figure 23.7. It has three helium lines. Each line has two valves in series. Three independent channels, labeled A, B and C send the trip signal to the valves. Signal A opens the two A valves, signal B opens the two B valves and signal C opens the two C valves.

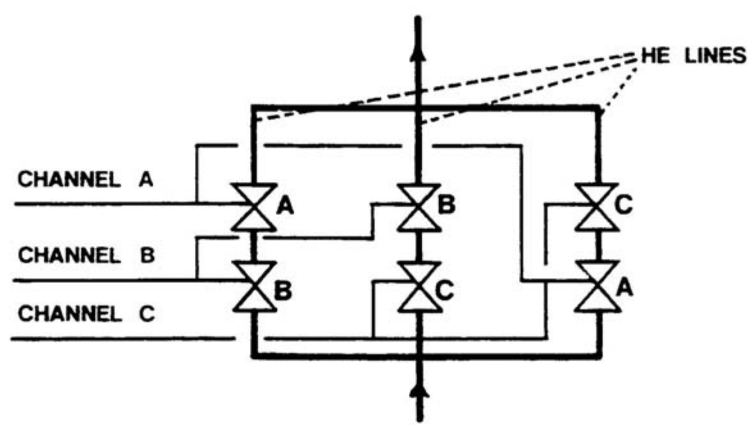


Figure 23.7
Typical Trip Logic

On a normal trip, all three channels simultaneously send trip signals. All valves open. Helium flows through all three lines, causing a shutdown by poison injection or by moderator dump.

The system also operates if only one or two helium lines open on a trip. Why then have three lines? There are three reasons for the arrangement of Figure 23.7

a) There is no reactor trip on a spurious signal in any one channel.

Suppose one channel fails, producing a spurious trip signal. For example, a fault in a signal transmitter in channel A could open the channel A valves. With just one set of open valves, helium cannot pass through any of the three lines. There is no reactor trip.

An unnecessary trip caused by this type of equipment failure requires simultaneous failures in two channels. Reliable equipment is used, and it is tested and maintained regularly. This makes a single fault unlikely. The chance of two channels failing simultaneously is extremely small.

b) A trip occurs even if one channel fails to respond to a real trip situation.

On a valid trip, if any one channel fails to provide a trip signal the system still operates. For example, a faulty transmitter in channel A could fail to send a signal to the channel A valves. The other four valves, operated by signals B and C, do open. Flow of helium through the line with no A valve will cause a reactor trip.

Again, reliable equipment that is carefully tested and maintained makes a single fault unlikely. Simultaneous failures in two channels, which could make the system fail, are extremely unlikely.

If one shutdown system does fail, the other shutdown system will shut down the reactor. Reactor shutdown in a real emergency is almost certain.

c) Two-out-of-three trip Logic allows for maintenance and testing at power without any loss of protection.

A single channel can be tested by tripping it to see if it works. This does not trip the reactor, provided testing is done on one channel at a time. There is no loss of trip coverage should a real emergency arise during testing. A trip signal on any other channel will trip the reactor.

There is an increased risk of an unnecessary shutdown, caused by a spurious trip on another channel during testing. This does not harm reactor safety, but it is expensive.

Some on power maintenance can be done, one channel at a time, with the channel tripped. In this state, a trip signal on either of the other channels will cause a shutdown. There is no loss of trip coverage should a real emergency arise during maintenance. Again, there is an increased risk of an unnecessary shutdown.

There is a safety advantage if equipment that fails causes a channel trip signal. Failures that cause the equipment to operate are called fail safe. The channel A transmitter failure example in a) above failed safe, causing the A valves to open. The transmitter failure in example b) was not fail safe. Nothing was observed until the system was needed, and then the A valves did not open.

A safety system built with components that fail-safe is more reliable. Suppose a real emergency occurs, producing trip conditions on all three channels. In the first example, because channel A is tripped, a signal on any second channel trips the reactor. In the second example, if either channel B or C fails to trip, the reactor will not trip.

When equipment fails safe, as in a), it is usually obvious that a failure has occurred. The operator can immediately order repairs. When equipment fails passively, as in b), the fault may not be noticed until it is discovered by routine testing. In the interval, the system is not quite as reliable. Frequent testing to discover faults improves reliability.

23.6 Summary of Key Ideas

- In a two-out-of-three trip system, the devices that trip the reactor are operated by channelized signals.

- The tripping devices and channelized trip signals are arranged so any two of three channelized signals will cause a reactor trip.

- A single equipment fault cannot trip the reactor. Two or three channel trips are required for a reactor trip. This limits the number of unnecessary shutdowns.

- An equipment single fault cannot prevent the reactor from tripping. The other two channel trip signals are enough to trip the reactor.

- Shutdown system maintenance and testing can be carried out one channel at a time without loss of trip coverage. The reactor tripping devices are activated in the channel under test or maintenance. Any one other channel trip signal will trip the reactor.

- Shutdown system equipment is very reliable. Careful maintenance keeps it that way so simultaneous equipment faults are rare. Testing finds single faults so maintenance can be done.

- Fail safe components are components that are designed to operate when they fail. Safety systems made from such components are more reliable. The fault is likely to be noticed immediately and quickly corrected, meanwhile, the system continues to provide full trip coverage even if the fault is not found and corrected.

23.7 Assignment

1. a) State two general functions of reactivity mechanisms.

   b) State two general principles of operation of reactivity mechanisms.

2. List five causes of reactivity variations other than reactivity mechanisms. Classify the effects as long, intermediate, or short term.

3. Make a table showing the function and principle of operation of each of the eight reactivity mechanisms.

4. Name the reactivity mechanism used especially for xenon override.

5. State 3 reasons for using a 2 out of 3 trip system.

6. Describe how the 2 out of 3 trip logic works to provide a reactor trip:

   a. When all the equipment is operating normally,

   b. When one channel fails to trip when required.

7. What is a fail-safe device, and how does fail-safe operation of safety system equipment contribute to reactor safety?

24 Emergency Coolant Injection & Containment

24.1 Introduction

Each CANDU has four special safety systems. The two shutdown systems, SDS#1 and SDS#2 have been discussed. This module introduces the emergency coolant injection system (ECIS) and the containment system. The plant design includes these four special safety systems to protect the public from a harmful radiation release.

In normal operation, five barriers stand between the main source of radiation and the public. The ceramic fuel pellets hold about 95% of the fission products. The fuel sheath contains the free fission product inventory. The heat transport system contains fission products released from failed fuel.

The containment system is the fourth barrier to release of radiation. Its purpose is to make and protect an envelope that holds radioactive material released from the heat transport system. This limits public exposure to radioactivity when the first three barriers fail.

The fifth barrier to public radiation exposure is the 1 km exclusion zone around each reactor core. This zone allows dilution of radioactive material that escapes from the containment system.

These five barriers protect the public if the reactor power is controlled, the fuel cooled and the radiation contained.

Rapid shutdown protects the first three barriers (pellet, sheath and piping). In most upsets, a rapid power reduction quickly matches the fuel heat output to available cooling. This limits the amount of steam generated. It keeps the system pressure from putting the piping at risk. The fuel stays wet and does not release fission products.

Some accidents do release fission products. For these accidents, rapid shutdown limits the fuel failures and radiation release. This limited release helps the two final barriers, containment and dilution, to do an effective job.

The emergency coolant injection system (ECIS) protects the fuel and heat transport system boundary when normal cooling fails. Its purpose is to refill the heat transport system and keep it full after a loss of coolant accident (LOCA). This sets up an alternative heat flow path for removing decay heat.

Both the emergency coolant injection system and the containment system must operate under conditions caused by a loss of coolant accident. Before we describe these systems, here is a brief description of a LOCA.

Coolant escapes from the heat transport system if pipes break, or pump seals fail. Various sizes of failure are possible.

For a small HTS break, the pressure regulating equipment keeps the heat transport system pressure normal. This is called a leak. The D_2O recovery system, is designed to cope with leaks. In a loss of coolant accident the heat transport pressure inevitably decreases by itself.

Here is an example. Just before noon on August 1, 1983, pressure tube G16 on Unit 2 at Pickering A suddenly split. Pressure in the annular space burst an annulus bellows. Coolant escaped slowly past the journal bearing through the break in the bellows.

Less than 1½ hours after the pressure tube ruptured, the operators had shut down the reactor, cooled the heat transport system and reduced its pressure. The operators controlled the HTS pressure reduction during this time. The operators managed a controlled shutdown, bringing in reserve D_2O from other units. None of the special safety systems was required. If they had left the system to itself, pressure would have dropped gradually on its own and special safety systems would have operated.

This example of coolant loss is near the boundary that distinguishes a LOCA from a leak. The pressure could not be kept high and would have fallen naturally, as in a LOCA. However, because pressure fell slowly; the operators controlled it, as in a leak.

During a LOCA, low-pressure produces steam in the heat transport system. If this condition persists, fuel fails and releases fission products through the break. A large pipe break (for example, a reactor header or pump suction piping) causes the poorest cooling conditions and the most fuel failures. In these accidents some coolant reverses flow to travel towards the low-pressure at the break. Depending on the break size and location, coolant flow stops briefly in some channels.

Emergency cooling limits the release, helping maintain the effectiveness of the final two barriers. For a small LOCA, rapid shutdown and ECIS operation may prevent any fuel failures.

24.2 Summary Of Key Ideas

- Four special safety systems protect the public from an accidental radiation release. These are SDS#1, SDS#2, ECIS and Containment.

- Rapid reactor shutdown matches heat output to available cooling, protecting the first three barriers to radiation release. These barriers are the fuel pellet, the fuel sheath and the heat transport pressure boundary.

- Rapid shutdown also helps the last two barriers (the containment system and the exclusion zone), by limiting the amount of radioactive material released.

- In a loss of coolant accident (LOCA), heat transport system pressure falls. Steam produced in the heat transport system impairs fuel cooling. On a large break, fission products escape into containment.

- The purpose of the emergency coolant injection system (ECIS) is to refill the heat transport system after a LOCA and keep it full. This sets up an alternative heat flow path for removing decay heat.

- The purpose of the containment system is to make and protect an envelope that limits the release of radiation to the environment.

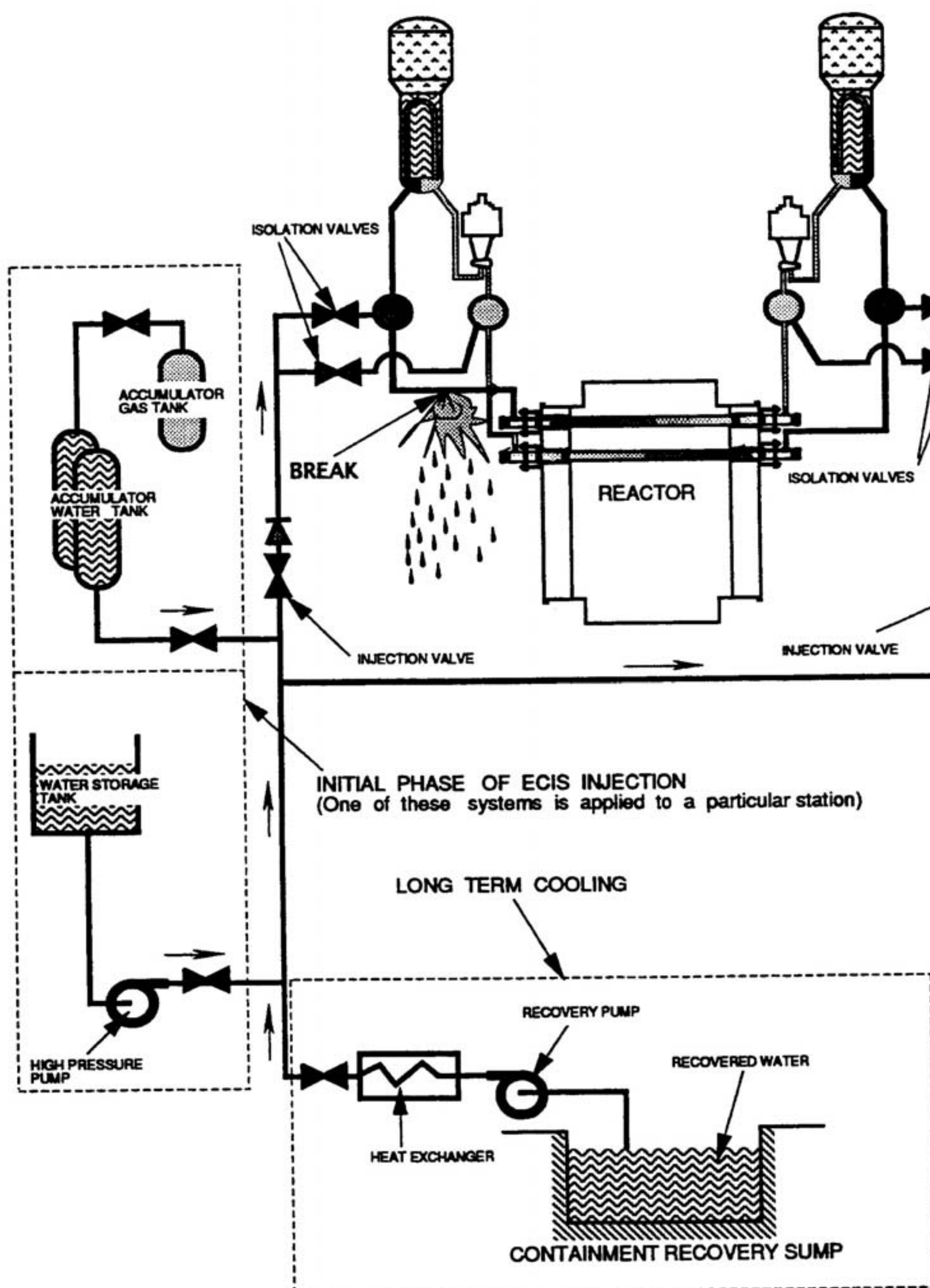


Figure 24.1
Simplified Emergency Cooling Injection System

24.3 Emergency Coolant Injection

On a LOCA, conditions in the reactor core or heat transport system trip the emergency shutdown systems. The emergency coolant injection system (ECIS) can remove decay heat, not fission heat. Without shutdown, high-pressure steam in the core may prevent effective injection of emergency cooling water. Heat removal will be inadequate and many fuel elements will fail.

Figure 24.1 shows the main equipment in a typical emergency coolant injection system. In a multi-unit station, a single system protects all the units. Figure 24.1 shows injection of cooling water into one unit. Some stations use gas pressure to push H_2O from the water tanks into the reactor. Some stations use high-pressure pumps for high-pressure injection.

Low HTS pressure automatically in conjunction with a second parameter triggers the emergency coolant injection system. The use of two parameters to initiate the system prevents spurious injections. The ECIS signal opens the isolation valves (also called injection valves) of the affected unit. These valves separate the ECIS H_2O from the coolant D_2O. The signal also connects the high-pressure source that forces the light water into the reactor inlet and outlet headers. Injection begins when the HTS pressure is lower than the ECIS injection pressure.

The ECIS injects light water because D_2O is too expensive to keep on hand for an emergency that should never happen. After a LOCA, upgrading of coolant would be one of many problems requiring solution before the unit could be returned to use.

The ECI systems in different stations are significantly different. In fact, there are two names used in the stations: ECI and ECC for emergency core cooling.

24.3.1 The Small LOCA

On a small loss of coolant accident, heat transport system pressure falls slowly because the leak rate is slow. The coolant begins to boil and the fuel channels gradually fill with steam. While the coolant boils, the system pressure falls only if the temperature drops.

For injection to start quickly, heat must be removed from the system. Large valves on the boilers open automatically to release steam, reducing the boiler temperature. The operator can open the steam valves from the control room if they do not open automatically. Crash cooling of the boilers causes rapid transfer of heat from the coolant to the boiler water. This quickly reduces the heat transport system temperature and pressure and injection begins. Cool water refills the system and rewets the fuel. The main heat transport pumps circulate the mixture of H_2O and D_2O to the boilers, which remove most of the decay heat.

Some steam and hot water escape from the break. This water collects in the ECIS recovery sump. If necessary, the recovery pumps return this water to the system to keep it full. The recovery heat exchangers cool the water before returning it to the HTS.

24.3.2 The Large LOCA

On a large loss of coolant accident, the main heat sink is the large amount of hot water and steam that escapes through the break. Pressure and temperature of the heat transport system drop quickly with or without boiler crash cooling.

The break is the lowest pressure point in the system. Coolant moves from the inlet and outlet headers towards the break. Water injected at the headers passes over the hot fuel and rewets it. A mixture of light and heavy water and steam escapes through the break.

For long-term fuel cooling, the operator sets up a recirculation cooling loop. The loop includes the heat transport system piping, a recovery sump, recovery pumps and recovery heat exchangers. Water cools the fuel and spills from the break to the containment floor. Hot water collects in the recovery sump. The recovery pumps return the recovered water to the reactor headers through the heat exchangers that cool it. This cooling loop can operate for an indefinite time.

On a big break the HTS cools down faster than the boilers. Crash cooling stops the boilers from dumping extra heat into the heat transport system.

The high-pressure water supply tanks may empty before water collects in the recovery sump. To protect against this, a low-pressure water supply injects water after high-pressure injection ends and before recovery begins. The low-pressure supply consists of the recovery pumps, drawing water from an emergency storage tank and pumping it to the reactor headers.

24.4 Summary Of Key Ideas

- SDS#l or SDS#2 shut down the reactor on a LOCA. This permits effective ECIS operation.

- ECIS equipment includes a high-pressure water supply, a low-pressure water supply, a recovery system and isolation/injection valves.

- Gas pressure supplies high-pressure injection to the reactor headers at some stations. Other stations use high-pressure pumps.

- The low-pressure water supply to the headers uses the recovery pumps, drawing water from an emergency storage tank.

- For long term cooling, recovery pumps deliver water to the headers from the recovery sump via the recovery heat exchangers.

- ECIS operation begins automatically when HTS pressure fails. The ECIS signal opens the isolation/injection valves and opens the valves that connect the high-pressure source.

- When the ECIS system triggers, large steam valves on the boilers open automatically to begin crash cooling. This is particularly important on a small LOCA, where the boilers are the main heat sink. The break is the main heat sink on a large LOCA.

- The control room operator can start ECIS operation and crash cooling from the control room if they do not trigger automatically.

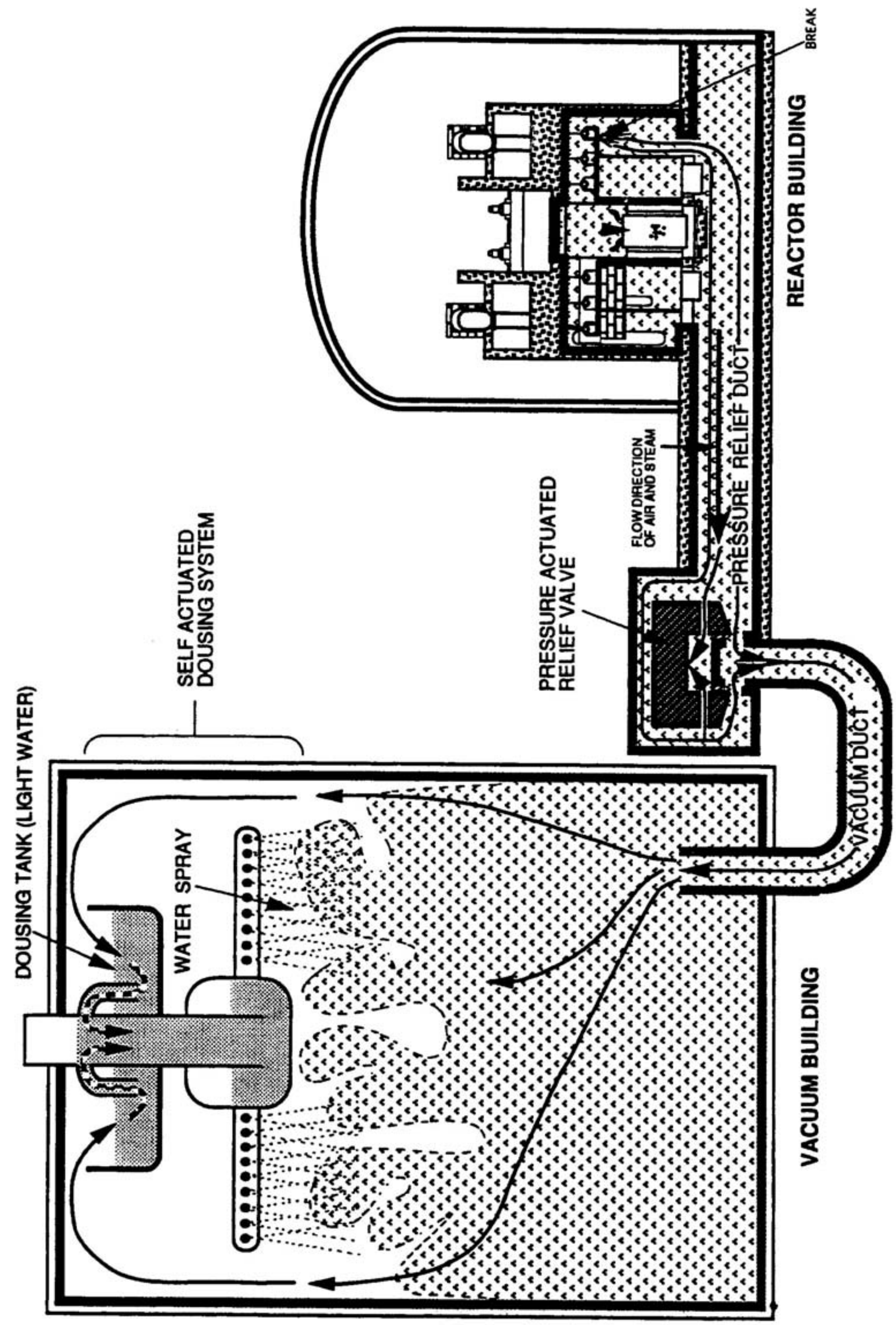


Figure 24.2
The Vacuum Containment Concept

24.5 Containment

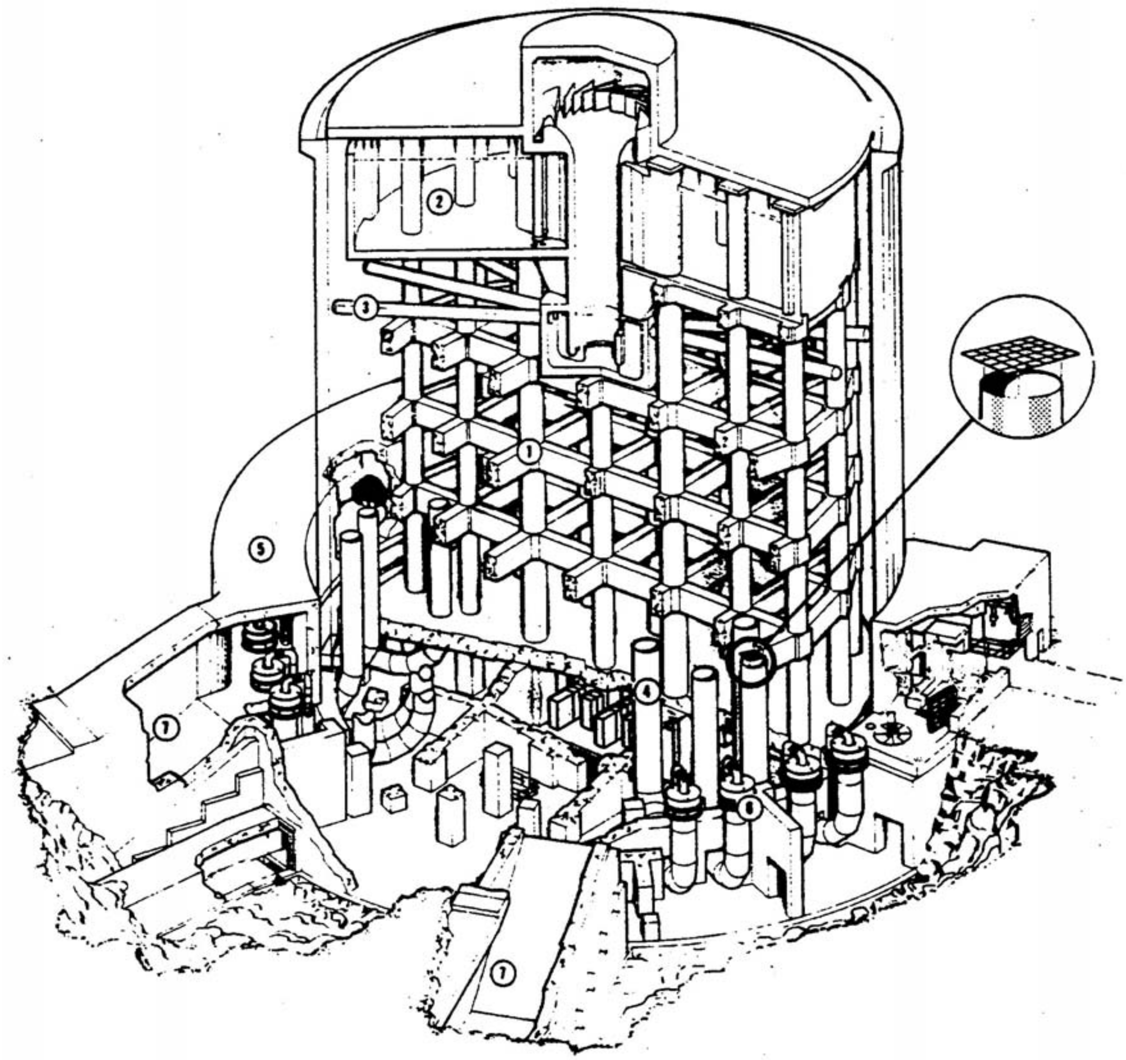
The containment system is actually a group of systems. First there is the envelope itself. This surrounds all nuclear systems that could release radiation. On a LOCA signal, the containment envelope boxes up or buttons up. That is, penetrations through the envelope close to prevent escape of radioactive material. Various energy sinks protect the envelope from over pressure. A dousing system condenses steam and cools the containment atmosphere. Air coolers remove heat. A clean air discharge system can be used to filter and discharge air, allowing relief of high-pressure in the building.

There are two different containment designs for CANDU reactors: the pressure suppression containment system and the negative pressure containment system.

Single unit stations use a pressure suppression containment system (Figure 24.6). The containment structure, which is the reactor building itself, complies with the pressure vessel code standards. It has a very low leak rate under pressure.

Units in a multi-unit station share a vacuum building. The vacuum building is part of a negative pressure containment system. Figure 24.2 shows that the vacuum building and reactor building are each part of the containment envelope. After a LOCA, pressure inside this envelope is sub-atmospheric, preventing leakage out.

Negative pressure systems and pressure suppression systems both limit radiation release to the public. The cost of negative pressure containment is reasonable when several units share a vacuum building. Pressure suppression containment requires reinforcement of each unit. This is less expensive than a vacuum building for a single unit. It is less cost effective for a multiple unit station.



1 Internal Structure
2 Emergency Water Storage Tank
3 Distribution and Spray Headers
4 Vacuum Duct
5 Valve Manifold
6 Pressure Relief Valve
7 Pressure Relief Duct

Figure 24.3
Vacuum Building, Relief Valves and Dousing

24.5.1 Negative Pressure Containment

Figure 24.4 shows a pressure relief duct connecting four units to a vacuum building via several vacuum ducts. (The pressure relief duct is on the reactor building side of the relief valves. The vacuum ducts are on the vacuum building side of these valves.) In some stations, the connecting duct runs underground and is not visible. Figure 24.3 show this arrangement, with the relief valves in a valve manifold surrounding the vacuum building.

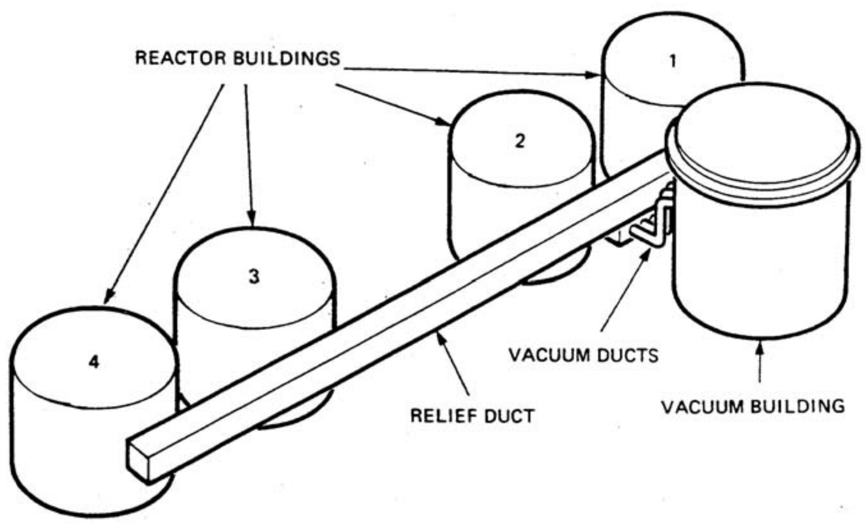


Figure 24.4
Multi-Unit Containment

Some stations with negative pressure containment systems put as much equipment as possible inside the containment envelope. This reduces the number of penetrations but results in a large containment volume. A large containment volume requires a large vacuum building.

Other stations place as much equipment as possible outside containment. This gives better equipment access and reduces the containment volume. It also increases the number of containment penetrations where radiation could leak. Figure 24.5 shows these differences.

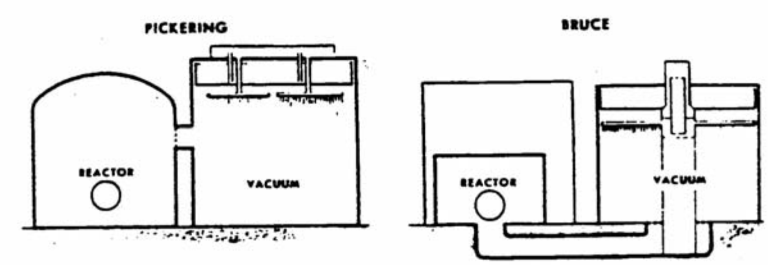


Figure 24.5
Sizes of Containment for Pickering & Bruce Stations

On a LOCA, pressure inside containment rises. High-pressure lifts the relief valves that connect the pressure relief duct to the vacuum building via the vacuum ducts. Heated air and steam escape into the low-pressure vacuum building. Increased pressure forces water from the dousing tank into the dousing spray headers. The dousing spray cools the air and condenses steam. Within a minute or two the pressure inside containment is again lower than the pressure outside.

The high-pressure of the accident opens the relief valves and starts dousing. A LOCA signal boxes up the containment envelope and shuts off the system that draws vacuum in the vacuum building.

In addition to the valves that are opened by the pressure of a LOCA, there are large and small control valves that modulate for long-term pressure control.

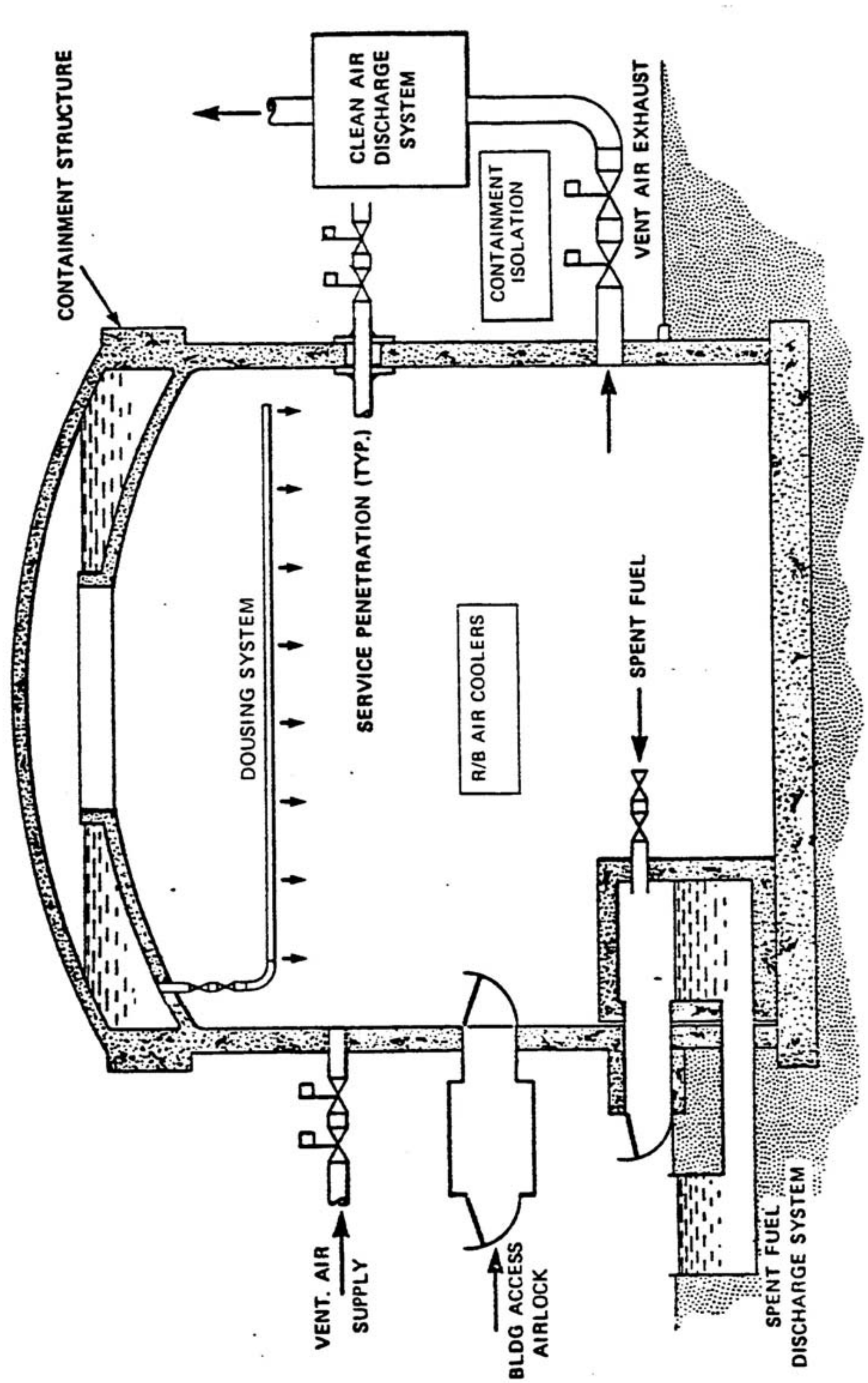


Figure 24.6
A Pressure Suppression Containment System

24.6 Pressure Suppression Containment

Figure 24.6 shows a pressure suppression containment system. The reactor building walls are reinforced concrete, lined with an epoxy liner. The leak rate from this structure is low if the pressure does not rise too high.

The dousing water tank is in the top of the reactor building. A LOCA signal boxes up the containment structure and opens the dousing valves. Dousing removes heat and condenses the steam. This helps lower the pressure and keeps the leak rate small.

24.7 Summary of Key Ideas

- The containment system is a group of systems. First there is the containment envelope and the system that boxes it up. Then there are energy sinks (dousing, air coolers and clean air discharge) that protect the envelope from damage.

- Both negative pressure containment systems and pressure suppression containment systems protect the public from release of radiation. Single unit stations use a pressure suppression system. Multi-unit stations share a vacuum building containment system.

- Negative pressure containment uses a vacuum building. Pressure relief into the vacuum building combines with dousing to drop pressure. Inside the containment envelope, the pressure is soon sub-atmospheric. This stops radiation leakage to the environment

- Pressure suppression containment uses a lined, reinforced concrete structure. This structure has a low leak rate when under pressure. Dousing helps keep a low leak rate.

- With negative pressure containment, pressure from the LOCA opens the pressure relief valves and causes dousing. The LOCA signal boxes up the envelope and turns off the vacuum system.

- With pressure suppression containment, the LOCA signal triggers box-up and dousing.

24.8 Assignment

1. What are the four special safety systems?

2. How does rapid shutdown of the reactor help:

   a. the emergency coolant injection system do its job?

   b. the containment system do its job?

3. Describe how the emergency coolant injection system helps prevent the escape of radiation to the environment.

4. List the various systems that make up the containment system.

5. Describe the way each system in question helps protect the public from radiation exposure.

6. Give the major differences between a vacuum building containment system and a pressure suppression containment system.

7. Outline the immediate response of the emergency coolant injection system and the containment system to a large LOCA.

8. Describe the medium and long-term response of the emergency coolant injection system when there is a LOCA.

25 The Conventional Side of the Station

25.1 Introduction

The basic similarity between typical electric generating stations is the conversion of shaft mechanical power to electrical power in a generator. The major difference comes from the method used to produce the shaft mechanical power. There are four principle methods of obtaining this shaft power in general use:

- hydraulic turbines,
- fossil fuel steam turbines,
- nuclear steam turbines, and
- gas turbines.

This module covers nuclear steam turbines and the different systems involved in the energy conversions. Electric power production requires the transfer and conversion of heat energy to mechanical energy. This is achieved by two basic energy transport systems: the heat transport system and the steam/feedwater system. These systems are sometimes called the primary heat transport system and the secondary heat transport system. The following sections discuss the steam/feedwater cycle, the equipment involved, energy transfers, control, and auxiliary equipment required. The values of pressure, temperature, etc., quoted in the text refer to full power operation and are approximate. Real values differ slightly from station to station.

25.2 The Boiler (Steam Generator)

During normal operation, the heat transport system transfers heat from the reactor to the secondary coolant by way of the boilers. The boilers thus act as the principal heat sink for the reactor. Reactor heat is transferred from the HTS to the boiler feedwater. As a result, the boiler produces steam to drive the turbine. To understand the production of steam in the boilers, it is useful to introduce some terminology.

Figure 25.1 shows the effect of adding heat to a kilogram of water. Adding more heat to the water will increase its temperature until it reaches the boiling point. At this point we say that the water has reached the saturation temperature. As we add more heat, latent heat of vapourization is added causing the water to boil. From this point, the water temperature does not change. The ratio of steam to water shifts to a higher steam percentage as more heat is added. The steam, being lighter than water, rises, leaving the water behind. This steam, which is free of moisture, is called saturated steam. Applying more heat to the saturated steam will increase the temperature above the boiling point.

When the steam temperature is above the boiling point, we have superheated steam.

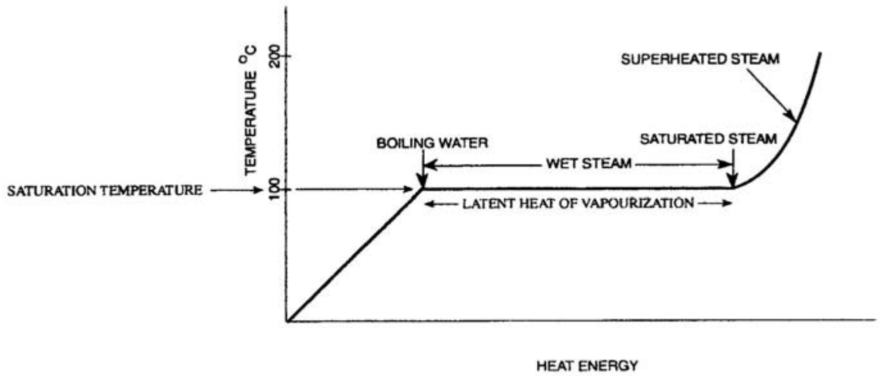


Figure 25.1

Figure 25.2 shows the effect of increasing the pressure in the boiler to 4000 kPa. Boiling does not occur until the temperature reaches 250°C. Saturated steam at about 250°C and 4000 kPa is typical in CANDU nuclear boilers. The temperature and pressure of the steam indicate its energy content. The higher the temperature and pressure the greater the heat energy will be. In boiler operations it is very important that steam pressure is maintained.

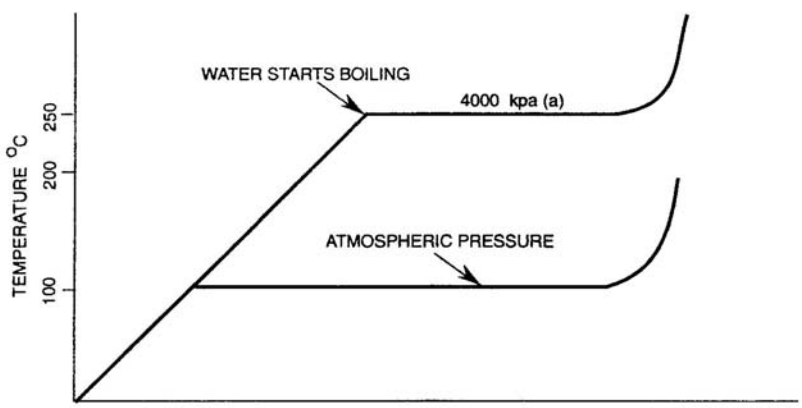


Figure 25.2

Figure 25.3 shows a nuclear power plant boiler typical of those used in large generating stations. Hot, pressurized heavy water enters the boiler and passes through the tube bundle. The heavy water inside the tube is hotter than the feedwater around the tubes. This allows heat transfer from the heavy water to the feedwater, causing the feedwater to boil.

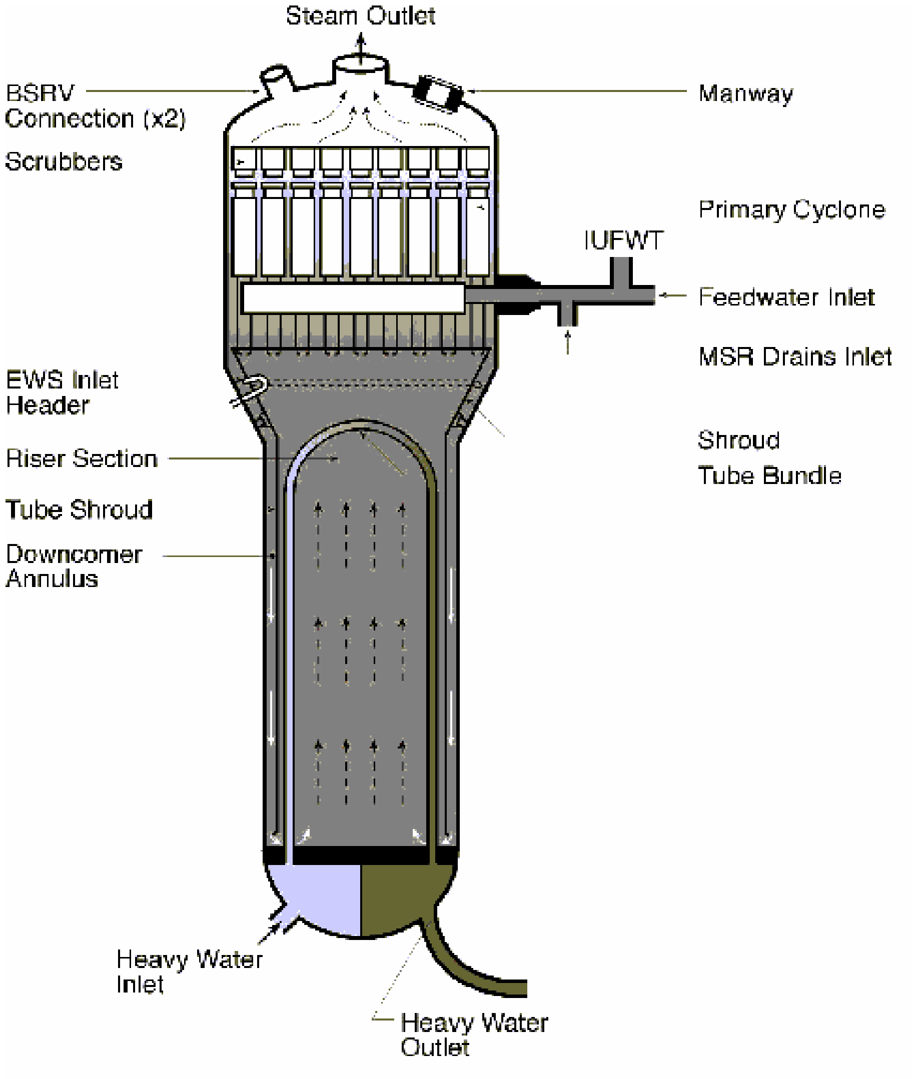


Figure 25.3
Typical Boiler

The steam leaving the top of the tube bundle is about 90% water. To prevent damage to the steam piping, valves and (most important) the turbine, only dry steam must leave the boiler. Cyclone separators, located above the tube bundle, dry the steam by giving the steam/water mixture a swirling centrifugal motion. The water, being denser than steam, moves to the outside area of the separator and is drained off. The steam that leaves the top of the cyclone separators has low moisture content but is still unacceptable for use in the turbine. The steam scrubbers, located above the cyclone separators, remove the last traces of moisture.

Water separated from the steam in the cyclone separator and steam scrubber drains to the outside of the boiler's tube shroud. The water flows down to the bottom of the boiler through the downcomer annulus and re-enters the tube bundle area enabling it to generate more steam. The amount of water cycling through the tube bundle, through the downcomer, is typically ten times as much as feedwater entering the boiler.

The water in the boiler moves through natural circulation without the use of pumps. The water and steam in the tube bundle move upward because of the decrease in density due to the addition of heat. The water that comes out from the cyclone separators is relatively dense, because it has no steam bubbles, and falls down the downcomer to begin the cycle again.

Simply, the feedwater flow in the boiler starts from the preheater. The preheater heats the feedwater to near saturation temperature. Inside the boiler, the feedwater circulates up around the tube bundle and down the downcomer many times while acquiring the latent heat of vapourization, and finally leaves the boiler as nearly saturated steam.

25.3 The Steam/Feedwater Cycle

The major functions of the steam/feedwater cycle are to provide cooling for the HTS and to convert heat into mechanical energy for the generator. This system operates as a continuous loop of demineralized light water. The major systems are the steam system and the boiler feedwater system.

25.3.1 The Steam System

Figure 25.4 shows a simplified schematic of the steam system and components typical of a large turbine unit. Safety valves protect the steam system components from over pressure. The pressure from the boilers drives the steam to the high-pressure (HP) turbine. On route to the turbine, the steam travels through several valves. Two, of interest, are the emergency stop valve and the governor valve. The governor valve varies the electrical output from no load to full load by controlling the quantity of steam flowing to the turbine. Before reaching the governor valve the steam passes through the emergency stop valve. The emergency stop valve quickly stops the steam flow to the turbine in the event of an emergency that could damage the turbine.

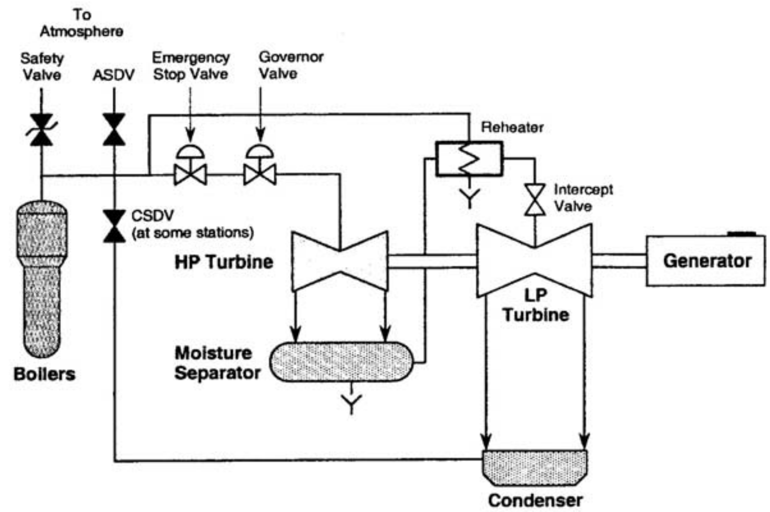


Figure 25.4
Steam Cycle

From the governor valve the steam passes through the HP turbine. The HP turbine converts the latent heat of the steam to mechanical energy. As the HP turbine uses the latent heat in the steam, the steam becomes wet (moist). Moisture content of more than 10% will cause excessive erosion on the turbine blades. Removing the moisture in the steam allows further conversion of the remaining available energy. The HP/LP arrangement of the turbine provides an opportunity at this stage to improve the quality of the steam to allow more energy to be converted without risk of damage to the turbine.

Steam leaves the high-pressure turbine at approximately 900 kPa and 170°C at 10% moisture. It passes to the moisture separator, which removes the moisture in the steam. Steam leaving the moisture separator has the same temperature and pressure as that at the turbine outlet but without moisture. It then passes through a reheater to heat the steam. This increases the work that the steam can do in the Low-pressure (LP) turbine. The reheater uses steam directly from the boiler to heat the steam from the moisture separator. The steam leaves the reheater in a superheated condition at about 230°C and 900 kPa. Before entering the LP turbine, the steam passes through intercept valves. In a fashion similar to the emergency stop valve, these valves shut off steam to the LP turbine in an emergency. Steam passes through the normally open intercept valve, passes through the low-pressure turbine, and is then exhausted to the condenser at approximately 5 kPaa (absolute pressure), 35°C and 10% moisture.

Stopping the flow of steam to the turbine results in increased boiler pressure. This can happen on a turbine trip, when the turbine is stopped due to mechanical failure. Reducing reactor power and getting rid of the steam prevents excessive boiler pressure build up. Adjusting the reactor power level too low can poison out the reactor. However, if the power level is kept above 60% full power, the reactor can keep operating. Providing an alternative heat sink, while operating at this power level, will prevent a boiler pressure increase. The alternate heat sink can be provided by blowing the steam to atmosphere or directly to the condenser. All CANDU units have large steam reject valves able to discharge steam either to the atmosphere or to the condenser with the reactor at 60% FP. They are also equipped with smaller steam reject valves that are able to discharge steam to the atmosphere at the decay heat power level, if the condenser is unavailable. The valves that direct steam to the condenser are called condenser steam discharge valves or CSDVs. The ones that send steam to atmosphere are ASDVs, an acronym for atmospheric steam discharge valves.

25.4 The Steam Turbine

The turbine uses steam from the boiler. It converts steam pressure to rotational energy. This conversion involves transformation of the heat energy (pressure) of steam into high velocity steam through fixed nozzles. A nozzle is a device that converts the heat energy of the steam to this high velocity kinetic energy. The fixed nozzles form the turbine fixed blades. The high velocity steam directs its kinetic energy on to the moving blades forcing them to move (rotate). Figure 25.5 shows how the high velocity steam leaves the fixed nozzles and drives the moving blades.

From the first set of fixed and moving blades, the steam then moves through succeeding sets to repeat the process of energy conversions. A set of fixed blade nozzles and moving blade constitutes a turbine stage. It is common to use a number of stages in a turbine to convert the useful heat energy in the steam into mechanical energy. The moving blades are attached to a blade wheel, as shown in Figure 25.6, and the blade wheel is mounted on the rotor shaft. The high velocity steam leaving the nozzle drives the wheel, which in turn rotates the shaft.

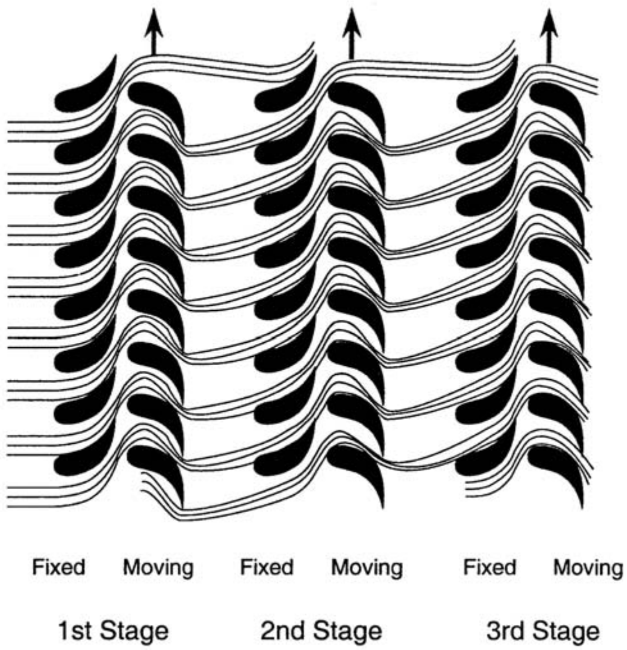


Figure 25.5
Fixed and Moving Blades Arrangement

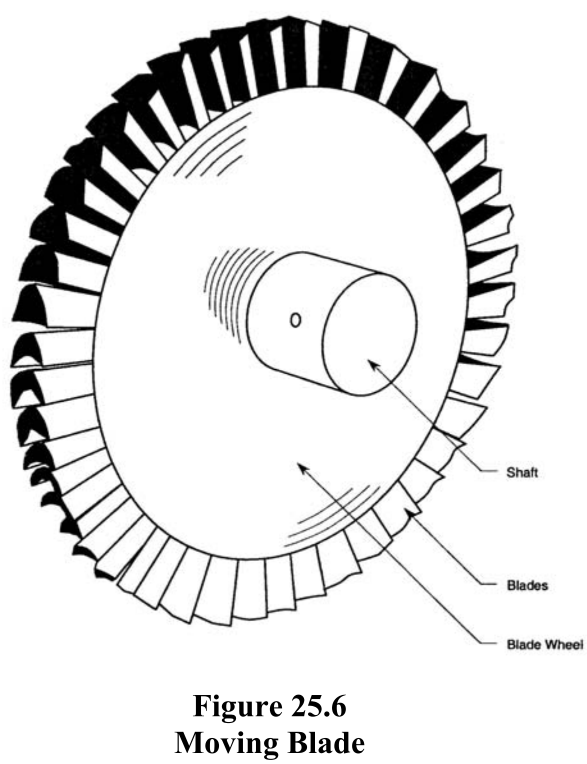


Figure 25.6
Moving Blade

Figure 25.7 is a simplified section view of a turbine. The steam turbine consists of a single rotating shaft that has a number of blade wheels attached to it. The steam passing through the turbine is contained within a casing. The casing is usually split into upper and lower halves that are bolted together. This allows the upper half to be raised for maintenance. Attached to the casing are diaphragms that support the fixed blade nozzles. Wherever the fixed and moving parts of the turbine come together, there is a need for sealing to prevent leakage from the high-pressure side to the low-pressure side. This leakage is prevented by gland seals installed on the turbine casing.

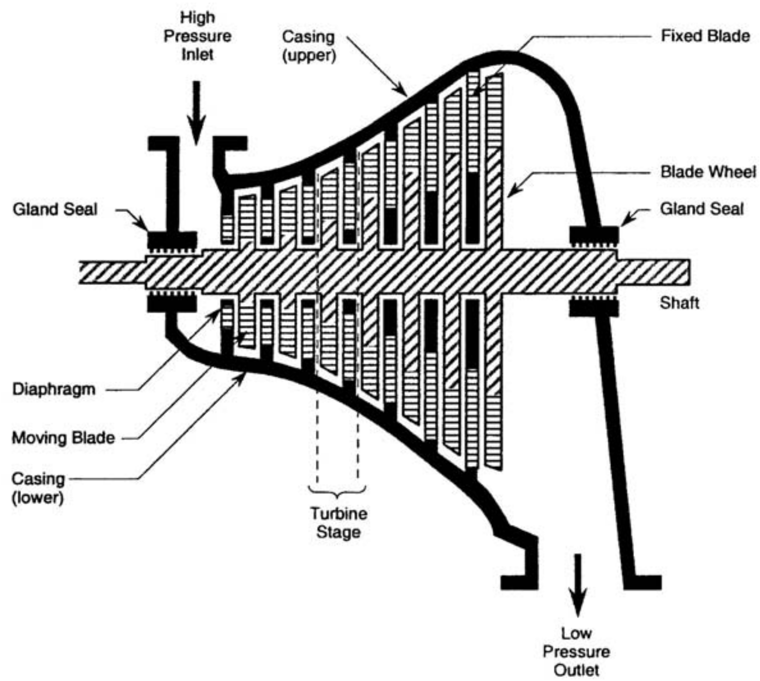


Figure 25.7
Turbine Section View

Figure 25.8 shows the construction of a typical diaphragm. The upper and lower halves of the diaphragm attach to the upper and lower casings respectively.

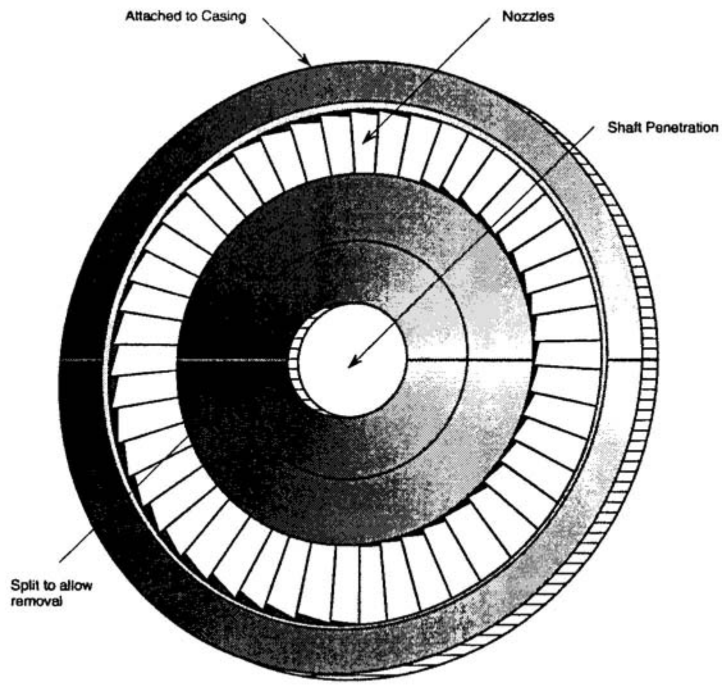


Figure 25.8
Turbine Diaphragm

Looking at Figure 25.7, you will note that the turbine casing gets progressively larger as the steam goes from the high-pressure end to the low-pressure end. This is necessary to accommodate the expansion of the steam as a direct result of pressure and temperature reduction. Steam entering the high-pressure end of a modern nuclear turbine set is typically around 250°C and 4000 kPa. At this temperature and pressure, one kilogram of steam occupies .05 m^3. The steam leaving the turbine unit and entering the condenser is typically around 35°C and 5 kPa(a). At this temperature and pressure one kilogram of steam occupies 25.2 m^3. The steam expands roughly five hundred times from the inlet to the exhaust. In a large turbine generator set it is usually not possible to accommodate the large volume of steam in one turbine unit. Normally one high-pressure turbine will exhaust to two or more low-pressure turbines in combination with the double flow design.

Figure 25.9 shows a turbine unit typical of those installed at a modern CANDU plant.

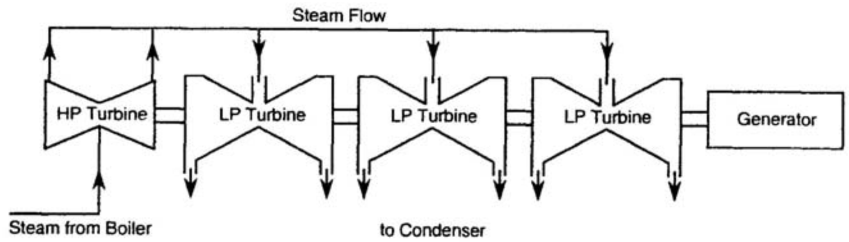


Figure 25.9
Typical Turbine Layout

Why use double flow steam turbines? The double flow turbine design not only provides double the expansion volume within a common casing, it also balances the large pressure drop between the turbine steam inlet and exhaust which tends to force the blade wheels from the high-pressure side towards the low-pressure side.

Figure 25.10 shows a double flow turbine. Steam enters the turbine in the middle of the casing and expands outward in both directions before exhausting at the ends of the turbine. In each half of the turbine, a very large thrust is generated. These thrusts oppose each other; the resultant force is significantly reduced. The resultant thrust is taken up by a thrust-bearing located between the high-pressure and first low-pressure turbine.

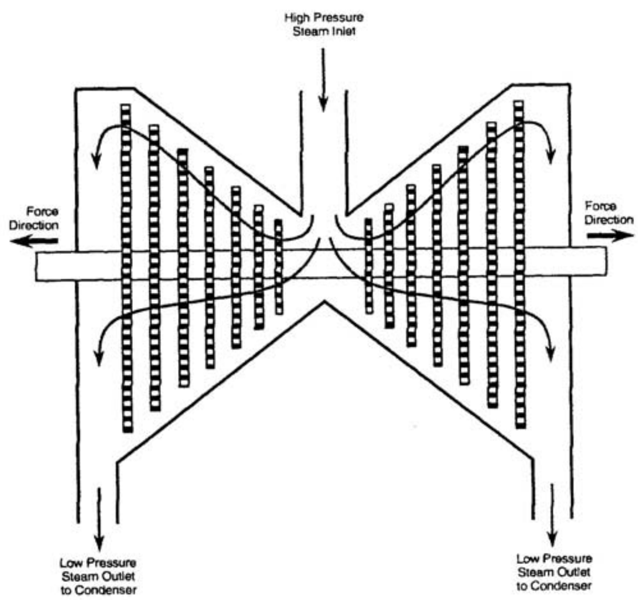


Figure 25.10
Double Flow Turbine

25.5 The Condenser

The condenser is the final destination for most of the steam produced in the boiler. The condenser removes latent heat, turning the steam back into water. The large decrease in volume creates a vacuum in the condenser. This permits steam flow from the high-pressure boiler to the low-pressure condenser so the turbine can extract mechanical energy efficiently.

Figure 25.11 shows a typical condenser used in most CANDU turbine units. The condenser cooling water (CCW) system supplies cooling water to the condenser. The water enters through the inlet water-box, passes through the condenser tubes and discharges to the lake through the outlet water-box. The turbine exhaust steam enters the condenser through the condenser exhaust trunk and reaches the outside surface of the condenser tubes. The steam condenses to a liquid by releasing its latent heat of vaporization through the tubes. The large volume of CCW absorbs the latent heat of vaporization from the steam. The condensate falls into the bottom of the condenser and collects in the hotwell.

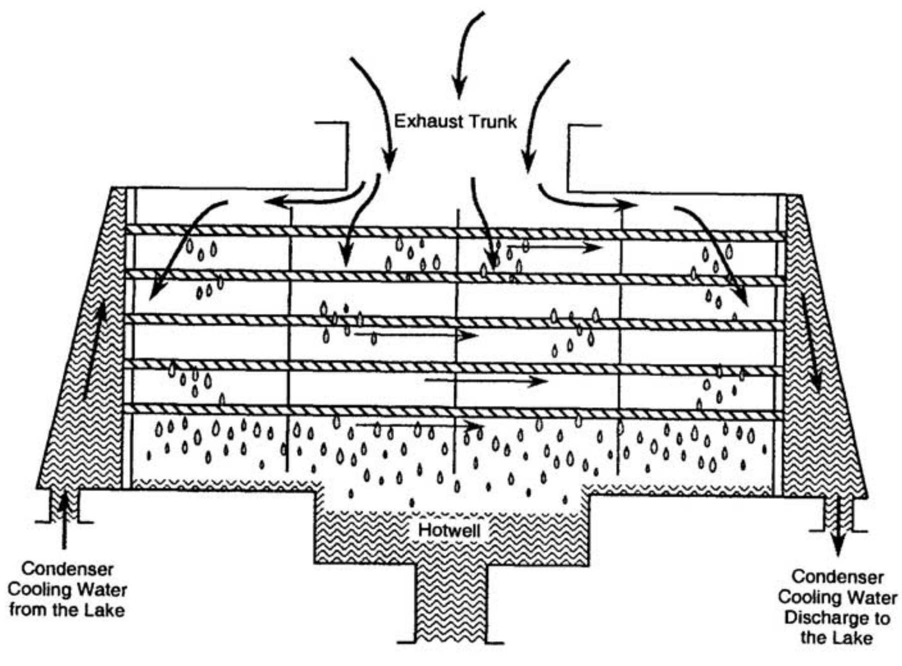


Figure 25.11
The Condenser

The CCW flow maintains the saturation temperature of the condensate. The exact condenser conditions (temperature and pressure) are dependent on the temperature of the lake water. Since the maximum power output of the unit depends on the pressure and temperature across the turbine, the output of the unit varies several percent with lake temperature.

The lower the turbine exhaust temperature and pressure the greater the amount of steam energy that can be converted into mechanical energy in the turbine. As mentioned, steam leaves the turbine at about 35°C and 5 kPaa. This condition is near a perfect vacuum. This allows roughly 35% more energy extraction than if the steam is left at atmospheric pressure (101.3 kPa). The condenser provides the means of maintaining this low absolute pressure at the turbine exhaust through condensation of steam (the 25.2 m^3 of steam reduces to 0.001 m^3 of water or 25200 times volume reduction).

The steam/feedwater system is a closed loop because it would be wasteful and expensive to reject the clean, chemically treated, demineralized water after it has completed its work in the turbine. It would also be wasteful to throw away the heat held by the 35°C condensate, especially as the CCW flow is adjusted to keep the condensate at saturation temperature and not cool it more than necessary for condensation.

25.6 Boiler Feedwater System

Figure 25.12 shows a simplified steam system and boiler feedwater system. The feedwater system is generally divided into three parts:

- low-pressure feedheating system,
- deaerator and storage tank, and
- high-pressure feedheating system.

The water leaving the condenser is at relatively low temperature and pressure. A series of heat exchangers raises the condensate temperature to 170°C. The preheaters then increase the temperature to 240°C (almost saturation temperature in the boiler). A set of pumps, known as boiler feed pumps (BFP), force the feedwater into the boilers operating at 4000 kPa.

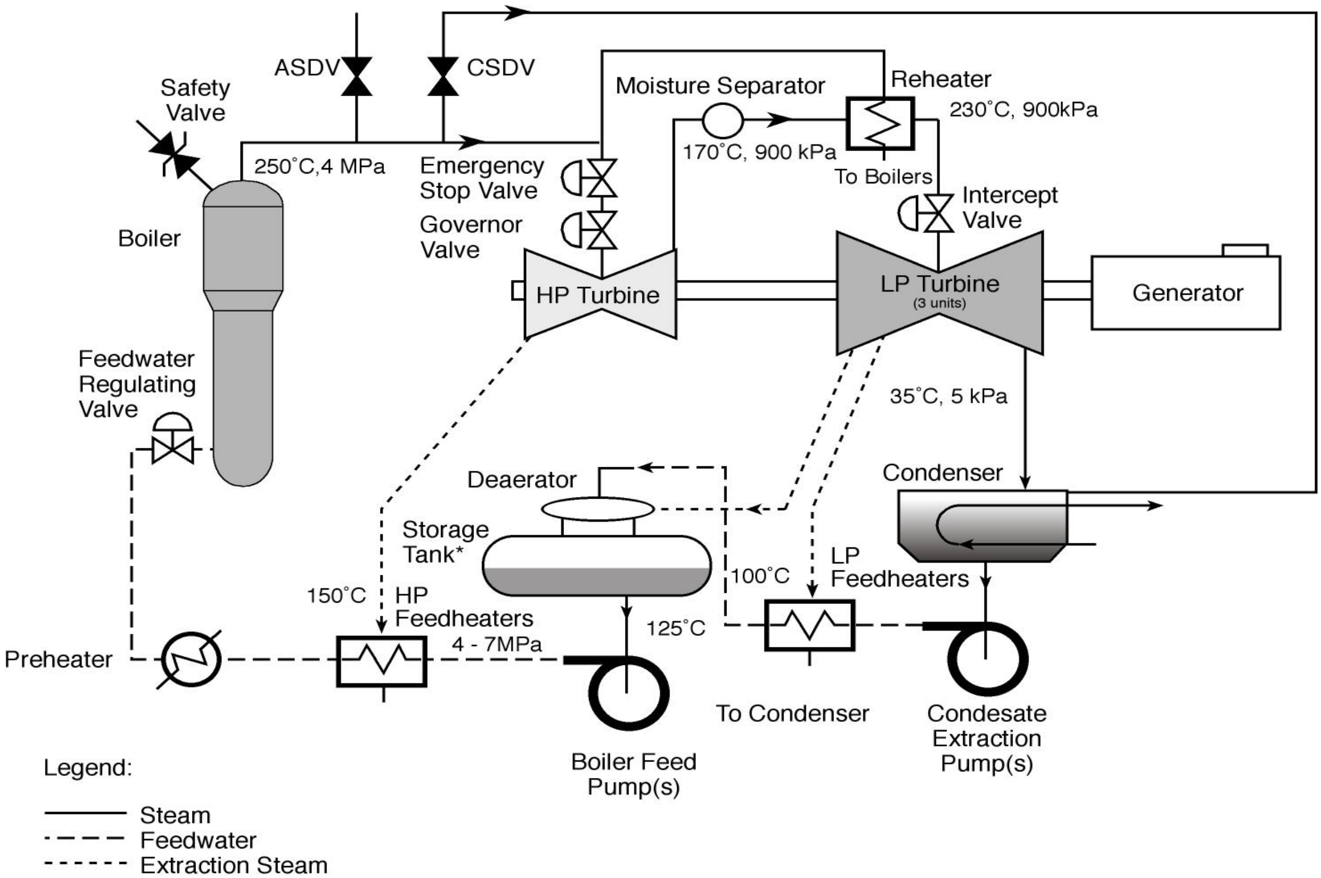


Figure 25.12
Simplified Feedheating System and Steam System

25.6.1 Low-pressure Feedheating System

The first stage in the boiler feedwater heating is through the LP feedheating system. The condensate extraction pump (CEP) delivers the condensate from the condenser hotwell to the LP feedheaters. The low-pressure feedheating system gets its name from the low-pressure condition of the feedwater, at about 1400 kPa, compared to the 4000 kPa in the boiler.

The LP feedheaters use extraction steam from the LP turbines as their heating medium. Extraction steam is wet steam removed from the turbine before it reaches the exhaust end because it could cause damage to turbine blades. The extraction steam transfers its latent heat of vapourization to the feedwater through a process similar to that in the condenser. A series of low-pressure feedheaters heat the feedwater. The extraction steam condenses in the shell of the heater. A separate pump recovers this condensate by pumping it to the condenser hotwell. The feedwater leaves the last LP feedheater at approximately 80°C to

100°C. The heated feedwater then goes to the next stage of the feedheating process.

25.6.2 The Deaerator and Storage Tank

The deaerator is the next stage in the feedwater heating process. This is the highest vessel in the system. The deaerator adds heat to and removes non-condensable gases from the feedwater. Some of these gases can increase the corrosion rate of the metals in the high-pressure feedheating system and boiler. All non-condensable gases will take-up space in the steam system. This reduces the amount of steam in the system and hence energy flows in the plant.

Figure 25.13 shows a typical deaerator and its associated storage tank. The incoming feedwater enters the deaerator near the top and sprays downward over cascade trays. Extraction steam from the LP turbine enters the deaerator near the bottom and passes upward. As a result the feedwater heats up to about 125°C. The steam passing over the water droplets scrubs the non-condensable gases off their surface. As the extraction steam condenses, the water droplets release the non-condensable gases, which are vented to the atmosphere. The deaerated feedwater and condensed steam drain from the deaerator into a storage tank. The storage tank supplies water for boiler operation.

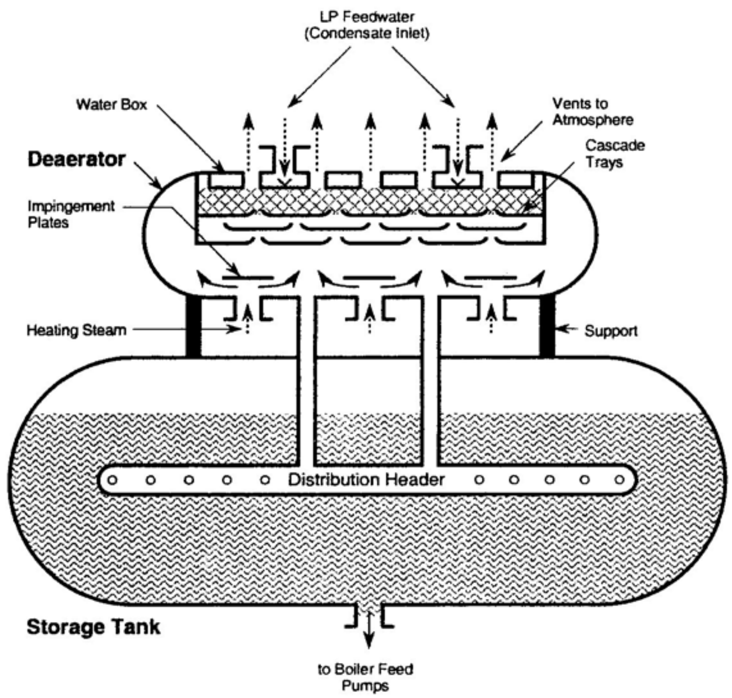


Figure 25.13
Simplified Deaerator and Storage Tank Assembly

25.6.3 High-pressure Feedheating System

From the deaerator storage tank, the feedwater undergoes the final preparation. The boiler feed pumps (BFP) take suction from the deaerator storage tank and raise the feedwater pressure to between 4 and 7 MPa. The pump discharges the high-pressure feedwater to the high-pressure (HP) feedheaters. The HP feedheaters heat the feedwater to about 170°C. HP feedheater operation and construction are similar to that of the LP feedheaters. Extraction steam from the HP turbine normally supplies the heating medium.

The feedwater regulating valve controls the flow of feedwater into the boiler. This valve allows sufficient feedwater to enter the boiler to match the steam flow out hence, maintaining a constant water mass in the boiler. To do this, the controller for the valve compares steam flow out of the boiler with feedwater flow into the boiler and positions the valve to make the two equal. It also compares the actual boiler water level with a predetermined programmed level and positions the valve to make these two equal.

It is critical to maintain a proper boiler level. If the boiler water level is too high, the cyclone separators and scrubbers will not operate properly. This results in wet steam being delivered to the turbine, which could lead to damage to the turbine blades. If water level were too low, there would not be enough inventory to cool the HTS coolant.

25.7 Lubrication System

Each unit of the turbine and the generator has its own rotor/shaft that is supported at each end by journal bearings. Journal bearings get hot due to friction and heat conduction along the shaft from hot parts of the turbine. The journal bearings are normally lined with white metal known as anti-friction metal or Babbitt, a lead-tin alloy with a melting point that can be as low as 182.2°C. A centralized lubricating system is employed to protect the bearings from damage due to metal-to-metal contact and high temperature. This extends the life of the bearings and reduces the chance of failure. A bearing failure is a very serious incident, as far as the turbine-generator is concerned and would cause extensive damage. For this reason it is always important to have sufficient oil flow through the bearings for lubrication and cooling purposes.

25.8 The Turning Gear

When a turbine comes to rest, after operating, the cooler denser steam tends to collect in the lower half of the cylinder and makes the lower half of the rotor cool quicker than the upper half. This causes the shaft to hog (bend upwards). When at rest and cool, the shaft will begin to sag under its own weight. If the turbine shaft is not rotated, hogging, especially above a critical temperature, can become permanent and the shaft would have to be sent to the manufacturer for heat treatment and straightening. Sagging does not usually become permanent but it takes time to recover the sag. To prevent a bent shaft due to sag or hog the shaft is rotated by a turning gear, which is a motor-driven gear-train mechanism on the turbine-generator shaft.

25.9 Chemical Control

Boiler steam and feedwater system construction in almost all CANDU stations uses carbon steel, copper alloys and nickel alloys. Each metal is susceptible to corrosion at a different pH level. A compromise pH of 8.8 to 9.3 is relatively safe for all metals involved. Chemical addition of morpholine and/or hydrazine into the feedwater maintains the appropriate pH level.

Other methods used to prevent corrosion are:

- oxygen removal from the system, and
- chemical addition to react with oxygen.

All stations use a common approach in removing oxygen from the system. Most oxygen is removed from the system by the condenser air extraction system and the scrubbing action of the deaerator. Hydrazine addition to the feedwater, after the deaerator, removes the remaining oxygen. Its reaction with oxygen produces non-corrosive nitrogen gas and water. Unfortunately, hydrazine also produces ammonia, which attacks copper alloys. Stations with boiler tubes that are a copper alloy have lower hydrazine levels than stations with stainless steel alloy tubes.

High quality feedwater and makeup water is vital, as low quality will produce deposits in the boiler and turbine causing:

- Reduced heat transfer because of an insulating scale layer on the boiler tube surface.
- An increased risk of stress corrosion cracking. This is a form of metal failure from low stresses in a corrosive environment.
- Corrosion of tubes and other components.

All will shorten the life of the boilers and turbines. Demineralization, deaeration, oxygen removal and pH control ensures high quality boiler water. A blowdown system in each boiler allows removal of any impurities that collect in the boilers. This system minimizes accumulation of impurities by draining the contaminated water out of the boiler. Blowdown can be intermittent or continuous, depending on the water condition.

25.10 Conventional and Radiological Hazards

25.10.1 .Chemical Energy

In the feedwater system, chemicals treat the boiler feedwater. These chemicals include hydrazine, ammonia and morpholine. Proper protection and careful handling of these chemicals will prevent injury.

Hydrogen used for generator cooling is potentially explosive in the range of 4% to 76% in air. Hydrogen can leak out of the generator or air can leak into the generator. This requires very good seals on the generator to minimize the explosion hazard.

25.10.2 Thermal Energy

Steam and hot feedwater leaks are thermal hazards. Steam leaks are often invisible. Therefore, once found a leak location should be roped off immediately. In most cases, it is necessary to shut down a unit before the steam leak can be repaired. Equipment that handles hot feedwater or steam will have high temperatures. Insulation and/or physical barriers are installed to prevent personnel from being exposed to the heat.

25.10.3 Electrical Energy

Most equipment and controls associated with the conventional side of the station will require electrical energy to operate. This includes numerous DC and AC electrical hazards. This energy is either produced by the generator or supplied by different sources. If not controlled and handled properly, it can cause shock, burns or even cardiac arrest.

25.10.4 Mechanical Energy

This hazard is found in anything that moves or is capable of moving. Generator and turbine rotating parts and other major components in pumps and motors are hazards that can cause cuts, abrasions, and crushing injuries.

25.10.5 Noise Energy

The main hazard here is excessive noise from unexpected opening of boiler safety valves, which could cause temporary loss of concentration and risk of injury. Personal hearing protection will reduce this risk somewhat.

25.10.6 Pressurized Fluid Energy

This hazard is found in most of the conventional systems. Uncontrolled release of pressurized fluid can cause punctures, fractures, abrasions and crushing injuries. Excessive pressure can cause equipment to fail. This could lead to broken components flying around causing injury to personnel and damage to other equipment.

Oil supplied to the hydraulic valves and bearings is pressurized. Any rupture in the oil lines can result in fire, if ignition sources are present. Fire Retardant Fluid (FRF) is used for operating the valves that control the turbine, making a fire around the valves unlikely.

25.10.7 Radiation Energy

There is a large pressure difference across the boiler tubes. The HTS is at 10 MPa and the boiler at 4 MPa, hence a leak will always occur into the feedwater. Failure of boiler tubes during operation will result in loss of D_2O to the boiler feedwater system, along with radioactive tritium, activated corrosion products and, possibly, fission products. In this case steam leaks from the main steam system could result in radiological hazards. Units are shutdown and the tube leaks repaired if a significant leak occurs.

25.11 Assignment

1. What is the function of the steam cycle and the feedwater cycle?

2. In the table, indicate the quality of steam at each location.

Location	Moisture Content	Temperature	Pressure
Outlet of HP Turbine			
Outlet of Moisture Separator			
Outlet of Reheater			
Outlet of LP Turbine			

3. Why is the turbine set divided into high and low-pressure units?

4. How does condenser vacuum improve the efficiency of turbine operation?

5. What is the purpose of each of the following components?

   a) Boiler
   b) Emergency Stop Valve
   c) Governor Valve
   d) HP and LP Turbines
   e) Boiler Feed Pumps
   f) Lubricating System
   g) Turning Gear

6. What is an alternative heat sink for the boiler if the turbine is tripped?

7. What are the three stages of heating for condensate water, and from where does each stage draw its heating source?

8. What are the major chemistry problems in the steam/feedwater cycle and how are they minimized?

9. What are the major hazards associated with the conventional side of the station?

26 Other Major Systems

26.1 The Generator

26.1.1 AC Generator Energy Conversion

A generator converts mechanical energy into electrical energy. The basic prerequisites to produce electricity from an AC generator are:

- there must be a conductor,
- there must be a magnetic field, and
- there must be a relative motion between the conductor and the magnetic field.

Whenever these three conditions are met, a voltage is induced in the conductor. In a practical generator, a large number of conductor coils multiply the effect.

Figure 26.1 shows a simplified arrangement of a generator coupled to a steam turbine drive. The stationary conductors (coils) and the associated iron cores are referred to as a stator. Conductors (coils) and the associated iron core, mounted on the shaft, are referred to as a rotor.

Insulated slip rings on the shaft transfer DC current to create a magnetic field in the rotor. The stator windings act as the conductors for the main generator current while the turbine provides the mechanical torque on the shaft of the generator. The rotating motion provided by the shaft produces the relative motion between the rotor magnetic field and the stator conductors. As a result, a voltage is induced in the stator conductors and transferred to the transmission lines through a step-up transformer.

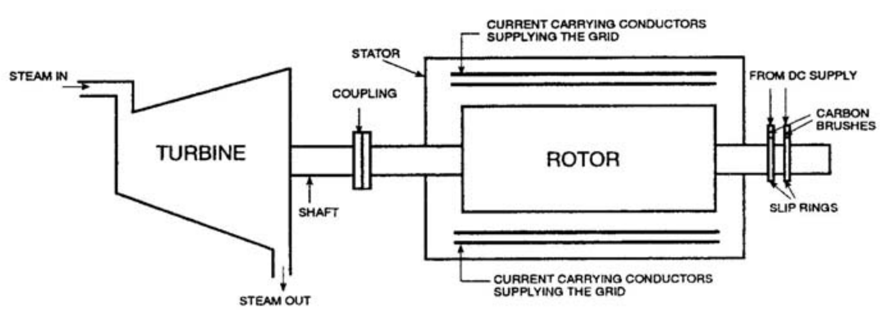


Figure 26.1
Simplified Arrangement of a Generator Coupled to a Turbine Drive

In a generator, the rotor velocity determines the frequency. When the generator is connected to the grid the frequency is fixed at 60 Hz. Since the frequency for the Ontario Grid is fixed at 60 Hz, the velocity of the rotor is kept constant. In nuclear plants this speed is generally 1800 rpm.

As electrical consumers use electricity, they create a load current on the Ontario grid, thereby increasing counter torque to the turbine shaft. The tendency of the turbine is to slow down as counter torque is increased which would decrease the frequency. To compensate for the increased counter torque more steam is admitted to the turbine to produce more shaft mechanical power and to maintain the generator speed.

26.1.2 Generator Cooling

The modern electric generator for a steam power station is an extremely efficient machine. Approximately 98% of the mechanical power delivered on the shaft from the turbine is converted to electrical power. The remaining 2% appears as heat in various places in the generator. Two percent does not appear to be very much until you consider that 2% of a 750 MW machine is equal to 15 MW. Since all of this 15 MW is converted to heat, it is like putting a heater of this size inside the generator.

The heat that is produced in a generator comes from several sources including windage (gas friction) between the rotor and the circulating cooling gas, the electrical heating due to the current resistance in the windings of both the rotor and stator, and the electrical heating due to current induced in the structural material of the rotor and stator.

Even small increases in the operating temperature of a generator will lead to rapid deterioration of the insulation on the windings. For this reason, two systems are provided to cool the generator. One system uses hydrogen circulated through the generator. Hydrogen has the advantages of:

- better thermal capacity than air,
- less damaging to insulation than air, and
- less dense than air so less heat is produced from windage.

The disadvantage is that it is explosive when mixed with air. To avoid this hazard, the generator requires very good seals to prevent air in-leakage or leakage of hydrogen out of the generator. Special procedures are required when filling and emptying the generator to prevent an explosive mixture.

By itself, the hydrogen cooling system is inadequate. To complement it, a stator cooling water system is also provided. The conductors in the stator are hollow and water is circulated through them. This water has to be exceptionally pure to prevent leakage of current from the stator conductors to ground through the coolant.

The combination of hydrogen and stator-water cooling is sufficient to cool generators as large as 1500 MW, which is far larger than any generator in service in a CANDU plant.

26.2 Electrical Systems

26.2.1 Major Components

Figure 26.2 shows a simplified diagram of a CANDU station's electrical interconnection with the power grid.

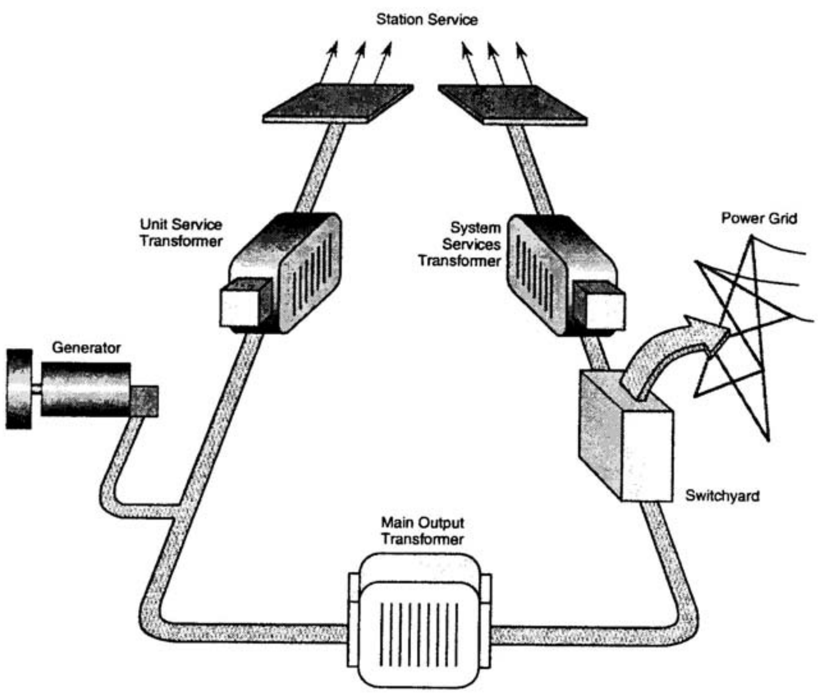


Figure 26.2
Main Power Output and Unit Distribution System

The main output transformer steps up the voltage from the unit generator to the level required by the Ontario Power Generation grid. A higher voltage means a reduced current and therefore reduced line losses over the long distances that power is transmitted.

A higher voltage means a reduced current and therefore reduced line losses over the long distances that power is transmitted.

This electrical power is delivered to the switchyard. The switchyard increases overall reliability by providing the means to switch generator output to available output lines, to isolate a faulty generator or line, and, if necessary, to draw a unit's energy requirements from the grid.

A reactor unit can draw power to meet its internal needs from two sources:

6. The unit service transformer (UST) connected directly to the unit generator. The UST is a step-down transformer that reduces generator voltage to the level appropriate for the unit.

7. The system service transformer (SST) is connected directly to the grid. This is a step-down transformer, which reduces grid voltage to the appropriate level for the unit.

The UST is the primary supply to the reactor unit and the SST is the alternate supply to the reactor unit as well as the primary supply to a portion of the common station loads.

26.2.2 Classes of Power

All power consumers (loads) in the station are not created equal. It is essential to ensure that some loads (e.g., protective relaying) never lose their power source, while others (e.g., office air conditioning) can go without power almost indefinitely. To handle the various needs, a hierarchy of four classes of power has been developed based on the urgency or importance of maintaining power to individual loads. Each class has both a normal power source and an emergency power source. The emergency power source takes over when the normal source is not available. Each class supplies power to odd and even buses. Equipment is divided up between odd and even buses to ensure independence; failure of one bus does not deprive all similar equipment of power. In the case of redundant equipment, for instance two 100% pumps, one piece would be supplied from an even bus and the other from an odd bus. Figure 26.3 graphically illustrates the four classes of power, the alternate sources for each and the odd/even arrangement.

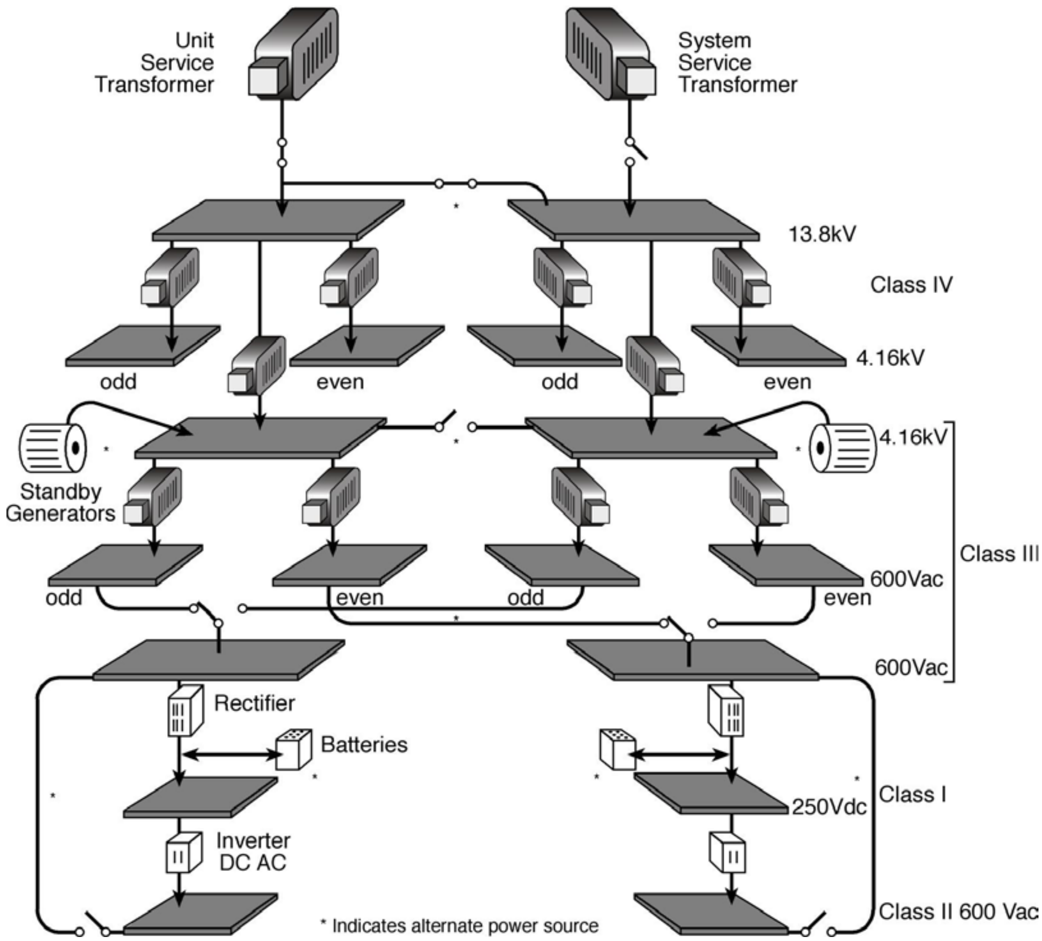


Figure 26.3
Classes of Power and Distribution

Class IV Power
Class IV power supplies AC loads that can be interrupted indefinitely without affecting personnel or plant safety. Typical loads on a Class IV system are normal lighting and the primary heat transport pump motors. During normal operation, the UST carries the reactor unit loads. Should the need arise; the loads can be supplied fully by the SST. Thus, the electrical grid system serves as an emergency power supply for the Class IV system.

Class III Power
Class III power supplies AC loads that can tolerate the short interruption (one to three minutes) required to start the standby generators without affecting personnel or plant safety, but are required for safe plant shutdown. Typical loads on a Class III system are the moderator main circulation pump motors and the pressurizing feed pump motors. Normally, Class III power is supplied from a Class IV

source. Should a supply path from the UST and the SST both fail, then one or more of the standby generators (gas turbine driven) will automatically start and begin picking up the loads. This process is initiated in approximately three minutes.

Class II Power

Class II power is considered uninterruptible. Class II supplies AC loads that cannot tolerate the short interruptions, which can occur in Class III. Typical loads on a Class II system are digital control computers and reactor safety systems. Class II power is normally fed from Class I via an inverter which changes DC to AC. Should the Class I supply fail, Class II power can be supplied from Class III while high priority action is taken to restore the Class I supply. In this situation, it is normal to start a standby generator and hook it to Class III.

Class I Power

Class I power is considered uninterruptible. Class I supplies DC loads that cannot tolerate the short interruptions which can occur in Class III. Typical loads on a Class I system are protective relaying, circuit breaker control, turbine lube oil emergency pump, emergency seal oil pump, and emergency stator conductor water cooling system pump. Class I is normally obtained from Class III, via a rectifier (battery charger). Should the rectifier or the Class III supply fail, then Class I is supplied from a battery bank, which is normally maintained fully charged by the rectifier.

26.2.3 Emergency Power Supply (EPS)

In the unlikely event of a loss of all Class IV and III power the emergency power supply (EPS) provides electrical power to certain nuclear safety-related systems that support the capability to control, cool, and contain. The EPS is started automatically on the loss of power to certain 600V buses on the common unit or it can be manually started from the EPS control room within 30 minutes of an identified need. It is seismically and environmentally qualified and has sufficient fuel stores to operate unaided for a seven-day period. Seismic Qualification requires that equipment and systems retain their specified performance capability following an earthquake. Environmental Qualification (EQ) requires that equipment must be protected against steam leaks, water flooding, high intensity fires or other mishaps, which could disable it. The EPS must be available whenever there is a significant amount of fission products in a reactor core. The EPS system in older stations may not be named the same or be as extensive as in new stations.

Some worst-case accidents, which could lead to a need for an emergency power supply, are:

- tornado,
- widespread fires, and
- earthquake.

The EPS is quite similar to other standby generators but is remotely located from the standby generators to reduce chances of it being disabled by the same incident. Cables and control equipment involved in switching the Emergency Power Supply into service are routed through areas that are considered to be at lowest risk of damage

26.3 Water and Air Systems

26.3.1 Light Water Systems

Water for all purposes (cooling, feedwater make-up, fire protection, domestic use, etc.) is drawn into the nuclear station's intake channel from the lake through a tunnel, which extends approximately 600 meters out under the lakebed. Each unit has its own pump-house to supply condenser cooling water and service water. Domestic water and demineralized water for feedwater make-up are supplied from a water treatment plant to all units in a station station. System interconnections are shown in Figure 26.4. The systems shown (except water treatment) are duplicated in each unit.

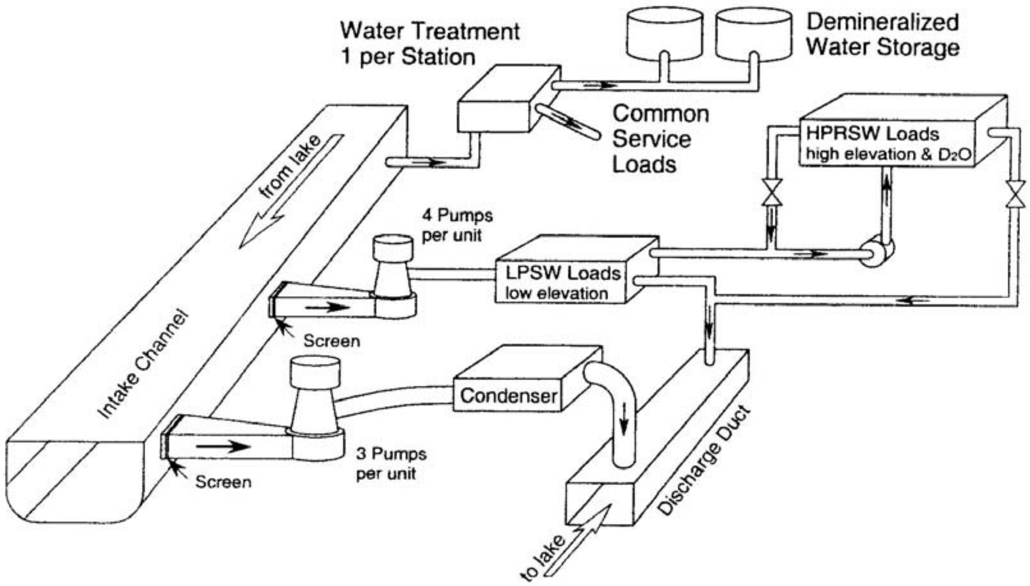


Figure 26.4
Water Systems

26.3.2 Water Treatment

Water treatment has two purposes:

- to remove harmful constituents from the water, and
- to treat the water with beneficial ingredients.

The water treatment plant produces demineralized water primarily for boiler feedwater makeup, but also for end-shield cooling, the closed loop demineralized water-cooling system, the irradiated fuel bay, and the chemistry lab. The chemical treatment process for each water system is varied. Demineralized water is essential in systems that must be protected from corrosion or the build-up of scale and crud.

26.3.3 Condenser Cooling Water

The purpose of the Condenser Cooling Water (CCW) system is to supply strained lake water to the condensers. The only treatment this water receives is filtering through screens to remove small debris such as entrained organic matter.

The CCW is one system required to have a certificate of approval. The CCW system must be capable of removing 70 per cent of the reactor's thermal power, but the certificate of approval requires it to do this without raising the discharge temperature more than 11°C above lake water temperature. To meet this requirement, the system must provide flows in the range of 31 m^3/s for each unit. Over 85% of a station's total cooling water flow is required for condenser cooling. About four MW of electricity is required to drive the pumps for each unit.

26.3.4 Common Service Water
The Common Service Water system (CSW) provides a continuous flow of water to the Central Service Area, the Water Treatment Plant, the Vacuum Building, and the Ancillary (Auxiliary) Services Building. Common service water is strained and filtered before being distributed. It provides water for cooling, waste dilution, lawn watering, etc.

26.3.5 Low-pressure Service Water
The Low-pressure Service Water system (LPSW) provides a continuous flow of strained lake water for specific cooling purposes such as to seals, bearings, and heat exchangers. The temperature of the LPSW ranges from 2°C to 27°C. The LPSW draws its supply from the intake channel.

26.3.6 High-pressure Recirculating Service Water
The High-pressure Recirculating Service Water System (HPRSW) is fed from the LPSW system. It increases the pressure and tempers the water to 15°C to 30ºC by directing some of its outlet flow back to its inlet. It serves all applications where potential D_2O freezing is a problem or where equipment is located at high elevations within the plant. D_2O has a freezing point of about 4ºC, 4 degrees warmer than a typical lake in Canada in January. Typical loads are the closed loop demineralized water cooling system heat exchangers, Moderator pump motors, H.T. feed pump oil coolers, maintenance cooling pumps, D_2O vapour recovery dryers, and heat transport pumps.

26.3.7 Closed Loop Demineralized Service Water System
The Closed Loop Demineralized Service Water system is used to provide cooling to plant equipment where the impact of corrosion is of particular concern. Typical loads are the H.T. bleed cooler and gland seal cooler, the delayed neutron water boxes, and the H.T. Pump neutron shields.

26.3.8 Emergency Water System
The Emergency Water system (EWS) is environmentally and seismically qualified. It provides cooling water to critical systems when the normal systems (boiler feedwater and LPSW, and/or Class IV and III power) are unavailable. It draws its power from the EPS. Emergency water can be routed to the boilers, to the ECI heat exchangers, the H.T. System, the vault coolers or the primary and secondary irradiated fuel bay heat exchangers as required. The EWS

draws its water from the station outfall, providing an independent source in the event that the supply from the forebay is not available.

26.3.9 Other Water Systems

The Fire Protection Water system provides water for fire fighting to areas such as fire hose cabinets in the station, hydrants, transformer deluge systems, turbine sprinkler systems, and an air foam system. In emergencies, it can supply water to the irradiated fuel bay, the ECI system, the vacuum building emergency water storage tank and the moderator heat exchangers. The system draws its supply from the common service water intake duct.

The Domestic Water Distribution system is different at each station but its uses are common. Hot and cold potable water is supplied to the plumbing fixtures (toilets, urinals, sinks, showers, drinking fountains, eyewash stations, safety showers) and laundry machines as required in the station, and ancillary buildings. Supply is drawn from a local pump house or the water treatment plant.

26.4 Air systems

A CANDU station has numerous uses for compressed air. The quality of the air required depends on the application. To handle the varying requirements, the station has a number of different compressed air systems.

26.4.1 Instrument Air

The Instrument Air system provides instrument quality compressed air to all parts of the station. There are actually a number of systems, one for the common areas of the station, and one each for the units. This air is used for control valve actuators, power operators, pneumatic controllers, and special applications in the chemistry lab and irradiated fuel bay where service air is of insufficient quality.

26.4.2 Service Air

The Service Air system provides general purpose compressed air to all parts of the station. This air is used for air-powered tools, cleaning, and water treatment plant regeneration.

26.4.3 Breathing Air

The Breathing Air system supplies breathing air to any areas of the plant in which personnel may require plastic suits. The primary use is for personnel working inside the reactor vault during shutdowns. Most of the air supplied to the plastic suits is for cooling.

26.5 Identification System

26.5.1 Equipment Identification

A standardized numbering system has been adopted as system of identification. This is supplemented in the field by colour coding and tagging, and on drawings (flowsheets) by equipment symbols. Although in principle the numbering system is identical in each stations, it does vary in detail. This system specifies all of the equipment and most operations in the station.

The system is subdivided into divisions. Figure 26.5 shows how some of these divisions relate to station systems.

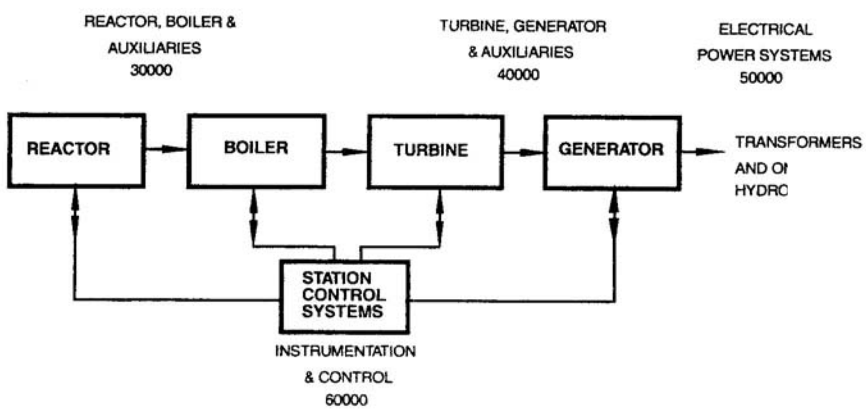


Figure 26.5 Equipment Numbering Divisions

The complete set of Divisions are:

Division 0	General Project
Division 1	Site and Improvements
Division 2	Buildings, Structures and Shielding
Division 3	Reactor, Boiler and Auxiliaries
Division 4	Turbine, Generator and Auxiliaries
Division 5	Electric Power Systems
Division 6	Instrumentation and Control
Division 7	Common Processes and Services
Division 8	Construction Indirects

Each division is further subdivided as shown below. A five-digit number allows the specification of an individual component of any system in a plant. An example from Division 4 illustrates the structure of the system.

It should be noted that below the system level, the numbering system may be changed to suit particular station needs.

Division	**4**0000	Turbine, Generator and Auxiliaries
Major System	4**2**000	Condensing System
System	42**1**00	Main Condensing System
Sub-System	421**2**0	Condenser Extraction System
Components	4212**1**	Ejectors
	4212**2**	Vacuum Pumps
	4212**3**	Valves
	4212**8**	Pipe Supports
	4212**9**	Piping

26.5.2 Field Identification

In the field, the system number and a brief written description are found either printed on the equipment or on a tag attached to the equipment. Where more than one component of the same kind (e.g., valves) is contained within a sub-system, a special device code is provided in place of the component digit in the system. This code consists of a descriptive letter (P for pump, V for valve, etc) and a unique number. For example, 42123 which indicate any valve in the condenser air extraction system could be changed to 42120-V2 or 42120-V15 to indicate a specific valve in the system.

26.5.3 Piping

For quick identification, piping is colour coded to indicate the type of fluid it contains. Also, an arrow is attached showing the direction of flow. Colours commonly used are:

Air	Blue
Heavy Water	Pink
Light Water	Green
Steam	Silver (Aluminium)
Oil	Yellow
Gases	Brown
Building Heating	White
Drains & Sewage	Black
Fire Protection	Red
Vacuum	Purple
Chemicals	Orange

26.5.4 Flowsheets

Each station maintains a complete set of system drawings called flowsheets. The flowsheets are graphical representations of the systems using standard symbols to represent equipment and devices. The flowsheets are labelled using the number system and the equipment and device code labels on the flowsheets are identical to the codes used in the field. Interrelationships between systems are indicated by reference to other flowsheet numbers. Complete sets of flowsheets and the legend of symbols are maintained by the records section of the station.

26.6 Waste Management

26.6.1 Liquid Waste Management

Like any large facility, a CANDU station has an extensive network of floor drains to collect spills and drainage from its various processes. Because of the nature of the business, it is necessary to subdivide the drainage system into:

- inactive drainage, and
- active drainage.

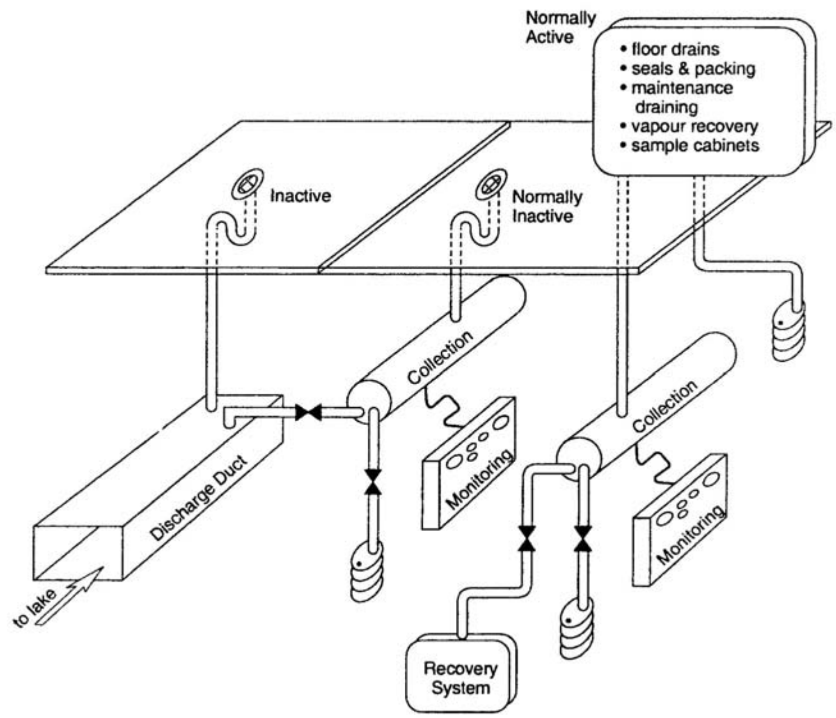


Figure 26.6
Drainage System

26.6.2 Inactive Drainage

The inactive drainage system collects drainage from the conventional side of the station. The waste discharges to the condenser cooling water discharge channel or intake channel depending on the location. Clean drains such as leakage collection from the main steam blowdown pipe trenches are returned to the lake through yard drains. The resin regenerant waste effluent from the water treatment plant is monitored for pH and discharged under controlled conditions to the condenser cooling water discharge channel.

26.6.3 Active Drainage

The active drainage system collects drainage from the reactor side of the station. Because the volume of water from these areas is quite large, the system further segregates the drainage into normally inactive drainage and normally active drainage to minimize the amount of water requiring treatment prior to disposal.

Normally Inactive Drainage

This drainage contains very little or no activity, but it is collected prior to discharge to ensure that it can be treated if contamination occurs. The major sources are reactor building floor drains, laundry drains, and non-active laboratory sinks and floor drains.

Normally Active Drainage

This drainage is expected to have activity so it is collected and sampled to determine the required treatment prior to release. The major sources of normally active waste are the reactor auxiliary bay floor drains, irradiated fuel bay drainage, spent ion exchange resin slurry water, auxiliary irradiated fuel bay drainage, active chemical laboratory drains, decontamination centre drains, fuelling machine maintenance shop drains, laundry first rinse cycle drains, and decontamination shower drains.

Reactor building drains are diverted to the reactor building liquid recovery system to recover heavy water.

26.7 Solid Waste Management

26.7.1 Irradiated Fuel Storage

Stations do not produce large amounts of high-level radioactive material. A four unit 850 MW station produces an average 20,000 irradiated fuel bundles per year (390 Mg per year).

Irradiated fuel is stored in pools of demineralized light water called Irradiated Fuel Bays (IFB). The water provides cooling, shielding for personnel, and visibility, and it also allows easy handling without removal. In the short to medium term, the IFBs at each station are more than adequate to handle the irradiated fuel, but in time, there may be a need to move the older fuel into dry storage on site. This is feasible because over time, the radiation levels and heat produced by the fuel drops off significantly. Several sites now have some of the fuel used at the site stored in dry storage containers at the site.

26.7.2 Waste Volume Reduction and Storage

A typical radioactive waste storage facility is designed to reduce, by incineration or compacting, the volume of waste that requires storage. These facilities handle both low and medium level radioactive waste.

Radioactive waste is trucked to the facility in specially designed metal containers. If possible, the waste is incinerated and the radioactive ash loaded into 200 litre drums with a volume reduction about 20 to 1. Waste that cannot be burnt because of metal content or high radioactivity is compacted with a volume reduction about 4 to 1.

After treatment, the radioactive waste is reclassified as low, medium or high level. Low-level waste is stored in a warehouse. Medium level waste is stored in deep trenches. High-level waste is stored in deep tile holes.

Examples of radioactive wastes are:

- rags and protective clothing (low, medium or high level),
- equipment components or tools (medium or high level),
- H.T. gland seal filters (high level),
- Moderator & H.T. spent ion exchange resin (high level), and neutron activated components from the reactor core (high level).

26.8 Heavy Water Management

Heavy water is very expensive to produce and when it is used in a reactor it becomes radioactive largely due to the production of tritium. It is therefore critical that heavy water is managed in such a way as to:

- minimize permanent losses,
- reduce our environmental emissions, and
- minimize the chronic hazard to personnel.

Figure 26.7 shows the systems used for managing D_2O in a CANDU station.

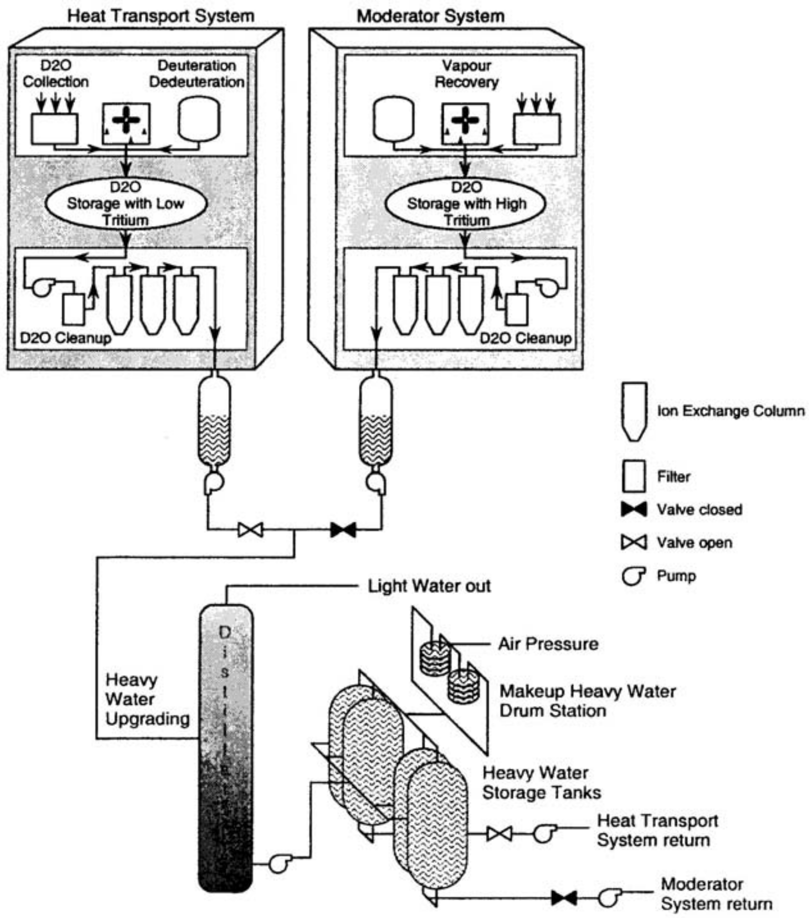


Figure 26.7
CANDU Station D_2O Management Systems

26.8.1 Loss Recovery

Part of the cost of operating a CANDU station is the D_2O upkeep cost. This consists both of replacement costs for D_2O lost permanently to the station, and the cost of upgrading D_2O that has been downgraded (isotopic below limit).

D_2O can be lost permanently through:

- vapour losses (the largest factor),
- discharge of wet fuel bundles to the irradiated fuel bay,
- resin deuteration and dedeuteration which produces some downgraded D_2O that is not recoverable,
- D_2O sampling and analysis,
- component decontamination,
- the top product from the upgrader which contains a small percentage of unrecoverable D_2O, and
- moderator heat exchanger leaks.

To minimize D_2O losses, vapour recovery and special collection systems are employed. The areas where vapour losses are most likely have closed loop ventilation systems containing vapour recovery driers, which reclaim most of the vapour. The recovered D_2O is always downgraded due to mixing with moisture in the air. Despite this recovery system, vapour loss makes up the largest fraction of permanent heavy water loss.

Deuteration is a process of exchanging the light water ions on IX column resins with heavy water ions to ensure no downgrading when the columns are placed in service. The resin must be dedeuterated prior to disposal to recover the heavy water.

There are two types of D_2O liquid collection system. The open method uses drip trays under potential leak points such as flanged joints (used rarely). The closed system conveys leakage directly to collection tanks without it coming in contact with the atmosphere to prevent downgrading from moisture in the air. This type of leakage generally occurs from double-packed valve stems or bellows-sealed valves.

26.8.2 Upgrading

If D_2O is downgraded below a certain isotopic then it is not worth recovering. If it is economical to recover the D_2O then it is passed through a station upgrader. The upgrader uses distillation to separate the heavy water from the light water. The output is 99.9% or greater D_2O.

26.8.3 Tritium Removal

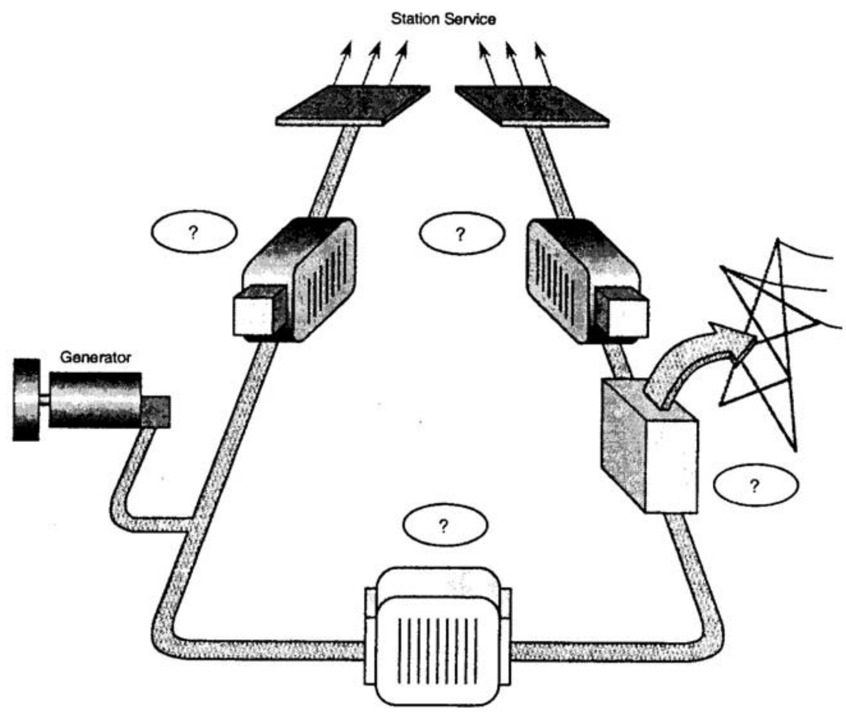
The Tritium Removal Facility (TRF) located at Darlington NGS forms a part of the heavy water management system. The TRF has the capability to remove tritium in batches of moderator and heat transport system D_2O. Replacing water in the Moderator and H.T. systems with this low tritium level water effectively dilutes the tritium levels in these systems and thereby reduces the tritium hazard to personnel. The extracted tritium is marketed commercially.

26.9 Assignment

1) What are the three basic prerequisites to producing electricity in an AC generator?

2) How is heat produced in an AC generator and what two methods are used for removing it?

3) In the basic CANDU power system diagram shown below, label and briefly explain the purpose of each component identified by a question mark (?).



4) Fill in the following table.

Class of Power	Length of Possible Interruption	Normal Power Source	Alternate Power Source	Example
Class IV				
Class III				
Class II				
Class I				

5) What is the purpose of the EPS?

6) State the purpose of the following systems:

a) Water Treatment Condenser
b) Cooling Water Common Service Water
c) Low-pressure Service Water
d) High-pressure Recirculating Service Water
e) Closed Loop Demineralized
f) Service Water System

5) What is the Emergency Water System and from where does it draw its supply?

6) State the purpose of the following systems:
   a. Instrument Air
   b. Service Air
   c. Breathing Air

7) Why is irradiated fuel stored in deep pools of water?

8) Why is it important to reduce the volume of solid wastes

9) Why is D_2O managed so carefully?

10) Where might losses of D_2O occur and what systems are in place to keep these D_2O losses to a minimum?

11) What function is served by the Tritium Removal Facility?