ATTACHMENT B

INTRODUCTORY DOCUMENTATION FOR NUCLEAR TECHNOLOGY PEBBLE BED MODULAR REACTOR

APPENDIX
INTRODUCTION TO
NUCLEAR POWER AND NTPBMR TECHNOLOGY
TABLE OF CONTENTS

ASCENTRUST LLC
The POWER of ENGINEERING

APPENDIX
INTRODUCTION TO
NUCLEAR POWER AND NTPBMR TECHNOLOGY

SECTION A: Introduction to Nuclear Physics

ASCENTRUST LLC
The POWER of ENGINEERING

INTRODUCTION TO NUCLEAR REACTOR PHYSICS

The purpose of this composite white papers on **Nuclear Reactor Physics** is the non-technical introduction of some of the most important concepts of physics required for a general understanding of the generation of energy through the use of nuclear power.

Inherent in any discussion on the structure of matter and certainly in the discussion of the production of electrical energy through the fission of nuclear material are some of the topics of modern physics: **Quantum Mechanics and Special Relativity.**

In this introduction to nuclear physics we shall consider only those laws of mechanics, thermodynamics and hydraulics which can appropriately be discussed in the guise of "Rational Mechanics", and we will use the concepts which form the basis of modern Physics: **Quantum Mechanics and Special Relativity** only when absolutely necessary to impart the information needed to the reader.

The discovery of fission in 1939 was and event of epochal significance in the annals of physics because it ushered in the age of the atom. This discovery opened up the prospect of an entirely new source of power utilizing the internal binding energy of the atom.

The operation of a nuclear reactor depends upon various interactions of neutrons with atomic nuclei. In order to appreciate the complexities of a nuclear reactor it is desirable to consider briefly some of the fundamental of atomic and nuclear physics. This paper was produced to provide such an introduction.

CHAPTER ONE: INTRODUCTION TO NUCLEAR PHYSICS

SECTION ONE: BASIC CONCEPTS

1.1. HISTORY OF STRUCTURE OF MATTER

Early Greek philosophers speculated that the earth was made up of different combinations of basic substances, or elements. They considered these basic elements to be earth, air, water, and fire. Modern science shows that the early Greeks held the correct concept that matter consists of a combination of basic elements, but they incorrectly identified the elements.

In 1661 the English chemist Robert Boyle published the modern criterion for an element. He defined an element to be a basic substance that cannot be broken down into any simpler substance after it is isolated from a compound, but can be combined with other elements to form compounds. To date, 105 different elements have been confirmed to exist, and researchers claim to have discovered three additional elements. Of the 105 confirmed elements, 90 exist in nature and 15 are man-made.

Another basic concept of matter that the Greeks debated was whether matter was continuous or discrete. That is, whether matter could be continuously divided and subdivided into ever smaller particles or whether eventually an indivisible particle would be encountered. Democritus in about 450 B.C. argued that substances were ultimately composed of small, indivisible particles that he labeled "**atoma**". He further suggested that different substances were composed of different atoms or combinations of atoms, and that one substance could be converted into another by rearranging the atoms. It was impossible to conclusively prove or disprove this proposal for more than 2000 years.

The modern proof for the atomic nature of matter was first proposed by the English chemist John Dalton in 1803. Dalton stated that each chemical element possesses a particular kind of atom, and any quantity of the element is made up of identical atoms of this kind. What distinguishes one element from another element is the kind of atom of which it consists, and the basic physical difference between kinds of atoms is their weight.

1.2. SUBATOMIC PARTICLES

For almost 100 years after Dalton established the atomic nature of atoms, it was considered impossible to divide the atom into even smaller parts. All of the results of chemical experiments during this time indicated that the atom was indivisible. Eventually, experimentation into electricity and radioactivity indicated that particles of matter smaller than the atom did indeed exist.

In 1906, J. J. Thompson won the Nobel Prize in physics for establishing the existence of electrons. **Electrons** are negatively-charged particles that have 1/1835 the mass of the hydrogen atom. Soon after the discovery of electrons, protons were discovered. **Protons** are relatively large particles that have almost the same mass as a hydrogen atom and a positive charge equal in magnitude (but opposite in sign) to that of the electron. The third subatomic particle to be

discovered, the **neutron**, was not found until 1932. The **neutron** has almost the same mass as the proton, but it is electrically neutral.

1.3. BOHR MODEL OF THE ATOM

The British physicist Ernest Rutherford postulated that the positive charge in an atom is concentrated in a small region called a nucleus at the center of the atom with electrons existing in orbits around it.

Niels Bohr, coupling Rutherford's postulation with the newly minted theories of quantum mechanics introduced by Max Planck, proposed that the atom consists of a dense nucleus of **protons** surrounded by **electrons** traveling in discrete orbits at fixed distances from the nucleus.

An electron in one of these stationary orbits or shells has a specific or discrete quantity of energy (quantum). When an electron moves from one allowed orbit to another allowed orbit, the energy difference between the two states is emitted or absorbed in the form of a single quantum of radiant energy called a **photon**.

The Quantum of energy emitted from this jump from one stationary state to another is given by the so called Plank formula:

$$\mathbf{E = h\nu}$$

Where **h** = Planck's constant = 6.63×10^{-34} J-s

ν= frequency of the photon.

Bohr's theory was the first to successfully account for the discrete energy levels of this radiation as measured in the laboratory. Although Bohr's atomic model was designed specifically to explain the hydrogen atom, his theories apply generally to the structure of all atoms. Additional information on electron shell theory can be found in any introductory book on **Quantum Mechanics.**

1.4. MEASURING UNITS ON THE ATOMIC SCALE

The size and mass of atoms are so small that the use of normal measuring units, while possible, is often inconvenient. Units of measure have been defined for mass and energy on the atomic scale to make measurements more convenient to express.

The unit of measure for mass is the **atomic mass unit** (**amu**). One atomic mass unit is equal to 1.66×10^{-24} grams. The reason for this particular value for the atomic mass unit will be made clear later in this introduction. Note that the mass of a neutron and a proton are both about 1 amu.

The unit for energy is the **electron volt** (**eV**). The electron volt is the amount of energy acquired by a single electron when it falls through a potential difference of one volt. One electron volt is equivalent to 1.602×10^{-19} joules or 1.18×10^{-19} foot-pounds.

1.5. NUCLIDES

The total number of protons in the nucleus of an atom is called the ***atomic number*** of the atom and is given the symbol **Z**. The number of electrons in an electrically-neutral atom is the same as the number of protons in the nucleus.

The number of neutrons in a nucleus is known as the neutron number and is given the symbol **N**. The **mass number** of the nucleus is the total number of nucleons, that is, protons and neutrons in the nucleus. The mass number is given the symbol **A** and can be found by the equation

$$A = Z + N.$$

Each of the chemical elements has a unique atomic number because the atoms of different elements contain a different number of protons. The atomic number of an atom identifies the particular element.

Each type of atom that contains a unique combination of protons and neutrons is called a **nuclide.** Not all combinations of numbers of protons and neutrons are possible, but about 2500 specific nuclides with unique combinations of neutrons and protons have been identified. Each nuclide is denoted by the chemical symbol of the element with the atomic number written as a subscript and the mass number written as a superscript.

Because each element has a unique name, chemical symbol, and atomic number, only one of the three is necessary to identify the element. For this reason nuclides can also be identified by either the chemical name or the chemical symbol followed by the mass number (for example, U-235 or uranium-235).

Another common format is to use the abbreviation of the chemical element with the mass number superscripted (for example, $\mathbf{U^{235}}$). In this white paper the format used will usually be the element's name followed by the mass number as a superscript.

1.6. ISOTOPES

Isotopes are nuclides that have the same atomic number and are therefore the same element, but differ in the number of neutrons. Most elements have a few stable isotopes and several unstable, radioactive isotopes. For example, oxygen has three stable isotopes that can be found in nature (**oxygen^{16}, oxygen^{17}**, **oxygen^{18},** and eight radioactive isotopes. Another example is hydrogen, which has two stable isotopes (**hydrogen-1**, **hydrogen^{1}** and hydrogen-2) and a single radioactive isotope (hydrogen-3, **hydrogen^{3}**).

The isotopes of hydrogen are unique in that they are each commonly referred to by a unique name instead of the common chemical element name. Hydrogen-1 is almost always referred to as hydrogen, but the term protium is infrequently used also. Hydrogen-2 is commonly called deuterium and Hydrogen-3 is commonly called tritium.

1.7. ATOMIC AND NUCLEAR RADII

The size of an atom is difficult to define exactly due to the fact that the electron cloud, formed by the electrons moving in their various orbitals, does not have a distinct outer edge. A reasonable measure of atomic size is given by the average distance of the outermost electron from the nucleus.

Except for a few of the lightest atoms, the average atomic radii are approximately the same for all atoms, about **2×10^{-8} cm**. Like the atom the nucleus does not have a sharp outer boundary. Experiments have shown that the nucleus is shaped like a sphere with a radius that depends on the atomic mass number of the atom.

1.8. NUCLEAR FORCES

In the Bohr model of the atom, the nucleus consists of positively-charged protons and electrically neutral neutrons. Since both protons and neutrons exist in the nucleus, they are both referred to as nucleons. One problem that the Bohr model of the atom presented was accounting for an attractive force to overcome the repulsive force between protons.

Two of the four forces present in the nucleus are;

1. Electrostatic forces between charged particles
2. Gravitational forces between any two objects that have mass.

It is possible to calculate the magnitude of the gravitational force and electrostatic force based upon principles from classical physics. Newton stated that the **gravitational force** between two bodies is directly proportional to the masses of the two bodies and inversely proportional to the square of the distance between the bodies. This relationship is shown in the equation below.

$$F_g = (G \times m_1 \times m_2)/ r_{12}$$

where:

F_g = gravitational force (newtons)
m_1 = mass of first body (kilograms)
m_2 = mass of second body (kilograms)
G = gravitational constant (6.67 x 10 -11 N-m2/kg2)
r = distance between particles (meters)

The equation illustrates that the larger the masses of the objects or the smaller the distance between the objects, the greater the gravitational force. So even though the masses of nucleons are very small, the fact that the distance between nucleons is extremely short may make the gravitational force significant. The gravitational force between two protons that are separated by a distance of 10^{-20} meters is about 10^{-24} newtons.

Coulomb's Law can be used to calculate the force between two protons. The **electrostatic force** is proportional to the electrical charges of the two particles and inversely proportional to the square of the distance between the particles.

Coulomb's Law is stated as the following equation:

$$F_e = \frac{(K \times Q_1 \times Q_2)}{r_{12}}$$

where:

F_e = Electrostatic force (newtons)
K = electrostatic constant (9.0 x 109 N-m2/C2)
Q_1 = charge of first particle (coulombs)
Q_2 = charge of second particle (coulombs)
r_{12} = Distance between particles (meters)

Using this equation, the electrostatic force between two protons that are separated by a distance of **10^{-20}** meters is about **10^{12}** newtons. Comparing this result with the calculation of the gravitational force (**10^{-24}** newtons) shows that the gravitational force is so small that it can be neglected.

If only the electrostatic and gravitational forces existed in the nucleus, then it would be impossible to have stable nuclei composed of protons and neutrons. The gravitational forces are much too small to hold the nucleons together compared to the electrostatic forces repelling the protons. Since stable atoms of neutrons and protons do exist, there must be another attractive force acting within the nucleus. This force is called the **nuclear force**.

The **nuclear force** is a strong attractive force that is independent of charge. It acts equally only between pairs of neutrons, pairs of protons, or a neutron and a proton. The nuclear force has a very short range; it acts only over distances approximately equal to the diameter of the nucleus (**10^{-13}** cm). The attractive nuclear force between all nucleons drops off with distance much faster than the repulsive electrostatic force between protons.

1.9. ATOMIC NATURE OF MATTER: SUMMARY

An atom consists of a positively charged **Nucleus** surrounded by a number of negatively charged particles, called **electrons**, so that the atom as a whole is electrically neutral. The atomic nuclei are built up of two kinds of primary particles, namely **protons** and **neutrons**, which can be ordinarily referred to as **Nucleons.** The masses of the protons and neutrons are similar and much heavier than an electron, by a factor of 1840. As the nucleus contains all the protons and neutrons, it follows that the mass of the atoms are concentrated in the nucleus.

The proton carries a single unit of charge and the neutron is electrically neutral. The unit of charge carried by the proton is equal in magnitude and opposite in sign, to the charge on the electron. This charge is often referred to in physics as the fundamental charge.

Each electron carries a negative charge equal to the charge on the proton. The number of orbital electrons is equal to the number of protons in the nucleus so that their charge balances and overall the atom is electrically neutral.

The number of protons in the nucleus determines the number of orbital electrons. It is the electronic structure of the atom, in particular the outermost orbiting electrons that give the atoms its chemical properties.

In this paper we will only mention that the electron, in a stable orbit around the nucleus occupies a stationary state, often referred to in Physics as an Eigenstate or an Eigunfunction of the orbital decomposition of the central field problem. The solution of the electronic motion of an electron in a central field of force will not be addressed here. The force which holds the electron to the nucleus is the electrostatic force of electromagnetic theory.

SECTION TWO: STRUCTURE OF THE NUCLEUS

2.1. ATOMIC NUMBER AND CHART OF NUCLIDES

As has been previously stated the nucleus consists of protons and neutrons. For a given element, the number of protons present in the atomic nucleus, which is the same as the number of positive charges it carries, is called the **atomic number.** This number is identical with the ordinal number of the element which is used in the familiar Periodic table of the elements. Thus the atomic number of hydrogen is 1, of helium 2, and lithium 3, up to 92 for **Uranium,** the element of highest atomic number existing in nature. A large number of heavier elements have been produced artificially, of these elements, plutonium, atomic number 94, is the most important because of its connection with nuclear weapons.

The total number of **Nucleons,** i.e. of protons and neutrons, in an atomic nucleus is referred to as the **mass number.** Since the masses of neutrons and protons and very nearly identical, it is evident that the mass number is the integer nearest to the atomic weight of the species under considerations.

It is the atomic number, i.e. the number of protons in the nucleus, which determines the chemical nature of an element. This is because the chemical properties depend on the (orbital) electrons, surrounding the nucleus, and their number must be equal to the number of protons in the nucleus since the atom must be electrically neutral. Consequently, atoms with nuclei containing the same numbers of protons, i.e. with the same atomic number, but with different numbers of neutrons, i.e. with different mass number, are essentially identical chemically. Such species having the same atomic number but different mass numbers are called **Isotopes.** Isotopes are, in general, chemically identical but have different atomic weight. They are, in general, indistinguishable chemically, but have different atomic weights. As of January 1, 1962, all atomic weights are expressed on a single scale which assigns a value of 12 to the common isotope of $\mathbf{C^{12}}$.

In nuclear physics, and related fields, the masses of atoms, of nuclei, and of nuclear particles are invariably expressed on the so-called physical scale. The **Atomic Mass unit** (amu) is then defined as exactly one-twelfth of the mass of the $\mathbf{C^{12}}$ atom.

2.2. CHART OF THE NUCLIDES

A tabulated chart called the **Chart of the Nuclides** lists the stable and unstable nuclides in addition to pertinent information about each one. This chart plots a box for each individual nuclide, with the number of protons (**Z**) on the vertical axis

and the number of neutrons (**N = A - Z**) on the horizontal axis. The chart indicates stable isotopes.

Some **isotopes** are **artificially radioactive**, meaning that they are produced by artificial techniques and do not occur naturally.

A. Information for Stable Nuclides

For the stable isotopes, in addition to the symbol and the atomic mass number, the number percentage of each isotope in the naturally occurring element is listed, as well as the thermal neutron activation cross section and the mass in atomic mass units (amu).

B. Information for Unstable Nuclides

For unstable isotopes the additional information includes the half life, the mode of decay (for example, b-, a), the total disintegration energy in **MeV** (**million electron volts**), and the mass in **amu** when available

C. Neutron - Proton Ratios

If you plot the number of protons on the x-axis and the number of neutrons on the y axis, then you will see that as the mass numbers become higher, the ratio of neutrons to protons in the nucleus becomes larger. For helium-4 (2 protons and 2 neutrons) and oxygen-16 (8 protons and 8 neutrons) this ratio is unity. For indium-115 (49 protons and 66 neutrons) the ratio of neutrons to protons has increased to 1.35, and for uranium-238 (92 protons and 146 neutrons) the neutron to-proton ratio is 1.59.

D. Natural Abundance of Isotopes

The relative abundance of an isotope in nature compared to other isotopes of the same element is relatively constant. The Chart of the Nuclides presents the relative abundance of the naturally occurring isotopes of an element in units of **atom percent**. **Atom percent** is the percentage of the atoms of an element that are of a particular isotope.

Atom percent is abbreviated as **a/o.** For example, if a cup of water contains 8.23×10^{24} atoms of oxygen, and the isotopic abundance of oxygen-18 is 0.20%, then there are 1.65×10^{22} atoms of oxygen-18 in the cup.

E. Atomic Weight

The **atomic weight** for an element is defined as the average atomic weight of the isotopes of the element. The atomic weight for an element can be calculated by summing the products of the isotopic abundance of the isotope with the atomic mass of the isotope.

SECTION THREE: MASS DEFECT AND BINDING ENERGY

3.1. MASS DEFECT

Careful measurements have shown that the mass of a particular atom is always slightly less than the sum of the masses of the individual neutrons, protons, and

electrons of which the atom consists. The difference between the mass of the atom and the sum of the masses of its parts is called the **mass defect (D_m)**.

The mass defect can be calculated using the equation show below. In calculating the mass defect it is important to use the full accuracy of mass measurements because the difference in mass is small compared to the mass of the atom. Rounding off the masses of atoms and particles to three or four significant digits prior to the calculation will result in a calculated mass defect of zero.

$$D_m = [\, Z(m_p + m_e) + (A-Z)m_n \,] - m_{atom}$$

where:

D_m = mass defect (**amu**)
m_p = mass of a proton (1.007277 **amu**)
m_n = mass of a neutron (1.008665 **amu**)
m_e = mass of an electron (0.000548597 **amu**)
m_{atom} = mass of nuclide (**amu**)
Z = atomic number (number of protons)
A = mass number (number of nucleons)

3.2. BINDING ENERGY

The loss in mass, or mass defect, is due to the conversion of mass to binding energy when the nucleus is formed. **Binding energy** is defined as the amount of energy that must be supplied to a nucleus to completely separate its nuclear particles (nucleons). It can also be understood as the amount of energy that would be released if the nucleus was formed from the separate particles.

Binding energy is the energy equivalent of the mass defect. Since the mass defect was converted to binding energy (BE) when the nucleus was formed, it is possible to calculate the binding energy using a conversion factor derived by the mass-energy relationship from **Einstein's Theory of Relativity.**

Einstein's famous equation relating mass and energy is;

$$E = mc^2$$

Where;

E = Energy in Joules
m = mass in kilograms
c = is the velocity of light (**$c = 3 \times 10^8$ meters/sec**).

The energy equivalent of **1 amu** can be determined by inserting this quantity of mass into Einstein's equation and applying conversion factors.
1 amu = 1.6606×10^{-27} kg

3.3. ENERGY LEVELS OF ATOMS

The electrons that circle the nucleus move in fairly well-defined orbits. Some of these electrons are more tightly bound in the atom than others. For example, only 7.38 eV is required to remove the outermost electron from a lead atom, while 88,000 eV is required to remove the innermost electron.

The process of removing an electron from an atom is called **ionization,** and the energy required to remove the electron is called the ionization energy. In a neutral atom (number of electrons **= Z**) it is possible for the electrons to be in a variety of different orbits, each with a different energy level. The state of lowest energy is the one in which the atom is normally found and is called the ground state. When the atom possesses more energy than its ground state energy, it is said to be in an excited state.

An atom cannot stay in the excited state for an indefinite period of time. An excited atom will eventually transition to either a lower-energy excited state, or directly to its ground state, by emitting a discrete bundle of electromagnetic energy called an x-ray. The energy of the x-ray will be equal to the difference between the energy levels of the atom and will typically range from several eV to 100,000 eV in magnitude.

3.4. ENERGY LEVELS OF THE NUCLEUS

The nucleons in the nucleus of an atom, like the electrons that circle the nucleus, exist in shells that correspond to energy states. The energy shells of the nucleus are less defined and less understood than those of the electrons.

There is a state of lowest energy (the ground state) and discrete possible excited states for a nucleus. Where the discrete energy states for the electrons of an atom are measured in **eV** or **keV,** the energy levels of the nucleus are considerably greater and typically measured in **MeV.**

A nucleus that is in the excited state will not remain at that energy level for an indefinite period. Like the electrons in an excited atom, the nucleons in an excited nucleus will transition towards their lowest energy configuration and in doing so emit a discrete bundle of electromagnetic radiation called a gamma ray (g-ray). The only differences between x-rays and g-rays are their energy levels and whether they are emitted from the electron shell or from the nucleus.

SECTION FOUR: MODES OF RADIOACTIVE DECAY

Most atoms found in nature are stable and do not emit particles or energy that change form over time. Some atoms, however, do not have stable nuclei. These atoms emit radiation in order to achieve a more stable configuration.

4.1. STABILITY OF NUCLEI

As mass numbers become larger, the ratio of neutrons to protons in the nucleus becomes larger for the stable nuclei. Non-stable nuclei may have an excess or deficiency of neutrons and undergo a transformation process known as **beta** (β) decay.

Non-stable nuclei can also undergo a variety of other processes such as alpha (α) or neutron (**n**) decay. As a result of these decay processes, the final nucleus is in a more stable or more tightly bound configuration.

4.2. NATURAL RADIOACTIVITY

In 1896, the French physicist Becquerel discovered that crystals of a uranium salt emitted rays that were similar to x-rays in that they were highly penetrating, could affect a photographic plate, and induced electrical conductivity in gases. Becquerel's discovery was followed in 1898 by the identification of two other radioactive elements, **polonium** and **radium,** by Pierre and Marie Curie.

Heavy elements, such as **uranium** or **thorium**, and their unstable decay chain elements emit radiation in their naturally occurring state. **Uranium** and **thorium**, present since their creation at the beginning of geological time, have an extremely slow rate of decay. All naturally occurring nuclides with atomic numbers greater than 82 are radioactive.

4.3. NUCLEAR DECAY

Whenever a nucleus can attain a more stable (i.e., more tightly bound) configuration by emitting radiation, a spontaneous disintegration process known as radioactive decay or nuclear decay may occur. In practice, this "radiation" may be electromagnetic radiation, particles, or both. Detailed studies of radioactive decay and nuclear reaction processes have led to the formulation of useful conservation principles.

The four principles of most interest in this document are discussed below.

1. **Conservation of electric charge:** Conservation of electric charge implies that sum of the charges in the beginning of a process is equal to the sum of the charges after the interaction has occurred.
2. **Conservation of mass number:** Conservation of mass number does not allow a net change in the number of nucleons. However, the conversion of a proton to a neutron and vice versa is allowed.
3. **Conservation of mass and energy:** Implies that the total of the kinetic energy and the energy equivalent of the mass in a system must be conserved in all decays and reactions. Mass can be converted to energy and energy can be converted to mass, but the sum of mass and energy must be constant.
4. **Conservation of momentum:** is responsible for the distribution of the available kinetic energy among product nuclei, particles, and/or radiation. The total amount is the same before and after the reaction even though it may be distributed differently among entirely different nuclides and/or particles.

4.4. ALPHA DECAY (α)

Alpha decay is the emission of **alpha particles** (**helium nuclei**) which may be represented as either $\mathbf{He^4}$ or **(**α**)**. When an unstable nucleus ejects an alpha particle, the atomic number is reduced by 2 and the mass number decreased by 4. An example is uranium-234, ($\mathbf{U^{234}}$) which decays by the ejection of an alpha particle accompanied by the emission of a 0.068 MeV **gamma ray photon**.

The combined **kinetic energy** of the new nucleus (Thorium-230, **Th** ^{230}) and the α particle is designated as **KE.** The sum of the **KE** and the **gamma energy** is equal to the difference in mass between the original nucleus **U**^{234} (Uranium-234) and the final particles (equivalent to the binding energy released, since Δ**m = BE).** The alpha particle will carry off as much as 98% of the kinetic energy and, in most cases, can be considered to carry off all the kinetic energy.

4.5. BETA DECAY (β)

Beta decay is the emission of electrons of nuclear rather than orbital origin. These particles are electrons that have been expelled by excited nuclei and may have a charge of either sign. If both energy and momentum are to be conserved, a third type of particle, the **neutrino** must be involved.

The **neutrino** is associated with positive electron emission, and its antiparticle, the **antineutrino**, is emitted with a negative electron. These uncharged particles have only the weakest interaction with matter, no mass, and travel at the speed of light. For all practical purposes, they pass through all materials with so few interactions that the energy they possess cannot be recovered.

The **neutrinos** and **antineutrinos** are included here only because they carry a portion of the kinetic energy that would otherwise belong to the beta particle, and therefore, must be considered for energy and momentum to be conserved. They are normally ignored since they are not significant in the context of nuclear reactor applications.

Negative electron emission, effectively converts a neutron to a proton, thus increasing the atomic number by one and leaving the mass number unchanged. This is a common mode of decay for nuclei with an excess of neutrons.

Positively charged electrons (beta-plus) are known as **positrons.** Except for sign, they are nearly identical to their negatively charged cousins. When a positron is ejected from the nucleus, the atomic number is decreased by one and the mass number remains unchanged. A proton has been converted to a neutron.

4.6. ELECTRON CAPTURE (EC, K-capture)

Nuclei having an excess of protons may capture an electron from one of the inner orbits which immediately combines with a proton in the nucleus to form a neutron. This process is called **electron capture** (**EC**). The electron is normally captured from the innermost orbit (the **K-shell**), and, consequently, this process is sometimes called **K-capture**.

A neutrino is formed at the same time that the neutron is formed, and energy carried off by it serves to conserve momentum. Any energy that is available due to the atomic mass of the product being appreciably less than that of the parent will appear as gamma radiation. Also, there will always be characteristic x-rays given off when an electron from one of the higher energy shells moves in to fill the vacancy in the K-shell. Electron capture and positron emission result in the

production of the same daughter product, and they exist as competing processes.

For positron emission to occur, however, the mass of the daughter product must be less than the mass of the parent by an amount equal to at least twice the mass of an electron. This mass difference between the parent and daughter is necessary to account for two items present in the parent but not in the daughter. One item is the positron ejected from the nucleus of the parent. The other item is that the daughter product has one less orbital electron than the parent. If this requirement is not met, then orbital electron capture takes place exclusively.

4.7. GAMMA EMISSION (γ)

Gamma radiation is a high-energy electromagnetic radiation that originates in the nucleus. It is emitted in the form of **photons,** discrete bundles of energy that have both wave and particle properties. Often a daughter nuclide is left in an excited state after a radioactive parent nucleus undergoes a transformation by **alpha decay**, **beta decay**, or **electron capture**. The nucleus will drop to the ground state by the emission of gamma radiation.

4.8. INTERNAL CONVERSION

The usual method for an excited nucleus to go from the excited state to the ground state is by emission of **gamma radiation**. However, in some cases the gamma ray (photon) emerges from the nucleus only to interact with one of the innermost orbital electrons and, as a result, the energy of the photon is transferred to the electron. The gamma ray is then said to have undergone **internal conversion**. The conversion electron is ejected from the atom with kinetic energy equal to the gamma energy minus the binding energy of the orbital electron. An orbital electron then drops to a lower energy state to fill the vacancy, and this is accompanied by the emission of characteristic x-rays.

4.9. ISOMERS AND ISOMERIC TRANSITION

Isomeric transition commonly occurs immediately after particle emission; however, the nucleus may remain in an excited state for a measurable period of time before dropping to the ground state at its own characteristic rate. A nucleus that remains in such an excited state is known as a **nuclear isomer** because it differs in energy and behavior from other nuclei with the same atomic number and mass number. The decay of an excited nuclear isomer to a lower energy level is called an **isomeric transition**. It is also possible for the excited isomer to decay by some alternate means, for example, by beta emission.

4.10. DECAY CHAINS

When an unstable nucleus decays, the resulting daughter nucleus is not necessarily stable. The nucleus resulting from the decay of a parent is often itself unstable, and will undergo an additional decay. It is possible to trace the steps of an unstable atom as it goes through multiple decays trying to achieve stability. The list of the original unstable nuclide, the nuclides that are involved as

intermediate steps in the decay, and the final stable nuclide is known as the **decay chain**.

4.11. PREDICTING TYPES OF DECAY

Radioactive nuclides tend to decay in a way that results in a daughter nuclide that lies closer to the line of stability. Due to this, it is possible to predict the type of decay that a nuclide will undergo based on its location relative to the line of stability.

SECTION FIVE: RADIOACTIVITY

The rate at which a sample of radioactive material decays is not constant. As individual atoms of the material decay, there are fewer of those types of atoms remaining. Since the rate of decay is directly proportional to the number of atoms, the rate of decay will decrease as the number of atoms decreases.

5.1. DEFINITION OF RADIOACTIVITY

Radioactivity is the property of certain nuclides of spontaneously emitting particles or gamma radiation. The decay of radioactive nuclides occurs in a random manner, and the precise time at which a single nucleus will decay cannot be determined. However, the average behavior of a very large sample can be predicted accurately by using statistical methods. These studies have revealed that there is a certain probability that in a given time interval a certain fraction of the nuclei within a sample of a particular nuclide will decay.

This probability per unit time that an atom of a nuclide will decay is known as the **radioactive decay constant**. The units for the decay constant are inverse time such as 1/second, 1/minute, 1/hour, or 1/year.

5.2. ACTIVITY

The **activity (A)** of a sample is the rate of decay of that sample. This rate of decay is usually measured in the number of disintegrations that occur per second. For a sample containing millions of atoms, the activity is the product of the decay constant and the number of atoms present in the sample.

The relationship between the activity, number of atoms, and decay constant is shown below;

$$\mathbf{A} = \lambda \mathbf{N}$$

where:

A = Activity of the nuclide (disintegrations/second)
λ = decay constant of the nuclide (second-1)
N = Number of atoms of the nuclide in the sample

Since λ is a constant, the activity and the number of atoms are always proportional.

5.3. UNITS OF MEASUREMENT FOR RADIOACTIVITY

Two common units to measure the activity of a substance are the **curie** (**Ci**) and **becquerel** (**Bq**). A **curie** is a unit of measure of the rate of radioactive decay equal to 3.7×10^{10} disintegrations per second. This is approximately equivalent to

the number of disintegrations that one gram of radium-226 will undergo in one second. A **becquerel** is a more fundamental unit of measure of radioactive decay that is equal to 1 disintegration per second.

Currently, the **curie** is more widely used in the United States, but usage of the **becquerel** can be expected to broaden as the metric system slowly comes into wider use. The conversion between curies and becquerels is shown below.

1 curie = 3.7×10^{10} becquerels

5.4. VARIATION OF RADIOACTIVITY OVER TIME

The rate at which a given radionuclide sample decays is stated in section **5.2** as being equal to the product of the number of atoms and the decay constant.

From this basic relationship it is possible to use calculus to derive an expression which can be used to calculate how the number of atoms present will change over time. The derivation is beyond the scope of this document but the following equation is the useful result of the solution of this important differential equation:

$$\mathbf{N_t = N_o\, e^{-\lambda t}}$$

where:

$\mathbf{N_t}$ = number of atoms present at time t
$\mathbf{N_o}$ = number of atoms initially present o
e = Natural logarithm
λ= decay constant (time-1)
t = time

5.5. RADIOACTIVE HALF-LIFE

One of the most useful terms for estimating how quickly a nuclide will decay is the **radioactive half-life**. The **radioactive half-life** is defined as the amount of time required for the activity to decrease to one-half of its original value.

A relationship between the half-life and decay constant can be developed from the equation developed in section **5.4.**

Assuming an initial number of atoms $\mathbf{N_o}$ the population, and consequently, the activity may be noted to decrease by one-half of this value in a time of one half-life. Additional decreases occur so that whenever one half-life elapses, the number of atoms drops to one-half of what its value was at the beginning of that time interval. After five half-lives have elapsed, only 1/32, or 3.1%, of the original number of atoms remains. After seven half-lives, only 1/128, or 0.78%, of the atoms remains. The number of atoms existing after 5 to 7 half-lives can usually be assumed to be negligible.

5.6. PLOTTING RADIOACTIVE DECAY

It is useful to plot the activity of a nuclide as it changes over time. Plots of this type can be used to determine when the activity will fall below a certain level. This plot is usually done showing activity on either a linear or a logarithmic scale. The decay of the activity of a single nuclide on a logarithmic scale will plot as a straight line because the decay is exponential. If a substance contains more than one radioactive nuclide, the total activity is the sum of the individual activities of

each nuclide. The initial activity of each of the nuclides would be the product of the number of atoms and the decay constant.

5.7. RADIOACTIVE EQUILIBRIUM

A. Radioactive equilibrium exists when a radioactive nuclide is decaying at the same rate at which it is being produced. Since the production rate and decay rate are equal, the number of atoms present remains constant over time.

B. Transient radioactive equilibrium occurs when the parent nuclide and the daughter nuclide decay at essentially the same rate. For transient equilibrium to occur, the parent must have a long half-life when compared to the daughter. An example of this type of compound decay process is barium-140, which decays by beta emission to **lanthanum-140,** which in turn decays by beta emission to stable **cerium-140**.

The decay constant for **barium-140** is considerably smaller than the decay constant for **lanthanum-140**. Remember that the rate of decay of both the parent and daughter can be represented as λ**N**. Although the decay constant for **barium-140** is smaller, the actual rate of decay (λ**N)** is initially larger than that of **lanthanum-140** because of the great difference in their initial concentrations. As the concentration of the daughter increases, the rate of decay of the daughter will approach and eventually match the decay rate of the parent. When this occurs, they are said to be in transient equilibrium.

C. Secular equilibrium occurs when the parent has an extremely long half-life. In the long decay chain for a naturally radioactive element, such as thorium-232, where all of the elements in the chain are in secular equilibrium, each of the descendants has built up to an equilibrium amount and all decay at the rate set by the original parent. The only exception is the final stable element on the end of the chain. Its number of atoms is constantly increasing.

6. NEUTRON INTERACTIONS

Neutrons can cause many different types of interactions. The neutron may simply scatter off the nucleus in two different ways, or it may actually be absorbed into the nucleus. If a neutron is absorbed into the nucleus, it may result in the emission of a gamma ray or a subatomic particle, or it may cause the nucleus to fission.

6.1. SCATTERING

A neutron **scattering** reaction occurs when a nucleus, after having been struck by a neutron, emits a single neutron. Despite the fact that the initial and final neutrons do not need to be (and often are not) the same, the net effect of the reaction is as if the projectile neutron had merely "bounced off," or scattered from, the nucleus. The two categories of scattering reactions, **elastic** and **inelastic** scattering, are described below:

A. ELASTIC SCATERING

In an **elastic scattering** reaction between a neutron and a target nucleus, there is no energy transferred into nuclear excitation. Momentum and kinetic energy of the "system" are conserved although there is usually some transfer of kinetic energy from the neutron to the target nucleus. The target nucleus gains the amount of kinetic energy that the neutron loses. Elastic scattering of neutrons by nuclei can occur in two ways:

1. The more unusual of the two interactions is the absorption of the neutron, forming a compound nucleus, followed by the re-emission of a neutron in such a way that the total kinetic energy is conserved and the nucleus returns to its ground state. This is known as **resonance elastic scattering** and is very dependent upon the initial kinetic energy possessed by the neutron. Due to formation of the compound nucleus, it is also referred to as compound elastic scattering.

2. The second, more usual method, is termed **potential elastic scattering** and can be understood by visualizing the neutrons and nuclei to be much like billiard balls with impenetrable surfaces. Potential scattering takes place with incident neutrons that have an energy of up to about **1 MeV.** In potential scattering, the neutron does not actually touch the nucleus and a compound nucleus is not formed. Instead, the neutron is acted on and scattered by the short range nuclear forces when it approaches close enough to the nucleus.

B. INELASTIC SCATTERING

In **inelastic scattering,** the incident neutron is absorbed by the target nucleus, forming a compound nucleus. The compound nucleus will then emit a neutron of lower kinetic energy which leaves the original nucleus in an excited state. The nucleus will usually, by one or more gamma emissions, emit this excess energy to reach its ground state.

For the nucleus that has reached its ground state, the sum of the kinetic energy of the exit Inelastic Scattering neutron, the target nucleus, and the total gamma energy emitted is equal to the initial kinetic energy of the incident neutron.

6.2. ABSORPTION REACTIONS

Most **absorption reactions** result in the loss of a neutron coupled with the production of a charged particle or gamma ray. When the product nucleus is radioactive, additional radiation is emitted at some later time. **Radiative capture, particle ejection**, and **fission** are all categorized as absorption reactions and are briefly described below.

A. Radiative Capture: In radiative capture the incident neutron enters the target nucleus forming a compound nucleus. The compound nucleus then decays to its ground state by gamma emission.

B. Particle Ejection: In a **particle ejection** reaction the incident particle enters the target nucleus forming a compound nucleus. The newly formed compound nucleus has been excited to a high enough energy level to cause it to eject a new particle while the incident neutron remains in the nucleus. After the new particle is ejected, the remaining nucleus may or may not exist in an excited state depending upon the mass-energy balance of the reaction.

C. Fission: One of the most important interactions that neutrons can cause is fission, in which the nucleus that absorbs the neutron actually splits into two similarly sized parts. Fission will be discussed in detail in the next chapter.

SECTION SEVEN: NUCLEAR FISSION

Nuclear fission is a process in which an atom splits and releases energy, fission products, and neutrons. The neutrons released by fission can, in turn, cause the fission of other atoms.

7.1. FISSION

In the fission reaction the incident neutron enters the heavy target nucleus, forming a compound nucleus that is excited to such a high energy level (E > E) that the nucleus "splits" (fissions) into two large fragments plus some neutrons. A large amount of energy is released in the form of radiation and fragment kinetic energy.

7.2. LIQUID DROP MODEL OF A NUCLEUS

The nucleus is held together by the attractive nuclear force between nucleons. The characteristics of the nuclear force are listed below:

1. Very short range, with essentially no effect beyond nuclear dimensions (10^{-13} cm)
2. Stronger than the repulsive electrostatic forces within the nucleus
3. Independent of nucleon pairing, in that the attractive forces between pairs of neutrons are no different than those between pairs of protons or a neutron and a proton
4. Saturable, that is, a nucleon can attract only a few of its nearest neighbors

One theory of fission considers the fission process of a nucleus to be similar in some respects to the splitting of a liquid drop. This analogy is justifiable to some extent by the fact that a liquid drop is held together by molecular forces that tend to make the drop spherical in shape and that try to resist any deformation in the same manner as nuclear forces are assumed to hold the nucleus together.

By considering the nucleus as a liquid drop, the fission process can be described. The nucleus in the ground state is undistorted, and its attractive nuclear forces are greater than the repulsive electrostatic forces between the protons within the nucleus. When an incident particle (in this instance a neutron) is absorbed by the target nucleus, a compound nucleus is formed. The compound nucleus temporarily contains all the charge and mass involved in the reaction and exists in an excited state. The excitation energy added to the compound nucleus is equal to the binding energy contributed by the incident particle plus the kinetic energy possessed by that particle.

The excitation energy thus imparted to the compound nucleus, which may cause it to oscillate and become distorted. If the excitation energy is greater than a certain critical energy, the oscillations may cause the compound nucleus to become dumbbell-shaped. When this happens, the attractive nuclear forces (short-range) in the neck area are small due to saturation, while the repulsive electrostatic forces (long-range) are only slightly less than before. When the repulsive electrostatic forces exceed the attractive nuclear forces, nuclear fission occurs,

7.3. CRITICAL ENERGY

The measure of how far the energy level of a nucleus is above its ground state is called the **excitation energy**. For fission to occur, the excitation energy must be above a particular value for that nuclide. The **critical energy** (E_{crit}) is the minimum excitation energy required for fission to occur.

7.4. FISSIONABLE, FISSILE AND FERTILE MATERIALS

Theoretically, all nuclei heavier than iron have the potential to undergo fission, however the energy barrier that needs to be exceeded before fission can occur is impossibly high for all but the heavier elements. It is only for mass numbers greater than about 230 that the fission activation energy may be less than 10 MeV.

A. Fissionable Materials

Consider the compound nucleus Uranium 239 formed by the absorption of a neutron by $\mathbf{U^{238}}$**.** For the neutron to induce fission the sum of the binding and kinetic energy transferred to the $\mathbf{U^{239}}$ compound nucleus must exceed its fission activation energy. The activation energy of $\mathbf{U^{239}}$ is 7 MeV; the difference between the binding energy of $\mathbf{U^{238}}$ and $\mathbf{U^{239}}$ is 5.5 MeV. Thus the kinetic energy of the incoming neutron must be at least 1.5 MeV.

Materials, such as $\mathbf{U^{238}}$**,** which may undergo fission following absorption of fast neutrons of a few MeV kinetic energy are called **Fissionable Materials.**

B. Fissile Materials

A **fissile material** is composed of nuclides for which fission is possible with neutrons of any energy level. What is especially significant about these nuclides is their ability to be fissioned with zero kinetic energy

neutrons (thermal neutrons). Thermal neutrons have very low kinetic energy levels (essentially zero) because they are roughly in equilibrium with the thermal motion of surrounding materials.

Therefore, in order to be classified as fissile, a material must be capable of fissioning after absorbing a thermal neutron. Consequently, they impart essentially no kinetic energy to the reaction. Fission is possible in these materials with thermal neutrons, since the change in binding energy supplied by the neutron addition alone is high enough to exceed the critical energy. Some examples of fissile nuclides are $\mathbf{U^{235}}$ (uranium-235), $\mathbf{U^{233}}$ (uranium-233), and $\mathbf{PU^{239}}$ (plutonium-239).

Consider the compound nucleus $\mathbf{U^{236}}$ formed from the absorption of a neutron by $\mathbf{U^{235}}$**.** In this case the $\mathbf{U^{236}}$ fission activation energy is 6.5 MeV whereas the difference in binding energy between $\mathbf{U^{235}}$ and $\mathbf{U^{236}}$ is 6.8 MeV. Thus neutrons of any kinetic energy can induce fission following absorption in $\mathbf{U^{235}}$. $\mathbf{U^{235}}$ is the only naturally occurring fissile material.

C. Fertile Materials

All of the neutron absorption reactions that do not result in fission lead to the production of new nuclides through the process known as **transmutation.** These nuclides can, in turn, be transmuted again or may undergo radioactive decay to produce still different nuclides. The nuclides that are produced by this process are referred to as transmutation products. Because several of the fissile nuclides do not exist in nature, they can only be produced by nuclear reactions (transmutation).

The target nuclei for such reactions are said to be fertile. **Fertile materials** are materials that can undergo transmutation to become fissile materials. The fertile nuclides, thorium-232 and uranium-238 can be bombarded with neutrons to produce uranium-233 and plutonium-239, respectively.

If a reactor contains fertile material in addition to its fissile fuel, some new fuel will be produced as the original fuel is burned up. This is called **conversion**. Reactors that are specifically designed to produce fissionable fuel are called "breeder" reactors. In such reactors, the amount of fissionable fuel produced is greater than the amount of fuel burnup. If less fuel is produced than used, the process is called conversion, and the reactor is termed a "converter."

A **fissionable material** is composed of nuclides for which fission with neutrons is possible. All fissile nuclides fall into this category. However, also included are those nuclides that can be fissioned only with high energy neutrons. The change in binding energy that occurs as the result of neutron absorption results in a nuclear excitation energy level that is less than the required critical energy. Therefore, the additional excitation energy must be supplied by the kinetic energy of the incident neutron.

The reason for this difference between fissile and fissionable materials is the so-called odd-even effect for nuclei. It has been observed that nuclei with even numbers of neutrons and/or protons are more stable than those with odd numbers. Therefore, adding a neutron to change a nucleus with an odd number of neutrons to a nucleus with an even number of neutrons produces an appreciably higher binding energy than adding a neutron to a nucleus already possessing an even number of neutrons. Some examples of nuclides requiring high energy neutrons to cause fission are **Th^{232}** (**thorium-232)**, **U^{238}** (**uranium-238)**, and **Pu^{240}** (**plutonium-240)**.

Uranium 239, which may be formed as the result of **U^{238}** absorbing a neutron, is radioactive and decays by β emission, with a half-life of 23-1/2 minutes, to **Neptunium 239.** This neptunium 239 decays by β emission, with half life of 2.3 days, to **Plutonium 239**, an α emitter of half life of 24,000 years. It turns out that plutonium 239 is a fissile material; that is, as in the case of **U^{235}**, it readily undergoes fission on absorption of neutrons of any energy including slow neutrons of very low energies. The uranium 238 is called a **Fertile Material** because the absorption of the neutrons, which we have seen previously it most readily does in the resonance capture mode, leads to the formation of the fissile material **Pu^{239}.**

Similarly thorium 232 is also a fertile material because neutron absorption leads, via **Protactinium 233,** to the fissile material **Uranium 233.**

Thus the fertile materials **U^{238}** and **Th^{232}** yield the fissile materials **Pu^{239}** and **U^{233}**, respectively.

7.5. NATURAL URANIUM

Natural Uranium is found in ore deposits in many places around the world. It is predominantly a mixture of the two isotopes 238, 234 and 235, in the proportions mentioned at the beginning of this white paper. All three isotopes are radioactive.
Therefore of the three fissile materials mentioned above, natural uranium is a direct source for one, **U^{235},** and an indirect source for a second, **Pu^{239}** via the fertile **U^{238}.** These facts underscore the importance of natural uranium in the production of Nuclear Power.

The third fissile material, **U^{233},** is of little significance at present, although of possibly important potential because of large ore reserves of the fertile thorium.

Before returning to our discussion on fission it will be useful to summarize some of the properties of natural uranium and its isotopes:

- Natural uranium consists of
  99.3% **U^{238}** α emitter half life 4.5×10^9 years
  0.7% **U^{235}** α emitter half life 7.1×10^8 years
  0.1% **U^{234}** α emitter half life 2.5×10^5 years

- **U^{238}** is a fissionable material; it can undergo fission provided the absorbed neutron has an incident kinetic energy of at least 1.1 MeV.
- **U^{238}** is a fertile material, forming fissile **Pu^{239}** following capture of a neutron. Neutrons of intermediate energy are readily captured in the resonance capture peaks of **U^{238}.**
- **U^{235}** is a fissile material; it can undergo fission with neutrons of any energy but is much more likely to do so the less energetic, or slower, the neutron.

The isotope uranium-235 is usually the desired material for use in reactors. A vast amount of equipment and energy are expended in processes that separate the isotopes of uranium (and other elements). The details of these processes are beyond the scope of this document. These processes are called enrichment processes because they selectively increase the proportion of a particular isotope. The enrichment process typically starts with feed material that has the proportion of isotopes that occur naturally. In the case of uranium, the natural uranium ore is 0.72 a/o uranium-235. The desired outcome of the enrichment process is to produce enriched uranium.

- **Enriched uranium** is defined as uranium in which the isotope uranium-235 has a concentration greater than its natural value. The enrichment process will also result in the byproduct of depleted uranium.
- **Depleted uranium** is defined as uranium in which the isotope uranium-235 has a concentration less than its natural value. Although depleted uranium is referred to as a by-product of the enrichment process, it does have uses in the nuclear field and in commercial and defense industries.

7.6. CRITICAL ENERGIES/BINDING ENERGY OF LAST NEUTRON

Uranium-235 fissions with thermal neutrons because the binding energy released by the absorption of a neutron is greater than the critical energy for fission; therefore uranium-235 is a fissile material. The binding energy released by uranium-238 absorbing a thermal neutron is less than the critical energy, so additional energy must be possessed by the neutron for fission to be possible. Consequently, uranium-238 is a fissionable material.

7.7. BINDING ENERGY PER NUCLEON (BE/A)

As the number of particles in a nucleus increases, the total binding energy also increases. The rate of increase, however, is not uniform. This lack of uniformity results in a variation in the amount of binding energy associated with each nucleon within the nucleus. This variation in the binding energy per nucleon (BE/A) is easily seen when the average BE/A is plotted versus atomic mass number (A).

This plot illustrates that as the atomic mass number increases, the binding energy per nucleon decreases for A > 60. The BE/A curve reaches a maximum

value of 8.79 MeV at A = 56 and decreases to about 7.6 MeV for A = 238. The general shape of the BE/A curve can be explained using the general properties of nuclear forces. The nucleus is held together by very short-range attractive forces that exist between nucleons. On the other hand, the nucleus is being forced apart by long range repulsive electrostatic (coulomb) forces that exist between all the protons in the nucleus.

As the atomic number and the atomic mass number increase, the repulsive electrostatic forces within the nucleus increase due to the greater number of protons in the heavy elements. To overcome this increased repulsion, the proportion of neutrons in the nucleus must increase to maintain stability. This increase in the neutron-to-proton ratio only partially compensates for the growing proton-proton repulsive force in the heavier, naturally occurring elements. Because the repulsive forces are increasing, less energy must be supplied, on the average, to remove a nucleon from the nucleus. The BE/A has decreased. The BE/A of a nucleus is an indication of its degree of stability. Generally, the more stable nuclides have higher BE/A than the less stable ones. The increase in the BE/A as the atomic mass number decreases from 260 to 60 is the primary reason for the energy liberation in the fission process. In addition, the increase in the BE/A as the atomic mass number increases from 1 to 60 is the reason for the energy liberation in the fusion process, which is the opposite reaction of fission.

The heaviest nuclei require only a small distortion from a spherical shape (small energy addition) for the relatively large coulomb forces forcing the two halves of the nucleus apart to overcome the attractive nuclear forces holding the two halves together. Consequently, the heaviest nuclei are easily fissionable compared to lighter nuclei.

SECTION EIGHT: ENERGY RELEASE FROM FISSION

Fission of heavy nuclides converts a small amount of mass into an enormous amount of energy. The amount of energy released by fission can be determined based on either the change in mass that occurs during the reaction or by the difference in binding energy per nucleon between the fissile nuclide and the fission products.

8.1. CALCULATION OF FISSION ENERGY

Nuclear fission results in the release of enormous quantities of energy. It is necessary to be able to calculate the amount of energy that will be produced. The logical manner in which to pursue this is to first investigate a typical fission reaction. When the compound nucleus splits, it breaks into two fission fragments, rubidium-93, cesium-140, and some neutrons. Both fission products then decay by multiple - emissions as a result of the high neutron-to-proton ratio possessed by these nuclides.

In most cases, the resultant fission fragments have masses that vary widely. The most probable pair of fission fragments for the thermal fission of the fuel uranium-235 have masses of about 95 and 140.

Referring now to the binding energy per nucleon curve (Figure 20), we can estimate the amount of energy released by our "typical" fission by plotting this reaction on the curve and calculating the change in binding energy (DBE) between the reactants on the left-hand side of the fission equation and the products on the right-hand side. Plotting the reactant and product nuclides on the curve shows that the total binding energy of the system after fission is greater than the total binding energy of the system before fission. When there is an increase in the total binding energy of a system, the system has become more stable by releasing an amount of energy equal to the increase in total binding energy of the system. Therefore, in the fission process, the energy liberated is equal to the increase in the total binding energy of the system.

8.2. BINDING ENERGY PER NUCLEON

The energy released will be equivalent to the difference in binding energy (**BE**) between the reactants and the products. The energy liberation during the fission process can also be explained from the standpoint of the conservation of mass-energy. During the fission process, there is a decrease in the mass of the system. There must, therefore, be energy liberated equal to the energy equivalent of the mass lost in the process.

Again, referring to the "typical" fission reaction. E, the instantaneous energy, is the energy released immediately after the fission process. It is equal to the energy equivalent of the mass lost in the fission process. The total energy released per fission will vary from the fission to the next depending on what fission products are formed, but the average total energy released per fission of uranium-235 with a thermal neutron is **200 MeV.**

The majority of the energy liberated in the fission process is released immediately after the fission occurs and appears as the kinetic energy of the fission fragments, kinetic energy of the fission neutrons, and instantaneous gamma rays. The remaining energy is released over a period of time after the fission occurs and appears as kinetic energy of the beta, neutrino, and decay gamma rays.

8.3. ESTIMATION OF DECAY ENERGY

In addition to this instantaneous energy release during the actual fission reaction, there is additional energy released when the fission fragments decay by - emission. This additional energy is called decay energy, E. The energy released during the decay for each chain will be equivalent to the mass difference between the original fission product and the sum of the final stable nuclide and the beta particles emitted.

8.4. DISTRIBUTION OF FISSION ENERGY

The average energy distribution for the energy released per fission with a thermal neutron in uranium-235 is shown below:

A. Instantaneous Energy from Fission

Kinetic Energy of Fission Products 167 Mev

Energy of Fission Neutrons 5 MeV
Instantaneous Gamma-ray Energy 5 MeV
Capture Gamma-ray Energy 10 MeV
Total Instantaneous Energy 187 MeV

B. Delayed Energy from Fission

Beta Particles From Fission Products 7 Mev
Gamma-rays from Fission Products 6 MeV
Neutrinos 10 MeV
Total Delayed Energy 23 MeV

All of the energy released, with the exception of the neutrino energy, is ultimately transformed into heat through a number of processes. The fission fragments, with their high positive charge and kinetic energy, cause ionization directly as they rip orbital electrons from the surrounding atoms. In this ionization process, kinetic energy is transferred to the surrounding atoms of the fuel material, resulting in an increase in temperature. The beta particles and gamma rays also give up their energy through ionization, and the fission neutrons interact and lose their energy through elastic scattering.

Of the 200 MeV released per fission, about seven percent (13 MeV) is released at some time after the instant of fission. When a reactor is shut down, fissions essentially cease, but energy is still being released from the decay of fission products. The heat produced by this decay energy is referred to as "decay heat." Although decay energy represents about seven percent of reactor heat production during reactor operation, once the reactor is shut down the decay heat production drops off quickly to a small fraction of its value while operating. The decay heat produced is significant, however, and systems must be provided to keep the reactor cool even after shutdown.

SECTION NINE: INTERACTION OF RADIATION WITH MATTER

Different types of radiation interact with matter in widely different ways. A large, massive, charged alpha particle cannot penetrate a piece of paper and even has a limited range in dry air. A neutrino, at the other extreme, has a low probability of interacting with any matter, even if it passed through the diameter of the earth.

9.1. IONIZATION

Radiation can be classified into two general groups, charged and uncharged; therefore, it may be expected that interactions with matter fall into two general types. Charged particles directly ionize the media through which they pass, while uncharged particles and photons can cause ionization only indirectly or by secondary radiation.

A moving charged particle has an electrical field surrounding it, which interacts with the atomic structure of the medium through which it is passing. This interaction decelerates the particle and accelerates electrons in the atoms of the medium. The accelerated electrons may acquire enough energy to escape from the parent atom. This process, whereby radiation "strips" off orbital electrons, is called ionization. Uncharged moving particles have no electrical field, so they can

only lose energy and cause ionization by such means as collisions or scattering. A photon can lose energy by the photoelectric effect, Compton effect, or pair production.

Because ionizing radiation creates ions in pairs, the intensity of ionization or the specific ionization is defined as the number of ion-pairs formed per centimeter of travel in a given material. The amount of ionization produced by a charged particle per unit path length, which is a measure of its ionizing power, is roughly proportional to the particle's mass and the square of its charge as illustrated in the equation below. where:

$$I = \frac{mz^2}{K.E.}$$

Where:
I is the ionizing power
m is the mass of the particle
z is the number of unit charges it carries
K.E. is its kinetic energy

Since m for an alpha particle is about 7300 times as large as m for a beta article, and z is twice as great, an alpha will produce much more ionization per unit path length than a beta particle of the same energy. This phenomenon occurs because the larger alpha particle moves slower for a given energy and thus acts on a given electron for a longer time.

9.2. ALPHA RADIATION

Alpha radiation is normally produced from the radioactive decay of heavy nuclides and from certain nuclear reactions. The alpha particle consists of 2 neutrons and 2 protons, so it is essentially the same as the nucleus of a helium atom. Because it has no electrons, the alpha particle has a charge of +2. This positive charge causes the alpha particle to strip electrons from the orbits of atoms in its vicinity. As the alpha particle passes through material, it removes electrons from the orbits of atoms it passes near. Energy is required to remove electrons and the energy of the alpha particle is reduced by each reaction. Eventually the particle will expend its kinetic energy, gain 2 electrons in orbit, and become a helium atom. Because of its strong positive charge and large mass, the alpha particle deposits a large amount of energy in a short distance of travel. This rapid, large deposition of energy limits the penetration of alpha particles. The most energetic alpha particles are stopped by a few centimeters of air or a sheet of paper.

9.3. BETA-MINUS RADIATION

A beta-minus particle is an electron that has been ejected at a high velocity from an unstable nucleus. An electron has a small mass and an electrical charge of -1. Beta particles cause ionization by displacing electrons from atom orbits. The ionization occurs from collisions with orbiting electrons. Each collision removes kinetic energy from the beta particle, causing it to slow down. Eventually the beta particle will be slowed enough to allow it to be captured as an orbiting electron in

an atom. Although more penetrating than the alpha, the beta is relatively easy to stop and has a low power of penetration. Even the most energetic beta radiation can be stopped by a few millimeters of metal.

9.4. POSITRON RADIATION

Positively charged electrons are called positrons. Except for the positive charge, they are identical to beta-minus particles and interact with matter in a similar manner. Positrons are very short-lived, however, and quickly are annihilated by interaction with a negatively charged electron, producing two gammas with a combined energy equal to the rest mass of the positive and negative electrons.

9.5. NEUTRON RADIATION

Neutrons have no electrical charge. They have nearly the same mass as a proton (a hydrogen atom nucleus). A neutron has hundreds of times more mass than an electron, but 1/4 the mass of an alpha particle. The source of neutrons is primarily nuclear reactions, such as fission, but they may also be produced from the decay of radioactive nuclides. Because of its lack of charge, the neutron is difficult to stop and has a high penetrating power.

Neutrons are attenuated (reduced in energy and numbers) by three major interactions, elastic scatter, inelastic scatter, and absorption. In elastic scatter, a neutron collides with a nucleus and bounces off. This reaction transmits some of the kinetic energy of the neutron to the nucleus of the atom, resulting in the neutron being slowed, and the atom receives some kinetic energy.

As the mass of the nucleus approaches the mass of the neutron, this reaction becomes more effective in slowing the neutron. Hydrogenous material attenuates neutrons most effectively. In the inelastic scatter reaction, the same neutron/nucleus collision occurs as in elastic scatter. However, in this reaction, the nucleus receives some internal energy as well as kinetic energy. This slows the neutron, but leaves the nucleus in an excited state.

When the nucleus decays to its original energy level, it normally emits a gamma ray. In the absorption reaction, the neutron is actually absorbed into the nucleus of an atom. The neutron is captured, but the atom is left in an excited state. If the nucleus emits one or more gamma rays to reach a stable level, the process is called radiative capture. This reaction occurs at most neutron energy levels, but is more probable at lower energy levels.

9.6. GAMMA RADIATION

Gamma radiation is electromagnetic radiation. It is commonly referred to as a gamma ray and is very similar to an x-ray. The difference is that gamma rays are emitted from the nucleus of an atom, and x-rays are produced by orbiting electrons. The x-ray is produced when orbiting electrons move to a lower energy orbit or when a fast-moving electron approaching an atom is deflected or decelerated as it reacts with the atom's electrical field (called **Bremsstrahlung**). The gamma ray is produced by the decay of excited nuclei and by nuclear reactions. Because the gamma ray has no mass and no charge, it is difficult to

stop and has a very high penetrating power. A small fraction of the original gamma stream will pass through several feet of concrete or several meters of water.

There are three methods of attenuating gamma rays:

1. **Photo-Electric Effect:** The first method is referred to as the **photo-electric effect.** When a low energy gamma strikes an atom, the total energy of the gamma is expended in ejecting an electron from orbit. The result is ionization of the atom and expulsion of a high energy electron. This reaction is most predominant with low energy gammas interacting in materials with high atomic weight and rarely occurs with gammas having an energy above 1 MeV. Annihilation of the gamma results. Any gamma energy in excess of the binding energy of the electron is carried off by the electron in the form of kinetic energy.

2. **Compton Scattering:** The second method of attenuation of gammas is called **Compton scattering**. The gamma interacts with an orbital or free electron; however, in this case, the photon loses only a fraction of its energy. The actual energy loss depending on the scattering angle of the gamma. The gamma continues on at lower energy, and the energy difference is absorbed by the electron. This reaction becomes important for gamma energies of about 0.1 MeV and higher.

3. **Pair Production:** At higher energy levels, a third method of attenuation is predominant. This method is **pair-production**. When a high energy gamma passes close enough to a heavy nucleus, the gamma completely disappears, and an electron and a positron are formed. For this reaction to take place, the original gamma must have at least 1.02 MeV energy. Any energy greater than 1.02 MeV becomes kinetic energy shared between the electron and positron. The probability of pair production increases significantly for higher energy

CHAPTER TWO:INTRODUCTION TO NEUTRON PHYSICS

SECTION ONE: NEUTRON SOURCES

Neutrons from a variety of sources are always present in a reactor core. This is true even when the reactor is shut down. Some of these neutrons are produced by naturally occurring (intrinsic) neutron sources, while others may be the result of fabricated (installed) neutron sources that are incorporated into the design of the reactor. The neutrons produced by sources other than neutron-induced fission are often grouped together and classified as source neutrons. Source neutrons are important because they ensure that the neutron population remains high enough to allow a visible indication of neutron level on the most sensitive monitoring instruments while the reactor is shutdown and during the startup sequence. This verifies instrument operability and allows monitoring of neutron population changes. Source neutrons can be classified as either intrinsic or installed neutron sources.

1.1. INTRINSIC NEUTRON SOURCES

Some neutrons will be produced in the materials present in the reactor due to a variety of unavoidable reactions that occur because of the nature of these materials. **Intrinsic neutron sources** are those neutron-producing reactions that always occur in reactor materials.

A limited number of neutrons will always be present, even in a reactor core that has never been operated, due to spontaneous fission of some heavy nuclides that are present in the fuel. Uranium-238, uranium-235, and plutonium-239 undergo spontaneous fission to a limited extent. Uranium-238, for example, yields almost 60 neutrons per hour per gram.

1.2. NEUTRON PRODUCTION BY SPONTANEOUS FISSION

Another intrinsic neutron source is a reaction involving natural boron and fuel. In some reactors, natural boron is loaded into the reactor core as a neutron absorber to improve reactor control or increase core life-time. Boron-11 (80.1% of natural boron) undergoes a reaction with the alpha particle emitted by the radioactive decay of heavy nuclides in the fuel to yield a neutron.

The boron-11 must be mixed with, or in very close proximity to, the fuel for this reaction because of the short path length of the alpha particle. For a reactor core with this configuration, this **(α,n)** reaction is an important source of neutrons for reactor startup. In a reactor that has been operated, another source of neutrons becomes significant. Neutrons may be produced by the interaction of a gamma ray and a deuterium nucleus. This reaction is commonly referred to as a photo-neutron reaction because it is initiated by electromagnetic radiation and results in the production of a neutron.

There is an abundant supply of high energy gammas in a reactor that has been operated because many of the fission products are gamma emitters. All water-cooled reactors have some deuterium present in the coolant in the reactor core because a small fraction of natural hydrogen is the isotope deuterium.

The atom percentage of deuterium in the water ranges from close to the naturally occurring value (0.015%) for light water reactors to above 90% deuterium for heavy water reactors. Therefore, the required conditions for production of photo-neutrons exist.

The supply of gamma rays decreases with time after shutdown as the gamma emitters decay; therefore, the photo-neutron production rate also decreases. In a few particular reactors, additional Deuterium Oxide, **DO_2** (heavy water) may be added to the reactor to increase the production of photo-neutrons following a long shutdown period.

1.3. INSTALLED NEUTRON SOURCES

Because intrinsic neutron sources can be relatively weak or dependent upon the recent power history of the reactor, many reactors have artificial sources of neutrons installed. These neutron sources ensure that shutdown neutron levels

are high enough to be detected by the nuclear instruments at all times. This provides a true picture of reactor conditions and any change in these conditions.

An **installed neutron source** is an assembly placed in or near the reactor for the sole purpose of producing source neutrons. One strong source of neutrons is the artificial nuclide californium-252, which emits neutrons at the rate of about **2 x 10^{12}** neutrons per second per gram as the result of spontaneous fission. Important drawbacks for some applications may be its high cost and its short half-life (2.65 years).

Many installed neutron sources use the (α**,n**,) reaction with beryllium. These sources are composed of a mixture of metallic beryllium (100% beryllium-9) with a small quantity of an alpha particle emitter, such as a compound of radium, polonium, or plutonium. The reaction that occurs produces a neutron flux. The beryllium is intimately (homogeneously) mixed with the alpha emitter and is usually enclosed in a stainless steel capsule.

Another type of installed neutron source that is widely used is a photo-neutron source that employs the **gamma/neutron**(γ**, n**) reaction with beryllium. Beryllium is used for photo-neutron sources because its stable isotope beryllium-9 has a weakly attached last neutron with a binding energy of only 1.66 MeV. Thus, a gamma ray with greater energy than 1.66 MeV can cause neutrons to be ejected by the (γ**, n**) reaction.

SECTION TWO: NUCLEAR CROSS SECTIONS AND NEUTRON FLUX

To determine the frequency of neutron interactions, it is necessary to describe the availability of neutrons to cause interaction and the probability of a neutron interacting with material. The availability of neutrons and the probability of interaction are quantified by the neutron flux and nuclear cross section.

2.1. INTRODUCTION

Fission neutrons are born with an average energy of about 2 MeV. These fast neutrons interact with the reactor core materials in various absorption and scattering reactions. Collisions that result in scattering are useful in slowing neutrons to thermal energies. Thermal neutrons may be absorbed by fissile nuclei to produce more fissions or be absorbed in fertile material for conversion to fissionable fuel. Absorption of neutrons in structural components, coolant, and other non-fuel material results in the removal of neutrons without fulfilling any useful purpose. To safely and efficiently operate a nuclear reactor it is necessary to predict the probability that a particular absorption or scattering reaction will

occur. Once these probabilities are known, if the availability of neutrons can be determined, then the rate at which these nuclear reactions take place can be predicted.

2.2. ATOMIC DENSITY

One important property of a material is the atom density. **The atom density** is the number of atoms of a given type per unit volume of the material. To calculate the atom density of a substance use the following equation:

$$\mathbf{N = \frac{\rho N_A}{M}}$$

where:

N = atom density (atoms/cm^3)
ρ = density (gram/cm^3)
$\mathbf{N_A}$ = Avogadro's number (6.022 x 10^{23} atoms/mole)
M = gram atomic weight

2.3. CROSS-SECTION FOR NUCLEAR REACTIONS

The probability of a neutron interacting with a nucleus for a particular reaction is dependent upon not only the kind of nucleus involved, but also the energy of the neutron. Accordingly, the absorption of a thermal neutron in most materials is much more probable than the absorption of a fast neutron. Also, the probability of interaction will vary depending upon the type of reaction involved.

The probability of a particular reaction occurring between a neutron and a nucleus is called the **microscopic cross section (σ)** of the nucleus for the particular reaction. This cross section will vary with the energy of the neutron. The microscopic cross section may also be regarded as the effective area the nucleus presents to the neutron for the particular reaction. The larger the effective area, the greater the probability for reaction. Because the microscopic cross section is an area, it is expressed in units of area, or square centimeters.

A square centimeter is tremendously large in comparison to the effective area of a nucleus, and it has been suggested that a physicist once referred to the measure of a square centimeter as being "as big as a barn" when applied to nuclear processes. The name has persisted and microscopic cross sections are expressed in terms of **barns**. The relationship between **barns** and $\mathbf{cm^2}$ is shown below.

$$\mathbf{1\ barn = 10^{-24}\ cm^2}$$

Whether a neutron will interact with a certain volume of material depends not only on the microscopic cross section of the individual nuclei but also on the number of nuclei within that volume. Therefore, it is necessary to define another kind of cross section known as the **macroscopic cross section (Σ)**.

The **macroscopic cross section** is the probability of a given reaction occurring per unit travel of the neutron.

A. Microscopic Cross Section

For neutrons of a specific energy one or more of the foregoing nuclear reactions can occur. It is necessary to have a method of calculating how many neutrons are undergoing which reactions.

Clearly the rate **R** at which neutrons undergo any reactions is dependent on the number of target nuclei and the number of bombarding neutrons. We are going to say that the rate **R** is proportional to $\mathbf{N}\phi$ where **N** is the number of nuclei per cubic meters for the target material and ϕ is the intensity of the neutron radiation measured in neutrons per square meters. ϕ is called the neutron flux and is the product of the number of neutrons per cubic meters and the average neutron speed, denoted ν, which may be in any direction. We can say that: $\phi = \mathbf{n}\nu$ neutrons per square meters. If we let this constant of proportionality by σ**,** so that:

$$\mathbf{R} = \sigma\mathbf{N}\phi$$

This equation is usually written in the order

$$\mathbf{R} = \mathbf{N}\sigma\phi$$

And the value of the proportionality constant σ for each type of nuclear reaction is dependent on the target material and the neutron speed. σ may be viewed as a measure of the likelihood, or the probability in the non-mathematical sense, of a given reaction occurring. Alternatively, because σ has the units of meters squared, it is also thought of as the 'effective area' presented to the incident neutron by the target nucleus. Hence σ is called the microscopic cross-section for the neutron reaction but it must be emphasized that it is not the physical area of the nucleus (σ can have values orders of magnitude larger or smaller than the physical area of the atom. The units for the microscopic cross-section if the **Barn.** The **Barn** is defined by 1 barn = 10^{-28} meters$^{2.}$

Since neutrons can have more than one type of reaction the total microscopic cross-section, the likelihood of the neutron interacting with the nucleus, can be expressed as the sum of a number of partial cross-sections.

B. Macroscopic Cross-Sections

When dealing with matter in bulk, the reaction rate $\mathbf{N}\sigma\phi$ is often written as: $\Sigma\phi$ where $\Sigma = \mathbf{N}\sigma$ and is known as the macroscopic cross-section. Σ (Capital Sigma) may be regarded as the total collision area presented by the target nuclei per unit volume of material. As it has the units of (area/volume), or $\mathbf{m^{-1}}$ **,** Σ may be interpreted as the probability per meter of tract length that a neutron will interact with the material.

Now the average distance that a neutron travels without interacting is known as the **Mean Free Path.** The mean free path is usually denoted by the Greek letter **Lambda:** λ. The mean free path is equal to the reciprocal of the macroscopic cross-section.

The difference between the microscopic and macroscopic cross sections is extremely important and is restated for clarity. The **microscopic cross section** (σ) represents the effective target area that a single nucleus presents to a bombarding particle. The units are given in **barns** or **cm^2**. The **macroscopic cross section** (Σ) represents the effective target area that is presented by all of the nuclei contained in **1 cm^3** of the material.

2.4. NUCLEAR REACTION

A neutron interacts with an atom of the material it enters in two basic ways:

- It will interact through a **scattering interaction**
- or through an **absorption reaction**.

The probability of a neutron being absorbed by a particular atom is the microscopic cross section for absorption, σ_a. The probability of a neutron scattering off of a particular nucleus is the microscopic cross section for scattering, σ_s. The sum of the microscopic cross section for absorption and the microscopic cross section for scattering is the total microscopic cross section:

$$\sigma_T = \sigma_a + \sigma_s$$

where:

σ_T = total microscopic cross section
σ_a = microscopic cross section for absorption
σ_s = microscopic cross section for scattering

Both the absorption and the scattering microscopic cross sections can be further divided. For instance, the scattering cross section is the sum of the elastic scattering cross section (σ_{se}) and the inelastic scattering cross section (σ_{si}).

$$\sigma_s = \sigma_{si} + \sigma_{se}$$

The microscopic absorption cross section (σ_a) includes all reactions except scattering. However, for most purposes it is sufficient to merely separate it into two categories, fission (σ_{af}) and capture (σ_{ac}).

$$\sigma_a = \sigma_{af} + \sigma_{ac}$$

The variation of absorption cross sections with neutron energy is often complicated. For many elements the absorption cross sections are small, ranging from a fraction of a barn to a few barns for slow (or thermal) neutrons.

For a considerable number of nuclides of moderately high (or high) mass numbers, an examination of the variation of the absorption cross section with the energy of the incident neutron reveals the existence of three regions on a curve of absorption cross section versus neutron energy.

A. First, the cross section decreases steadily with increasing neutron energy in a low energy region, which includes the thermal range (**E < 1 eV**). In this region the absorption cross section, which is often high, is inversely proportional to the **velocity** (**v**). This region is frequently referred to as the "1/v region," because the **absorption cross section** is proportional to **1/v,** which is the reciprocal of neutron velocity.

B. Following the 1/v region, there occurs the "**resonance region**" in which the cross sections rise sharply to high values called "resonance peaks" for neutrons of certain energies, and then fall again. These energies are called resonance energies and are a result of the affinity of the nucleus for neutrons whose energies closely match its discrete, quantum energy levels. That is, when the binding energy of a neutron plus the kinetic energy of the neutron are exactly equal to the amount required to raise a compound nucleus from its ground state to a quantum level, resonance absorption occurs. The typical heavy nucleus will have many closely spaced resonances starting in the low energy (eV) range. This is because heavy nuclei are complex and have more possible configurations and corresponding energy states. Light nuclei, being less complex, have fewer possible energy states and fewer resonances that are sparsely distributed at higher energy levels.

C. For higher neutron energies, the absorption cross section steadily decreases as the energy of the neutron increases. This is called the "fast neutron region." In this region the absorption cross sections are usually less than 10 barns. With the exception of hydrogen, for which the value is fairly large, the elastic scattering cross sections are generally small, for example, 5 barns to 10 barns. This is close to the magnitude of the actual geometric cross sectional area expected for atomic nuclei. In potential scattering, the cross section is essentially constant and independent of neutron energy. Resonance elastic scattering and inelastic scattering exhibit resonance peaks similar to those associated with absorption cross sections. The resonances occur at lower energies for heavy nuclei than for light nuclei. In general, the variations in scattering cross sections are very small when compared to the variations that occur in absorption cross sections.

2.5. MEAN FREE PATH

If a neutron has a certain probability of undergoing a particular interaction in one centimeter of travel, then the inverse of this value describes how far the neutron will travel (in the average case) before undergoing an interaction. This average distance traveled by a neutron before interaction is known as the **mean free path** for that interaction and is represented by the symbol λ. The relationship between the mean free path (λ) and the macroscopic cross section (Σ) is shown below.

$$\Sigma = \frac{1}{\lambda}$$

2.6. CALCULATION OF MACROSCOPIC CROSS SECTION

Most materials are composed of several elements, and because most elements are composed of several isotopes, most materials involve many cross sections, one for each isotope involved. Therefore, to include all the isotopes within a given material, it is necessary to determine the macroscopic cross section for each isotope and then sum all the individual macroscopic cross sections.

2.7. EFFECTS OF TEMPERATURE ON CROSS SECTION

The microscopic absorption cross section varies significantly as neutron energy varies. The microscopic cross sections provided on most charts and tables are measured for a standard neutron velocity of **2200 meters/second,** which corresponds to an ambient temperature of **68°F**. Therefore, if our material is at a higher temperature, the absorption cross section will be lower than the value for **68°F.**

2.8. NEUTRON FLUX

Macroscopic cross sections for neutron reactions with materials determine the probability of one neutron undergoing a specific reaction per centimeter of travel through that material. If one wants to determine how many reactions will actually occur, it is necessary to know how many neutrons are traveling through the material and how many centimeters they travel each second. It is convenient to consider the number of neutrons existing in one cubic centimeter at any one instant and the total distance they travel each second while in that cubic centimeter. The number of neutrons existing in a **cm^3** of material at any instant is called **neutron density** and is represented by the symbol **n** with units of **neutrons/cm^3**. The total distance these neutrons can travel each second will be determined by their velocity.

A good way of defining **neutron flux (ϕ)** is to consider it to be the total path length covered by all neutrons in one cubic centimeter during one second. Mathematically, this is the equation below:

$$\phi = n v$$

where:

ϕ = neutron flux (neutrons/cm^2-sec)
n = neutron density (neutrons/cm^3)
v = neutron velocity (cm/sec)

The term neutron flux in some applications (for example, cross section measurement) is used as parallel beams of neutrons traveling in a single direction. The **intensity (I)** of a neutron beam is the product of the neutron density times the average neutron velocity. The directional beam intensity is equal to the number of neutrons per unit area and time (neutrons/cm^2-sec) falling on a surface perpendicular to the direction of the beam.

One can think of the neutron flux in a reactor as being comprised of many neutron beams traveling in various directions. Then, the neutron flux becomes the scalar sum of these directional flux intensities. Since the atoms in a reactor do not interact preferentially with neutrons from any particular direction, all of these directional beams contribute to the total rate of reaction. In reality, at a given point within a reactor, neutrons will be traveling in all directions.

2.9. SELF-SHIELDING

In some locations within the reactor, the flux level may be significantly lower than in other areas due to a phenomenon referred to as **neutron shadowing** or **self-shielding**. For example, the interior of a fuel pin or pellet will "see" a lower

average flux level than the outer surfaces since an appreciable fraction of the neutrons will have been absorbed and therefore cannot reach the interior of the fuel pin.

SECTION THREE: REACTION RATES

It is possible to determine the rate at which a nuclear reaction will take place based on the neutron flux, cross section for the interaction, and atom density of the target. This relationship illustrates how a change in one of these items affects the reaction rate.

3.1. REACTION RATES

If the total path length of all the neutrons in a cubic centimeter in a second is known, (**neutron flux** (ϕ)), and if the probability of having an interaction per centimeter path length is also known (**macroscopic cross section** (Σ)), multiply them together to get the number of interactions taking place in that cubic centimeter in one second. This value is known as the **reaction rate** and is denoted by the symbol **R.** The reaction rate can be calculated by the equation shown below:

$$\mathbf{R} = \phi \mathbf{x} \Sigma$$

where:

R = reaction rate (reactions/sec)
ϕ = neutron flux (neutrons/cm^2-sec)
Σ = macroscopic cross section (cm^{-1})

Substituting the fact that:

$$\Sigma = \mathbf{N} \mathbf{x} \sigma$$

where:

Σ = macroscopic cross section (cm^{-1})
N = atom density (atoms/cm^3)
σ = microscopic cross section (cm^2)

we get: $\mathbf{R} = \phi \mathbf{x} \mathbf{N} \mathbf{x} \sigma$

The **reaction rate** calculated will depend on which macroscopic cross section is used in the calculation. Normally, the **reaction rate** of greatest interest is the fission reaction rate.

3.2. REACTOR POWER CALCULATION

Multiplying the reaction rate per unit volume by the total volume of the core results in the total number of reactions occurring in the core per unit time. If the amount of energy involved in each reaction were known, it would be possible to determine the rate of energy release (power) due to a certain reaction.

In a reactor where the average energy per fission is 200 MeV, it is possible to determine the number of fissions per second that are necessary to produce one watt of power using the following conversion factors.

1 fission = 200 MeV

1 MeV = 1.602 x 10^{-6} ergs
1 erg = 1 x 10^{-7} watt-sec

This is equivalent to stating that 3.12 x 10^{10} fissions release 1 watt-second of energy. The power released in a reactor can be calculated based upon this equation. Multiplying the reaction rate by the volume of the reactor results in the total fission rate for the entire reactor. Dividing by the number of fissions per watt-sec results in the power released by fission in the reactor in units of watts.

This relationship is shown mathematically in the equation below:

$$P = \frac{\phi_{th} \Sigma_f V}{3.12 \times 10^{10}}$$

where:

P = power (watts)
ϕ_{th} = thermal neutron flux (neutrons/cm -sec)
Σ_f= macroscopic cross section for fission (**cm^{-1}**)
V = volume of core (cm3)

3.3. RELATIONSHIP BETWEEN NEUTRON FLUX AND REACTOR POWER

In an operating reactor the volume of the reactor is constant. Over a relatively short period of time (days or weeks), the number density of the fuel atoms is also relatively constant. Since the atom density and microscopic cross section are constant, the macroscopic cross section must also be constant. Examining the power equation above it is apparent that if the reactor volume and macroscopic cross section are constant, then the reactor power and the neutron flux are directly proportional.

This is true for day-to-day operation. The neutron flux for a given power level will increase very slowly over a period of months due to the **burnup** of the fuel and resulting decrease in atom density and macroscopic cross section.

SECTION FOUR: NEUTRON MODERATION

In thermal reactors, the neutrons that cause fission are at a much lower energy than the energy level at which they were born from fission. In this type of reactor, specific materials must be included in the reactor design to reduce the energy level of the neutrons in an efficient manner.

4.1. NEUTRON SLOWING DOWN/THERMALIZATION

Fission neutrons are produced at an average energy level of 2 MeV and immediately begin to slow down as the result of numerous scattering reactions with a variety of target nuclei. After a number of collisions with nuclei, the speed of a neutron is reduced to such an extent that it has approximately the same average kinetic energy as the atoms (or molecules) of the medium in which the neutron is undergoing elastic scattering.

This energy, which is only a small fraction of an electron volt at ordinary temperatures (0.025 eV at 20(C), is frequently referred to as the thermal energy,

since it depends upon the temperature. Neutrons whose energies have been reduced to values in this region (< 1 eV) are designated **thermal neutrons**. The process of reducing the energy of a neutron to the thermal region by elastic scattering is referred to as **thermalization,** slowing down, or moderation. The material used for the purpose of thermalizing neutrons is called a **moderator.**

A good moderator reduces the speed of neutrons in a small number of collisions, but does not absorb them to any great extent. Slowing the neutrons in as few collisions as possible is desirable in order to reduce the amount of neutron leakage from the core and also to reduce the number of resonance absorptions in non-fuel materials. The ideal moderating material (**moderato**r) should have the following nuclear properties.

- large scattering cross section
- small absorption cross section
- large energy loss per collision

A convenient measure of energy loss per collision is the logarithmic energy decrement. The **average logarithmic energy decrement** is the average decrease per collision in the logarithm of the neutron energy. This quantity is represented by the symbol ξ.

$$\xi = \ln E_i - \ln E_f$$

where:

ξ = average logarithmic energy decrement
E_i = average initial neutron energy
E_f = average final neutron energy

The symbol ξ is commonly called the average logarithmic energy decrement because of the fact that a neutron loses, on the average, a fixed fraction of its energy per scattering collision. Since the fraction of energy retained by a neutron in a single elastic collision is a constant for a given material, is also a constant. Because it is a constant for each type of material and does not depend upon the initial neutron energy, is a convenient quantity for assessing the moderating ability of a material. The values for the lighter nuclei are tabulated in a variety of sources.

Since ξ represents the average logarithmic energy loss per collision, the total number of collisions necessary for a neutron to lose a given amount of energy may be determined by dividing ξ into the difference of the natural logarithms of the energy range in question. The number of collisions (**N)** to travel from any energy, E_{high}, to any lower energy, E_{low}, can be calculated as shown below.

$$N = \frac{\ln E_i - \ln E_f}{\xi}$$

or

$$N = \frac{\ln (E_i / E_f)}{\xi}$$

4.2. MACROSCOPIC SLOWING DOWN POWER

Although the logarithmic energy decrement is a convenient measure of the ability of a material to slow neutrons, it does not measure all necessary properties of a moderator. A better measure of the capabilities of a material is the macroscopic slowing down power. The **macroscopic slowing down power** (**MSDP**) is the product of the **logarithmic energy decrement** and the **macroscopic cross section for scattering** in the material.

4.3. MODERATING RATIO

Macroscopic slowing down power indicates how rapidly a neutron will slow down in the material in question, but it still does not fully explain the effectiveness of the material as a moderator. An element such as boron has a high logarithmic energy decrement and a good slowing down power, but it is a poor moderator because of its high probability of absorbing neutrons. The most complete measure of the effectiveness of a moderator is the moderating ratio. The **moderating ratio** is the ratio of the macroscopic slowing down power to the macroscopic cross section for absorption. The higher the moderating ratio, the more effectively the material performs as a moderator.

$$\mathbf{MR} = \frac{\xi \Sigma_s}{\Sigma_a}$$

SECTION FIVE: PROMPT AND DELAYED NEUTRONS

Not all neutrons are released at the same time following fission. Most neutrons are released virtually instantaneously and are called prompt neutrons. A very small fraction of neutrons are released after the decay of fission products and are called delayed neutrons. Although delayed neutrons are a very small fraction of the total number of neutrons, they play an extremely important role in the control of the reactor.

5.1. NEUTRON CLASSIFICATION

The great majority (over 99%) of the neutrons produced in fission are released within about $\mathbf{10^{-13}}$ seconds of the actual fission event. These are called **prompt neutrons**. A small portion of fission neutrons are **delayed neutrons**, which are produced for some time after the fission process has taken place. The delayed neutrons are emitted immediately following the first beta decay of a fission fragment known as a delayed neutron precursor. An example of a delayed neutron precursor is bromine-87, shown below.

For most applications, it is convenient to combine the known precursors into groups with appropriately averaged properties. These groups vary somewhat depending on the fissile material in use. The fraction of all neutrons that are produced by each of these precursors is called the delayed neutron fraction for that precursor. The total fraction of all neutrons born as delayed neutrons is called the **delayed neutron fraction** (β). The fraction of delayed neutrons produced varies depending on the predominant fissile nuclide in use. The delayed neutron fractions (β) for the fissile nuclides of most interest are as follows: **uranium^{233}** (0.0026), uranium^{235}, (0.0065), uranium-238^{238} (0.0148), and plutonium^{239}(0.0021).

5.2. NEUTRON GENERATION TIME

The neutron generation time is the time required for neutrons from one generation to cause the fissions that produce the next generation of neutrons. The generation time for prompt neutrons (ℓ^* - pronounced "ell-star") is the total time from birth to rebirth. Three time intervals are involved:

1. The time it takes a fast neutron to slow down to thermal energy,
2. The time the now thermal neutron exists prior to absorption in fuel,
3. The time required for a fissionable nucleus to emit a fast neutron after neutron absorption.

Fast neutrons slow to thermal energies or leak out of the reactor in 10^{-4} seconds to 10^{-6} seconds, depending on the moderator. In water moderated reactors, thermal neutrons tend to exist for about 10^{-4} seconds before they are absorbed.

Fission and fast neutron production following neutron absorption in a fissionable nucleus occurs in about 10^{-13} seconds. Thus, fast reactors have an ℓ^* of about 10^{-6} seconds, while thermal reactors have an ℓ^* of about 10^{-6} seconds + 10^{-4} seconds, which is about 10^{-4} seconds to 10^{-5} seconds.

On the other hand, the average generation time for the six delayed neutron groups is the total time from the birth of the fast neutron to the emission of the delayed neutron. Again, three time intervals are involved:

1. The time it takes a fast neutron to slow down to thermal energy
2. The time the thermal neutron exists prior to absorption
3. The average time from neutron absorption to neutron emission by the six precursor groups. The average time for decay of precursors from $\mathbf{U^{235}}$ is 12.5 seconds. The other terms in the delayed neutron generation time are insignificant when compared to this value, and the average delayed neutron generation time becomes approximately 12.5 seconds.

   A neutron generation time in the range of 10^{-4} seconds to 10^{-5} seconds or faster could result in very rapid power excursions, and control would not be possible without the dependence upon delayed neutrons to slow down the rate of the reaction. The average generation time, and hence the rate that power can rise, is determined largely by the delayed neutron generation time.

SECTION SIX: NEUTRON FLUX SPECTRUM

The number of neutrons that exist at a given energy level varies. A plot of either the fraction of neutrons or the neutron flux at a given energy versus the energy level is called a **neutron energy spectrum**. The neutron energy spectrum varies widely for different types of reactors.

6.1. PROMPT NEUTRON ENERGIES

The neutrons produced by fission are high energy neutrons, and almost all fission neutrons have energies between 0.1 MeV and 10 MeV. The neutron energy distribution, or spectrum, may best be described by plotting the fraction of neutrons per MeV as a function of neutron energy. The most probable neutron energy is about 0.7 MeV, and the average energy of fission neutrons is about 2 MeV.

6.2. REACTOR NEUTRON SPECTRA

The spectrum of neutron energies produced by fission varies significantly from the energy spectrum, or flux, existing in a reactor at a given time. The neutron flux spectra of a thermal reactor and a fast breeder reactor are essentially the same. The differences in the curve shapes may be attributed to the neutron moderation or slowing down effects. No attempt is made to thermalize or slow down neutrons in the fast breeder reactor (liquid metal cooled); therefore, an insignificant number of neutrons exist in the thermal range.

For the thermal reactor (water moderated), the spectrum of neutrons in the fast region (> 0.1 MeV) has a shape similar to that for the spectrum of neutrons emitted by the fission process. In the thermal reactor, the flux in the intermediate energy region (1 eV to 0.1 MeV) has approximately a **1/E** dependence. That is, if the **energy (E**) is halved, the flux doubles. This **1/E** dependence is caused by the slowing down process, where elastic collisions remove a constant fraction of the neutron energy per collision (on the average), independent of energy; thus, the neutron loses larger amounts of energy per collision at higher energies than at lower energies. The fact that the neutrons lose a constant fraction of energy per collision causes the neutrons to tend to "pile up" at lower energies, that is, a greater number of neutrons exist at the lower energies as a result of this behavior.

In the thermal region the neutrons achieve a thermal equilibrium with the atoms of the moderator material. In any given collision they may gain or lose energy, and over successive collisions will gain as much energy as they lose. These thermal neutrons, even at a specific temperature, do not all have the same energy or velocity; there is a distribution of energies, usually referred to as the Maxwell distribution. The energies of most thermal neutrons lie close to the most probable energy, but there is a spread of neutrons above and below this value.

6.3. MOST PROBABLE NEUTRON VELOCITIES

Neutrons in thermal equilibrium with the surrounding material are referred to as **Thermal Neutrons**. It may be shown, through the use of the kinetic theory of perfect gases, that the average kinetic energy of a gas particle in thermal equilibrium, at a temperature **T** is given by:

$$\tfrac{1}{2}\, mV^2 = 3/2\, kT$$

which gives us ½ kT of energy per degree of freedom. In this equation **m** is the mass of the neutron and **k** is the Boltzmann's constant. (1.38×10^{-23} J/K).

Nuclear data are often given for the 'standard' temperature of 293K (20 ^{o}C). Calculations for thermal neutrons in surroundings at a temperature of 293K gives a mean velocity of 2200 meters per second and corresponds to a kinetic energy of **0.025 eV.** Thus for Uranium 235 and for thermal neutrons at the 'standard' 2200 meters per second we can calculate a macroscopic cross-section of: **690 Barns.**

Sometime the term 'slow neutron' is used as if synonymous with 'thermal neutron' – this is not the case. Also, thermal neutrons are sometimes regarded as being necessarily at 20^{o}C – again this is not the case. For example, neutrons at, say, 250^{o}C would be thermal neutrons with corresponding speed and energy value of (2940 meters/second) and (0.145 eV).

The **most probable velocity** (v_p) of a thermal neutron is determined by the temperature of the medium and can be determined by the following Equation

$$v_p = \sqrt{\frac{3kT}{m}}$$

where:
v_p = most probable velocity of neutron (cm/sec)
k = Boltzman's constant (1.38 x 10^{-16} erg/ K)
T = absolute temperature in degrees Kelvin (K)
m = mass of neutron (1.66 x 10^{-24} grams)

From these calculations it is evident that the most probable velocity of a thermal neutron increases as temperature increases. The most probable velocity at 20 C is of particular importance since reference data, such as nuclear cross sections, are tabulated for a neutron velocity of 2200 meters per second.

END OF CHAPTER 2-NEUTRON PHYSICS

CHAPTER THREE: INTRODUCTION TO REACTOR PHYSICS

SECTION ONE: NEUTRON LIFE CYCLE

Some number of the fast neutrons produced by fission in one generation will eventually cause fission in the next generation. The series of steps that fission neutrons go through as they slow to thermal energies and are absorbed in the reactor is referred to as the neutron life cycle. The neutron life cycle is markedly different between fast reactors and thermal reactors. This section of this white paper presents the neutron life cycle for thermal reactors.

1.1. THE CHAIN REACTION

Although the binary fission of a nucleus by an impinging neutron would no doubt be of academic interest, it would in itself be of little significance in the context of large scale power production. It is the accompanying release of neutrons which makes power production possible in that the neutrons may induce further nuclei to fission and thus lead to a chain reaction.

1.2. INFINITE MULTIPLICATION FACTOR, k_∞

Not all of the neutrons produced by fission will have the opportunity to cause new fissions because some neutrons will be absorbed by non-fissionable material. Some will be absorbed parasitically in fissionable material and will not cause fission, and others will leak out of the reactor.

For the maintenance of a self-sustaining chain reaction, however, it is not necessary that every neutron produced in fission initiate another fission. The minimum condition is for each nucleus undergoing fission to produce, on the average, at least one neutron that causes fission of another nucleus. This condition is conveniently expressed in terms of a multiplication factor.

The number of neutrons absorbed or leaking out of the reactor will determine the value of this multiplication factor, and will also determine whether a new generation of neutrons is larger, smaller, or the same size as the preceding generation. Any reactor of a finite size will have neutrons leak out of it.

Generally, the larger the reactor, the lower the fraction of neutron leakage. For simplicity, we will first consider a reactor that is infinitely large, and therefore has no neutron leakage. A measure of the increase or decrease in neutron flux in an infinite reactor is the infinite multiplication factor, $\mathbf{k_\infty}$**.** The **infinite multiplication factor** is the ratio of the neutrons produced by fission in one generation to the number of neutrons lost through absorption in the preceding generation. This can be expressed mathematically as shown below:

$$\mathbf{k_\infty} = \frac{\textbf{Neutron Production from Fission in one Generation}}{\textbf{neutron absorption in the preceding generation}}$$

For simplicity, at the beginning of our discussion, it is advantageous to consider the presupposition that there are no losses of neutrons out of the system; that is, the system – the nuclear reactor – is assumed to be of infinite size. The symbol $\mathbf{k_\infty}$ is now used for the multiplication constant, the subscript infinity (∞) being a

reminder of the imposed condition of an infinite sized nuclear reactor. From our definition of $\mathbf{k}_\infty$, the multiplication constant it is clear that:

- If $\mathbf{k}_\infty > 1$ the chain reaction is divergent and the nuclear reactor is said to be supercritical. As the power produced is related to the number of fission events taking place, this is the condition for power raising. The larger the value of $\mathbf{k}_\infty$ the quicker is the rate of power rise.

- If $\mathbf{k}_\infty = 1$ the chain reaction is self-sustained and the reactor is said to be critical. A nuclear power station producing power at a steady rate for perhaps months or years – a base load power station – is therefore a critical reactor. The word is unfortunate, but understandable when it is recalled that the term 'critical' dates back to 12 December 1942. On that day Enrico Fermi and his colleagues first achieved a self-sustained chain reaction, and created the first nuclear pile.

- If $\mathbf{k}_\infty < 1$ the chain reaction is converging and the reactor power is decreasing. As before the rate of power decrease is determined by how much less than unity is the value of $\mathbf{k}_\infty$.

1.3 . CHAIN REACTION IN NATURAL URANIUM

Let us first consider an infinite mass of natural uranium, representing the simplest conceptual design for a nuclear reactor. Assume $\mathbf{n_1}$ fast neutrons (i.e., with energy of fission neutrons $\cong$ 2 MeV) are introduced into the reactor. To determine the value of $\mathbf{k}_\infty$ for the reactor it is necessary to calculate the number of neutrons in the generation following the absorption in the uranium of all the original generation of $\mathbf{n_1}$ fast neutrons.

In considering the possible fate of the $\mathbf{n_1}$ neutrons it is important to recall that natural uranium consists overwhelmingly of $\mathbf{U^{338}}$**.** There are three possibilities.

1.3.1. FISSION

Fast neutrons can induce fission in $\mathbf{U^{238}}$ provided they have energy in excess of the threshold 1.1 MeV, the fission cross-section being much the same value as for $\mathbf{U^{235}}$**.** Thus the fission event taking place in the reactor can be regarded as being predominantly of $\mathbf{U^{238}}$ nuclei and giving rise to the next generation of fast neutrons, $\mathbf{n_2}$ say.

1.3.2. CAPTURE

Some of the $\mathbf{n_1}$ neutrons will undergo direct capture in the natural uranium, i.e., non-fission absorptions in the $\mathbf{U^{238}}$ and $\mathbf{U^{235}}$.

1.3.3. SCATTER

Some of the $\mathbf{n_1}$ neutrons will be scattered but only those undergoing inelastic scattering are significant. In this case the neutrons will emerge from the inelastic scattering event with energy less than the threshold value of 1.1 MeV necessary for $\mathbf{U^{238}}$ fission and will subsequently be captured in the resonance capture

peaks of the U^{238}. In this context therefore inelastic scattering may be regarded as leading indirectly to capture.

Elastic scattering events may be ignored because the neutrons are effectively unchanged by the collision and are still identified as being of the original n_1 neutron generation.

1.4. k_∞ for Natural Uranium and Fast Neutrons

The initial n_1 neutrons will therefore either cause fission or be captured (directly or indirectly via inelastic scattering collision). The fraction of causing fission may be calculated using the cross-section values for the nuclear reactions and hence the next generation of neutrons, n_2, may be determined analytically . This solution for n_2, yields $n_2 = 0.26 n_1$

Therefore $k_\infty = 0.26 < 1$. Hence natural Uranium, no matter what the geometry cannot sustain a chain reaction.

1.5. To Achieve $k_\infty > 1$

The value of is determined by the balance between neutron **Production** in the reactor fuel and neutron loss by **Absorption** in the reactor materials and by **Leakage** out of the reactor for a finite system. In this paper we have discussed the impossibility of having a chain reaction in natural uranium. There are two ways in which we can increase the value of k_∞ .

1. Change the properties of the fuel
2. Change the properties of the impinging neutrons.

1.5.1. FUEL ENRICHMENT

k_∞ is less than unity, in our previous discussion, largely because the inelastic collisions of the fast neutrons in the natural uranium fuel leads to the subsequent loss of the neutrons in the U^{238} resonance capture peaks. The fast fission cross-section of U^{238} and U^{235} are much the same and too low to counter the neutron losses.

It follows that if the proportion of U^{238} in the fuel is reduced, thus increasing the proportion of U^{235} , the effect is to replace a high neutron capture material U^{238} by another material U^{235} which has much the same fast fission cross-section but less neutron capture characteristics. This process of increasing the proportion of U^{235} is known as **Fuel Enrichment.**

As U^{235} has no energy threshold for fission, the net result of fuel enrichment is that the neutron from fission increases and neutron absorption decreases. Calculation shows that U^{235} enrichment of at least 20% is necessary to give $k_\infty > 1$ for fast neutrons. This enrichment process forms the basis of 'fast reactors', the title fast reactor being derived from the fact that the neutrons inducing fission are largely energetic fast neutrons.

1.5.2. NEUTRON MODERATION

The alternative to large fuel enrichment and fast neutrons is to retain, slightly enriched natural uranium as the fuel but provide the nuclear chain reaction with

slow neutrons, and thus take advantage of the greatly increased likelihood of U^{235} fission occurring.
Instead of having n_1 fast neutrons, we consider the chain reaction to consist of n_1 **slow** neutrons introduced into our infinite mass of nuclear material. We then repeat our calculations to obtain the next generation of n_2 neutrons. In this instance there is no possibility with slow neutrons of inelastic collisions occurring. The neutrons are simply either captured or cause fission.

1.6. FOUR FACTOR FORMULA

A group of fast neutrons produced by fission can enter into several reactions. Some of these reactions reduce the size of the neutron group while other reactions allow the group to increase in size or produce a second generation. There are four factors that are completely independent of the size and shape of the reactor that give the inherent multiplication ability of the fuel and moderator materials without regard to leakage. This **four factor formula** accurately represents the infinite multiplication factor as shown in the equation below.

$$k_\infty = \varepsilon\, p\, f\, \eta$$

where:

ε = fast fission factor
p = resonance escape probability
f = thermal utilization factor
η = reproduction factor

Each of these four factors, which are explained in the following subsections, represents a process that adds to or subtracts from the initial neutron group produced in a generation by fission.

1.6.1. Fast Fission Factor, (ε)

The first process that the neutrons of one generation may undergo is **fast fission. Fast fission** is fission caused by neutrons that are in the fast energy range. Fast fission results in the net increase in the fast neutron population of the reactor core. The cross section for fast fission in **uranium-235, U^{235}** or **uranium-238, U^{238}** is small; therefore, only a small number of fast neutrons cause fission.

The fast neutron population in one generation is therefore increased by a factor called the fast fission factor. **The fast fission factor** (ε) is defined as the ratio of the net number of fast neutrons produced by all fissions to the number of fast neutrons produced by thermal fissions.

The mathematical expression of this ratio is shown below.

$$\varepsilon = \frac{\text{number of fast neutrons produced by all fission}}{\text{number of fast neutrons produced by thermal fissions}}$$

In order for a neutron to be absorbed by a fuel nucleus as a fast neutron, it must pass close enough to a fuel nucleus while it is a fast neutron. The value of ε will be affected by the arrangement and concentrations of the fuel and the

moderator. The value of is essentially 1.00 for a homogenous reactor where the fuel atoms are surrounded by moderator atoms.

However, in a heterogeneous reactor, all the fuel atoms are packed closely together in elements such as pins, rods, or pellets. Neutrons emitted from the fission of one fuel atom have a very good chance of passing near another fuel atom before slowing down significantly. The arrangement of the core elements results in a value of about 1.03 for in most heterogeneous reactors. The value of ε is not significantly affected by variables such as temperature, pressure, enrichment, or **neutron poison** concentrations. **Poison**s are non-fuel materials that easily absorb neutrons and will be discussed in more detail later.

1.6.2. Resonance Escape Probability, (p)

After increasing in number as a result of some fast fissions, the neutrons continue to diffuse through the reactor. As the neutrons move they collide with nuclei of fuel and non-fuel material and moderator in the reactor losing part of their energy in each collision and slowing down.

While they are slowing down through the resonance region of uranium-238, ($\mathbf{U^{238}}$) which extends from about 6 eV to 200 eV, there is a chance that some neutrons will be captured. The probability that a neutron will not be absorbed by a resonance peak is called the **resonance escape probability**. The **resonance escape probability** (**p**) is defined as the ratio of the number of neutrons that reach thermal energies to the number of fast neutrons that start to slow down. This ratio is shown below.

$$\mathbf{p = \frac{\text{Number of nuetonrs that reach thermal energy}}{\text{number of fast neutrons that start to slow down}}}$$

The value of the resonance escape probability is determined largely by the fuel-moderator arrangement and the amount of enrichment of uranium-235 ($\mathbf{U^{235}}$). To undergo resonance absorption, a neutron must pass close enough to a uranium-238 ($\mathbf{U^{238}}$) nucleus to be absorbed while slowing down.

In a homogeneous reactor the neutron does its slowing down in the region of the fuel nuclei, and this condition is easily met. This means that a neutron has a high probability of being absorbed by uranium-238 ($\mathbf{U^{238}}$) while slowing down; therefore, its escape probability is lower. In a heterogeneous reactor, however, the neutron slows down in the moderator where there are no atoms of uranium-238 ($\mathbf{U^{238}}$) present. Therefore, it has a low probability of undergoing resonance absorption, and its escape probability is higher.

The value of the resonance escape probability is not significantly affected by pressure or poison concentration. In water moderated, low uranium-235 ($\mathbf{U^{235}}$) enrichment reactors, raising the temperature of the fuel will raise the resonance absorption in uranium-238 ($\mathbf{U^{238}}$) due to the Doppler effect (an apparent broadening of the normally narrow resonance peaks due to thermal motion of nuclei). The increase in resonance absorption lowers the resonance escape probability, and the fuel temperature coefficient for resonance escape is negative

(explained in detail later). The temperature coefficient of resonance escape probability for the moderator temperature is also negative. As water temperature increases, water density decreases. The decrease in water density allows more resonance energy neutrons to enter the fuel and be absorbed. The value of the resonance escape probability is always slightly less than one (normally 0.95 to 0.99). The product of the fast fission factor and the resonance escape probability (p) is the ratio of the number of fast neutrons that survive slowing down (thermalization) compared to the number of fast neutrons originally starting the generation.

1.6.3. Thermal Utilization Factor, (f)

Once thermalized, the neutrons continue to diffuse throughout the reactor and are subject to absorption by other materials in the reactor as well as the fuel. The thermal utilization factor describes how effectively thermal neutrons are absorbed by the fuel, or how well they are utilized within the reactor. The **thermal utilization factor** (**f**) is defined as the ratio of the number of thermal neutrons absorbed in the fuel to the number of thermal neutrons absorbed in any reactor material. This ratio is shown below.

$$f = \frac{\text{number of thermal neutrons absorbed in the fuel}}{\text{number of thermal neutrons absorbed in all reactor materials}}$$

The thermal utilization factor will always be less than one because some of the thermal neutrons absorbed within the reactor will be absorbed by atoms of non-fuel materials.

An equation can be developed for the thermal utilization factor in terms of reaction rates as follows:

$$f = \frac{\text{number of thermal neutrons absorbed in the fuel}}{\text{number of thermal neutrons absorbed in all reactor materials}}$$

$$f = \frac{\Sigma_a^u \phi^u v^u}{\Sigma_a^u \phi^u v^u + \Sigma_a^m \phi^m v^m + \Sigma_a^p \phi^p v^p}$$

Σ = macroscopic cross section (cm^{-1})
ϕ = neutron flux (neutrons/cm^2-sec)
n = neutron density (neutrons/cm^3)
v = neutron velocity (cm/sec)

The superscripts u, m, and p refer to uranium, moderator, and poison, respectively.

In a heterogeneous reactor, the flux will be different in the fuel region than in the moderator region due to the high absorption rate by the fuel. Also, the volumes of fuel, moderator, and poisons will be different. Although not shown in the above equation, other non-fuel materials, such as core construction materials, may absorb neutrons in a heterogeneous reactor. These other materials are often lumped together with the superscript designation OS, for "other stuff." To be

completely accurate, the above equation for the thermal utilization factor should include all neutron-absorbing reactor materials when dealing with heterogeneous reactors.

In a homogeneous reactor the neutron flux seen by the fuel, moderator, and poisons will be the same. Also, since they are spread throughout the reactor, they all occupy the same volume. This allows the previous equation to be rewritten as shown below:

$$f = \frac{\Sigma_a^u}{\Sigma_a^u + \Sigma_a^m + \Sigma_a^p}$$

The equation above gives an approximation for a heterogeneous reactor if the fuel and moderator are composed of small elements distributed uniformly throughout the reactor.

Since absorption cross sections vary with temperature, it would appear that the thermal utilization factor would vary with a temperature change. But, substitution of the temperature correction formulas in the above equation will reveal that all terms change by the same amount, and the ratio remains the same.

In heterogeneous water-moderated reactors, there is another important factor. When the temperature rises, the water moderator expands, and a significant amount of it will be forced out of the reactor core. This means that the number of moderator atoms per **cm^3**, will be reduced, making it less likely for a neutron to be absorbed by a moderator atom. This reduction in $\mathbf{N_m}$ results in an increase in thermal utilization as moderator temperature increases because a neutron now has a better chance of hitting a fuel atom. Because of this effect, the temperature coefficient for the thermal utilization factor is positive.

The amount of enrichment of uranium-235 (**U^{235}**) and the poison concentration will affect the thermal utilization factor in a similar manner as can be seen from the equation above.

1.6.4. Reproduction Factor, (η)

Most of the neutrons absorbed in the fuel cause fission, but some do not . **The reproduction factor** (η) is defined as the ratio of the number of fast neutrons produces by thermal fission to the number of thermal neutrons absorbed in the fuel. The reproduction factor is shown below:

$$\eta = \frac{\text{number of fast neutrons produced by thermal fission}}{\text{number of thermal neutrons absorbed in the fuel}}$$

The reproduction factor can also be stated as a ratio of rates as shown below:

$$\eta = \frac{\text{rate of production of fast neutrons by thermal fission}}{\text{rate of absorption of thermal neutrons by the fuel}}$$

The rate of production of fast neutrons by thermal fission can be determined by the product of the fission reaction rate ($\Sigma_f^u \phi^u$) and the average number of

neutrons produced per fission (ν). The average number of neutrons released in thermal fission of uranium-235 ($\mathbf{U^{235}}$) is 2.42.

The rate of absorption of thermal neutrons by the fuel is $\Sigma_a^u \phi^u$. Substituting these terms into the equation above results in the following equation:

$$\eta = \frac{\Sigma_f^u \phi^u \nu}{\Sigma_a^u \phi^u}$$

η = reproduction factor
Σ = macroscopic cross section (cm^{-1})
ϕ = neutron flux **(**neutrons/cm^2-sec)
ν = average number of neutrons per fission

As temperature varies, each absorption and fission microscopic cross section varies according to the **1/v** relationship and therefore, η changes only as uranium-235 ($\mathbf{U^{235}}$) enrichment changes. η increases with enrichment because there is less uranium-238 ($\mathbf{U^{238)}}$ in the reactor making it more likely that a neutron absorbed in the fuel will be absorbed by uranium-235 ($\mathbf{U^{235}}$) and cause fission.

1.7. EFFECTIVE MULTIPLICATION FACTOR

The infinite multiplication factor can fully represent only a reactor that is infinitely large, because it assumes that no neutrons leak out of the reactor. To completely describe the neutron life cycle in a real, finite reactor, it is necessary to account for neutrons that leak out.

The multiplication factor that takes leakage into account is the **effective multiplication factor (k_{eff})**, which is defined as the ratio of the neutrons produced by fission in one generation to the number of neutrons lost through absorption and leakage in the preceding generation. The effective multiplication factor may be expressed mathematically as shown below:

$$k_{eff} = \frac{\text{neutron production from fission in one reneration}}{\text{neutron absorption in the preceding generation} + \text{neutron leakage in the preceding generation}}$$

So, the value of k_{eff} for a self-sustaining chain reaction of fissions, where the neutron population is neither increasing nor decreasing, is one. The condition where the neutron chain reaction is self-sustaining and the neutron population is neither increasing nor decreasing is referred to as the **critical condition** and can be expressed by the simple equation:

$$k_{eff} = 1 .$$

If the neutron production is greater than the absorption and leakage, the reactor is called supercritical. In a **supercritical reactor**, k_{eff} is greater than one, and the neutron flux increases in each generation.

If, on the other hand, the neutron production is less than the absorption and leakage, the reactor is called **subcritical.** In a **subcritical reactor**, k_{eff} is less than one, and the flux decreases each generation.

When the multiplication factor of a reactor is not equal to exactly one, the neutron flux will change and cause a change in the power level. Therefore, it is essential to know more about how this factor depends upon the contents and construction of the reactor. The balance between production of neutrons and their absorption in the core and leakage out of the core determines the value of the multiplication factor.

If the leakage is small enough to be neglected, the multiplication factor depends upon only the balance between production and absorption, and is called the **infinite multiplication factor** $\mathbf{k_\infty}$ since an infinitely large core can have no leakage.
When the leakage is included, the factor is called the effective multiplication factor $\mathbf{k_{eff}}$. The effective multiplication factor $\mathbf{k_{eff}}$ for a finite reactor may be expressed mathematically in terms of the infinite multiplication factor and two additional factors which account for neutron leakage as shown below:

$$\mathbf{k_{eff} = k_\infty \times \mathscr{L}_f \times \mathscr{L}_t}$$

where:

$\mathbf{k_{eff}}$ = effective multiplication factor
$\mathbf{k_\infty}$ = infinite multiplication factor
$\mathscr{L}_f$ = fast non-leakage probability
$\mathscr{L}_t$ = thermal non-leakage probability

1.7.1. FAST NON-LEAKAGE PROBABILITY ($\mathscr{L}_f$)

In a realistic reactor of finite size, some of the fast neutrons leak out of the boundaries of the reactor core before they begin the slowing down process. The **fast non-leakage probability** ($\mathscr{L}_f$) is defined as the ratio of the number of fast neutrons that do not leak from the reactor core to the number of fast neutrons produced by all fissions. This ratio is stated as follows:

$$\mathscr{L}_f = \frac{\text{number of fast neutrons that do not leak from reactor}}{\text{number of fast neutrons produced by all fissions}}$$

1.7.2. THERMAL NON-LEAKAGE PROBABILITY ($\mathscr{L}_t$)

Neutrons can also leak out of a finite reactor core after they reach thermal energies. The **thermal non-leakage probability** ($\mathscr{L}_t$) is defined as the ratio of the number of thermal neutrons that do not leak from the reactor core to the number of neutrons that reach thermal energies. The thermal non-leakage probability is represented by the following equation:

$$\mathscr{L}_t = \frac{\text{number of thermal neutrons that do not leak from reactor}}{\text{number of neutrons that reach thermal energies}}$$

The fast non-leakage probability ($\mathscr{L}_f$) and the thermal non-leakage probability ($\mathscr{L}_t$) may be combined into one term that gives the fraction of all neutrons that do not leak out of the reactor core. This term is called the **total non-leakage probability** and is given the symbol $\mathscr{L}_T$.

where $\mathscr{L}_T = \mathscr{L}_t \times \mathscr{L}_f$, $\mathscr{L}_f$ and $\mathscr{L}_t$ t are both effected by a change in coolant temperature in a heterogeneous water-cooled, water-moderated reactor.

As coolant temperature rises, the coolant expands. The density of the moderator is lower; therefore, neutrons must travel farther while slowing down. This effect increases the probability of leakage and thus decreases the non-leakage probability. Consequently, the temperature coefficient (defined later) for the non-leakage probabilities is negative, because as temperature rises, $\mathscr{L}_f$ and $\mathscr{L}_t$ decrease.

1.8. SIX FACTOR FORMULA

With the inclusion of these last two factors it is possible to determine the fraction of neutrons that remain after every possible process in a nuclear reactor. The effective multiplication factor ($\mathbf{k_{ef}}$) can then be determined by the product of six terms.

$$\mathbf{k_{eff}} = \varepsilon \mathscr{L}_f \mathbf{p} \mathscr{L}_t \mathbf{f} \eta$$

where:

$\mathbf{k_{eff}}$ = effective multiplication factor
ε = fast fission factor
$\mathscr{L}_f$ = fast non-leakage probability
p = resonance escape probability
$\mathscr{L}_t$ = thermal non-leakage probability
f = thermal utilization factor
η = reproduction factor

The equation above is called the **six factor formula**. Using this six factor formula, it is possible to trace the entire neutron life cycle.

1.9. NEUTRON LIFE CYCLE OF A FAST REACTOR

The neutron life cycle in a fast reactor is markedly different than that for a thermal reactor. In a fast reactor, care is taken during the reactor design to minimize thermalization of neutrons. Virtually all fissions taking place in a fast reactor are caused by fast neutrons. Due to this, many factors that are taken into account by the thermal reactor neutron life cycle are irrelevant to the fast reactor neutron life cycle. The resonance escape probability is not significant because very few neutrons exist at energies where resonance absorption is significant.

The thermal non-leakage probability does not exist because the reactor is designed to avoid the thermalization of neutrons. A separate term to deal with fast fission is not necessary because all fission is fast fission and is handled by the reproduction factor.

The thermal utilization factor is modified to describe the utilization of fast neutrons instead of thermal neutrons. The reproduction factor is similarly modified to account for fast fission instead of thermal fission.

SECTION TWO: REACTIVITY

Reactivity is a measure of the departure of a reactor from criticality. The reactivity is related to the value of $\mathbf{k_{eff}}$. Reactivity is a useful concept to predict how the neutron population of a reactor will change over time.

2.1. APPLICATION OF THE EFFECTIVE MULTIPLACATION FACTOR

When k_{eff} remains constant from generation to generation, it is possible to determine the number of neutrons beginning any particular generation by knowing only the value of k_{eff} and the number of neutrons starting the first generation. If N_0 neutrons start the first generation, then $N_0 \times k_{eff}$ neutrons start the second generation. The following equation can be used to calculate the number of neutrons after the completion of "**n**" generations:

$$N_n = N_0 \times (k_{eff})^n$$

Where:

N_n = Number of neutrons after the completion of "**n**" generations
N_0 = Number of neutrons in the first generation
k_{eff} = effective multiplication factor
n = Number of generations

2.2. REACTIVITY

If there are N_0 neutrons in the preceding generation, then there are $N_0 \times k_{eff}$ neutrons in the present generation. The numerical change in neutron population is $(N_0 \times k_{eff} - N_0)$. The gain or loss in neutron population $(N_0 \times k_{eff} - N_0)$, expressed as a fraction of the present generation $(N_0 \times k_{eff})$, is shown below:

$$\rho = \frac{N_0 \times k_{eff} - N_0}{N_0 \times k_{eff}}$$

Where:

ρ = reactivity: The fractional change in neutron population pear generation
N_0 = Number of neutrons in the first generation
k_{eff} = effective multiplication factor

This relationship represents the fractional change in neutron population per generation and is referred to as **reactivity (ρ).** Cancelling out the term N_0 from the numerator and denominator, the reactivity is determined as shown in the equation below:

$$\rho = \frac{k_{eff} - 1}{k_{eff}}$$

Where:

ρ = reactivity:
k_{eff} = effective multiplication factor

From the equation above it may be seen that ρ may be positive, zero, or negative, depending upon the value of k_{eff}. The larger the absolute value of reactivity in the reactor core, the further the reactor is from criticality. It may be convenient to think of reactivity as a measure of a reactor's departure from criticality.

2.3. UNITS OF REACTIVITY

Reactivity is a dimensionless number. It is simply a ratio of two dimensionless quantities which is often a small decimal value. In order to make this value easier to express, artificial units are defined. By definition, the value for **reactivity** that results directly from the calculation of the equation above is in units of Δ**k/k**.

Alternative units for reactivity are **%Δk/k** and **pcm** (percent millirho). The conversions between these units of reactivity are shown below:

$$\%\Delta k = \frac{0.01 \times \Delta k}{k}$$

and:

$$pcm = \frac{0.00001 \times \Delta k}{k}$$

Another unit of reactivity that is used at some reactors is equivalent to 10^{-4} **Δk/k.** This unit of reactivity does not have a unique name. Special units for reactivity that do have unique names are dollars and cents. These units and their applications will be described later in this white paper.

2.4. REACTIVITY COEFFICIENTS AND REACTIVITY DEFECTS

The amount of reactivity (ρ) in a reactor core determines what the neutron population, and consequently the reactor power, are doing at any given time. The reactivity can be effected by many factors (for example, fuel depletion, temperature, pressure, or poisons). In this white paper we will discuss the factors affecting reactivity and how they are used to control or predict reactor behavior.

To quantify the effect that a variation in parameter (that is, increase in temperature, control rod insertion, increase in neutron poison) will have on the reactivity of the core, **reactivity coefficients** are used. **Reactivity coefficients** are the amount that the reactivity will change for a given change in the parameter. For instance, an increase in moderator temperature will cause a decrease in the reactivity of the core. The amount of reactivity change per degree change in the moderator temperature is the moderator temperature coefficient. Typical units for the moderator temperature coefficient are **pcm/°F**.

Reactivity coefficients are generally symbolized by α_x, where **x** represents some variable reactor parameter that affects reactivity. The definition of a reactivity coefficient in equation format is shown below:

$$\alpha_x = \frac{\Delta\rho}{\Delta x}$$

If the parameter **x** increases and positive reactivity is added, then α_x is positive. If the parameter **x** increases and negative reactivity is added, then α_x is negative.

Reactivity defects $\Delta\rho$ are the total reactivity change caused by a variation in a parameter. Reactivity defects can be determined by multiplying the change in the parameter by the average value of the reactivity coefficient for that parameter. The equation below shows the general method for relating reactivity coefficients to reactivity defects:

$$\Delta\rho = \alpha_x \Delta x$$

SECTION THREE: REACTIVITY COEFFICIENTS

Changes in the physical properties of the materials in the reactor will result in changes in the reactivity. Reactivity coefficients are useful in quantifying the reactivity change that will occur due to the change in a physical property such as the temperature of the moderator or fuel.

3.1. MODERATOR EFFECTS

As discussed in other introductory white papers, a moderator possesses specific desirable characteristics:

- Large neutron scattering cross section
- Low neutron absorption cross section
- large neutron energy loss per collision

With the exception of the **Liquid Metal Fast Breeder Reactor** (**LMFBR**), the remaining major reactor types that are currently employed use moderating materials to reduce fission neutron energies to the thermal range. Light moderators (composed of light nuclei) are found to be more effective than heavy moderators because the light moderator removes more energy per collision than a heavy moderator. Therefore, the neutrons reach thermal energy more rapidly and they are less likely to be lost through resonance absorption.

As discussed in other white papers, the ability of a given material to slow down neutrons is referred to as the **macroscopic slowing down power** (**MSDP**) and is defined as the product of the **logarithmic energy decrement per collision** (ξ) times the **macroscopic scattering cross section** for neutrons as follows:

$$\mathbf{MSDP} = \xi \times \Sigma_s$$

Where:

MSDP = macroscopic slowing down power
ξ = logarithmic energy decrement per collision
Σ_s = macroscopic scattering cross section (cm^{-1})

Macroscopic slowing down power indicates how rapidly slowing down occurs in the material in question, but it does not completely define the effectiveness of the material as a moderator. An element such as boron has a high logarithmic energy decrement and a good slowing down power, but is a poor moderator. It is a poor moderator because of its high probability of absorbing neutrons, and may be accounted for by dividing the macroscopic slowing down power by the macroscopic absorption cross section. This relationship is called the **moderating ratio** (**MR**):

$$\mathbf{MR} = \frac{\xi \times \Sigma_s}{\Sigma_a}$$

Where:

MR = moderator ratio
ξ = logarithmic energy decrement per collision
Σ_s = macroscopic scattering cross section (cm^{-1})
Σ_a = macroscopic absorption cross section (cm^{-1})

The moderating ratio is merely the ratio of slowing down power to the **macroscopic absorption cross section**. The higher the **moderating ratio**, the more effectively the material performs as a moderator.

Another ratio, the **moderator-to-fuel ratio** ($\mathbf{N^m/N^u}$), is very important in the discussion of moderators. As the reactor designer increases the amount of moderator in the core (that is, $\mathbf{N^m/N^u}$ increases), neutron leakage decreases.

Neutron absorption in the moderator (Σ_a^m) increases and causes a decrease in the thermal utilization factor. Having insufficient moderator in the core (that is, $\mathbf{N^m/N^u}$ decreases) causes an increase in slowing down time and results in a greater loss of neutrons by resonance absorption. This also causes an increase in neutron leakage. This effect is beneficial in a thermal reactor in the event of a **LOCA** (Loss of Coolant Accident).

Because the moderator-to-fuel ratio affects the thermal utilization factor and the resonance escape probability, it also affects $\mathbf{k_{eff}}$. The remaining factors in the six factor formula are also affected by the moderator-to-fuel ratio, but to a lesser extent than **f** and **p**.

For a Nuclear Power Plant with a large core fueled with low-enriched fuel, there is an optimum point above which increasing the moderator-to-fuel ratio decreases $\mathbf{k_{eff}}$ due to the dominance of the decreasing thermal utilization factor. Below this point, a decrease in the moderator-to-fuel ratio decreases $\mathbf{k_{eff}}$ due to the dominance of the increased resonance absorption in the fuel. If the ratio is above this point, the core is said to be over moderated, and if the ratio is below this point, the core is said to be under moderated.

In practice, water-moderated reactors are designed with a moderator-to-fuel ratio so that the reactor is operated in an under moderated condition. The reason that some reactors are designed to be under moderated is if the reactor were over moderated, an increase in temperature would decrease the $\mathbf{N^m/N^u}$ due to the expansion of the water as its density became lower. This decrease in $\mathbf{N^m/N^u}$ would be a positive reactivity addition, increasing $\mathbf{k_{eff}}$ and further raising power and temperature in a dangerous cycle. If the reactor is under moderated, the same increase in temperature results in the addition of negative reactivity, and the reactor becomes more self-regulating.

3.2. MODERATOR TEMPERATURE COEFFICIENT

The change in reactivity per degree change in temperature is called the **temperature coefficient of reactivity**. Because different materials in the reactor have different reactivity changes with temperature and the various materials are at different temperatures during reactor operation, several different temperature coefficients are used.

Usually, the two dominant temperature coefficients are:

- the moderator temperature coefficient
- the fuel temperature coefficient.

The change in reactivity per degree change in moderator temperature is called the **moderator temperature coefficient**. The magnitude and sign (+ or -) of the moderator temperature coefficient is primarily a function of the moderator-to-fuel ratio. If a reactor is under moderated, it will have a negative moderator temperature coefficient. If a reactor is over moderated, it will have a positive moderator temperature coefficient. A negative moderator temperature coefficient is desirable because of its self-regulating effect. For example, an increase in reactivity causes the reactor to produce more power. This raises the temperature of the core and adds negative reactivity, which slows down, or turns, the power rise.

3.3. FUEL TEMPERATURE COEFFICIENT

Another temperature coefficient of reactivity, the fuel temperature coefficient, has a greater effect than the moderator temperature coefficient for some reactors. The **fuel temperature coefficient** is the change in reactivity per degree change in fuel temperature. This coefficient is also called the "prompt" temperature coefficient because an increase in reactor power causes an immediate change in fuel temperature.

A negative fuel temperature coefficient is generally considered to be even more important than a negative moderator temperature coefficient because fuel temperature immediately increases following an increase in reactor power. The time for heat to be transferred to the moderator is measured in seconds. In the event of a large positive reactivity insertion, the moderator temperature cannot turn the power rise for several seconds, whereas the fuel temperature coefficient starts adding negative reactivity immediately.

Another name applied to the fuel temperature coefficient of reactivity is the **fuel Doppler reactivity coefficient**. This name is applied because in typical low enrichment, light water-moderated, thermal reactors the fuel temperature coefficient of reactivity is negative and is the result of the **doppler effect**, also called **doppler broadening**. The phenomenon of the Doppler effect is caused by an apparent broadening of the resonances due to thermal motion of nuclei. Stationary nuclei absorb only neutrons of energy $\mathbf{E_0}$. If the nucleus is moving away from the neutron, the velocity (and energy) of the neutron must be greater than $\mathbf{E_0}$ to undergo resonance absorption. Likewise, if the nucleus is moving toward the neutron, the neutron needs less energy than $\mathbf{E_0}$ to be absorbed. Raising the temperature causes the nuclei to vibrate more rapidly within their lattice structures, effectively broadening the energy range of neutrons that may be resonantly absorbed in the fuel. Two nuclides present in large amounts in the fuel of some reactors with large resonant peaks that dominate the doppler effect are **Uranium-238** ($\mathbf{U^{238}}$) and **Plutonium-240** ($\mathbf{Pu^{240}}$).

3.4. PRESSURE COEFFICIENT

The reactivity in a reactor core can be affected by the system pressure. The **pressure coefficient** of reactivity is defined as the change in reactivity per unit change in pressure. The pressure coefficient of reactivity for the reactor is the result of the effect of pressure on the density of the moderator. For this reason, it is sometimes referred to as the moderator density reactivity coefficient.

As pressure increases, density correspondingly increases, which increases the moderator-to-fuel ratio in the core. In the typical under moderated core the increase in the moderator-to-fuel ratio will result in a positive reactivity addition. In reactors that use water as a moderator, the absolute value of the pressure reactivity coefficient is seldom a major factor because it is very small compared to the moderator temperature coefficient of reactivity.

3.5. VOID COEFFICIENT

In systems with boiling conditions, such as **boiling water reactors** (**BWR**), the pressure coefficient becomes an important factor due to the larger density changes that occur when the vapor phase of water undergoes a pressure change. Of prime importance during operation of a **BWR**, and a factor in some other water-moderated reactors, is the **void coefficient**. The **void coefficient** is caused by the formation of steam voids in the moderator.

The **void coefficient** of reactivity is defined as the change in reactivity per percent change in void volume. As the reactor power is raised to the point where the steam voids start to form, voids displace moderator from the coolant channels within the core. This displacement reduces the moderator-to-fuel ratio, and in an under moderated core, results in a negative reactivity addition, thereby limiting reactor power rise. The void coefficient is significant in water-moderated reactors that operate at or near saturated conditions.

SECTION FOUR: NEUTRON POISONS

In some reactors, neutron-absorbing materials called poisons are intentionally designed into the reactor for specific purposes. Some of these poisons are depleted as they absorb neutrons during reactor operation, and others remain relatively constant.

4.1. FIXED BURNABLE POISONS

During operation of a reactor the amount of fuel contained in the core constantly decreases. If the reactor is to operate for a long period of time, fuel in excess of that needed for exact criticality must be added when the reactor is first loaded with fuel. The positive reactivity due to the excess fuel must be balanced with negative reactivity from neutron-absorbing material.

Moveable control rods containing neutron-absorbing material are one method used to offset the excess fuel. Control rods will be discussed in detail in another white paper. Using control rods alone to balance the excess reactivity may be undesirable or impractical for several reasons. One reason for a particular core

design may be that there is physically insufficient room for the control rods and their large mechanisms.

To control large amounts of excess fuel without adding additional control rods, **burnable poisons** are loaded into the core. **Burnable poisons** are materials that have a high neutron absorption cross section that are converted into materials of relatively low absorption cross section as the result of neutron absorption. Due to the burnup of the poison material, the negative reactivity of the burnable poison decreases over the useful life of the fuel. Ideally, these poisons should decrease their negative reactivity at the same rate the fuel's excess positive reactivity is depleted.

Fixed burnable poisons are generally used in the form of compounds of **boron** or **gadolinium** that are shaped into separate lattice pins or plates, or introduced as additives to the fuel. Since they can usually be distributed more uniformly than control rods, these poisons are less disruptive to the core power distribution.

4.2. SOLUBLE POISONS

Soluble poisons, also called **chemical shim,** produce a spatially uniform neutron absorption when dissolved in the water coolant. The most common soluble poison in **commercial pressurized water reactors** (**PWR**) is boric acid, which is often referred to as "soluble boron," or simply "solbor." The boric acid in the coolant decreases the thermal utilization factor, causing a decrease in reactivity. By varying the concentration of boric acid in the coolant (a process referred to as boration and dilution), the reactivity of the core can be easily varied.

If the boron concentration is increased, the coolant/moderator absorbs more neutrons, adding negative reactivity. If the boron concentration is reduced (dilution), positive reactivity is added. The changing of boron concentration in a **PWR** is a slow process and is used primarily to compensate for fuel burnout or poison buildup. The variation in boron concentration allows control rod use to be minimized, which results in a flatter flux profile over the core than can be produced by rod insertion. The flatter flux profile is due to the fact that there are no regions of depressed flux like those that would be produced in the vicinity of inserted control rods.

Some reactors include emergency shutdown systems that inject solutions containing neutron poisons into the system that circulates reactor coolant. Various solutions, including sodium polyborate and gadolinium nitrate, are used.

Fixed burnable poisons possess some advantages over chemical shim. Fixed burnable poisons may be discretely loaded in specific locations in order to shape or control flux profiles in the core. Also, fixed burnable poisons do not make the moderator temperature reactivity coefficient less negative as chemical shim does. With chemical shim, as temperature rises and the moderator expands, some moderator is pushed out of the active core area. Boron is also moved out, and this has a positive effect on reactivity. This property of chemical shim limits

the allowable boron concentration because any greater concentration makes the moderator temperature coefficient of reactivity positive.

4.3. NON-BURNABLE POISONS

A **non-burnable poison** is one that maintains a constant negative reactivity worth over the life of the core. While no neutron poison is strictly non-burnable, certain materials can be treated as non-burnable poisons under certain conditions. One example is **hafnium**. The removal (by absorption of neutrons) of one isotope of hafnium leads to the production of another neutron absorber, and continues through a chain of five absorbers. This absorption chain results in a long-lived burnable poison which approximates non-burnable characteristics.

Absorbers with low neutron absorption cross sections can also be treated as non-burnable under most conditions. It is possible to make the reactivity of a poison material that is usually a burnable poison more uniform over core life through the use of self-shielding. In self-shielding, the poison material is thick enough that only the outer layer of the poison is exposed to the neutron flux. The absorptions that take place in the outer layers reduce the number of neutrons that penetrate to the inner material. As the outer layers of poison absorb neutrons and are converted to non-poison materials, the inner layers begin absorbing more neutrons, and the negative reactivity of the poison is fairly uniform. The normal use of fixed non-burnable poisons is in power shaping, or to prevent excessive flux and power peaking near moderator regions of the reactor.

SECTION FIVE: XENON

Xenon-135 (**Xe^{135}**) has a tremendous impact on the operation of a nuclear reactor. It is important to understand the mechanisms that produce and remove xenon from the reactor to predict how the reactor will respond following changes in power level.

5.1. FISSION PRODUCT POISONS

Fission fragments generated at the time of fission decay to produce a variety of fission products. Fission products are of concern in reactors primarily because they become parasitic absorbers of neutrons and result in long term sources of heat. Although several fission products have significant neutron absorption cross sections, **Xenon-135** (**Xe^{135}**) and **Samarium-149** (**Sm^{149}**) have the most substantial impact on reactor design and operation. Because these two fission product poisons remove neutrons from the reactor, they will have an impact on the thermal utilization factor and thus $\mathbf{k_{eff}}$ and reactivity.

5.2. PRODUCTION AND REMOVAL OF XENON-135 (**Xe^{135}**)

Xenon-135 (**Xe^{135}**) has a 2.6 x 10^6 barns neutron absorption cross section. It is produced directly by some fissions, but is more commonly a product of the **tellurium-135** (**Te^{135}**) decay chain shown below:

$$_{52}Te^{135} \xrightarrow[19.0\ \text{seconds}]{\beta^-} {}_{53}I^{135} \xrightarrow[6.57\ \text{Hrs.}]{\beta^-} {}_{54}Xe^{135} \xrightarrow[9.10\ \text{Hrs.}]{\beta^-} {}_{55}Ce^{135} \xrightarrow[2.3 \times 10^6\ \text{years}]{\beta^-} {}_{56}Ba^{135}$$

The fission yield (γ) for **Xenon-135** (Xe^{135}) is about 0.3%, while γ for **Tellurium-135** (Te^{135}) is about 6%. The half-life for tellurium-135 (Te^{135}) is so short compared to the other half-lives that it can be assumed that iodine-135 (I^{135}) is produced directly from fission. Iodine-135(I^{135}) is not a strong neutron absorber, but decays to form the neutron poison xenon-135 (Xe^{135}). Ninety-five percent of all the xenon-135 (Xe^{135}) produced comes from the decay of iodine-135(I^{135}). Therefore, the half-life of iodine-135(I^{135}) plays an important role in the amount of xenon-135 (Xe^{135}). present. The rate of change of iodine concentration is equal to the rate of production minus the rate of removal. This can be expressed in the equation below.

rate of iodine concentration = yield from fission - decay rate - burnup rate

or

$$\frac{dN_I}{dt} = \gamma_I \Sigma_f^{fuel} \varphi - \lambda_I N_I - \sigma_a^I N_I \varphi$$

where:

N_I = I^{135} concentration
γ_I = fission yield of I^{135}
Σ_f^{fuel} = macroscopic fission cross section fuel
φ = thermal neutron flux
λ_I = decay constant for I^{135}
σ_a^I = microscopic absorption cross section I^{135}

Since the σ_a^I (decay constant for Iodine) is very small, the burn up rate term may be ignored, and the expression for the rate of change of iodine concentration is modified as shown below:

$$\frac{dN_I}{dt} = \gamma_I \Sigma_f^{fuel} \varphi - \lambda_I N_I$$

When the rate of production of iodine equals the rate of removal of iodine, equilibrium exists. The iodine concentration then remains constant and is designated $N^I(eq)$. The following equation for the equilibrium concentration of iodine can be determined from the preceding equation by setting the two terms equal to each other and solving for $N^I(eq)$.

$$N_I = \frac{\gamma_I \Sigma_f^{fuel} \varphi}{\lambda_I}$$

Since the equilibrium iodine concentration is proportional to the fission reaction rate, it is also proportional to reactor power level.

The rate of change of the xenon concentration is equal to the rate of production minus the rate of removal. Recall that 5% of xenon comes directly from fission

and 95% comes from the decay of iodine. The rate of change of xenon concentration is expressed by the following equations:

$$\begin{pmatrix}\text{rate of change of}\\ \text{Xe}^{135}\text{ present}\end{pmatrix} = \begin{pmatrix}\text{Xe}^{135}\text{ yield}\\ \text{from fission}\end{pmatrix} + \begin{pmatrix}\text{I}^{135}\\ \text{decay}\end{pmatrix} - \begin{pmatrix}\text{Xe}^{135}\\ \text{decay}\end{pmatrix} + \begin{pmatrix}\text{Xe}^{135}\\ \text{Burn-up}\end{pmatrix}$$

$$\frac{dN_{Xe}}{dt} = \gamma_{Xe}\Sigma_f^{fuel}\varphi + \lambda_I N_I - \lambda_{Xe} N_{Xe} - \sigma_a^{Xe} N_{Xe}\varphi$$

where:

N_{Xe} = Xe^{135} concentration
γ_{Xe} = fission yield of Xe^{135}
Σ_f^{fuel} = macroscopic fission cross section fuel
φ = thermal neutron flux
λ_I = decay constant for I^{135}
N_I = I^{135} concentration
λ_{Xe} = decay constant for Xe^{135}
σ_a^{Xe} = microscopic absorption cross section Xe^{135}

The xenon burn-up term above refers to neutron absorption by **Xenon-135** (Xe^{135}) by the following reaction:

$$_{54}Xe^{135} + {}_0n^1 \rightarrow {}_{54}Xe^{136} + \gamma$$

Xenon-136 (Xe^{136}) is not a significant neutron absorber; therefore, the neutron absorption by **Xenon-135** (Xe^{135}) constitutes removal of poison from the reactor. The burnup rate of **Xenon-135** (Xe^{135}) is dependent upon the neutron flux and the **Xenon-135** (Xe^{135}) concentration.

The equilibrium concentration of **Xenon-135** (Xe^{135}) is designated $N_{Xe}(eq)$, and is represented as shown below:

$$N_{Xe}(eq) = \frac{\gamma_{Xe}\Sigma_f^{fuel}\varphi + \lambda_I N_I}{\lambda_{Xe} + \sigma_a^{Xe}\varphi}$$

where:

N_{Xe} = Xe^{135} equilibrium concentration
γ_{Xe} = fission yield of Xe^{135}
Σ_f^{fuel} = macroscopic fission cross section fuel
φ = thermal neutron flux
λ_I = decay constant for I^{135}
N_I = I^{135} concentration
λ_{Xe} = decay constant for Xe^{135}
σ_a^{Xe} = microscopic absorption cross section Xe^{135}

For **Xenon-135** (Xe^{135}) to be in equilibrium, **Iodine-135** (I^{135}) must also be in equilibrium. Substituting the expression for equilibrium **Iodine-135** (I^{135}) concentration into the equation for equilibrium xenon results in the following:

$$N_{Xe}(eq) = \frac{(\gamma_{Xe} + \gamma_I)\Sigma_f^{fuel}\varphi + \lambda_I N_I}{\lambda_{Xe} + \sigma_a^{Xe}\varphi}$$

where:

N_{Xe} = Xe^{135} equilibrium concentration
γ_{Xe} = fission yield of Xe^{135}
γ_I = fission yield of I^{135}
Σ_f^{fuel} = macroscopic fission cross section fuel
φ = thermal neutron flux
λ_I = decay constant for I^{135}
N_I = I^{135} concentration
λ_{Xe} = decay constant for Xe^{135}
σ_a^{Xe} = microscopic absorption cross section Xe^{135}

From this equation it can be seen that the equilibrium value for **Xenon-135** (**Xe^{135}**) increases as power increases. Thermal flux is also in the denominator; therefore, as the thermal flux exceeds 10^{12} neutrons/cm^2-sec, the term begins to dominate, and at approximately 10^{15} neutrons/cm^2-sec, the **Xenon-135** (**Xe^{135}**) concentration approaches a limiting value.

The higher the power level, or flux, the higher the equilibrium **Xenon-135** (**Xe^{135}**) concentration, but equilibrium **Xenon-135** (**Xe^{135}**) is not directly proportional to power level. For example, equilibrium **Xenon-135** (**Xe^{135}**) at 25% power is more than half the value for equilibrium **Xenon-135** (**Xe^{135}**) at 100% power for many reactors. Because the **Xenon-135** (**Xe^{135}**) concentration directly affects the reactivity level in the reactor core, the negative reactivity due to the xenon concentrations for different power levels or conditions are frequently plotted instead of the xenon concentration.

5.3. XENON-135 RESPONSE TO REACTOR SHUTDOWN

When a reactor is shutdown, the neutron flux is reduced essentially to zero. Therefore, after shutdown, **Xenon-135** (**Xe^{135}**) is no longer produced by fission and is no longer removed by burnup. The only remaining production mechanism is the decay of the , **Iodine-135** (**I^{135}**) which was in the core at the time of shutdown. The only removal mechanism for **Xenon-135** (**Xe^{135}**) is decay.

$$\frac{dN_{Xe}}{dt} = \lambda_I N_I - \lambda_{Xe} N_{Xe}$$

where:

N_{Xe} = Xe^{135} concentration
λ_I = decay constant for I^{135}
N_I = I^{135} concentration
λ_{Xe} = decay constant for Xe^{135}

Because the decay rate of iodine-135 is faster than the decay rate of xenon-135, the xenon concentration builds to a peak. The peak is reached when the product of the terms IINI is equal to IXeNXe (in about 10 to 11 hours). Subsequently, the production from iodine decay is less than the removal of xenon by decay, and the concentration of **Xenon-135** (**Xe^{135}**) decreases. The greater the flux level prior to shutdown, the greater the concentration of **Iodine-135** (**I^{135}**) at shutdown;

therefore, the greater the peak in **Xenon-135** (**Xe^{135}**) concentration after shutdown.

Negative **Xenon-135** (**Xe^{135}**) reactivity, also called **Xenon Poisoning,** may provide sufficient negative reactivity to make the reactor inoperable because there is insufficient positive reactivity available from control rod removal or chemical shim dilution (if used) to counteract it. The inability of the reactor to be started due to the effects of **Xenon-135** (**Xe^{135}**) is sometimes referred to as a **xenon precluded startup**.

The period of time where the reactor is unable to "override" the effects of **xenon** is called **xenon dead time**. Because the amount of excess core reactivity available to override the negative reactivity of the **Xenon-135** (**Xe^{135}**) is usually less than 10%Δ**k/k,** thermal power reactors are normally limited to flux levels of about 5 x 10^{13} neutrons/cm^2 -sec so that timely restart can be ensured after shutdown. For reactors with very low thermal flux levels (~5 x 10^{12} neutrons/cm^2-sec or less), most **Xenon-135** (**Xe^{135}**) is removed by decay as opposed to neutron absorption. For these cases, reactor shutdown does not cause any **Xenon-135** (**Xe^{135}**) peaking effect.

Following the peak in **Xenon-135** (**Xe^{135}**) concentration about 10 hours after shutdown, the **Xenon-135** (**Xe^{135}**) concentration will decrease at a rate controlled by the decay of **Iodine-135** (**I^{135}**) into **Xenon-135** (**Xe^{135}**) and the decay rate of **Xenon-135** (**Xe^{135}**). For some reactors, the **Xenon-135** (**Xe^{135}**) concentration about 20 hours after shutdown from full power will be the same as the equilibrium **Xenon-135** (**Xe^{135}**) concentration at full power. About 3 days after shutdown, the **Xenon-135** (**Xe^{135}**) concentration will have decreased to a small percentage of its pre-shutdown level, and the reactor can be assumed to be **Xenon-135** (**Xe^{135}**) free without a significant error introduced into reactivity calculations.

5.4. XENON-135 (**Xe^{135}**) OSCILLATIONS

Large thermal reactors with little flux coupling between regions may experience spatial power oscillations because of the non-uniform presence of **Xenon-135** (**Xe^{135}**). The mechanism is described in the following four steps:

5.4.1. An initial lack of symmetry in the core power distribution (for example, individual control rod movement or misalignment) causes an imbalance in fission rates within the reactor core, and therefore, in the iodine-135 buildup and the **Xenon-135** (**Xe^{135}**) xenon-135 absorption.

5.4.2. In the high-flux region, **Xenon-135** (**Xe^{135}**) xenon-135 burnout allows the flux to increase further, while in the low-flux region, the increase in **Xenon-35** (**Xe^{135}**) causes a further reduction in flux. The iodine concentration increases where the flux is high and decreases where the flux is low.

5.4.3. As soon as the **Iodine-135** (I^{135}) levels build up sufficiently, decay to **Xenon** reverses the initial situation. Flux decreases in this area, and the former low-flux region increases in power.

5.4.4. Repetition of these patterns can lead to **Xenon** oscillations moving about the core with periods on the order of about 15 hours. With little change in overall power level, these oscillations can change the local power levels by a factor of three or more. In a reactor system with strongly negative temperature coefficients, the **Xenon-135** (Xe^{135}) oscillations are damped quite readily. This is one reason for designing reactors to have negative moderator-temperature coefficients.

5.5. XENON-135 (Xe^{135}) RESPONSE TO REACTOR POWER CHANGES

During periods of steady state operation, at a constant neutron flux level, the **Xenon-135** (Xe^{135}) concentration builds up to its equilibrium value for that reactor power in about 40 to 50 hours. Most **Xenon-135** (Xe^{135}) transients occurs as a result of a change in reactor power level. At time zero, reactor power is raised from 50% power to 100% power. When the reactor power is increased, **Xenon-135** (Xe^{135}) xenon concentration initially decreases because the burnup is increased at the new higher power level.

Because 95% of the **Xenon-135** (Xe^{135}) production is from **Iodine-135** (I^{135}) decay, which has a 6 to 7 hour half-life, the production of **Xenon-135** (Xe^{135}) remains constant for several hours. After a few hours (roughly 4 to 6 hours depending on power levels) the rate of production of **Xenon-135** (Xe^{135}) from iodine and fission equals the rate of removal of **Xenon-135** (Xe^{135}) by burnup and decay. At this point, the **Xenon-135** (Xe^{135}) concentration reaches a minimum.

The **Xenon-135** (Xe^{135}) xenon concentration then increases to the new equilibrium level for the new power level in roughly 40 to 50 hours. It should be noted that the magnitude and the rate of change of **Xenon-135** (Xe^{135}) concentration during the initial 4 to 6 hours following the power change is dependent upon the initial power level and on the amount of change in power level. The **Xenon concentration change** is greater for a larger change in power level.

When reactor power is decreased from 100% to 50% power (t = 55 hours), the process is reversed. There is an immediate decrease in **Xenon** burnup, which results in an increase in **Xenon-135** (Xe^{135}) concentration. The **Iodine-135** (I^{135}) concentration is still at the higher equilibrium level for 100% power and is therefore still producing **Xenon-135** (Xe^{135}) at the higher rate. The **Xenon-135** (Xe^{135}) concentration continues to rise until the rate of production of **Xenon-135** (Xe^{135}) becomes equal to the rate of removal (roughly 7 to 8 hours after the initial reduction in power level). The **Xenon-135** (Xe^{135}) concentration then gradually decreases to the new equilibrium level in about 50 to 60 hours.
The magnitude of the **Xenon** peak is greatest if the initial power level is very high. Maximum peak **Xenon** occurs when a reactor that is operating at 100% equilibrium **Xenon** concentration is suddenly shut down. The most rapid possible

burnout of **Xenon** occurs when a reactor is started up and operated at full power while this maximum peak xenon condition exists.

SECTION SIX: SAMARIUM AND OTHER FISSION PRODUCT POISONS

6.1. PRODUCTION AND REMOVAL OF SAMARIUM-149 (Sm^{149})

The fission product poison that has the most significant effect on reactor operations other than **Xenon-135** (Xe^{135}) is **Samarium-149** (Sm^{149}). **Samarium-149** (Sm^{149}) behaves significantly different from **Xenon-135** (Xe^{135}) due to its different nuclear properties.

Samarium-149 (Sm^{149}) is the second most important fission-product poison because of its high thermal neutron absorption cross section of 4.1×10^4 barns. **Samarium-149** (Sm^{149}) is produced from the decay of the **Neodymium-149** (Nd^{149}) fission fragment. For the purpose of examining the behavior of **Samarium-149** (Sm^{149}), the 1.73 hour half-life of **Neodymium-149** (Nd^{149}) is sufficiently shorter than the 53.1 hour value for **Promethium-149** (Pm^{149}) that the **Promethium-149** (Pm^{149}) may be considered as if it were formed directly from fission. This assumption, and neglecting the small amount of **Promethium-149** (Pm^{149}) burnup, allows the situation to be described as follows:

Rate of change of Pm^{149} = yield from fission - decay Pm^{149} concentration

therefore:

$$\frac{dN_{Pm}}{dt} = \gamma_{Pm} \Sigma_f^{fuel} \varphi + \lambda_{Pm} N_{Pm}$$

where:

N_{Pm} = Pm^{149} equilibrium concentration
γ_{Pm} = fission yield of Pm^{149}
Σ_f^{fuel} = macroscopic fission cross section fuel
φ = thermal neutron flux
λ_{Pm} = decay constant for Pm^{149}

Solving for the equilibrium value of **Promethium-149** (Pm^{149}) gives the following:

$$N_{Pm}(eq.) = \frac{\gamma_{Pm} \Sigma_f^{fuel} \varphi}{\lambda_{Pm}}$$

where:

$N_{Pm}(eq.)$ = Pm^{149} equilibrium concentration
γ_{Pm} = fission yield of Pm^{149}
Σ_f^{fuel} = macroscopic fission cross section fuel
φ = thermal neutron flux
λ_{Pm} = decay constant for Pm^{149}

The rate of **Samarium-149** (Sm^{149}) formation is described as follows:

Rate of change of Sm^{149} = yield from fission + Pm^{149} decay - Sm^{149} burnup

therefore:

$$\frac{d N_{Sm}}{dt} = \gamma_{Sm} \Sigma_f^{fuel} \varphi + \lambda_{Pm} N_{Pm} - \sigma_a^{Sm} N_{Sm} \varphi$$

where:

N_{Sm} = **Sm^{149}** concentration
γ_{Sm} = fission yield of **Sm^{149}**
Σ_f^{fuel} = macroscopic fission cross section fuel
φ = thermal neutron flux
λ_{Pm} = decay constant for **Pm^{149}**
N_{Pm} = **Pm^{149}** concentration
λ_{Xe} = decay constant for **Xe^{135}**
σ_a^{Sm} = microscopic absorption cross section **Sm^{149}**

The fission yield of **Samarium-149** (**Sm^{149}**), however, is nearly zero; therefore, the equation becomes the following:

$$\frac{d N_{Sm}}{dt} = \lambda_{Pm} N_{Pm} - \sigma_a^{Sm} N_{Sm} \varphi$$

Solving this equation for the equilibrium concentration of **Samarium-149** (**Sm^{149}**) and substituting

$$N_{Pm}(eq.) = \frac{\gamma_{Pm} \Sigma_f^{fuel} \varphi}{\lambda_{Pm}}$$

for the equilibrium concentration of **Promethium-149** (**Pm^{149}**), yields:

$$N_{Sm}(eq.) = \frac{\gamma_{Pm} \Sigma_f^{fuel} \varphi}{\lambda_{Sm}}$$

This expression for equilibrium **Samarium-149** (**Sm^{149}**) concentration during reactor operation illustrates that equilibrium **Samarium-149** (**Sm^{149}**), concentration is independent of neutron flux and power level. The **Samarium** concentration will undergo a transient following a power level change, but it will return to its original value.

6.2. SAMARIUM-149 RESPONSE TO REACTOR SHUTDOWN

Since the neutron flux drops to essentially zero after reactor shutdown, the rate of samarium-149 production becomes the following.

$$\frac{d N_{Sm}}{dt} = \lambda_{Pm} N_{Pm}$$

where:

N_{Sm} = **Sm^{149}** concentration
λ_{Pm} = decay constant for **Pm^{149}**
N_{Pm} = **Pm^{149}** concentration

Because **Samarium-149** (**Sm^{149}**) is not radioactive and is not removed by decay, it presents problems somewhat different from those encountered with **Xenon-135** (**Xe^{135}**). The equilibrium concentration and the poisoning effect build to an

equilibrium value during reactor operation. This equilibrium is reached in approximately 20 days (500 hours), and since **Samarium-149** (**Sm^{149}**) is stable, the concentration remains essentially constant during reactor operation.

When the reactor is shutdown, the **Samarium-149** (**Sm^{149}**) concentration builds up as a result of the decay of the accumulated **Promethium-149** (**Pm^{149}**). The buildup of **Samarium-149** (**Sm^{149}**) after shutdown depends upon the power level before shutdown. **Samarium-149** (**Sm^{149}**) does not peak as **Xenon-135** (**Xe^{135}**) does, but increases slowly to a maximum value.

After shutdown, if the reactor is then operated at power, **Samarium-149** (**Sm^{149}**) is burned up and its concentration returns to the equilibrium value. **Samarium-149** (**Sm^{149}**) poisoning is minor when compared to **Xenon-135** (**Xe^{135}**) poisoning. Although **Samarium-149** (**Sm^{149}**) has a constant poisoning effect during long-term sustained operation, its behavior during initial startup and during post-shutdown and restart periods requires special considerations in reactor design.

The **Xenon-135** (**Xe^{135}**) and **Samarium-149** (**Sm^{149}**) mechanisms are dependent on their very large thermal neutron cross sections and only affect thermal reactor systems. In fast reactors, neither these nor any other fission products have a major poisoning influence.

6.3. OTHER NEUTRON POISONS

There are numerous other fission products that, as a result of their concentration and thermal neutron absorption cross section, have a poisoning effect on reactor operation. Individually, they are of little consequence, but "lumped" together they have a significant impact. These are often characterized as **lumped fission product poisons** and accumulate at an average rate of 50 barns per fission event in the reactor. In addition to fission product poisons, other materials in the reactor decay to materials that act as neutron poisons. An example of this is the decay of tritium to helium-3. Since tritium has a half-life of 12.3 years, normally this decay does not significantly affect reactor operations because the rate of decay of tritium is so slow. However, if tritium is produced in a reactor and then allowed to remain in the reactor during a prolonged shutdown of several months, a sufficient amount of tritium may decay to helium-3 to add a significant amount of negative reactivity.

SECTION SEVEN: CONTROL RODS

Most reactors contain control rods made of neutron absorbing materials that are used to adjust the reactivity of the core. Control rods can be designed and used for coarse control, fine control, or fast shutdowns.

7.1. SELECTION OF CONTROL ROD MATERIALS

Rods of neutron-absorbing material are installed in most reactors to provide precise, adjustable control of reactivity. These rods are able to be moved into or out of the reactor core and typically contain elements such as silver, indium, cadmium, boron, or hafnium. The material used for the control rods varies depending on reactor design. Generally, the material selected should have a

good absorption cross section for neutrons and have a long lifetime as an absorber (not burn out rapidly).

The ability of a control rod to absorb neutrons can be adjusted during manufacture. A control rod that is referred to as a "black" absorber absorbs essentially all incident neutrons. A "grey" absorber absorbs only a part of them. While it takes more grey rods than black rods for a given reactivity effect, the grey rods are often preferred because they cause smaller depressions in the neutron flux and power in the vicinity of the rod. This leads to a flatter neutron flux profile and more even power distribution in the core.

If grey rods are desired, the amount of material with a high absorption cross section that is loaded in the rod is limited. Material with a very high absorption cross section may not be desired for use in a control rod, because it will burn out rapidly due to its high absorption cross section. The same amount of reactivity worth can be achieved by manufacturing the control rod from material with a slightly lower cross section and by loading more of the material. This also results in a rod that does not burn out as rapidly.

Another factor in control rod material selection is that materials that resonantly absorb neutrons are often preferred to those that merely have high thermal neutron absorption cross sections. Resonance neutron absorbers absorb neutrons in the epithermal energy range. The path length traveled by the epithermal neutrons in a reactor is greater than the path length traveled by thermal neutrons. Therefore, a resonance absorber absorbs neutrons that have their last collision farther (on the average) from the control rod than a thermal absorber. This has the effect of making the area of influence around a resonance absorber larger than around a thermal absorber and is useful in maintaining a flatter flux profile.

7.2. TYPES OF CONTROL RODS

There are several ways to classify the types of control rods. One classification method is by the purpose of the control rods. Three purposes of control rods are listed below.

A. **Shim rods**: used for coarse control and/or to remove reactivity in relatively large amounts.

B. **Regulating rods:** used for fine adjustments and to maintain desired power or temperature.

C. **Safety rods:** provide a means for very fast shutdown in the event of an unsafe condition. Addition of a large amount of negative reactivity by rapidly inserting the safety rods is referred to as a "scram" or "trip."

Not all reactors have different control rods to serve the purposes mentioned above. Depending upon the type of reactor and the controls necessary, it is possible to use dual-purpose or even triple-purpose rods. For example, consider a set of control rods that can insert enough reactivity to be used as shim rods. If the same rods can be operated at slow speeds, they will function as regulating

rods. Additionally, these same rods can be designed for rapid insertion, or scram. These rods serve a triple function yet meet other specifications such as precise control, range of control, and efficiency.

7.3. CONTROL ROD EFFECTIVENESS

The effectiveness of a control rod depends largely upon the value of the ratio of the neutron flux at the location of the rod to the average neutron flux in the reactor. The control rod has maximum effect (inserts the most negative reactivity) if it is placed in the reactor where the flux is maximum. If a reactor has only one control rod, the rod should be placed in the center of the reactor core.

If additional rods are added to this simple reactor, the most effective location is where the flux is maximum. Numerous control rods are required for a reactor that has a large amount of excess reactivity (that amount of reactivity in excess of that needed to be critical). The exact amount of reactivity that each control rod inserts depends upon the reactor design. The change in reactivity caused by control rod motion is referred to as control rod worth.

7.4. INTEGRAL AND DIFFERENTIAL CONTROL ROD WORTH

The exact effect of control rods on reactivity can be determined experimentally. For example, a control rod can be withdrawn in small increments, such as 0.5 inch, and the change in reactivity can be determined following each increment of withdrawal. By plotting the resulting reactivity versus the rod position, a graph is obtained. The graph depicts integral control rod worth over the full range of withdrawal. The **integral control rod worth** is the total reactivity worth of the rod at that particular degree of withdrawal and is usually defined to be the greatest when the rod is fully withdrawn.

The slope of the curve ($\Delta\rho/\Delta$**x**), and therefore the amount of reactivity inserted per unit of withdrawal, is greatest when the control rod is midway out of the core. This occurs because the area of greatest neutron flux is near the center of the core; therefore, the amount of change in neutron absorption is greatest in this area. If the slope of the curve for integral rod worth is taken, the result is a value for rate of change of control rod worth as a function of control rod position. A plot of the slope of the integral rod worth curve, also called the differential control rod worth.

At the bottom of the core, where there are few neutrons, rod movement has little effect so the change in rod worth per inch varies little. As the rod approaches the center of the core its effect becomes greater, and the change in rod worth per inch is greater. At the center of the core the differential rod worth is greatest and varies little with rod motion. From the center of the core to the top, the rod worth per inch is basically the inverse of the rod worth per inch from the center to the bottom.

Differential control rod worth is the reactivity change per unit movement of a rod and is normally expressed as ρ**/inch**, Δ**k/k per inch**, or **pcm/inch**. The integral rod worth at a given withdrawal is merely the summation of all the

differential rod worths up to that point of withdrawal. It is also the area under the differential rod worth curve at any given withdrawal position.

7.5. ROD CONTROL MECHANISM

The control rod insertion rates on a scram are designed to be sufficient to protect the reactor against damage in all transients that are expected to occur during the life of the reactor. During normal rod motion, the control rods must be able to move rapidly enough to compensate for the most rapid rate at which positive reactivity is expected to build within the reactor in order to provide positive control. The transient that is normally considered when setting this minimum rod speed is the burnout of maximum peak xenon while at full power. Xenon burnout is usually the most rapid, non-accident transient expected. The maximum rod speed is normally limited in order to reduce the severity of an accident involving the continuous withdrawal of control rods.

END OF CHAPTER THREE: INTRODUCTION TO REACTOR PHYSICS

CHAPTER FOUR: INTRODUCTION TO REACTOR OPERATIONS

SECTION ONE: SUB-CRITICAL MULTIPLICATION

1.1. SUB-CRITICAL MULTIPLICATION FACTOR

When a reactor is in a shutdown condition, neutrons are still present to interact with the fuel. These source neutrons are produced by a variety of methods that were discussed in the white paper on Neutron Physics. If neutrons and fissionable material are present in the reactor, fission will take place. Therefore, a reactor will always be producing a small number of fissions even when it is shutdown.

Consider a reactor in which $\mathbf{k_{eff}}$ is 0.6. If 100 neutrons are suddenly introduced into the reactor, these 100 neutrons that start the current generation will produce 60 neutrons (100 x 0.6) from fission to start the next generation. The 60 neutrons that start the second generation will produce 36 neutrons (60 x 0.6) to start the third generation.

Because the reactor is sub-critical, neutrons introduced in the reactor will have a decreasing effect on each subsequent generation. The addition of source neutrons to the reactor containing fissionable material has the effect of maintaining a much higher stable neutron level due to the fissions occurring than the neutron level that would result from the source neutrons alone. A neutron source strength of 100 neutrons per generation will result in 250 neutrons per generation being produced from a combination of sources and fission in a shutdown reactor with a $\mathbf{k_{eff}}$ of 0.6. If the value of $\mathbf{k_{ef}}$ were higher, the source neutrons would produce a greater number of fission neutrons and their effects would be felt for a larger number of subsequent generations after their addition to the reactor.

The effect of fissions in the fuel increasing the effective source strength of a reactor with a $\mathbf{k_{eff}}$ of less than one is **sub-critical multiplication**. For a given value of $\mathbf{k_{eff}}$ there exists a **sub-critical multiplication factor** (**M**) that relates the source level to the steady-state neutron level of the core.

If the value of $\mathbf{k_{eff}}$ is known, the amount that the neutron source strength will be multiplied (M) can easily be determined by the Equation below:

$$\mathbf{M = \frac{1}{1 - k_{eff}}}$$

where:

M = sub-critical multiplication factor
$\mathbf{k_{eff}}$ = effective multiplication factor

The equation above illustrates that the sub-critical multiplication factor will increase as positive reactivity is added to a shutdown reactor, increasing the value of $\mathbf{k_{eff}}$. If the source strength of this reactor were 1000 neutrons/sec, the neutron level would increase from 2500 neutrons/second at a $\mathbf{k_{eff}}$ keff of 0.6 to a neutron level of 71,400 neutrons/sec at a keff of 0.986.

1.2. Effect of Reactivity Changes on Sub-critical Multiplication

In a sub-critical reactor, the neutron level is related to the source strength by the following equation:

$N = S \times M$

where:

N = neutron level
S = neutron source strength
M = sub-critical multiplication factor

If the term **M** in the previous equation is replaced by the expression:

$$M = \frac{1}{1 - k_{eff}}$$

the following expression results:

$$N = S \times \frac{1}{(1 - k_{eff})}$$

To this point it has been necessary to know the neutron source strength of the reactor in order to use the concept of **sub-critical multiplication**. In most reactors the actual strength of the neutron sources is difficult, if not impossible, to determine. Even though the actual source strength may not be known, it is still possible to relate the change in reactivity to a change in neutron level.

Consider a reactor at two different times when k_{eff} is two different values, k_{eff1} and k_{eff2}. The neutron level at each time can be determined based on the neutron source strength and the **sub-critical multiplication factor** using our previous equation:

$$N_1 = \frac{S \times 1}{(1 - k_{eff1})} \quad \text{and}$$

$$N_2 = \frac{S \times 1}{(1 - k_{eff2})}$$

The equation for N_1 can be divided by N_2 and

$$\frac{N_1}{N_2} = \frac{(1 - k_{eff2})}{(1 - k_{eff1})}$$

Because the source strength appears in both the numerator and denominator, it cancels out of the equation. Therefore, the neutron level at any time can be determined based on the neutron level present at any other time provided the values of k_{eff} or reactivity for both times are known.

The neutron level in a shutdown reactor is typically monitored using instruments that measure the neutron leakage out of the reactor. The neutron leakage is proportional to the neutron level in the reactor. Typical units for displaying the instrument reading are **counts per second** (**cps**). Because the instrument count rate is proportional to the neutron level, the above equation can be restated as shown below:

$$\frac{CR_1}{CR_2} = \frac{(1 - k_{eff2})}{(1 - k_{eff1})}$$

where:

CR_1 = count rate at time 1
CR_2 = count rate at time 2
k_{eff1} = k_{eff} at time 1
k_{eff2} = k_{eff} at time 2

This equation is very useful during the shutdown operation of a reactor. Before adding positive reactivity to a reactor, it is possible to predict the effect the reactivity addition will have on the neutron level.

1.3 USE OF 1/M PLOTS

Because the sub-critical multiplication factor is related to the value of k_{eff}, it is possible to monitor the approach to criticality through the use of the sub-critical multiplication factor. As positive reactivity is added to a sub-critical reactor, k_{eff} will get nearer to one. As k_{eff} gets nearer to one, the sub-critical multiplication factor (**M**) gets larger.

The closer the reactor is to criticality, the faster **M** will increase for equal step insertions of positive reactivity. When the reactor becomes critical, **M** will be infinitely large. For this reason, monitoring and plotting **M** during an approach to criticality is impractical because there is no value of **M** at which the reactor clearly becomes critical. Instead of plotting **M** directly, its inverse (**1/M**) is plotted on a graph of **1/M** versus rod height:

$$M = \frac{1}{1 - k_{eff}}$$

$$\frac{1}{M} = 1 - k_{eff}$$

As control rods are withdrawn and k_{eff} approaches one and **M** approaches infinity, **1/M** approaches zero. For a critical reactor, 1/M is equal to zero. A true 1/M plot requires knowledge of the neutron source strength. Because the actual source strength is usually unknown, a reference count rate is substituted, and the calculation of the factor **1/M** is through the use of the following equation:

$$\frac{1}{M} = \frac{CR_0}{CR}$$

where: **1/M** = inverse multiplication factor
CR_0 = reference count rate
CR = current count rate

In practice, the reference count rate used is the count rate prior to the beginning of the reactivity change. The startup procedures for many reactors include instructions to insert positive reactivity in incremental steps with delays between the reactivity insertions to allow time for sub-critical multiplication to increase the steady-state neutron population to a new, higher level and allow more accurate plotting of **1/M.**

The neutron population will typically reach its new steady-state value within 1-2 minutes, but the closer the reactor is to criticality, the longer the time will be to stabilize the neutron population.

SECTION TWO: REACTOR KINETICS

The response of neutron flux and reactor power to changes in reactivity is much different in a critical reactor than in a sub-critical reactor. The reliance of the chain reaction on delayed neutrons makes the rate of change of reactor power controllable.

2.1. REACTOR PERIOD ( )

The **reactor period** is defined as the time required for reactor power to change by a factor of "e," where "e" is the base of the natural logarithm and is equal to about 2.718. The reactor period is usually expressed in units of seconds. From the definition of reactor period, it is possible to develop the relationship between reactor power and reactor period that is expressed by the following equation:

$$P = P_0 \times e^{t/\tau}$$

where:

P = transient reactor power
P_0 = initial reactor power
τ = reactor period (seconds)
t = time during the reactor transient (seconds)

The smaller the value of τ, the more rapid the change in reactor power. If the reactor period is positive, reactor power is increasing. If the reactor period is negative, reactor power is decreasing.

There are numerous equations used to express reactor period, but the following equation will be useful in most situations. The first term in this equation is the prompt term and the second term is the delayed term:

$$\tau = \frac{\ell^*}{\rho} + \frac{\beta_{eff} - \rho}{\lambda_{eff}\,\rho - (d\rho/dt)}$$

where:

τ = Reactor Period
ℓ^* = prompt generation lifetime
β_{eff} = effective delayed neutron fraction
ρ = reactivity
λ_{eff} = effective delayed neutron precursor decay constant
$d\rho/dt$ = rate of change of reactivity

2.2. EFFECTIVE DELAYED NEUTRON FRACTION

Recall that β, the **delayed neutron fraction**, is the fraction of all fission neutrons that are born as delayed neutrons. The value of β depends upon the actual nuclear fuel used. As discussed in our first white paper, the delayed neutron precursors for a given type of fuel are grouped on the basis of half-life.

The term $\bar{\beta}$ (pronounced beta-bar) is the **average delayed neutron fraction**. The value of $\bar{\beta}$ is the weighted average of the total delayed neutron fractions of the individual types of fuel. Each total delayed neutron fraction value for each

type of fuel is weighted by the percent of total neutrons that the fuel contributes through fission. If the percentage of fissions occurring in the different types of fuel in a reactor changes over the life of the core, the average delayed neutron fraction will also change.

For a light water reactor using low enriched fuel, the average delayed neutron fraction can change from 0.0070 to 0.0055 as **Uranium-235** ($\mathbf{U^{235}}$) is burned out and **Plutonium-239** ($\mathbf{Pu^{239}}$) is produced from **Uranium-238** ($\mathbf{U^{238}}$).

Delayed neutrons do not have the same properties as prompt neutrons released directly from fission. The average energy of prompt neutrons is about 2 MeV. This is much greater than the average energy of delayed neutrons (about 0.5 MeV). The fact that delayed neutrons are born at lower energies has two significant impacts on the way they proceed through the neutron life cycle.

1. Delayed neutrons have a much lower probability of causing fast fissions than prompt neutrons because their average energy is less than the minimum required for fast fission to occur.
2. Delayed neutrons have a lower probability of leaking out of the core while they are at fast energies, because they are born at lower energies and subsequently travel a shorter distance as fast neutrons.

These two considerations (lower fast fission factor and higher fast non-leakage probability for delayed neutrons) are taken into account by a term called the **importance factor** (**I**). The importance factor relates the **average delayed neutron fraction** to the **effective delayed neutron fraction.**

The **effective delayed neutron fraction** ($\bar{\beta}_{\mathbf{eff}}$) is defined as the fraction of neutrons at thermal energies which were born delayed. The effective delayed neutron fraction is the product of the average delayed neutron fraction and the importance factor.

$$\bar{\beta}_{\mathbf{eff}} = \bar{\beta} \times \mathbf{I}$$

where: $\bar{\beta}_{\mathbf{eff}}$ = effective delayed neutron fraction
$\bar{\beta}$ = average delayed neutron fraction
I = importance factor

In a small reactor with highly enriched fuel, the increase in fast non-leakage probability will dominate the decrease in the fast fission factor, and the importance factor will be greater than one. In a large reactor with low enriched fuel, the decrease in the fast fission factor will dominate the increase in the fast non-leakage probability and the importance factor will be less than one (about 0.97 for a commercial **PWR**).

2.3. EFFECTIVE DELAYED NEUTRON PRECURSOR DECAY CONSTANT

Another new term has been introduced in the **reactor period** (τ) equation. That term is $\lambda_{\mathbf{eff}}$ (pronounced lambda effective), the **effective delayed neutron precursor decay constant.** The decay rate for a given delayed neutron precursor can be expressed as the product of precursor concentration and the decay constant (λ) of that precursor. The decay constant of a precursor is simply

the fraction of an initial number of the precursor atoms that decays in a given unit time.

A decay constant of 0.1 sec^{-1}, for example, implies that one-tenth, or ten percent, of a sample of precursor atoms decays within one second. The value for the effective delayed neutron precursor decay constant, λ_{eff} , varies depending upon the balance existing between the concentrations of the precursor groups and the nuclide(s) being used as the fuel.

If the reactor is operating at a constant power, all the precursor groups reach an equilibrium value. During an up-power transient, however, the shorter-lived precursors decaying at any given instant were born at a higher power level (or flux level) than the longer-lived precursors decaying at the same instant. There is, therefore, proportionately more of the shorter-lived and fewer of the longer-lived precursors decaying at that given instant than there are at constant power. The value of λ_{eff} is closer to that of the shorter-lived precursors.

During a down-power transient the longer-lived precursors become more significant. The longer-lived precursors decaying at a given instant were born at a higher power level (or flux level) than the shorter-lived precursors decaying at that instant. Therefore, proportionately more of the longer-lived precursors are decaying at that instant, and the value of λ_{eff} approaches the values of the longer-lived precursors.
Approximate values for λ_{eff} are 0.08 sec for steady-state operation, 0.1 sec for a power increase, and 0.05 sec^{-1} for a power decrease. The exact values will depend upon the materials used for fuel and the value of the reactivity of the reactor core.

$$\tau = \frac{\ell^*}{\rho} + \frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\,\rho - (d\rho/dt)}$$

where:

τ = Reactor Period
ℓ^* = prompt generation lifetime
$\bar{\beta}_{eff}$ = effective delayed neutron fraction
ρ = reactivity
λ_{eff} = effective delayed neutron precursor decay constant
$d\rho/dt$ = rate of change of reactivity

The first term of this equation is the prompt term and the second term is the delayed term of the Nuclear Reaction. If the positive reactivity added is less than the value of $\bar{\beta}_{eff}$, the emission of prompt fission neutrons alone is not sufficient to overcome losses to non-fission absorption and leakage. If delayed neutrons were not being produced, the neutron population would decrease as long as the reactivity of the core has a value less than the effective delayed neutron fraction. The positive reactivity insertion is followed immediately by a small immediate power increase called the *prompt jump*. This power increase occurs because the rate of production of prompt neutrons changes abruptly as the reactivity is added.

Recall from an earlier module that the generation time for prompt neutrons is on the order of 10-13 seconds. The effect can be seen in Figure 2.

After the prompt jump, the rate of change of power cannot increase any more rapidly than the built-in time delay the precursor half-lives allow. Therefore, the power rise is controllable, and the reactor can be operated safely.

Conversely, in the case where negative reactivity is added to the core there will be a prompt drop in reactor power. The **prompt drop** is the small immediate decrease in reactor power caused by the negative reactivity addition. After the prompt drop, the rate of change of power slows and approaches the rate determined by the delayed term of the equation above.

2.4. PROMPT CRITICALITY

It can be readily seen from the equation above that if the amount of positive reactivity added equals the value of $\bar{\beta}_{\mathbf{eff}}$, the reactor period equation becomes the following:

$$\tau = \frac{\ell^*}{\rho}$$

where: τ = Reactor Period

ℓ^* = prompt generation lifetime

In this case, the production of prompt neutrons alone is enough to balance neutron losses and increase the neutron population. The condition where the reactor is critical on prompt neutrons, and the neutron population increases as rapidly as the prompt neutron generation lifetime allows is known as **prompt critical**.

The **prompt critical** condition does not signal a dramatic change in neutron behavior. The reactor period changes in a regular manner between reactivities above and below this reference. **Prompt critical** is, however, a convenient condition for marking the transition from delayed neutron to prompt neutron time scales. A reactor whose reactivity even approaches **prompt critical** is likely to suffer damage due to the rapid rise in power to a very high level. For example, a reactor which has gone prompt critical could experience a several thousand percent power increase in less than one second.

Because the prompt critical condition is so important, a specific unit of reactivity has been defined that relates to it. The unit of reactivity is the dollar (**$**), where one dollar of reactivity is equivalent to the effective delayed neutron fraction . A reactivity unit related to the dollar is the **cent**, where one **cent** is one-hundredth of a dollar. **If the reactivity of the core is one dollar, the reactor is prompt critical**. Because the effective delayed neutron fraction is dependent upon the nuclides used as fuel, the value of the dollar is also dependent on the nuclides used as fuel.

2.5. STABLE PERIOD EQUATION

For normal reactor operating conditions, the value of positive reactivity in the reactor is never permitted to approach the effective delayed neutron fraction, and the reactor period equation is normally written as follows:

$$\tau = \frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\,\rho - (d\rho/dt)}$$

where: τ = Reactor Period
$\bar{\beta}_{eff}$ = effective delayed neutron fraction
ρ = reactivity
λ_{eff} = effective delayed neutron precursor decay constant
$d\rho/dt$ = rate of change of reactivity

This equation is referred to as the **transient period equation** since it incorporates the term ($d\rho/dt$) to account for the changing amount of reactivity in the core. The ℓ^*/ρ term (prompt period) is normally negligible with respect to the remainder of the equation and is often not included. For conditions when the amount of reactivity in the core is constant ($d\rho/dt = 0$), and the reactor period is unchanging, the equation above can be simplified further to the following form, which is known as the **stable period equation**.

$$\tau = \frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\,\rho}$$

where: τ = Reactor Period
$\bar{\beta}_{eff}$ = effective delayed neutron fraction
ρ = reactivity
λ_{eff} = effective delayed neutron precursor decay constant

2.6. REACTOR STARTUP RATE (SUR)

The **reactor startup rate** (**SUR**) is defined as the number of factors of ten that power changes in one minute. The units of **SUR** are powers of ten per minute, or **decades per minute** (**DPM**). The following equation shows the relationship between **reactor power** and **startup rate**:

$$P = P_0 \times 10^{SUR(t)}$$

where:

P = transient reactor power
P_0 = initial reactor power
SUR = reactor start-up rate (Decades per minute (DPM))
t = time during the reactor transient (minutes)

The relationship between reactor period and startup rate can be developed by considering the following equation;

$$P = P_0 \times e^{t/\tau}$$

where:

τ = reactor period (seconds)
t = time during the reactor transient (seconds)

and

$$P = P_0 \times 10^{SUR(t)}$$

Yields:

$$\underline{P} = e^{t/\tau} = P_0 \times 10^{SUR(t)}$$

Changing the base of the exponential term on the right side of this equation to the natural logarithm "**e**" ($10 = e^{2.303}$) and solving the result yields the following:

$$e^{t(sec)/\tau} = e^{2.303SUR(t(min))}$$

therefore:

$$t(sec)/\tau = 2.303SUR(t(min)) \text{ and}$$

$$60/\tau = 2.303SUR$$

$$SUR = 26.06/\tau$$

2.7. DOUBLING TIME

Sometimes it is useful to discuss the rate of change of reactor power in terms similar to those used in radioactive decay calculations. **Doubling or halving time** are terms that relate to the amount of time it takes reactor power to double or be reduced to one-half the initial power level. If the stable reactor period is known, doubling time can be determined as follows:

$$\text{Doubling Time (DT)} = \tau \text{ (ln 2)}$$

where: τ = stable reactor period
ln 2 = natural logarithm of 2

When the doubling time is known, the power level change from P_0 is given by the following equation:

$$P = P_0 \times e^{t/DT}$$

where:

P = transient reactor power
P_0 = initial reactor power
DT = reactor doubling time (seconds)
t = time during the reactor transient (seconds)

SECTION THREE: REACTOR OPERATION

It is important to understand the principles that determine how a reactor responds during all modes of operation. Special measures must be taken during the startup of a reactor to ensure that expected responses are occurring. During power operation, control of the flux shape is necessary to ensure operation within limits and maximum core performance. Even when a reactor is shut down, the

fact that the fission products created by the fission process continue to generate heat results in a need to monitor support systems to ensure adequate cooling of the core.

3.1. STARTUP

When a reactor is started up with un-irradiated fuel, or on those occasions when the reactor is restarted following a long shutdown period, the source neutron population will be very low. In some reactors, the neutron population is frequently low enough that it cannot be detected by the nuclear instrumentation during the approach to criticality.

Installed neutron sources are frequently used to provide a safe, easily monitored reactor startup. The neutron source, together with the sub-critical multiplication process, provides a sufficiently large neutron population to allow monitoring by the nuclear instruments throughout the startup procedure. Without the installed source, it may be possible to withdraw the control rods to the point of criticality, and then continue withdrawal without detecting criticality because the reactor goes critical below the indicating range. Continued withdrawal of control rods at this point could cause reactor power to rise at an uncontrollable rate before neutron level first becomes visible on the nuclear instruments.

An alternative to using a startup source is to limit the rate of rod withdrawal, or require waiting periods between rod withdrawal increments. By waiting between rod withdrawal increments, the neutron population is allowed to increase through **sub-critical multiplication**. **Sub=critical multiplication** is the process where source neutrons are used to sustain the chain reaction in a reactor with a multiplication factor ($\mathbf{k_{eff}}$) of less than one. The chain reaction is not "self-sustaining," but if the neutron source is of sufficient magnitude, it compensates for the neutrons lost through absorption and leakage. This process can result in a constant, or increasing, neutron population even though $\mathbf{k_{eff}}$ is less than one.

3.2. ESTIMATED CRITICAL POSITION

The **1/M** plots, previously discussed, are useful for monitoring the approach to criticality and predicting when criticality will occur based on indications received while the startup is actually in progress. Before the reactor startup is initiated, the operator calculates an estimate of the amount of rod withdrawal that will be necessary to achieve criticality. This process provides an added margin of safety because a large discrepancy between actual and estimated critical rod positions would indicate that the core was not performing as designed.

Depending upon a reactor's design or age, the buildup of **Xenon** within the first several hours following a reactor shutdown may introduce enough negative reactivity to cause the reactor to remain shutdown even with the control rods fully withdrawn. In this situation it is important to be able to predict whether criticality can be achieved, and if criticality cannot be achieved, the startup should not be attempted. For a given set of conditions (such as time since shutdown, temperature, pressure, fuel burnup, **samarium** and **xenon poisoning**) there is only one position of the control rods (and boron concentrations for a reactor with chemical shim) that results in criticality, using the normal rod withdrawal

sequence. Identification of these conditions allows accurate calculation of control rod position at criticality. The calculation of an **estimated critical position** (**ECP)** is simply a mathematical procedure that takes into account all of the changes in factors that significantly affect reactivity that have occurred between the time of reactor shutdown and the time that the reactor is brought critical again.

For most reactor designs, the only factors that change significantly after the reactor is shut down are the average reactor temperature and the concentration of fission product poisons. The reactivities normally considered when calculating an **ECP** include the following:

- **Basic Reactivity of the Core:** The reactivity associated with the critical control rod position for a **xenon-**free core at normal operating temperature. This reactivity varies with the age of the core (amount of fuel burnup).
- **Direct Xenon Reactivity:** The reactivity related to the **Xenon** that was actually present in the core at the time it was shutdown. This reactivity is corrected to allow for **Xenon** decay.
- **Indirect Xenon Reactivity:** The reactivity related to the **Xenon** produced by the decay of **Iodine** that was present in the core at the time of shutdown.
- **Temperature Reactivity:** The reactivity related to the difference between the actual reactor temperature during startup and the normal operating temperature.

To arrive at an **ECP (Estimated Critical Position)** of the control rods, the basic reactivity, direct and indirect **Xenon reactivity**, and **temperature reactivity** are combined algebraically to determine the amount of positive control rod reactivity that must be added by withdrawing control rods to attain criticality. A graph of control rod worth versus rod position is used to determine the estimated critical position.

3.3. CORE POWER DISTRIBUTION

In order to ensure predictable temperatures and uniform depletion of the fuel installed in a reactor, numerous measures are taken to provide an even distribution of flux throughout the power producing section of the reactor. This shaping, or flattening, of the neutron flux is normally achieved through the use of **reflectors** that affect the flux profile across the core, or by the installation of poisons to suppress the neutron flux where desired. The last method, although effective at shaping the flux, is the least desirable since it reduces neutron economy by absorbing the neutrons.

A reactor core is frequently surrounded by a "reflecting" material to reduce the ratio of peak flux to the flux at the edge of the core fuel area. Reflector materials are normally not fissionable, have a high scattering cross section, and have a low absorption cross section. Essentially, for thermal reactors a good moderator is a good reflector. Water, heavy water, beryllium, zirconium, or graphite are commonly used as reflectors. In fast reactor systems, reflectors are not composed of moderating materials because it is desired to keep neutron energy

high. The reflector functions by scattering some of the neutrons, which would have leaked from a bare (unreflected) core, back into the fuel to produce additional fissions.

the general effect of reflection in the thermal reactor system where core power is proportional to the thermal flux is to raise the power density of the core periphery and thus increase the core average power level without changing the peak power. the thermal flux in the reflector may actually be higher than that in the outermost fuel since there are very few absorptions in the reflector.

Varying the fuel enrichment or fuel concentrations in the core radially, axially, or both, can readily be used to control power distribution. Varying fuel concentrations or poison loading for flux shaping is frequently referred to as zoning.

3.4. POWER TILT

A **power tilt**, or **flux tilt**, is a specific type of core power distribution problem. It is a non-symmetrical variation of core power in one quadrant of the core relative to the others. The power in one portion might be suppressed by over-insertion of control rods in that portion of the core, which, for a constant overall power level, results in a relatively higher flux in the remainder of the core. This situation can lead to xenon oscillations, which were previously discussed.

3.5. SHUTDOWN MARGIN

Shutdown margin is the instantaneous amount of reactivity by which a reactor is sub-critical or would be sub-critical from its present condition assuming all control rods are fully inserted except for the single rod with the highest integral worth, which is assumed to be fully withdrawn. Shutdown margin is required to exist at all times, even when the reactor is critical. It is important that there be enough negative reactivity capable of being inserted by the control rods to ensure complete shutdown at all times during the core lifetime.

A shutdown margin in the range of one to five percent reactivity is typically required. The stuck rod criterion refers to the fact that the shutdown margin does not take credit for the insertion of the highest worth control rod. The application of the stuck rod criterion ensures that the failure of a single control rod will not prevent the control rod system from shutting down the reactor.

3.6. OPERATION

During reactor operation, numerous parameters such as temperature, pressure, power level, and flow are continuously monitored and controlled to ensure safe and stable operation of the reactor. The specific effects of variations in these parameters vary greatly depending upon reactor design, but generally the effects for thermal reactors are as follows.

3.6.1. TEMPEARATURE

The most significant effect of a variation in temperature upon reactor operation is the addition of positive or negative reactivity. As previously discussed, reactors are generally designed with **negative temperature coefficients of reactivity**

(moderator and fuel temperature coefficients) as a self-limiting safety feature. A rise in reactor temperature results in the addition of **negative reactivity**.

If the rise in temperature is caused by an increase in reactor power, the negative reactivity addition slows, and eventually turns the increase in reactor power. This is a highly desirable effect because it provides a negative feedback in the event of an undesired power excursion.

Negative temperature coefficients can also be utilized in water cooled and moderated power reactors to allow reactor power to automatically follow energy demands that are placed upon the system. For example, consider a reactor operating at a stable power level with the heat produced being transferred to a heat exchanger for use in an external closed cycle system. If the energy demand in the external system increases, more energy is removed from reactor system causing the temperature of the reactor coolant to decrease. As the reactor temperature decreases, positive reactivity is added and a corresponding increase in reactor power level results. As reactor power increases to a level above the level of the new energy demand, the temperature of the moderator and fuel increases, adding negative reactivity and decreasing reactor power level to near the new level required to maintain system temperature.

Some slight oscillations above and below the new power level occur before steady state conditions are achieved. The final result is that the average temperature of the reactor system is essentially the same as the initial temperature, and the reactor is operating at the new higher required power level. The same inherent stability can be observed as the energy demand on the system is decreased.

3.6.2. PRESSURE

The pressure applied to the reactor system can also affect reactor operation by causing changes in reactivity. The reactivity changes result from changes in the density of the moderator in response to the pressure changes. For example, as the system pressure rises, the moderator density increases and results in greater moderation, less neutron leakage, and therefore the insertion of positive reactivity. A reduction in system pressure results in the addition of negative reactivity. Typically, in pressurized water reactors (**PWR**), the magnitude of this effect is considerably less than that of a change in temperature. In two-phase systems such as boiling water reactors (**BWR**), however, the effects of pressure changes are more noticeable because there is a greater change in moderator density for a given change in system pressure.

3.6.3. POWER LEVEL

A change in reactor power level can result in a change in reactivity if the power level change results in a change in system temperature. The power level at which the reactor is producing enough energy to make up for the energy lost to ambient is commonly referred to as **the point of adding heat.** If a reactor is operating well below the point of adding heat, then variations in power level

produce no measurable variations in temperature. At power levels above the point of adding heat, temperature varies with power level, and the reactivity changes will follow the convention previously described for temperature variations.

The inherent stability and power turning ability of a negative temperature coefficient are ineffective below the point of adding heat. If a power excursion is initiated from a very low power level, power will continue to rise unchecked until the point of adding heat is reached, and the subsequent temperature rise adds negative reactivity to slow, and turn, the rise of reactor power. In this region, reactor safety is provided by automatic reactor shutdown systems and operator action.

3.6.4. FLOW

At low reactor power levels, changing the flow rate of the coolant through the reactor does not result in a measurable reactivity change because fuel and moderator temperatures and the fraction of steam voids occurring in the core are not changed appreciably. When the flow rate is varied, however, the change in temperature that occurs across the core (outlet versus inlet temperature) will vary inversely with the flow rate. At higher power levels, on liquid cooled systems, increasing flow will lower fuel and coolant temperatures slightly, resulting in a small positive reactivity insertion. A positive reactivity addition also occurs when flow is increased in a two-phase (steam-water) cooled system. Increasing the flow rate decreases the fraction of steam voids in the coolant and results in a positive reactivity addition. This property of the moderator in a two-phase system is used extensively in commercial BWRs. Normal power variations required to follow load changes on BWRs are achieved by varying the coolant/moderator flow rate.

3.7. CORE BURNUP

As a reactor is operated, atoms of fuel are constantly consumed, resulting in the slow depletion of the fuel frequently referred to as **core burnup**. There are several major effects of this fuel depletion. The first, and most obvious, effect of the fuel burnup is that the control rods must be withdrawn or chemical shim concentration reduced to compensate for the negative reactivity effect of this burnup.

Some reactor designs incorporate the use of supplemental burnable poisons in addition to the control rods to compensate for the reactivity associated with excess fuel in a new core. These fixed burnable poisons burn out at a rate that approximates the burnout of the fuel and they reduce the amount of control rod movement necessary to compensate for fuel depletion early in core life.

As control rods are withdrawn to compensate for fuel depletion, the effective size of the reactor is increased. By increasing the effective size of the reactor, the probability that a neutron slows down and is absorbed while it is still in the reactor is also increased. Therefore, neutron leakage decreases as the effective reactor size is increased.

The magnitude of the moderator negative temperature coefficient is determined in part by the change in neutron leakage that occurs as the result of a change in moderator temperature. Since the fraction of neutrons leaking out is less with the larger core, a given temperature change will have less of an effect on the leakage. Therefore, the magnitude of the moderator negative temperature coefficient decreases with fuel burnup.

There is also another effect that is a consideration only on reactors that use dissolved boron in the moderator (chemical shim). As the fuel is burned up, the dissolved boron in the moderator is slowly removed (concentration diluted) to compensate for the negative reactivity effects of fuel burnup. This action results in a larger (more negative) moderator temperature coefficient of reactivity in a reactor using chemical shim. This is due to the fact that when water density is decreased by rising moderator temperature in a reactor with a negative temperature coefficient, it results in a negative reactivity addition because some moderator is forced out of the core. With a coolant containing dissolved poison, this density decrease also results in some poison being forced out of the core, which is a positive reactivity addition, thereby reducing the magnitude of the negative reactivity added by the temperature increase. Because as fuel burnup increases the concentration of boron is slowly lowered, the positive reactivity added by the above poison removal process is lessened, and this results in a larger negative temperature coefficient of reactivity. The following effect of fuel burnup is most predominant in a reactor with a large concentration of uranium-238. As the fission process occurs in a thermal reactor with low or medium enrichment, there is some conversion of uranium-238 into plutonium-239. Near the end of core life in certain reactors, the power contribution from the fission of plutonium-239 may be comparable to that from the fission of uranium-235. The value of the delayed neutron fraction (b) for uranium-235 is 0.0064 and for plutonium-239 is 0.0021. Consequently, as core burnup progresses, the effective delayed neutron fraction for the fuel decreases appreciably. It follows then that the amount of reactivity insertion needed to produce a given reactor period decreases with burnup of the fuel.

3.8. SHUTDOWN

A reactor is considered to be shut down when it is sub-critical and sufficient shutdown reactivity exists so there is no immediate probability of regaining criticality. Shutdown is normally accomplished by insertion of some (or all) of the control rods, or by introduction of soluble neutron poison into the reactor coolant.

The rate at which the reactor fission rate decays immediately following shutdown is similar for all reactors provided a large amount of negative reactivity is inserted. After a large negative reactivity addition the neutron level undergoes a rapid decrease of about two decades (prompt drop) until it is at the level of production of delayed neutrons. Then the neutron level slowly drops off as the delayed neutron precursors decay, and in a short while only the longest-lived precursor remains in any significant amount.

This precursor determines the final rate of decrease in reactor power until the neutron flux reaches the steady state level corresponding to the sub-critical multiplication of the neutron source. The half-life of the longest lived delayed neutron precursor results in a reactor period of around -80 seconds or a startup rate of -1/3 **DPM** for most reactors after a reactor shutdown. One noticeable exception to this is a heavy water reactor. In a heavy water reactor, the photo-neutron source is extremely large after shutdown due to the amount of deuterium in the moderator and the large number of high energy gammas from short-lived fission product decay. The photo-neutron source is large enough to have a significant impact on neutron population immediately after shutdown. The photo-neutron source has the result of flux levels decreasing more slowly so that a heavy water reactor will have a significantly larger negative reactor period after a shutdown. Throughout the process of reactor shutdown the nuclear instrumentation is closely monitored to observe that reactor neutron population is decreasing as expected, and that the instrumentation is functioning properly to provide continuous indication of neutron population. Instrumentation is observed for proper overlap between ranges, comparable indication between multiple instrument channels, and proper decay rate of neutron population.

A distinction should be made between indicated reactor power level after shutdown and the actual thermal power level. The indicated reactor power level is the power produced directly from fission in the reactor core, but the actual thermal power drops more slowly due to decay heat production as previously discussed. Decay heat, although approximately 5 to 6% of the steady state reactor power prior to shutdown, diminishes to less than 1% of the pre-shutdown power level after about one hour. After a reactor is shutdown, provisions are provided for the removal of decay heat. If the reactor is to be shut down for only a short time, operating temperature is normally maintained. If the shutdown period will be lengthy or involves functions requiring cooldown of the reactor, the reactor temperature can be lowered by a number of methods. The methods for actually conducting cooldown of the reactor vary depending on plant design, but in all cases limitations are imposed on the maximum rate at which the reactor systems may be cooled. These limits are provided to reduce the stress applied to system materials, thereby reducing the possibility of stress induced failure. Although a reactor is shut down, it must be continuously monitored to ensure the safety of the reactor. Automatic monitoring systems are employed to continuously collect and assess the data provided by remote sensors. It is ultimately the operator who must ensure the safety of the reactor.

3.9. DECAY HEAT

About 7 percent of the 200 MeV produced by an average fission is released at some time after the instant of fission. This energy comes from the decay of the fission products. When a reactor is shut down, fission essentially ceases, but decay energy is still being produced. The energy produced after shutdown is referred to as decay heat. The amount of decay heat production after shutdown is directly influenced by the power history of the reactor prior to shutdown. A reactor operated at full power for 3 to 4 days prior to shutdown has much higher

decay heat generation than a reactor operated at low power for the same period. The decay heat produced by a reactor shutdown from full power is initially equivalent to about 5 to 6% of the thermal rating of the reactor. This decay heat generation rate diminishes to less than 1% approximately one hour after shutdown. However, even at these low levels, the amount of heat generated requires the continued removal of heat for an appreciable time after shutdown. Decay heat is a long-term consideration and impacts spent fuel handling, reprocessing, waste management, and reactor safety.

END OF CHAPTER FOUR: REACTOR OPERATIONS

ASCENTRUST LLC
The POWER of ENGINEERING

APPENDIX
INTRODUCTION TO
NUCLEAR POWER AND NTPBMR TECHNOLOGY

SECTION B: Health and Safety in Nuclear Systems

ASCENTRUST LLC
The POWER of ENGINEERING

HEALTH AND SAFETY IN NUCLEAR SYSTEMS
AN INTRODUCTION TO NUCLEAR POWER PRODUCTION
FOR
STAKEHOLDERS AND CONSULTANTS
OF
NUCLEAR TECHNOLOGY PEBBLE BED MODULAR REACTOR
(NTPBMR) PROJECT

HEALTH AND SAFETY ASPECTS OF NUCLEAR POWER PLANTS

Health and Safety Aspects of Nuclear Power Plants

SECTION 0: INTRODUCTION

0.1. PREAMBLE

This document started off as a short introductory paper on radiation and was part of a large group of introductory papers prepared by the Principal Engineer of the **Nuclear Technology Pebble Bed Modular Reactor (NTPBMR) Project** for the primary stakeholders to the project. This document will attempt to present an integrated, positive view of the safety characteristic of existing nuclear power facilities and make the case for the inclusion of the **NTPBMR** in the energy production of the twenty-first century.

This document is a compilation of a series of white papers undertaken by the Principal Engineer covering many aspects of the production of electrical power with the use of nuclear energy. The objective of these white papers was to provide a knowledge base consisting of factual, up-to-date information on aspects of nuclear power that are needed in order to make the case for the safety of the existing nuclear power production fleet in the U.S. and overseas.

The method used was a systematic and selective collection, condensation, and presentation of existing information for public dissemination and for the use by generalist as well as by the technical staff associated with our stakeholders and funders. These papers were not intended to provide design guidelines but rather as background sources to support detailed or specialized analyses and reviews of our **NTPBMR** nuclear power project.

The scope of this analysis includes some of the topics and issues deemed importance to the funding and licensing of the **NTPBMR Project.**. Prime among these issues is the safety of nuclear reactors, a subject of intense controversy and debate among technical experts, political groups, and the public at large. This issue, together with those covered in the other papers, was in fact responsible for the near halt in the development of nuclear power in most countries.

Since **AscenTrust, LLC. (The Company)** is now actively involved in the funding of the **NTPBMR Prototype Project,** we are engaged in discussions with an Investment Group and several countries that have a desire to include a nuclear component in their power system. This document was undertaken to familiarize the staff of our stakeholders with the scientific facts, and provide answers to the questions which are often posed:

1. Why nuclear Power?
2. How safe is the operation of your Nuclear Power Technology?
3. What are the radiation hazards of nuclear power?
4. What about Three Mile Island, Chernobyl and Fukushima?
5. How does your reactor technology mitigate the **LOCA** (Loss of Coolant Accident)
6. What is your solution to the nuclear waste problem?
7. What should be done about nuclear proliferation?

This document will provide the reader with a fairly complete answer to the first four question stated above. Section Zero contains the preamble which includes a description of the history of this document. Section Zero contains an introductions to the magnitude of the Nuclear Power generation industry in the world.

Section One includes a description of the three reactor concepts which were involved in nuclear events. The two most important **Light Water reactor technologies** are the pressure water reactor and the boiling water reactor. The third reactor technology which was involved in a major reactor incident is the Russian RBMK. Finally section one also includes an introduction to the Pebble Bed Reactor Technology.

Section Two begins with an elucidation of the public concerns regarding the production of energy with the use of nuclear fuel and then introduces the most important elements of the nature of radiation. Section two then addresses the current state of scientific understanding of the health effects of radiation, both somatic and genetic, originating from routine as well as accidental releases of radioactivity from nuclear power plant operations.

Section Three addresses routine emissions of radioactive material that result from normal plant operations. Operational data shows that the average radiation dose to the general public resulting from normal operation of nuclear power plants is a small fraction of the average individual dose received from natural sources of radiation in the environment, and that average doses resulting from the operation of nuclear power plants are a small fraction of the variation in natural background radiation dose received in different geographic regions of the world. Recent epidemiological studies have produced no statistically significant evidence of an excess rate of cancer incidence as a result of living near nuclear installations; nor have they produced statistically significant evidence of adverse health effects among nuclear power plant workers whose occupational doses (received in normal plant operations) can often exceed natural background radiation levels.

(It must be noted that although these statements are endorsed by the consensus opinion of the medical, scientific and technical communities and by official country and international bodies, there exist dissenting reports and views on the matter.)

Section Four reviews the basic principles of nuclear power plant safety, and describes engineering systems that are designed to prevent or mitigate nuclear accidents in typical pressurized water reactors operating in the U.S. Within this context, the range of possible accidents that could occur in a power plant is discussed together with goals for power plant safety that have been established by regulatory authorities in the United States.

Three major accidents have occurred in the history of commercial nuclear power, **Three Mile Island** (TMI), **Chernobyl** and **Fukushima** In the first accident which occurred in 1979, the reactor core suffered major damage and enormous loss of investment and cleanup cost ensued. However, the releases of radioactivity to the environment were

minor (owing to the effectiveness of the containment structure) and no measurable direct health effects were observed other than psychologically induced illnesses.

In the accident which occurred at Chernobyl of the Soviet Union in April of 1986, 31 persons, principally from the firefighting team, died within a month from acute exposure of radiation, large scale evacuation of population was necessitated, and large land contamination resulted. The long-term effects of radiation exposure resulting in increased cancer incidence is a matter of intense debate. The number of cancer deaths in the Soviet Union due to such exposure is estimated to be in the range of 10,000 - 40,000 during the period of the next 30 years but may not be statistically observable among the 70,000,000 deaths from naturally occurring cancer expected in the same time period. The design of the Chernobyl reactor (15 of which were operational at the time of the accident) was not exported by the Soviet Union and is no longer used for future units. About twenty units are still being operated in the USSR.

Section Five reviews the causes and consequences of the **TMI, Chernobyl** and **Fukushima** accidents, as well as those of an accident that occurred at a weapons material production plant in the United Kingdom in 1957.

0.2. CURRENT NUCLEAR POWER ANALYSIS

The American public has become aware that a sharp increase in the number of Nuclear Power Plants is the only viable solution to the global carbon dioxide emission crisis. Interest in nuclear power has recently increased because of the **Global Warming** issue. Nuclear energy is now a mature industry. In 2007, nuclear power provided 16.8% of total worldwide electricity generation, and is expected to increase that contribution in the years to come.

A. Global Nuclear Power Production: Today, there are some 439 nuclear power reactors operating in 30 countries, with a combined capacity of about 370 billion watts of electricity (GWe), which produce only about 16.8% of the world's electricity but do so without directly producing any greenhouse gases.

B. According to the US **Dept of Energy** (**DOE)** projections, world net electricity generation will nearly double from about 17.3 trillion kilowatt hours (kWh) in 2005 to 24.4 trillion kWh in 2015 to 33.3 trillion kWh by 2030

C. Globally, power generation emits nearly 10 billion tons of Carbon Dioxide (**CO_2)** per year. The United States, with over 8,000 of the more than 50,000 power plants worldwide, accounts for about 25 percent of that total or 2.8 billion tons.

D. More than 67% of electricity generated worldwide is produced by burning fossil fuels, primarily coal (42%), which is the most carbon-intensive fuel. In addition to carbon dioxide, burning fossil fuels also produces oxides of sulfur and nitrogen, and other air pollutants that contribute to problems like acid rain.

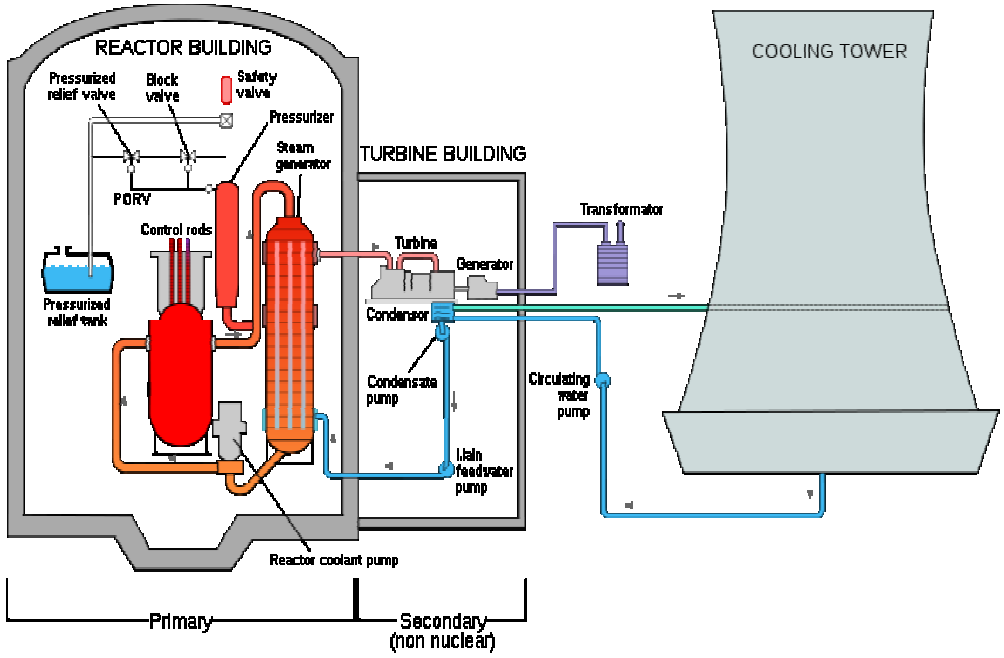
0.3. CURRENT NUCLEAR POWER SYSTEMS

The large majority of the nuclear power plants in the world today use light water as the coolant and are known generically as **light water reactors** (**LWR's**). They consist **PWR** and **BWR.** We will also be including a discussion of the **RBMK** because of the effect on the nuclear industry of the accident which occurred at Chernobyl. Finally we will also be including a detailed discussion on the **Pebble Bed Modular Reactor.**

SECTION ONE: DETAILS OF CURRENT POWER SYSTEMS

1.1. PRESSURIZED WATER REACTOR

SCHEMATIC OF THE PRESSURE WATER REACTOR



1.1.1. INTRODUCTION

Pressurized water reactors (**PWR**s) comprise a majority of all western nuclear power plants. In a **PWR** the primary coolant (superheated water) is pumped under high pressure to the reactor core, then the heated water transfers thermal energy to a steam generator. The Nuclear Reactors at Three Mile Island are all **PWR,** types of reactors.

1.1.2. PRESSURIZED WATER REACTOR OPERATION

1. The reactor core transfers the fission energy, primarily kinetic energy created by recoil of the fission fragments in the fuel rods into thermal energy of the water which is both the moderator and the cooling agent in a Light Water Reactor

2. Pressurized-water in the primary coolant loop carries the heat to the steam generator.
3. Inside the steam generator heat from the primary coolant loop vaporizes the water in the secondary loop producing steam.
4. The steam line directs the steam to the main turbine causing it to turn the turbine which is connected to the generator to create electrical power.
5. The unused steam is condensed into water.
6. The resulting water is pumped out of the condenser with a series of pumps, reheated and pumped back to the steam generator.

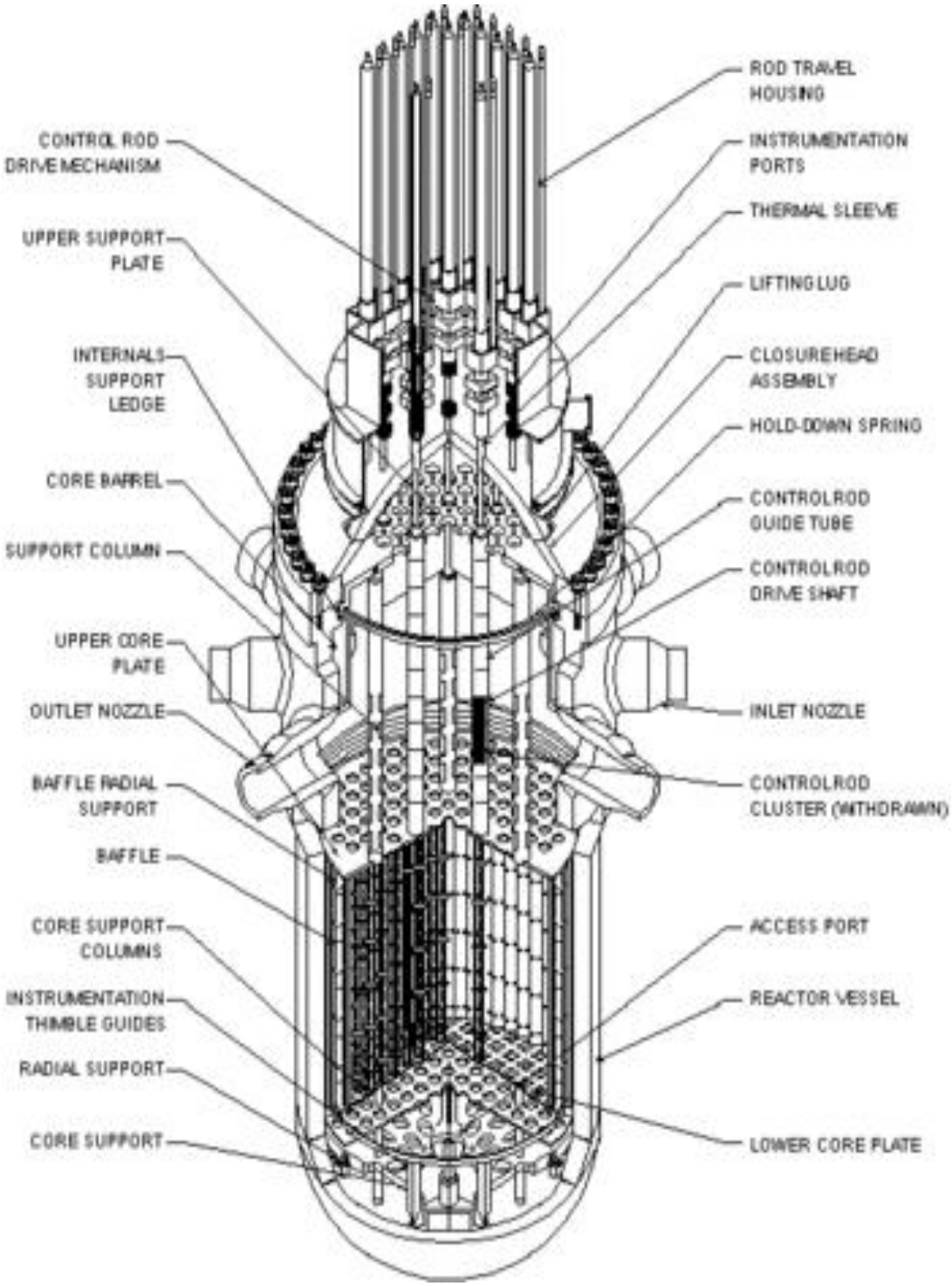
1.1.3. NUCLEAR STEAM SUPPLY SYSTEM:

Nuclear fuel in the reactor vessel is engaged in a fission chain reaction, which produces heat, heating the water in the primary coolant loop by thermal conduction through the fuel cladding. The hot primary coolant is pumped into a heat exchanger called steam generator, where heat is transferred across a set of tubes to the lower pressure secondary coolant, which evaporates to pressurized steam. The transfer of heat is accomplished without mixing the two fluids, which is desirable since the primary coolant might become radioactive.

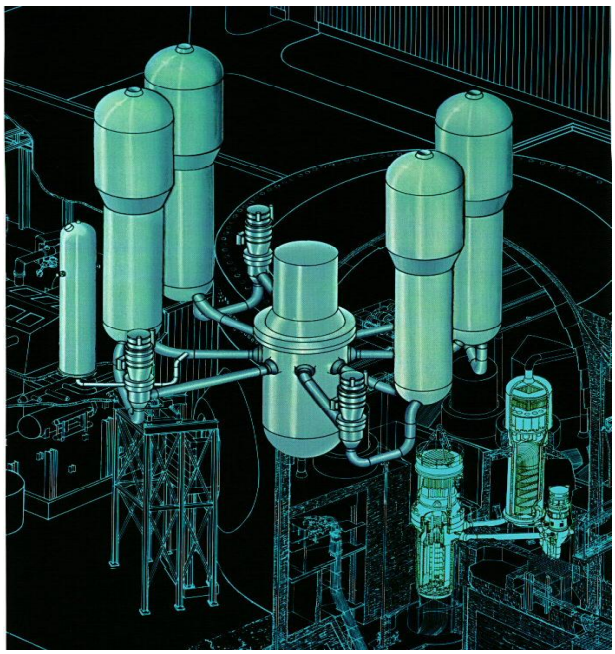
Two things are characteristic for the **pressurized water reactor** (**PWR**) when compared with other reactor types: coolant loop separation from the steam system and pressure inside the primary coolant loop. In a **PWR**, there are two separate coolant loops (primary and secondary), which are both filled with demineralized/deionized water. The pressure in the primary coolant loop is typically 15–16 megapascals (153 atmospheres, 2,250 psig,150–160 bar), which is notably higher than in other nuclear reactors. As an effect of this, only localized boiling occurs and steam will recondense promptly in the bulk fluid.

Pressure in the primary circuit is maintained by a Pressurizer, a separate vessel that is connected to the primary circuit and partially filled with water which is heated to the saturation temperature (boiling point) for the desired pressure by submerged electrical heaters. To achieve a pressure of 155 bar, the pressurizer temperature is maintained at 345 °C, which gives a sub-cooling margin (the difference between the pressurizer temperature and the highest temperature in the reactor core) of 30 °C. Thermal transients in the reactor coolant system result in large swings in pressurizer liquid volume, total pressurizer volume is designed around absorbing these transients without uncovering the heaters or emptying the pressurizer. Pressure transients in the primary coolant system manifest as temperature transients in the pressurizer and are controlled through the use of automatic heaters and water spray, which raise and lower pressurizer temperature, respectively. To achieve maximum heat transfer, the primary circuit temperature, pressure and flow rate are arranged such that subcooled nucleate boiling takes place as the coolant passes over the nuclear fuel rods.

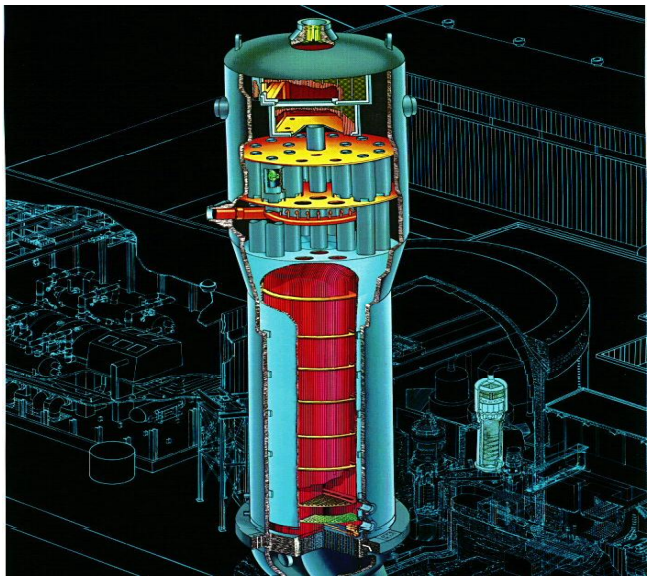
PRESSURIZED WATER REACTOR: REACTOR PRESSURE VESSEL



PRESSURIZED WATER REACTOR: REACTOR PRESSURE VESSEL CONFIGURATION OF REACTOR AND STEAM GENERATORS IN PRESSURIZED WATER REACTOR



STEAM GENERTOR IN WESTINGHOUSE PRESURIZED WATER REACTOR



The coolant is pumped around the primary circuit by powerful pumps, which can consume up to 6 MW each. After picking up heat as it passes through the reactor core, the primary coolant transfers heat in a steam generator to water in a lower pressure secondary circuit, evaporating the secondary coolant to saturated steam — in most designs 6.2 MPa (60 atm, 900 psia), 275 °C (530 °F) — for use in the steam turbine. The cooled primary coolant is then returned to the reactor vessel to be heated again.

Pressurized water reactors, like all thermal reactor designs, require the fast fission neutrons to be slowed down (a process called moderation or thermalization) in order to interact with the nuclear fuel and sustain the chain reaction. In **PWR**s the coolant water is used as a moderator by letting the neutrons undergo multiple collisions with light hydrogen atoms in the water, losing speed in the process. This "moderating" of neutrons will happen more often when the water is denser (more collisions will occur). The use of water as a moderator is an important safety feature of PWRs, as any increase in temperature causes the water to expand and become less dense; thereby reducing the extent to which neutrons are slowed down and hence reducing the reactivity in the reactor. Therefore, if reactivity increases beyond normal, the reduced moderation of neutrons will cause the chain reaction to slow down, producing less heat. This property, known as the negative temperature coefficient of reactivity, makes PWR reactors very stable.

PWRs are designed to be maintained in an undermoderated state, meaning that there is room for increased water volume or density to further increase moderation, because if moderation were near saturation, then a reduction in density of the moderator/coolant could reduce neutron absorption significantly while reducing moderation only slightly, making the void coefficient positive. Also, light water is actually a somewhat stronger moderator of neutrons than heavy water, though heavy water's neutron absorption is much lower. Because of these two facts, light water reactors have a relatively small moderator volume and therefore have compact cores. One next generation design, the supercritical water reactor, is even less moderated. A less moderated neutron energy spectrum does worsen the capture/fission ratio for ^{235}U and especially ^{239}Pu, meaning that more fissile nuclei fail to fission on neutron absorption and instead capture the neutron to become a heavier non-fissile isotope, wasting one or more neutrons and increasing accumulation of heavy transuranic actinides, some of which have long half-lives.

In a nuclear power station, the pressurized steam is fed through a steam turbine which drives an electrical generator connected to the electric grid for distribution. After passing through the turbine the secondary coolant (water-steam mixture) is cooled down and condensed in a condenser. The condenser converts the steam to a liquid so that it can be pumped back into the steam generator, and maintains a vacuum at the turbine outlet so that the pressure drop across the turbine, and hence the energy extracted from the steam, is maximized. Before being fed into the steam generator, the condensed steam (referred to as feed-water) is sometimes preheated in order to minimize thermal shock.

1.1.4. CONTROL SYSTEM FOR PRESSURIZED WATER REACTOR

Reactor power is controlled via two methods: by inserting or withdrawing control rods and by adjusting the concentration of Boron (called a chemical shim) in the primary cooling circuit. Positioning (withdrawing or inserting) control rods is the normal method for controlling power when starting up a **PWR**. As control rods are withdrawn, neutron absorption decreases in the control material and increases in the fuel, so reactor power increases. As control rods are inserted, neutron absorption increases in the control material and decreases in the fuel, so reactor power decreases.

In **PWR**s reactor power can be viewed as following steam (turbine) demand due to the reactivity feedback of the temperature change caused by increased or decreased steam flow. Boron and control rods are used to maintain primary system temperature at the desired point. In order to decrease power, the operator throttles shut turbine inlet valves. This would result in less steam being drawn from the steam generators. This results in the primary loop increasing in temperature. Reactivity adjustment to maintain 100% power as the fuel is burned up in most commercial **PWR**s is normally achieved by varying the concentration of boric acid dissolved in the primary reactor coolant. Boron readily absorbs neutrons and increasing or decreasing its concentration in the reactor coolant will therefore affect the neutron activity correspondingly. An entire control system involving high pressure pumps (usually called the charging and letdown system) is required to remove water from the high pressure primary loop and re-inject the water back in with differing concentrations of boric acid. The reactor control rods, inserted through the reactor vessel head directly into the fuel bundles, are moved for the following reasons:

- To start up the reactor.
- To shut down the reactor.
- To accommodate short term transients such as changes to load on the turbine.

The control rods can also be used:

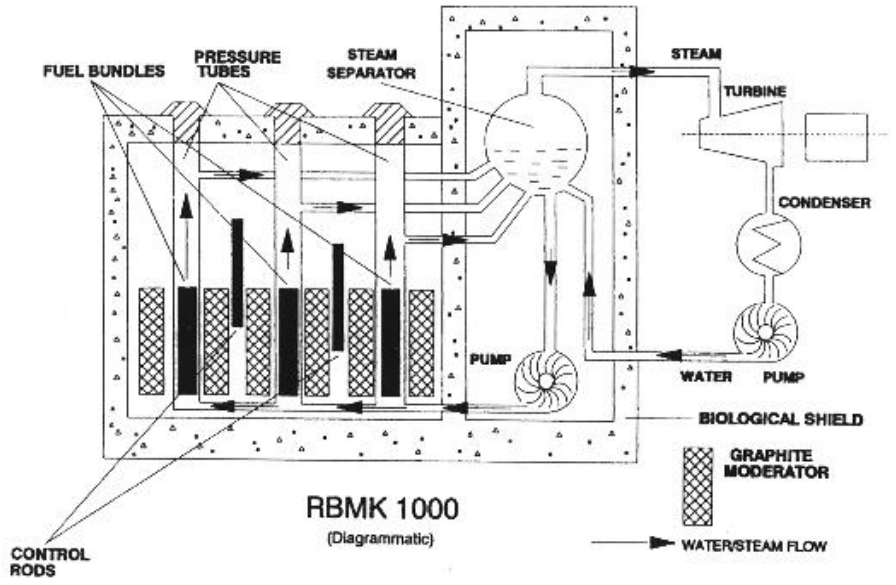
- To compensate for nuclear poison inventory.
- To compensate for nuclear fuel depletion.

1.2. THE RBMK NUCLEAR REACTOR

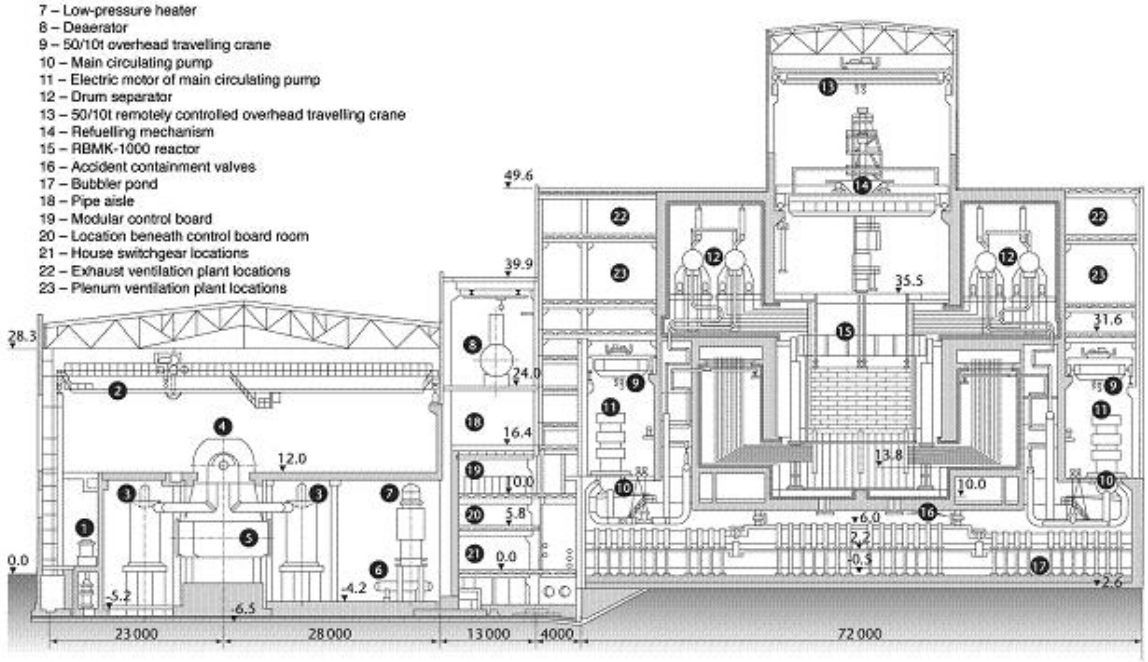
The Soviet-designed RBMK (*reaktor bolshoy moshchnosty kanalny*, high-power channel reactor) is a pressurized water-cooled reactor with individual fuel channels and using graphite as its moderator. It is very different from most other power reactor designs as it derived from a design principally for plutonium production and was intended and used in Russia for both plutonium and power production.

The combination of graphite moderator and water coolant is found in no other power reactors in the world. As the Chernobyl accident showed, several of the RBMK's design characteristics – in particular, the control rod design and a positive void coefficient – were unsafe.

GRAPHIC REPRESENTATION OF THE RBMK



Cross-sectional view of the RBMK-1000 main building
Key:
1 – First-stage condensate pump
2 – 125/20t overhead travelling crane
3 – Separator-steam superheater
4 – K-500-65/3000 steam turbine
5 – Condenser
6 – Additional cooler
7 – Low-pressure heater
8 – Deaerator
9 – 50/10t overhead travelling crane
10 – Main circulating pump
11 – Electric motor of main circulating pump
12 – Drum separator
13 – 50/10t remotely controlled overhead travelling crane
14 – Refuelling mechanism
15 – RBMK-1000 reactor
16 – Accident containment valves
17 – Bubbler pond
18 – Pipe aisle
19 – Modular control board
20 – Location beneath control board room
21 – House switchgear locations
22 – Exhaust ventilation plant locations
23 – Plenum ventilation plant locations



The RBMK is an unusual reactor design, one of two to emerge in the Soviet Union in the 1970s. The design had several shortcomings, and was the design involved in the 1986 Chernobyl disaster. Major modifications have been made to RBMK reactors still operating.

1.2.1. FEATURES OF THE RBMK

1. **Fuel:** Pellets of slightly-enriched uranium oxide are enclosed in a zircaloy tube 3.65m long, forming a fuel rod. A set of 18 fuel rods is arranged cylindrically in a carriage to form a fuel assembly. Two of these end on end occupy each pressure tube.
2. **Pressure tubes:** Within the reactor each fuel assembly is positioned in its own vertical pressure tube or channel about 7 m long. Each channel is individually cooled by pressurized water which is allowed to boil in the tube and emerges at about 290°C.
3. **Refueling**: When fuel channels are isolated, the fuel assemblies can be lifted into and out of the reactor, allowing fuel replenishment while the reactor is in operation.
4. **Graphite moderator:** A series of graphite blocks surround, and hence separate, the pressure tubes. They act as a moderator to slow down the neutrons released during fission so that a continuous fission chain reaction can be maintained. Heat conduction between the blocks is enhanced by a mixture of helium and nitrogen gas.
5. **Control rods:** Boron carbide control rods absorb neutrons to control the rate of fission. A few short rods, inserted upwards from the bottom of the core, even the distribution of power across the reactor. The main control rods are inserted from the top down and provide automatic, manual, or emergency control. The automatic rods are regulated by feedback from in-core detectors. If there is a deviation from normal operating parameters (*e.g.* increased reactor power level), the rods can be dropped into the core to reduce or stop reactor activity. A number of rods remain in the core during operation.
6. **Coolant:** Two separate water coolant loops each with four pumps circulate water through the pressure tubes to remove most of the heat from fission. There is also an emergency core cooling system designed to come into operation if either coolant circuit is interrupted.
7. **Steam separator:** Each of the two loops has two steam drums, or separators, where steam from the heated coolant is fed to the turbine to produce electricity in the generator (each loop has a turbo-generator associated with it). The steam is then condensed and fed back into the circulating coolant.
8. **Containment:** There is no secure containment in the sense accepted in the West. The reactor core is located in a reinforced concrete lined cavity that acts

as a radiation shield. The core sits on a heavy steel plate, with a 1000 tonne steel cover plate on the top. The extensions of the fuel channels penetrate the lower plate and the cover plate and are welded to each. The steam separators of the coolant systems are housed in their own concrete shields.

9. **Positive void coefficient:** The term 'positive void coefficient' is often associated with the RBMK reactors. Reactors cooled by boiling water will contain a certain amount of steam in the core. Because water is both a more efficient coolant and a more effective neutron absorber than steam, a change in the proportion of steam bubbles, or 'voids', in the coolant will result in a change in core reactivity. The ratio of these changes is termed the void coefficient of reactivity. When the void coefficient is negative, an increase in steam will lead to a decrease in reactivity. In those reactors where the same water circuit acts as both moderator and coolant, excess steam generation reduces the slowing of neutrons necessary to sustain the nuclear chain reaction. This leads to a reduction in power, and is a basic safety feature of most Western reactors. In reactor designs where the moderator and coolant are of different materials, excess steam reduces the cooling of the reactor, but as the moderator remains intact the nuclear chain reaction continues. In some of these reactors, most notably the RBMK, the neutron absorbing properties of the cooling water are a significant factor in the operating characteristics. In such cases, the reduction in neutron absorption as a result of steam production, and the consequent presence of extra free neutrons, enhances the chain reaction. This leads to an increase in the reactivity of the system. The void coefficient is only one contributor to the overall power coefficient of reactivity, but in RBMK reactors it is the dominant component, reflecting a high degree of dependence of reactivity on the steam content of the core.

   At the time of the accident at Chernobyl, the void coefficient of reactivity was so positive that it overwhelmed the other components of the power coefficient, and the power coefficient itself became positive. When the power began to increase, more steam was produced, which in turn led to an increase in power. The additional heat resulting from the increase in power raised the temperature in the cooling circuit and more steam was produced. More steam means less cooling and less neutron absorption, resulting in a rapid increase in power to around 100 times the reactor's rated capacity. The value of the void coefficient is largely determined by the configuration of the reactor core. In RBMK reactors, an important factor affecting this is the operating reactivity margin.

10. **Operating reactivity margin:** Although the definition is not precise, the operating reactivity margin (ORM) is essentially the number of 'equivalent' control rods of nominal worth remaining in the reactor core. The operators at Chernobyl seemed to believe that safety criteria would be met so long as the lower limit for the ORM of 15 equivalent rods was adhered to, regardless of the actual configuration of the core. The operators were not aware of the 'positive scram'

effect where, following a scram signal, the initial entry of the control rods actually added reactivity to the lower region of the core (see section below on *Post accident changes to the RBMK*). The ORM could have an extreme effect on the void coefficient of reactivity, as was the case for the core configuration of Chernobyl 4 in the run-up to the accident. Unacceptably large void coefficients were prevented for initial cores by increasing fuel enrichment levels, with the excess reactivity balanced by fixed absorbers. However, with increasing fuel burn-up, these absorbers could be removed to maintain the fuel irradiation levels - shifting the void coefficient in the positive direction and increasing the sensitivity of the coefficient to the extent of insertion of the control and protection rods.

1.2.2. POST ACCIDENT CHANGES TO THE RBMK

After the accident at Chernobyl, several measures were taken to improve the safety of RBMK plants. All operating RBMK reactors in the former Soviet Union had the following changes implemented to improve operating safety:

- Reduction of the void coefficient of reactivity.
- Improvement of the response efficiency of the emergency protection system.
- Introduction of calculation programs to provide an indication of the value of the **operating reactivity margin** (ORM, *i.e.* the effective number of control rods remaining in the core) in the control room.
- Prevention of the emergency safety systems from being bypassed while the reactor is operating.
- In order to ensure adequate sub-cooling at the core inlet, the avoidance of modes of operation that cause a reduction in the departure from nuclear boiling (DNB) ratio of the coolant at the reactor inlet.

Measures to reduce the void coefficient of reactivity were carried out by:

- The installation of 80-90 additional fixed absorbers in the core to inhibit operation at low power.
- Increasing the ORM from 26-30 rods (in steady state operational mode) to 43-48.
- An increase in fuel enrichment from 2% to 2.4%.

The increase in the number of fixed absorbers and the ORM reduced the value of the void coefficient of reactivity to +ß (where ß is the effective delayed neutron fraction). The additional absorbers require the use of higher fuel enrichment to compensate for the increased neutron absorption. The efficiency and speed of the emergency protection system was improved by implementing three independent retrofitting operations:

- Retrofitting of control rods with a design that does not give rise to water columns at the bottom of the channels.
- Scram (shut down) rod insertion time cut from 18 to 12 seconds.

- The installation of a fast-acting emergency protection (FAEP) system.

1.2.3. THE FAST-ACTING EMERGENCY PROTECTION (FAEP) SYSTEM

One of the most important post-accident changes to the RBMK was the retrofitting of the control rods. A graphite 'displacer' is attached to each end of the length of absorber of each rod (except for 12 rods used in automatic control). The lower displacer prevents coolant water from entering the space vacated as the rod is withdrawn, thus augmenting the reactivity worth of the rod. However, the dimensions of the rod and displacers were such that, with the rod fully withdrawn, the 4.5 m displacer sat centrally within the fuelled region of the core with 1.25 m of water at either end. On a scram signal, as the rod falls, the water at the lower part of the channel is replaced by the bottom of the graphite displacer, thus initially adding reactivity to the bottom part of the core. Following the Chernobyl accident, this 'positive scram' effect was mitigated by retrofitting the control rods so that, with the rods fully retracted, there would not be a region containing water at the bottom of the core.

The FAEP system was designed so that 24 emergency protection control rods would insert negative reactivity of at least 2ß in under 2.5 seconds. Tests in 1987-'88 at the Ignalina and Leningrad plants (the first RBMKs to be fitted with the new FAEP system) confirmed these characteristics.

In addition to the above changes, several further modifications have been implemented at RBMK plants. These measures consist of:

- Replacement of the fuel channels at all units (except Smolensk 3).
- Replacement of the group distribution headers and addition of check valves.
- Improvements to the emergency core cooling systems.
- Improvements of the reactor cavity over-pressure protection systems.
- Replacement of the SKALA process computer.

1.2.4. OPERATING RBMK PLANTS

There are currently 11 operating RBMKs in the world, all of which are in Russia. One more was under construction in Russia (Kursk 5), but it is unlikely to be completed. All operating RBMKs began operation between 1973 (Leningrad 1) and 1990 (Smolensk 3). There are currently three distinct generations of reactors having significant differences with respect to their safety design features:

The four first-generation units are Leningrad 1 and 2, and Kursk 1 and 2. They were designed and brought on line in the early-to-mid 1970s, before new standards on the design and construction of nuclear power plants, the OPB-82 General Safety Provisions, were introduced in the Soviet Union in 1982.

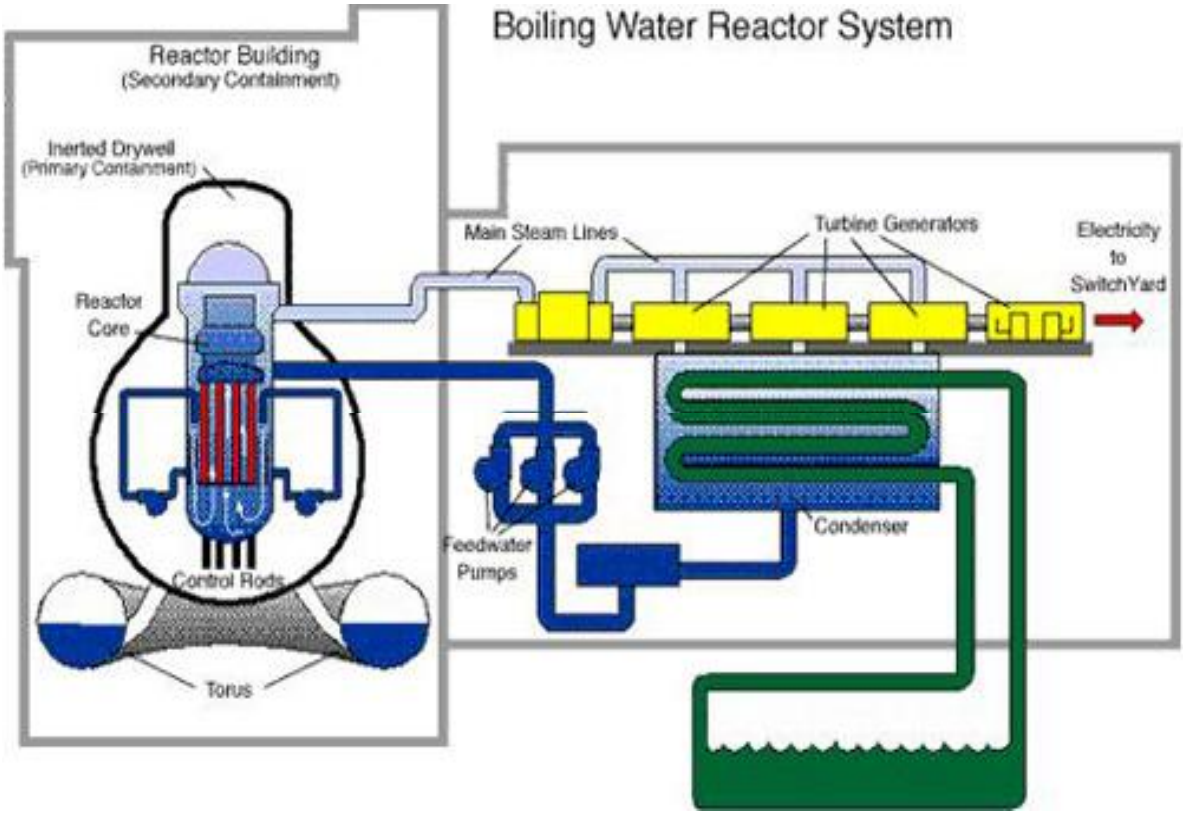
Second-generation RBMKs, brought on line since the late 1970s and early 1980s include Leningrad 3 and 4; Kursk 3, and 4; Ignalina 1 (now closed); and Smolensk 1 and

2. Ignalina 2 (now closed) had safety features beyond those of other second generation units. These units conform to the OPB-82 standards.

1.3. BOILING WATER REACTOR

1.3.1. INTRODUCTION

The **Boiling Water Reactor (BWR)** uses demineralized water (light water) as a coolant and neutron moderator. Heat is produced by nuclear fission in the reactor core, and this causes the cooling water to boil, producing steam. The steam is directly used to drive a turbine, after which it is cooled in a condenser and converted back to liquid water. This water is then returned to the reactor core, completing the loop. The cooling water is maintained at about 75 atm (7.6 MPa, 1000-1100 psi) so that it boils in the core at about 285°C (550°F). In comparison, there is no significant boiling allowed in a **PWR** because of the high pressure maintained in its primary loop - approximately 158 atm (16 MPa, 2300 psi). The recent nuclear event in Fukushima, Japan occurred in a General Electric Mark I Nuclear Power Plant



BOILING WATER REACTOR SCHEMATIC

1.3.2. BOILING WATER REACTOR OPERATIONS

1. The reactor core transfers the fission energy, primarily kinetic energy created by recoil of the fission fragments in the fuel rods into thermal energy of the water which is both the moderator and the cooling agent in a Light Water Reactor.
2. A steam-water mixture is produced when very pure water (reactor coolant) move upward through the core absorbing heat.
3. The steam-water mixture leaves the top of the core and enters the two stages of water seperation where the water droplets are removed and the steam is dried before entering the steam lines.
4. The steam line directs the steam to the main turbine causing it to turn the turbine which is connected to the generator to create electrical power.
5. The unused steam is condensed into water.
6. The resulting water is pumped out of the condenser with a series of pumps, reheated and pumped back to the reactor vessel.
7. The reactor's core contains fuel assemblies which are cooled by water, which is force-circulated by electrically powered pumps.
8. Emergency cooling water is supplied by other water sources which can be powered by onsite diesel generators.

1.3.3. NUCLEAR STEAM SUPPLY SYSTEM

Steam exiting from the turbine flows into condensers located underneath the low pressure turbines where the steam is cooled and returned to the liquid state (condensate). The condensate is then pumped through feedwater heaters that raise its temperature using extraction steam from various turbine stages. Feedwater from the feedwater heaters enters the reactor pressure vessel (RPV) through nozzles high on the vessel, well above the top of the nuclear fuel assemblies (these nuclear fuel assemblies constitute the "core") but below the water level.

The feedwater enters into the downcomer region and combines with water exiting the water separators. The feedwater subcools the saturated water from the steam separators. This water now flows down the downcomer region, which is separated from the core by a tall shroud. The water then goes through either jet pumps or internal recirculation pumps that provide additional pumping power (hydraulic head). The water now makes a 180 degree turn and moves up through the lower core plate into the nuclear core where the fuel elements heat the water. Water exiting the fuel channels at the top guide is about 12 to 15% saturated steam (by mass), typical core flow may be 45,000,000 kg/hr (100,000,000 lb/hr) with 6,500,000 kg/hr (14,500,000 lb/hr) steam flow. However, core-average void fraction is a significantly higher fraction (~40%). These sort of values may be found in each plant's publicly available Technical Specifications, Final Safety Analysis Report, or Core Operating Limits Report.

The heating from the core creates a thermal head that assists the recirculation pumps in recirculating the water inside of the RPV. A BWR can be designed with no recirculation pumps and rely entirely on the thermal head to recirculate the water inside of the RPV. The forced recirculation head from the recirculation pumps is very useful in controlling power, however. The thermal power level is easily varied by simply increasing or decreasing the forced recirculation flow through the recirculation pumps.

The two phase fluid (water and steam) above the core enters the riser area, which is the upper region contained inside of the shroud. At the top of the riser area is the water separator. By swirling the two phase flow in cyclone separators, the steam is separated and rises upwards towards the steam dryer while the water remains behind and flows horizontally out into the downcomer region. In the downcomer region, it combines with the feedwater flow and the cycle repeats.The saturated steam that rises above the separator is dried by a chevron dryer structure. The steam then exits the RPV through four main steam lines and goes to the turbine.

1.3.4. CONTROL SYSTEM

Reactor power is controlled via two methods: by inserting or withdrawing control rods and by changing the water flow through the reactor core. Positioning (withdrawing or inserting) control rods is the normal method for controlling power when starting up a BWR. As control rods are withdrawn, neutron absorption decreases in the control material and increases in the fuel, so reactor power increases. As control rods are inserted, neutron absorption increases in the control material and decreases in the fuel, so reactor power decreases. Some early BWRs and the proposed ESBWR (Economic Simplified BWR made by General Electric Hitachi) designs use only natural circulation with control rod positioning to control power from zero to 100% because they do not have reactor recirculation systems. Fine reactivity adjustment would be accomplished by modulating the recirculation flow of the reactor vessel.

Changing (increasing or decreasing) the flow of water through the core is the normal and convenient method for controlling power. When operating on the so-called "100% rod line," power may be varied from approximately 30% to 100% of rated power by changing the reactor recirculation system flow by varying the speed of the recirculation pumps. As flow of water through the core is increased, steam bubbles ("voids") are more quickly removed from the core, the amount of liquid water in the core increases, neutron moderation increases, more neutrons are slowed down to be absorbed by the fuel, and reactor power increases. As flow of water through the core is decreased, steam voids remain longer in the core, the amount of liquid water in the core decreases, neutron moderation decreases, fewer neutrons are slowed down to be absorbed by the fuel, and reactor power decreases.

1.3.5. POWER PRODUCTION CAPACITY

a. **Steam Turbines:** Steam produced in the reactor core passes through steam separators and dryer plates above the core and then directly to the turbine, which is part of the reactor circuit. Because the water around the core of a

reactor is always contaminated with traces of radionuclides, the turbine must be shielded during normal operation, and radiological protection must be provided during maintenance. The increased cost related to operation and maintenance of a BWR tends to balance the savings due to the simpler design and greater thermal efficiency of a BWR when compared with a PWR. Most of the radioactivity in the water is very short-lived (mostly N-16, with a 7-second half-life), so the turbine hall can be entered soon after the reactor is shut down.

b. Size: A modern BWR fuel assembly comprises 74 to 100 fuel rods, and there are up to approximately 800 assemblies in a reactor core, holding up to approximately 140 tons of uranium.

1.3.6. PLANT AND REACTOR SAFETY SYSTEMS

a. Safety Systems: Like the pressurized water reactor, the BWR reactor core continues to produce heat from radioactive decay after the fission reactions have stopped, making a core damage incident possible in the event that all safety systems have failed and the core does not receive coolant. Also like the pressurized water reactor, a boiling water reactor has a negative void coefficient, that is, the neutron (and the thermal) output of the reactor decreases as the proportion of steam to liquid water increases inside the reactor. However, unlike a pressurized water reactor which contains no steam in the reactor core, a sudden increase in BWR steam pressure (caused, for example, by the actuation of the main steam isolation valve (MSIV) from the reactor) will result in a sudden decrease in the proportion of steam to liquid water inside the reactor. The increased ratio of water to steam will lead to increased neutron moderation, which in turn will cause an increase in the power output of the reactor. This type of event is referred to as a "pressure transient".

The BWR is specifically designed to respond to pressure transients, having a "pressure suppression" type of design which vents overpressure using safety relief valves to below the surface of a pool of liquid water within the containment, known as the "wetwell" or "torus". There are 11 safety overpressure relief valves on BWR/1-BWR/6 models (7 of which are part of the ADS) and 18 safety overpressure relief valves on ABWR models, only a few of which have to function to stop the pressure rise of a transient. In addition, the reactor will have already have rapidly shut down before the transient affects the RPV (as described in the Reactor Protection System section below).

Because of this effect in BWRs, operating components and safety systems are designed to ensure that no credible scenario can cause a pressure and power increase that exceeds the systems' capability to quickly shutdown the reactor before damage to the fuel or to components containing the reactor coolant can occur. In the limiting case of an ATWS derangement, high neutron power levels (~ 200%) can occur for less than a second, after which actuation of SRVs will

cause the pressure to rapidly drop off. Neutronic power will fall to far below nominal power (the range of 30% with the cessation of circulation, and thus, void clearance) even before ARI or SLCS actuation occurs. Thermal power will be barely affected.

b. **Reactor Core Isolation Cooling System (RCIC):** The Reactor Core Isolation Cooling System is not a safety-related system proper, but is included because it can help cool the reactor in the event of a contingency, and it has additional functionality in advanced versions of the BWR. RCIC is designed to remove the residual heat of the fuel from the reactor once it has been shut down. It injects approximately 2,000 L/min (600 gpm) into the reactor core for this purpose, at high pressure. It also takes less time to start than the HPCI system, approximately 5 seconds from an initiating signal.

c. **Containment system:** The ultimate safety system inside and outside of every BWR are the numerous levels of physical shielding that both protect the reactor from the outside world and protect the outside world from the reactor. There are five levels of shielding:

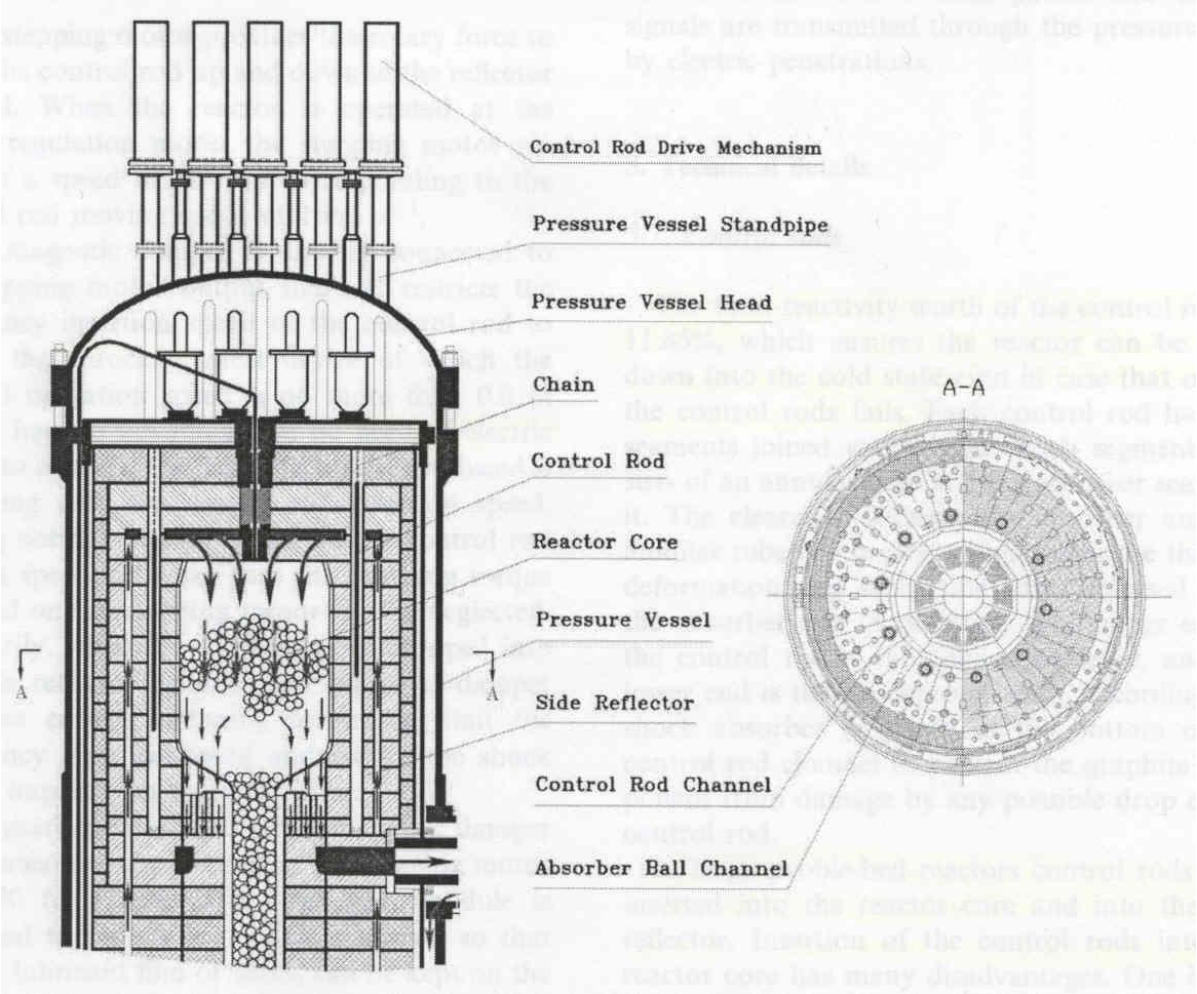
1. The fuel rods inside the reactor pressure vessel are coated in thick Zircalloy shielding;
2. The reactor pressure vessel itself is manufactured out of 6 inch thick steel, with extremely temperature, vibration, and corrosion resistant surgical stainless steel grade grade 316L plate on both the inside and outside;
3. The primary containment structure is made of steel 1 inch thick;
4. The secondary containment structure is made of steel-reinforced, pre-stressed concrete 1.2 - 2.4 meters (4 – 8 feet) thick.
5. The reactor building (the shield wall/missile shield) is also made of steel-reinforced, pre-stressed concrete .3 m to 1 m (1 – 3 feet) thick.

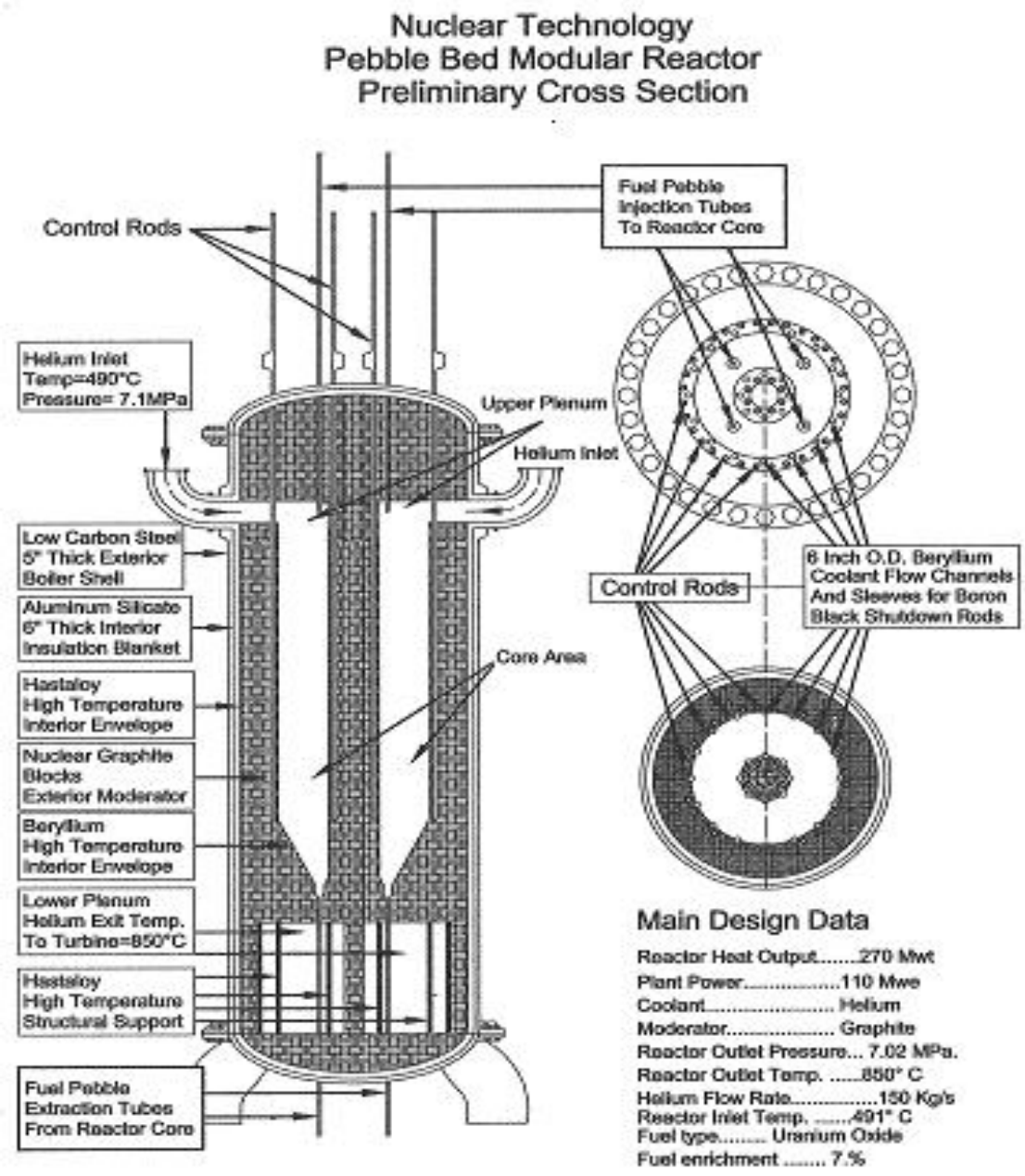
If every possible measure standing between safe operation and core damage fails, the containment can be sealed indefinitely, and it will prevent any substantial release of radiation to the environment from occurring in nearly any circumstance.

1.4. THE NUCLEAR TECHNOLOGY PEBBLE BED REACTOR

THE NTPBMR SOLUTION: The Nuclear Technology Pebble Bed Modular Reactor (**NTPBMR**) offers a future for the existing nuclear power plants. The **NTPBMR** is the only nuclear technology which can be used to replace the production capacity of an existing Nuclear Power Plant without increasing the diameter of its evacuation zone. They are gas-cooled, small, modular, inherently safe, flexible in design and operation, use a demonstrated nuclear technology and as the prices of natural gas increases, will become competitive with natural gas-fired turbine generators.

CROSS-SECTION OF CHINESE PEBBLE-BED REACTOR

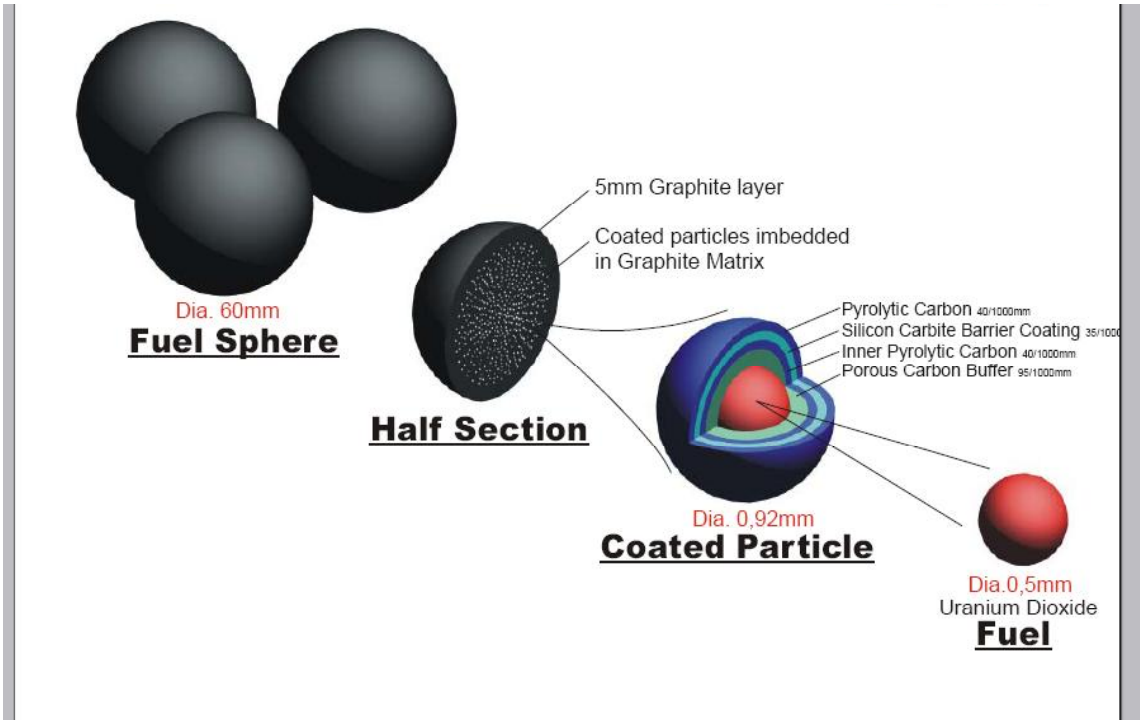




The **NTPBMR** technology consists of extensions and refinements of very well documented and successfully operated, helium cooled reactors which were built by the Germans in the 1970's and 1980's.

1.4.1. THE FUEL ELEMENT

The fuel element is a completely ceramic pebble containing low enriched Uranium Oxide (UO_2) as fuel.



Each individual **NTPBMR** reactor modules will be engineered and licensed as a process with all the major systems and sub-systems of the power plant fabricated in an off-site manufacturing facility. In addition each process involving a system or a sub-system will be manufactured under a set of code standards registered with the **NRC,** International Standards Organization (**ISO)**, International Atomic Energy Agency (**IAEA**).

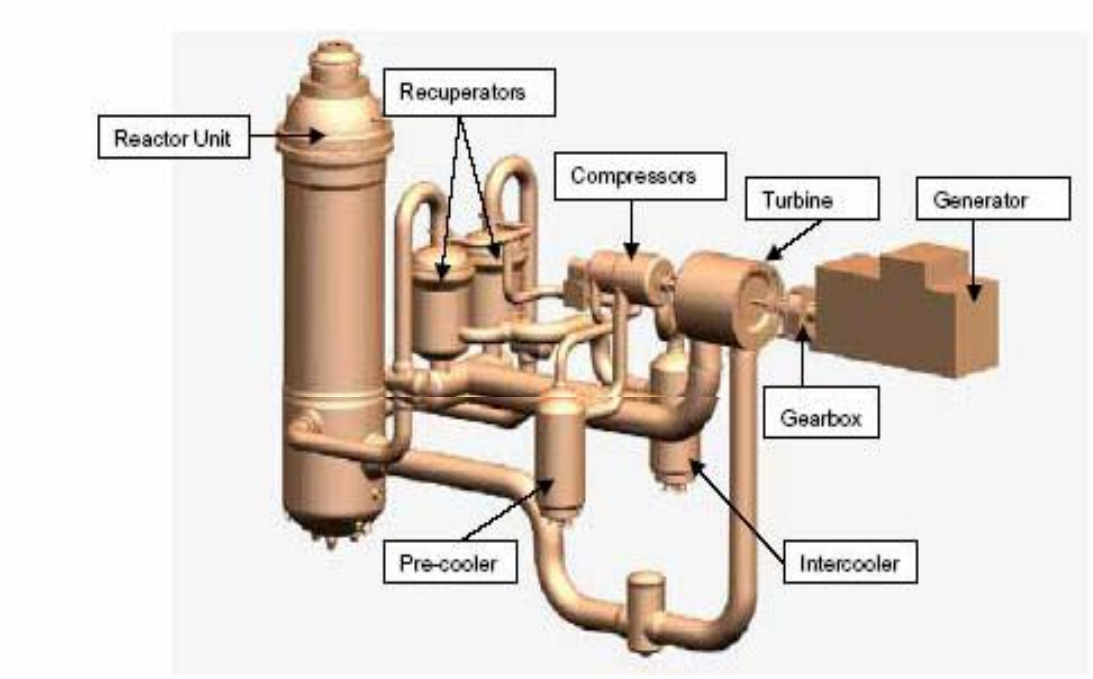
- A Nuclear reactor using gas as the core coolant will eliminate completely the types of problem which occurred at Three Mile Island and Chernobyl, in their Water-cooled Nuclear Reactor.
- Advances in gas turbine technologies will allow us to use helium as the coolant. Helium is an ideal cooling agent for a Nuclear reactor since it is completely inert chemically and its neutron absorption cross-sections are quite low.
- The reactor core contains approximately 360,000 uranium fueled pebbles about the size of tennis balls. Each pebble contains about 9 grams of low enriched Uranium Oxide (UO_2) in 10,000 to 15,000 (depending on the design) tiny grains of sand-like micro-sphere coated particles each with its own hard silicon carbide shell.

- The particle fuel consists of a spherical kernel of fissile or fertile fuel material encapsulated in multiple coating layers. The multiple coating layers form a miniature, highly corrosion resistant pressure vessel and an essentially impermeable barrier to release of gaseous and metallic fission products. This capability has been demonstrated at temperatures in excess of those that are predicted to be achieved under worst-case accident conditions.
- The micro-spheres are tri-coated with coatings of pyrolytic carbon and silicon carbide. The pyrolytic carbon layer absorbs the fission fragments and the Silicon Carbide coating retains these fission fragments and radioactive gasses within the micro-sphere. These micro-spheres are embedded in a graphite matrix material.
- The Uranium Oxide (**UO_2**) fuel has a melting temperature of approximately 2800^{o}C while the ceramic coating does not have a melting point and begins to degrade approximately at 2100^{o}C, and the degradation of the ceramic shell in the 50 or so hours required to empty the reactor would require temperatures in excess of 4000^{o}C. The temperature buildup in the core of the reactor in the event of a **Loss of Coolant** is not expected to exceed 1800^{o}C
- Another unique feature of pebble bed reactors is the online refueling capability in which the pebbles are re-circulated with checks on integrity and consumption of uranium. This system allows new fuel to be inserted during operation and used or damaged fuel to be discharged and stored on site for the life of the plant.
- The online refueling capability allows for the extraction of all the nuclear fuel in the event of a **LOCA.** Extraction of all the nuclear elements in the core will mitigate the possibility of melting the fuel pebbles.
- The comparatively small size and the lack of complexity in the design of a pebble-bed reactor adds to their economic feasibility. Each power module will produce approximately 100-120 megawatts (electric).
- The simplicity of design of our power plant is dramatic. These units will have only two dozen major plant subsystems which we believe can all be plant manufactured, licensed separately and moved to the proposed nuclear site. .

The **NTPBMR** modules are designed to produce 100-120MWe each. To place this in context a 100Mwe generator would produce the electricity consumed by 30,000 average homes. A single **NTPBMR** module would consist typically of a single main building, covering an area of approximately 13,000 square feet(130 x 100 feet).. The height of the building would be approximately 120 feet, the majority of the structure will be below ground level. The part of the building that would be visible above ground is equivalent to a four story building. There would be a unit control room, a high voltage switch yard, and a cooling tower for inland facilities. More than one **NTPBMR** module can be located on an existing licensed site.

The Nuclear Helium Supply System: Helium gas is used as the core coolant. Helium has a very small cross-section for neutron absorption, is inert and operating in a closed-loop, brayton cycle, single phase thermodynamic cycle which can power a turbine with high cycle efficiency.

ISOMETRIC VIEW: NUCLEAR TECHNOLOGY PEBBLE BED MODULAR REACTOR



1.4.2. THE THERMODYNAMIC CYCLE OF THE NTPBMR

1. Fission in the Triso-coated microspheres creates kinetic energy through the recoil of the Uranium atoms which are split by the injection of thermal neutrons.
2. The kinetic energy of recoil is transformed into thermal energy in the microspheres.
3. The thermal energy of the microsphere diffuses throughout the pebble and is transferred to the Helium Coolant by convective heat transfer.
4. The high pressure and high temperature helium is directed into the high pressure turbine. The high pressure turbine operates the compressors for the return of the helium to the reactor pressure vessel.
5. The helium is then directed to the low pressure turbine which operates the generator.

6. The helium is then cooled through a heat exchanger and the residual heat is exhausted to the atmosphere through an air powered radiator very much like an air conditioning unit on a house.

7. The cooled and compressed helium then re-enters the reactor pressure vessel.

1.4.3. UNIQUE CHARACTERISTICS OF THE NTPBMR TECHNOLOGY

A. On-line refueling capability: A unique feature of pebble bed reactors is the online refueling capability in which the pebbles are re-circulated with checks on integrity and consumption of uranium. This system allows new fuel to be inserted during operation and used or damaged fuel to be discharged and stored on site for the life of the plant.

The online refueling capability allows for the extraction of all the nuclear fuel in the event of a **LOCA.** Extraction of all the nuclear elements in the core will mitigate the melting of the fuel pebbles.

The online refueling capability allows for the insertion of graphite pebbles into the core as the extraction of all the nuclear fuel occurs in the event of a **LOCA.** The insertion of the graphite pebbles will increase the thermal mass of the core and thereby reduce the in-core temperature.

B. Graphite Moderator: The moderating environment of the **NTPBMR** is nuclear graphite. The **Reactor Pressure Vessel (RPV)** will house several hundred tons of Nuclear Graphite. The Nuclear graphite has high thermal mass and will allow for passive cooling of the reactor core in the loss of coolant event.

C. Solid Graphite Central Core: The moderating environment of the **NTPBMR** is will be greatly enhanced by the presence of a solid central core. The graphite central core will also be used to position the central control rod. This will greatly enhance the control ability of the **NTPBMR**.

D. Carbon Dioxide Emergency Core Fire Suppression System (ECFSS): The **ECFSS** is liquefied carbon dioxide. The carbon dioxide fire suppression system will mitigate the risk of a graphite fire of the type which occurred at Windscale, in England, in the early days of the English gas-cooled Magnox program. The carbon dioxide will also act as a passive emergency core cooling system to extract heat from the core.

D. Modular Design: The **NTPBMR** is modular in design and the comparatively small size and the lack of complexity in the design of the reactor adds to their economic feasibility. Each power module will produce approximately 110 megawatts (electric), with the use of two 55 MWe cooling loops.

The simplicity of design of our power plant is dramatic. These units will have only two dozen major plant subsystems which we believe can all be plant manufactured, licensed separately and moved to the proposed nuclear site. .

Each power module will produce approximately 110 megawatts electric, with the use of two 55 MWe cooling loops operating two closed loop brayton cycle gas turbines. The modules can easily be configured, in an energy park to produce up to 1.10 Gigawatts electrical power.

The technology can also be scaled down to 55 megawatts by employing only one leg of the Helium cooling system.

These reactor modules can be place in remote areas and connected to each other by the installation of smart, composite transmission lines to create a power grid or connect to an existing grid which need an increase in production capacity.

E. Safety Characteristics: The **NTPBMR** has the highest level of safety available in a Nuclear Power Plant. Its safety is a result of the design, the materials used and the physical processes rather than engineered safety systems. The peak temperature that can be reached in the reactor core (1,600 degrees Centigrade under the most severe conditions) is far below any sustained temperature (2,000 degrees Centigrade) that will damage the fuel elements.

F. Economic Benefits: The **NTPBMR** modules will all be built in a factory. This will allow the Company to capture the cost curve in the construction of Nuclear Power facilities, where the stakeholders have an equity position in the manufacturing of the components of the modules of the power plants. The Company's goal is to be able to design and build a Nuclear power Plant for less than $2000.00 per KW of electrical production.

SECTION TWO: NUCLEAR POWER AND RADIATION DOSES

2.1. PUBLIC CONCERN ABOUT NUCLEAR POWER:

The safety of nuclear power stations is a matter of considerable public concern. Past accidents such as the 1986 explosion of the reactor at Chernobyl in the Soviet Union and the 1979 accident at Three Mile Island (TMI) in the United States have contributed greatly to public fears (while also providing numerous lessons to the nuclear industry and its regulators that should lead to safety improvements at nuclear facilities). Public consciousness of the risk from nuclear power and associated activities appears to be much higher than for similar activities that pose comparable risks.

The reason for this is a subject of long-standing debate; one reason could be a mental association of nuclear power with nuclear weapons. (In this regard, it is important to note that a nuclear explosion such as that which occurs in an atomic bomb cannot occur in a nuclear power plant because, a nuclear power plant contains neither the necessary type nor configuration of nuclear materials.) Another reason could be fear of the invisible risk posed by radiation, and the involuntary nature of the risk.

Regardless of the specific reasons for public fears of nuclear power and for higher aversion of nuclear than other hazardous industrial activities, it is clear that continued use of nuclear power in developed and developing countries, and the prospects of its further development, requires not only firm assurance that technical and institutional measures will be effective in protecting public health and safety but also that public confidence and broad political support can be obtained. The technical complexity of nuclear power technology is one barrier to public understanding, making it difficult for many members of the public to evaluate safety questions for themselves.

This analysis does not address the question of public acceptance of Nuclear Power Production directly but rather seeks to characterize its health and safety risks, describe safety technology and regulations, and place the risk in perspective, so as to provide an adequate defense for the future use of Nuclear Power to produce electricity.

2.2. THE NATURE OF RADIATION:

To assist the reader in understanding portions of this analysis dealing with radiation exposure, risks and consequences, a comprehensive discussion of the nature of radiation and some of the units of radiation exposure and radioactivity is presented here. It is important to clarify the difference between "radioactive materials" and "radiation." In this context, radiation refers to the energy emitted from radioactive materials in the form of waves or particles as such materials decay due to instabilities within the atom.

Radioactive materials that are routinely emitted from nuclear power plants into air and water or that could be emitted in an accident may be ingested or inhaled by humans, after which they will continue to decay and thus cause a radiation dose to be delivered to the exposed person. Various pathways for such internal consumption of radioactive material are the principal concern with respect to limiting radiation doses from nuclear power plants.

The actual direct emanation of radiation from a nuclear power plant -- radiation "shine" -- is limited to the close proximity of the reactor itself and is of little concern to the general public living in the vicinity of the plant, although it is the principal concern with respect to personnel working at the plant (occupational exposure).

The radiation dose received by a nuclear power plant worker or a member of the public in the vicinity of a plant is measured in terms of the amount of energy deposited by the radiation in live tissue. The "rem" is one of the standard units for measuring radiation dose and is the unit used throughout this report.

Rem (Roentgen Equivalent Man): The acronym for roentgen equivalent man is a standard unit that measures the effects of ionizing radiation on humans. The dose equivalent in rems is equal to the absorbed dose in rads multiplied by the quality factor of the type of radiation (see 10 CFR 20.1004).

Röntgen or **roentgen (R):** Unit of exposure measuring the ionizing ability of g radiation; one röntgen produces one electric charge ($1.6x10^{-19}$ Coulombs) per 10^6 m^3 of dry air at 0° C and atmospheric pressure. This corresponds to an energy loss of 0.0877 joule per kilogram in air. The röntgen is no longer accepted for use with the International System.

Another important measure of radiation dose is the collective dose, which expresses radiation dose integrated over the population.

Collective Dose: The sum of the individual doses received in a given period by a specified population from exposure to a specified source of radiation.

Collective dose may be expressed in units of person-rem. As an example of the relationship between individual dose and collective dose, an exposure which causes 10 individuals to receive a dose of 1 rem each would produce a collective dose of 10 person-rem.

Finally. radioactivity is measured in terms of the number of radioactive emissions (or disintegrations) from a given quantity of material per unit time. The standard unit for measuring quantities of radioactivity that is used in this report is the curie.

Curie (Ci): The basic unit used to describe the intensity of radioactivity in a sample of material. The curie is equal to 37 billion (3.7×10^{10}) disintegrations per second, which is approximately the activity of 1 gram of radium. A curie is also a quantity of any radionuclide that decays at a rate of 37 billion disintegrations per second. It is named for Marie and Pierre Curie, who discovered radium in 1898. This unit is no longer recognized as part of the International System of units. It has been replaced by the becquerel.

Becquerel (Bq): Unit of activity in the International System–one disintegration per second; 1 Bq = 27 pCi. The unit of radioactive decay equal to 1 disintegration per second. 37 billion (3.7×10^{10}) becquerels = 1 curie (Ci).

Each radioisotope has its own characteristic rate at which it "decays" as a result of these emissions. The time required for any given radioisotope to decrease to one half of its initial quantity is a measure of the speed with which the radioisotope undergoes radioactive transformation. This period of time is known as the half-life and is characteristic of that particular radioisotope. Materials which decay slowly -- i.e., that have a long half-life -- are thus less radioactive than materials that have a short half-life. Half-lives vary tremendously, from microseconds to billions of years. For example, krypton-90 has a half-life of 33 seconds, whereas the half-life of potassium-40 (which is naturally present in food, e.g. orange juice) is 1.3 billion years.

2.3. NATURAL BACKGROUND RADIATION

The most significant sources of radiation to which humans are routinely exposed exist naturally in the environment. These include cosmic rays; radiation emanating from the ground and building materials and from radon gas and its decay products; and radio-elements in food and in the body.

Radon (Rn): A radioactive element that is one of the heaviest gases known. Its atomic number is 86. It is a daughter of radium.

As a benchmark, the **United States National Council on Radiation Protection and Measurements** (**NCRP**) estimates that the average American receives from these natural background sources of radiation a dose of roughly 300 millirem/year (i.e., 0.3 rem/year), 200 millirem of which is the result of radon gas and its decay products.

Actual individual doses due to natural background sources of radiation vary over a wide range depending on each person's place of residence and activities, which determine the exposure to these sources. For example, a person living in Colorado might receive an additional 100 millirem/year mainly because of the high altitude and the resulting higher levels of natural cosmic radiation. Similarly, a person taking a coast-to-coast airplane flight in the US would receive a dose of about 5 millirem due to cosmic radiation. Natural background radiation levels depend not only on variations in cosmic radiation with altitude, but also on variations in terrestrial radiation levels from different rock types.

Medical and dental radiation exposure brings the average person's radiation dose up significantly. The **NCRP** estimates a US average annual exposure from these sources of 53 millirem. Consumer products contribute on average an additional 10 millirem per year.

As a result, the average total radiation dose to individuals in the US resulting from natural background, medical, dental and consumer product radiation is about 360 millirem. It should be noted that smokers are also exposed to the radionuclide polonium-210 which occurs naturally in tobacco, resulting in a radiation dose to the lungs of up to 20 rem, which is a very large dose compared to natural background radiation exposure.

2.4. HISTORICAL PERSPECTIVE AND BACKGROUND

Research on the effects of ionizing radiation on populations extends back to the end of the last century when medical scientists were beginning to realize that the use of radioactive materials (radium) and radiation (x-rays) in the diagnosis and treatment of patients might well lead to hitherto unknown side effects. Some side effects were already well known; for example, skin erythema and the loss of hair showed up rather promptly after exposure to the relatively large doses from therapeutic use of x-ray machines. However, the delayed effects of radiation, such as cancer, were unknown until they began to appear at a much later date in patients who had been treated with radiation or radioactive materials and in workers occupationally exposed.

One of the early examples of the effects of radio-nuclides taken into the body comes from the ingestion of radium by painters of fluorescent watch dials who consistently wetted and pointed their brush tips with their lips during the period 1915 to 1935 when the practice was stopped. High incidence of bone cancer- and head carcinomas were observed among these workers and also among patients who had been treated internally with radium for tuberculosis of the bone.

During the 1920's and early 1930's a significant increase in mortality from leukemia among radiologists also began to be noticed. The actual doses received by such persons are very uncertain since at that time it was not common practice to monitor for radiation dose. It is, however, significant that these cancer excesses have not been observed in radiologists who entered practice after the 1930's when greater protective measures were taken. One example of the latent effects of x rays in specific populations was the discovery later in life of an excessive likelihood of tumors in individuals who had been treated with x rays in their childhood for scalp ringworm or enlarged thymus glands.

These observations clearly demonstrated the delayed effects of internal and external radiation in fairly large doses and, together with other observations, led the medical scientists of the day to speculate on whether exposure to much lower doses of radiation might lead proportionally to similar latent effects particularly when exposure was prolonged over a period of time.

The most comprehensive information relating health effects to radiation exposure over a wide range of dose arose much later from the medical histories of the surviving populations who were exposed to the atomic bombing of Hiroshima and Nagasaki in 1945. The effect of high doses was immediately visible, but it was not until some time later that an excess of leukemias began to be observed among the survivors. In the ensuing years, medical follow-up of the survivors has revealed excesses of other cancers including cancers of the lung, stomach, thyroid, and breast.

There is little doubt that exposure to high doses of radiation increases the potential for cancer in humans as these experiences have demonstrated. However, extrapolating latent effects of radiation to demonstrate that an increased risk exists at lower radiation doses is quite another matter and much more problematic. Even in exposed populations such as those in Japan, the effects of exposure are not easy to quantify. For example,

the medical records of the 80,000 atomic bomb survivors who were followed up from 1950 to 1978 showed that 23,500 persons had died, of which 4,750 had died of cancer. It has been estimated that only 250 of these were attributable to radiation exposure, i.e. in excess of the expected number of cancer deaths in a similar, but not exposed population. To obtain an estimate of the effects of low doses, such as those experienced in and around nuclear installations, requires extrapolations from high dose observations based on either:

1. empirical evidence at high doses, i.e. epidemiology, and then extrapolation to low doses;
2. laboratory data on the effects of low radiation doses on animals
3. theoretical formulations which seek to quantify the relationship between dose and effect. Significant uncertainties are associated with these techniques as explained below.

2.5. THE EFFECTS OF HIGH-LEVEL VERSUS LOW-LEVEL RADIATION DOSES

While doses in the range of a few rads are generally regarded by the scientific community as being low-level doses, and doses in excess of 100 rads are regarded as being high-level doses, the demarcation line between high and low level radiation is not a scientifically defined one. Nevertheless, the United Kingdom Radiological Board regard low doses as being less than 1 rad per year or low dose rates as less than 10 rad per day.

- **Rad (Radiation Absorbed Dose):** The special unit for radiation absorbed dose, which is the amount of energy from any type of ionizing radiation (e.g., alpha, beta, gamma, neutrons, etc.) deposited in any medium (e.g., water, tissue, air). A dose of one rad means the absorption of 100 ergs (a small but measurable amount of energy) per gram of absorbing tissue (100 rad = 1 gray).

Part of the reason for this is that health physics is concerned with two types of exposure:

1. a single accidental exposure to a high dose of radiation during a short period of time, which is commonly called acute exposure, and which may produce biological effects within a short time after exposure and a relatively higher probability of latent effects such as cancer and genetic damage;
2. Long-term, low-level exposure, commonly called continuous or chronic exposure, where the results of the exposure is of a much lower probability and will not become apparent for years. Any such exposures are the result of improper or inadequate protective measures. Exposure of the whole body to an extremely high dose of radiation (of the order of 1000 rads) is almost certain to result in death within a matter of weeks but if a limited area of the body is briefly exposed to a very high dose, this may not be fatal. In fact hundreds of rads are used in many therapy regimes. For example, an instantaneous whole body dose greater than 500 rads would probably be lethal, provided no treatment was given, as a result of damage to the bone marrow and gastrointestinal tract but if the

same total dose is received over a period of weeks or months, there is more opportunity for cellular repair and there may be no early signs of injury although damage to tissues may have occurred and may be manifested later in life, or possibly in the irradiated person's descendants. However, if only a limited area of the body is briefly exposed to a high dose of this nature, it may not be lethal but early effects may occur such as reddening of the skin (erythema) in a week or so.

The most important long term effect of radiation is cancer but the fundamental processes by which it is induced are not fully understood and, moreover, there is no way at present of medically distinguishing cancers caused by radiation from those occurring naturally and those caused by other carcinogens.

The main source of information on the risk of cancer following whole-body exposure to radiation comes from studies on the survivors from the atomic bombings of the cities of Hiroshima and Nagasaki. The risks derived from studies of these populations are based largely on exposure to high doses delivered over a short period of time (tens of rads or more), whereas most people are only exposed to low levels of radiation over long periods of time.

Because of the relatively low probability of effects occurring due to exposure to low-level radiation over long periods of time, the risks of such exposure can only be calculated from the data available on exposure to high levels of radiation. It is generally assumed for radiation protection purposes that there is a simple proportional relationship between dose and risk and for radiations from alpha particles this appears to be the case. However, for beta and gamma radiations and x rays there is considerable evidence that the risk is less at low doses and low dose rates than at high doses given at high dose rates. In either case, there is no sound basis for assuming the existence of a threshold below which no cancers or other health effects are induced.

2.6. NATURE OF EPIDEMIOLOGICAL STUDIES

One of the major problems in conducting epidemiological studies to estimate the effects of low-level radiation is the difficulty of identifying the influences of the multiple factors which have to be taken into account in order to estimate the effects. These factors include:

A. the quality of the exposure and medical data being used;
B. the selection of appropriate controls;
C. the methodology and scientific design of the analyses;
D. occupational conditions and personal habits and,
E. the validity of the statistics for a given population size as discussed in the following sections.

A. Quality of Data: One of the more complicated problems is the quality of the data used as input into the study. Epidemiology studies usually employ mortality data (death rates from a disease) or incidence data (the occurrence rate of a disease). Mortality data is usually based on death certificate information where it is often not known for certain whether the primary cause of death is cancer and

whether it has been accurately represented on the certificate. For example, the type of cancer reported on the certificate may be the result of metastasis and not the site of origin. A similar problem exists in incidence data where diagnoses are based on "registrations" of a disease occurring, and might not be accurate.

B. **Availability of Realistic Exposure Measurements:** A second problem is the limited availability of realistic exposure data. In most cases involving the deaths from, or diagnosis of, cancer in persons who are not occupationally exposed to radiation, valid exposure data almost never exists.

C. **Validity and Significance:** Third, for an epidemiological study to have a high degree of statistical significance, it is necessary to have a large enough data base. Detecting a subtle rise in cancer incidence above the "normal" and correctly attributing such an increase to low levels of radiation exposure requires the study of very large populations.

D. **Age-Dependence:** Another complication is that cancer is well known to be an age-dependent disease which is rarer in young people and much more prevalent in older populations. Consequently any analysis of cancer rates needs to account for the age distribution in the area in question and age-adjusted corrections made.

E. **Other Factors:** Other factors which affect cancer mortality and need to be considered in the analyses are duration and age when exposure begins; sex ratios; racial and ethnic factors; exposure to other environmental agents (viruses, carcinogens, smoking); and even social structure.

2.7. ESTIMATION OF RISK OF CANCER FROM EXPOSURE TO RADIATION

The derivation of the estimates of the risk of induction of cancer from exposure to radiation are carried out by various national and international bodies composed of internationally renowned experts in the fields of radiobiology, radiation epidemiology, health physics, medical radiology, statistics, genetics, etc. These bodies conduct comprehensive reviews of the evidence relating to risk of cancer, hereditary effects and other diseases as a result of radiation exposure.

Such bodies include:

- The **International Commission on Radiological Protection (ICRP),**
- The **United Nations Scientific Committee on the Effects of Atomic Radiation** (**UNSCEAR**),
- The **National Academy of Sciences Committee** on the **Biological Effects of Ionizing Radiation** (**BEIR**)
- The **National Council on Radiation Protection** and Measurements (**NCRP**) in the United States
- The **National Radiological Protection Board** (**NRPB**) in the U.K.

In 1977 the **ICRP** published its basic recommendations on the effects of exposure to radiation in which it assumed that there is no threshold below which radiation effects are

not harmful and took the position that the probability of harmful effects increases with dose. Also in 1977, **UNSCEAR** published a comprehensive review of animal and human exposure to radiation and the induction of cancers, from which it derived an estimate of the risk of the induction of leukemia as being in the range of 2 cases per million of population per 100 mrem for low doses. This value was used by **ICRP** in its 1977 report which calculated the risk factor for the induction of all fatal cancers as being in the range of 10 cases per million of population per 100 mrem, or 5 times the leukemia risk.

On this basis if a population of 50-million people receive a dose of 100 mrem above natural background radiation as a result of exposure to medical and other sources, 500 additional cancer deaths due to this "extra" radiation can be expected some time in the future, 100 of which might be a leukemia, out of an annual total of about 125,000 deaths normally expected from cancer. The **UNSCEAR** risk estimates can also be used to compare the risks from exposure to natural background radiation with the number that might be expected from the operation of a power station which, under normal circumstances and conservative assumptions, would deliver an increment of 4 mrem per year over natural background (see Section 3 for further discussion of typical doses received routinely by persons living near nuclear power plants). In this case, if the risk factor assessed by the ICRP in 1977 is used for the calculation of the number of deaths expected from the natural background radiation in a local population of 100,000 persons, the estimated number of deaths would be 0.2, and the additional number from the operation of the power plant would be 0.008 deaths (or 1 death in 125 years) which is virtually impossible to statistically detect even by a massive study covering the whole of the regional population for a very large number of years.

In 1990 the **BEIR** Committee published its fifth update of its report (BEIR V). This suggests that these earlier estimates are too low and that the risk factor should be increased by a factor of four, or even by a factor of 10 as some have suggested. Even if the higher figure was used, it would still require a very substantial period of follow-up or very large population groups to be able to state with any certainty that the additional deaths were due to the effects of radiation. Furthermore, it would still be extremely difficult to detect the additional number of deaths due to man-made radiation among those arising from natural background radiation.

2.8. BEIR VII

BEIR VII develops the most up-to-date and comprehensive risk estimates for cancer and other health effects from exposure to low-level ionizing radiation. It is among the first reports of its kind to include detailed estimates for cancer incidence in addition to cancer mortality. In general, **BEIR VII** supports previously reported risk estimates for cancer and leukemia, but the availability of new and more extensive data have strengthened confidence in these estimates. A comprehensive review of available biological and biophysical data supports a "**linear-no-threshold**" (**LNT**) risk model—that the risk of cancer proceeds in a linear fashion at lower doses without a threshold and that the smallest dose has the potential to cause a small increase in risk to humans.

This report is the seventh in a series of publications from the National Academies concerning radiation health effects called the **Biologic Effects of Ionizing Radiation** (**BEIR**) reports. **BEIR VII** focuses on the health effects of low levels of low **linear energy transfer** (low-**LET**) ionizing radiation such as x-rays and gamma rays. The most recent **BEIR** report to address low level low-LET radiation was the BEIR V report published in 1990. Humans are exposed to ionizing radiation from both natural and man-made sources (see Figure 1). Very high doses can produce damaging effects in tissues that can be evident within days after exposure. Late effects such as cancer, which can occur after more modest doses including the low dose exposures that are the subject of this report, may take many years to develop. Most radiation sources have a mixture of high- and low-LET radiation. Compared to high-LET radiation, low- LET radiation deposits less energy in the cell along the radiation path and is considered less destructive per radiation track. The BEIR VII report defines low doses as those in the range of near zero up to about 100 mSv (0.1 Sv) of low-LET radiation. People in the United States are exposed to average annual background radiation levels of about 3 mSv; exposure from a chest X-ray is about 0.1 mSv and exposure from a whole body computerized tomography (CT) scan is about 10 mSv.

There are many challenges associated with understanding the health effects of low doses of low-LET radiation, but current knowledge allows several conclusions. The BEIR VII report concludes that the current scientific evidence is consistent with the hypothesis that, at the low doses of interest in this report, there is a linear dose-response relationship between exposure to ionizing radiation and the development of solid cancers in humans. It is unlikely that there is a threshold below which cancers are not induced, but at low doses the number of radiation induced cancers will be small. Other health effects (such as heart disease and stroke) occur at higher radiation doses, but additional data must be gathered before an assessment of any possible dose response can be made between low doses of radiation and non-cancer health effects. The report also concludes that with low dose or chronic exposures to low-LET irradiation, the risk of adverse heritable health effects to children conceived after their parents have been exposed is very small compared to baseline frequencies of genetic diseases in the population.

Naturally-occurring genetic (i.e., hereditary) diseases arise as a result of alterations (mutations) occurring in the genetic material (DNA) contained in the germ cells (sperm and eggs) and are heritable (i.e., they can be transmitted to the offspring and subsequent generations). The concern over whether exposure to ionizing radiation would cause an increase in the frequencies of genetic diseases launched extensive research programs to examine the adverse genetic effects of radiation in the children of A-bomb survivors and other studies focusing on mammals that could be bred in the laboratory, primarily the mouse. Studies of 30,000 children of exposed A-bomb survivors show a lack of significant adverse genetic effects. During the past 10 years, major advances have occurred in our understanding of the molecular nature and mechanisms underlying naturally occurring genetic diseases and radiation-induced mutations in

experimental organisms including the mouse. The risk estimates presented in this report have incorporated all these advances. They show that, at low or chronic doses of low-LET irradiation, the genetic risks are very small compared to the baseline frequencies of genetic diseases in the population.

Given BEIR VII estimates, one would not expect to see an excess in adverse hereditary effects in a sample of about 30,000 children (the number of children evaluated in Hiroshima and Nagasaki). One reason that genetic risks are low is that only those genetic changes compatible with embryonic development and viability will be recovered in live births.

2.9. STUDIES OF THE EFFECTS OF NATURAL BACKGROUND RADIATION

Since there are substantial variations in natural background radiation levels from one geographic region to another, as discussed in Section 1.0, it might be expected that some difference in the incremental health impact of this type of radiation exposure would be observable.

In an attempt to determine whether such difference exist, a number of studies have been carried out in the USA, the UK and several other countries to investigate cancer incidence in different regions where the population is exposed to different levels of natural background radiation. The results of these studies have been inconclusive. Since the level of radiation that is routinely emitted from nuclear power plants (see Section 3) is substantially less than the variations in natural background radiation levels from one geographic region to another, it is not surprising that the incremental health impact of this type of radiation exposure, if any, is also very difficult to detect.

2.10. LEUKEMIA CLUSTERS

There have been persistent reports of leukemia in young persons living near nuclear facilities which have been described as "leukemia clusters". This issue is briefly addressed in this section. In the context of a disease such as .leukemia, the word "cluster" is generally used to describe an observation of an unusually high incidence of the disease in a small geographical area within a relatively short time period. It has also been used to refer to the persistent increased occurrence of the disease in a small area, such as might occur if the population of that area was permanently exposed to risk from a causative agent in the environment. The actual rates arid number of cases which occur in the locality being studied may be higher or lower than the national average. If the number is higher, then it may be described by some people as a "cluster". The definition of a leukemia cluster is complex but most specialists agree that it involves an unusually high incidence of leukemia in a small area for a limited time period. For example, it has been reported that a number of people living on the same road or in a small town developed leukemia within a few years of each other.

Reports of clusters of leukemia, due to unknown causes, have appeared in the medical literature for many years, one of the earliest being in the British Journal of Childhood Disease in 1917. Since the 1960's systematic searches, most not associated with nuclear energy, have been carried out in many parts of the world in an effort to

determine whether leukemia cases tend to occur more closely together than would be expected by chance.

The results of these searches revealed the occurrence of leukemia "clusters" in the following places:

- Leukemia San Francisco 1948-55
- Childhood leukemia Buffalo, N.Y. 1943-56
- Childhood cancer Buffalo, N.Y. 1943-56
- Childhood leukemia Northumberland 1951-60
- Childhood leukemia Liverpool, U.K. 1955-64
- Childhood leukemia Portland,Oregon 1950-61
- Childhood leukemia New Zealand 1953-64
- Childhood leukemia Atlanta, Georgia 1958-88
- Leukemia/lymphoma Bahrain 1966-76
- Hodgkin's disease King County, 1974-79 Washington, USA

The clusters listed in the table above have been ascribed to a variety of causes including radiation, association with the dairy industry, and a flood disaster in New York. It is still unclear whether leukemia occurs in clusters to a greater extent than would be expected by chance but it is clear that leukemia clusters can occur randomly without an apparent cause. This occurrence is consistent with statistical theory since a random distribution is not uniform and apparent clusters are the rule rather than the exception.'

2.11. GENETIC EFFECT

Ionizing Radiation: Any radiation capable of displacing electrons from atoms or molecules, thereby producing ions. Some examples are alpha, beta, gamma, x-rays, neutrons, and ultraviolet light. High doses of ionizing radiation may produce severe skin or tissue damage.

Ionizing radiation can result in damage to the genetic material (DNA) in reproductive cells leading to mutations which may be transmitted to subsequent generations. Such mutations are not seen in irradiated individuals, but only in their immediate or generational offspring. As the **BEIR VII** report points out, mutations in reproductive cells may occur spontaneously due to natural causes, including those which can be associated with exposure to natural background radiation. It is extremely difficult to estimate what small increments of mutations effects may be induced by man-made radiation above this spontaneous occurrence rate. Estimates of genetic effects in humans must rely more on results from experimental animal studies than on human epidemiology studies that are extremely sparse. Genetic effects of ionizing radiation are detected through the study of certain endpoints such as chromosome abnormalities, spontaneous abortions, congenital malformations, or premature death.

It must be emphasized that mutations caused by radiation do not lead to the grossly deformed offspring as portrayed by popular science fiction. As the **BEIR VII** report states: "Some mutations have drastic effects that are expressed immediately, and these are eliminated from the population quite rapidly."

Although it is generally believed by the scientific and medical community that there is a need to assess the genetic effects of radiation exposure, the perspective has changed since the 1950s. As the **BEIR** V report states: "...in regard to the induction of mutations, the greater current risk seems to result from exposure to chemical mutagens in the environment rather than from exposure of populations to radiation." It is now clear that the more significant risk of health consequences in persons exposed to radiation is that of cancer, with genetic effects of lesser concern than earlier considered. As a consequence, substantial efforts have been made to limit personal exposure to reduce the risk of cancer and this in turn has limited genetically significant exposures.

2.12. CONCLUSIONS

Recent studies from the United States and the United Kingdom have reported increases in mortality from leukemia in young children, especially under the age of 10, living near certain nuclear installations. The reasons for these increases are not clear and there is no convincing evidence that they are connected with exposure to low-level radiation. Nevertheless, because of the concerns raised by these observations, epidemiologic studies have been, and are continuing to be, carried out in the United Kingdom and a number of other countries with nuclear facilities in an attempt to determine whether there are any health effects on workers and populations living in the vicinity of those nuclear facilities which are explainable as a consequence of radioactive emissions from those facilities.

The most exhaustive of these studies that has been completed so far has been that carried out by the **National Cancer Institute** (**NC**I) in the United States. This survey encompassed all 62 nuclear facilities that went into service in the United States prior to 1982 and evaluated over 900,000 cancer deaths occurring between 1950 and 1984 in 107 counties.

The results of this survey were evaluated by the Ad-Hoc Advisory Committee of medical and epidemiological experts set up by the NCI who concluded that the survey had:

"produced no evidence that an excess occurrence of cancer has resulted from living near nuclear facilities. Further, measurements of radioactive releases from nuclear facilities indicate that the dose from routine operations is generally much below natural background radiation, and hence may be unlikely to produce observable effects on the health of surrounding populations."

The type of study undertaken by the NCI should help to provide the public with the reassurance that the normal operation of nuclear facilities does not pose undue public health risks.

SECTION THREE: ROUTINE RADIOACTIVE EMISSIONS

3.1. INTRODUCTION

Routine emissions of radioactive material in solid, liquid and gaseous forms result from the operation of nuclear power plants. These releases increase the amount of radioactivity in the biosphere; hence their impact on public health must be considered. Such emissions are an inevitable result of normal plant operations and must be clearly distinguished from non-routine, accidental releases from power plants. This section discusses the origins and quantities of such emissions, the doses involved, and their health impact. Furthermore, routine exposure of nuclear power plant employees to radiation must be taken into account and is reviewed in this section.

In addition to nuclear power plants, nuclear fuel cycle facilities such as uranium mines and fuel reprocessing plants also routinely release small amounts of radioactivity to the environment which might be larger than the emissions from nuclear power plants. Such facilities also cause occupational radiation exposure.

3.2. SOURCES OF ROUTINE EMISSIONS

- **Fission:** The splitting of a heavy nucleus into two roughly equal parts (which are nuclei of lower-mass elements), accompanied by the release of a relatively large amount of energy in the form of kinetic energy of the two parts and in the form of emission of neutrons and gamma rays.
- **Fission products:** Nuclei formed by the fission of higher mass elements. They are of medium atomic mass and almost all are radioactive. Examples: ^{90}Sr, ^{137}Ce.

There are three principal categories of radioactive materials produced as a result of the nuclear fission process in light water reactors: fission products, neutron activation products and tritium. Fission products are produced when the uranium atoms in the nuclear fuel fission into two smaller atoms. They are produced in both solid and gaseous forms. Neutron activation products, in contrast with fission products, are produced outside of the fuel material in either the fuel cladding material, fuel assembly structural materials or the reactor structure itself. Neutron activation products result when neutrons emitted in a fission reaction are absorbed by these materials, thereby making them radioactive. Finally, tritium, which is the radioactive isotope of hydrogen, is produced in a variety of ways, including by neutron capture in the reactor's coolant water. These radioactive materials find their way into nuclear power plant effluents in the following manner.

A. During reactor operation, almost all fission products are retained within the uranium fuel material and the metal cladding within which fuel elements are encased. (This fuel material is eventually removed from the reactor and either reprocessed into new fuel or disposed of in solid form as high-level nuclear waste.) However, a small percentage of the fission products may escape from the fuel rods through hairline cracks that may develop in the cladding material. Such cracks result either from welding defects or localized corrosion that occurs during reactor operation. As a result, the reactor's internal coolant water may

become contaminated with gaseous and, to a lesser extent, solid fission products;

B. similarly, small quantities of neutron activation products and tritium -- which are formed not in the fuel rods but in the coolant water or in structural materials that come into contact with the coolant water -- also contaminate the coolant water.

3.3. RADWASTE CONTROL SYSTEMS

Regulations require the application of "rad-waste" systems (described below) whose purpose is to reduce radioactivity levels in plant effluents to what are believed to be safe levels, based on the current understanding of the effects of radiation, as discussed in Section 2.0. **NRC** regulations place numerical limits on such effluents, and require radioactive emissions to be reduced to levels that are "as low as reasonably achievable" (**ALARA**). The radiation doses to the general public that result from nuclear power plant operation after such reductions are made are discussed in this Section and are compared to natural background levels of radiation.

Radwaste systems consist of liquid and gaseous waste processing systems. Radioactive liquids are decontaminated in two ways: evaporation and demineralization. Both are methods of filtering the liquid effluents so as to separate the radio-nuclides from the water that will be discharged to the environment. Similarly, gaseous emissions are passed through a particulate filter to remove solid radioactive particles. Gaseous radioactivity is reduced by storing gaseous wastes in holdup tanks before discharge to air, thus allowing radioactivity levels to reduce by natural radioactive decay. As a result of these decontamination procedures, the vast majority of the radioactivity that reaches the reactor coolant water is removed from that water, solidified by cementation or other methods, and shipped off-site for disposal as low-level radioactive waste. Effluents discharged from the plant into air and water contain only very small amounts of radioactivity that was not removed by these processes. 'The discharges must not exceed levels allowed under regulation. Tritium is particularly difficult to remove because it has chemical properties identical to hydrogen, and thus becomes an integral part of the reactor water.

The **United Nations Scientific Committee on the Effects of Atomic Radiation** (**UNSCEAR**) has compiled data on the quantities of various radioactive materials that are discharged from reactors worldwide, which vary depending on the reactor type.

The principal categories of effluents are:

- Fission product noble gases (such as krypton and xenon), which have short half-lives and high radioactivity levels but produce low radiation doses because they are chemically inert;
- Activation gases produced in gas-cooled reactor operation, especially argon- 41 and sulfur-35, which also have high activity levels but cause low doses
- Tritium, which, as mentioned above, has chemical properties identical to hydrogen;

- Carbon-14, which has a long half life (5730 years) and therefore is of concern in terms of long-term dose commitment. It is produced mainly from reactions with nitrogen and oxygen in the fuel and moderator.
- heavy water reactors (HWRs) produce relatively high levels of C-14 due to presence of oxygen in the moderator;
- Iodine-131, which has a half-life of 8 days, is mobile in the environment, and selectively migrates to and irradiates the thyroid;
- Particulates, which either arise directly, as decay products of fission product noble gases, or from corrosion of materials in the primary coolant circuit; and Other liquid effluents.

3.4. DOSE LEVELS TO THE GENERAL PUBLIC AND HEALTH IMPACT

The principal pathways for human exposure to radioactive effluents are:

- Inhalation;
- Ingestion of food crops and animal products
- Ingestion of drinking water
- Ingestion of fish and invertebrates
- Air submersion
- Ground irradiation.

Other pathways which have been found to cause generally much smaller doses include:

- Direct exposure from waterborne activities (swimming, boating, shoreline recreation)
- Ingestion of crops that were irrigated with contaminated water.

Based on known airborne releases from nuclear power plants in the United States, estimates of the doses received by persons residing near those plants have been calculated by the US **Nuclear Regulatory Commission (**NRCQ). The average distribution of doses among the estimated population of 140-million living within 2 to 80 km around each site for 70 nuclear power plants in the US are as follows:

- About 84% of the population at risk from airborne releases has been estimated as receiving a dose commitment of between 0.000003 and 0.001 mrem.
- About 0.4% of the population at risk received a dose of between 0.003 and 0.01 mrem.

The study did not estimate the maximum dose received by an individual, but licensee calculations at sites with the highest emissions indicated values of up to approximately 100 times the average individual doses, i.e., of the order of a few millirem per year.

Similarly, using international effluent data, the **United Nations Scientific Committee on the Effects of Atomic Radiation** (**UNSCEAR**) calculated doses to populations living near nuclear power plants resulting from each category of emissions and for each type of reactor. It should be noted that whereas the **NRC** figures given above are stated in terms of dose to the average individual, the **UNSCEAR** figures are given in terms of total population dose, which is equal to the average individual dose multiplied by the size of

the exposed population. **UNSCEAR** also estimates that when these figures for nuclear power plants are combined with comparable figures for other nuclear fuel cycle facilities, the total population dose is 400 person-rem per gigawatt-year, so that with current nuclear power generation at about 190 gigawatt-years per year, the annual dose to the world's population resulting from nuclear power plants and nuclear fuel cycle facilities is about 76,000 person-rem. For comparison purposes, it should also be noted that **UNSCEAR** has estimated that radioactive material emitted to the atmosphere from the burning of coal in coal-fired electricity generating plants results in a population dose of about 400 person-rem per gigawatt-year also.

For purposes of comparison with the individual and population doses due to nuclear power given above, it should be noted that the average person receives a dose due to natural background radiation of roughly 300 mrem/year, including the dose from cosmic rays, naturally occurring radioactive materials in the ground and building materials (including radon gas), and radio elements in the body. As also noted, medical and dental radiation exposure, plus exposure to radiation from consumer products, bring a person's radiation dose up significantly; the average total radiation dose to individuals in the US resulting from natural background radiation and these other sources is about 360 millirem. (In addition, smokers are also exposed to the radionuclide polonium-210 which occurs naturally in tobacco, resulting in a radiation dose to the lungs of up to 20 rem.)

Furthermore, by multiplying the average individual background dose figure of 300 millirem/year by the world population of approximately 5 billion, the total worldwide population dose due to natural background radiation can be determined to be about *1.5* billion person-rem/year, which can be compared with the figure of 76,000 person-rem/year for world population dose due to nuclear power and nuclear fuel cycle facilities.

This comparison of the radiation doses routinely received by the population as a result of nuclear power and those received from natural and medical sources strongly suggests that routine emissions from nuclear power plants should have little or no impact on public health. To corroborate this assumption, a number of studies have been done to assess whether or not the incidence of cancer is greater in populations living near nuclear power plants than populations living in other locations. A report published in 1987 by the British Office of Population Censuses and Surveys on cancer risk in the vicinity of nuclear facilities in England and Wales found that, overall, there was no evidence to conclude that cancer mortality near UK nuclear installations was higher than elsewhere in the U.K.. It did, however, note an increase in deaths from leukemia in young persons under age ten in the vicinity of the Sellafield fuel reprocessing plant but the reasons for this were not clear.

Subsequently, the **National Cancer Institute** in the US initiated a large-scale survey of the incidence of cancer in persons living near nuclear facilities in the United States, published in July 1990. The authors conclude that the survey produced no evidence that an excess occurrence of cancer has resulted from living near nuclear facilities. They also conclude that measurements of radioactive releases from such facilities indicate that

doses due to routine emissions are generally much lower than doses from natural background radiation and therefore may be unlikely to produce observable effects on the health of surrounding populations.

3.5. OCCUPATIONAL EXPOSURE

While the doses to the general public from nuclear power plants have been generally quite low and the health impact thus far undetectable, the doses to workers are naturally higher and often exceed natural background radiation dose levels. **UNSCEAR** has tabulated data on worldwide occupational radiation exposure at nuclear power plants during the period 1980 to 1984. For the same period, **UNSCEAR** also determined the following average collective occupational dose levels on a per-gigawatt-year basis, grouped by reactor type: Collective Dose Per Unit Energy Generated (person-rem per gigawatt-year):

- Light Water Reactor (LWR) 1300
- Heavy Water Reactor (HWR) 400
- Gas-Cooled Reactor (GCR) 500
- High Temperature Gas-Cooled Reactors (HTGR) 10

Among light water reactors, it was found that the collective dose in **BWR'**s can be up to a factor of two higher than in **PWR'**s, possibly because more maintenance work in radiation areas is necessary in **BWR'**s. The occupational exposure to radiation for the **NTPBMR (**A High Temperature Helium-Cooled Gas Reactor**)** is significantly lower than both types of Light Water Reactors.

According to figures recently published by the US Nuclear Regulatory Commission, in 1989 nuclear power plant workers in the US received an average dose of 340 mrem that year (somewhat higher than the average natural background radiation dose level)." The average collective dose per reactor was 344 person-rems. These figures are based on exposure data at 108 **PWR**s and **BWR**s, and represent a 14% decline from 1988 level-. Broken down further, the average PWR work force received a collective dose of 296 person-rems and the average BWR work force received a collective dose of 439 person-rems.

SECTION FOUR: THE SAFETY OF NUCLEAR POWER REACTORS

4.1. INTRODUCTION

The energy produced in the core of a nuclear reactor comes about as a result of the fissioning of the nuclei of uranium' atoms in the nuclear fuel. To convert the heat released through nuclear fission into electricity, a coolant flows through the reactor core to absorb the heat and make steam, which spins turbines that power electrical generators. The materials produced from the fissioning of uranium (the "fission products") are highly radioactive and have to be strictly contained to prevent them from being released to the environment, as do other categories of radioactive materials produced in a nuclear power plant.

As discussed in Section 3.0, small quantities of fission products and other radioactive materials contaminate reactor coolant water under normal operating conditions. This

contamination is largely removed using specially designed systems which ensure that routine releases of contaminated water to the environment remain strictly within regulatory limits. More importantly, accidental releases of radioactive material to the environment must be prevented. There are many possible events which could lead to accidental releases; for example, in a **pressurized water reactor** (**PWR**), the rupture of one of the many tubes in the heat exchangers would allow radioactive water to escape from the primary coolant into the secondary coolant. The large pipes of the secondarv cooling circuit penetrate the containment structure of a **PWR.** Consequently, once outside the containment, radioactive material leaked from the primary system through the ruptured tube could then potentially escape into the environment.

However, the principal event which could lead to large quantities of radioactive material being released to the environment is the **loss of coolant accid**ent (**LOCA**).

A **LOCA** could be initiated by a number of means:

- steam line breaks
- sudden expulsion of control rods
- loss of offsite power
- severe natural phenomena such as earthquakes, tornadoes and hurricanes.

The most serious event which has been postulated would be a break of a major pipe in the circuit that provides coolant to the reactor core, which could lead to the loss of substantial quantities of coolant. It is essential that the core be kept completely covered with coolant at all times. Otherwise the core, or parts of it, could overheat and the fuel elements could degrade and possibly melt, resulting in the release of large quantities of radioactive materials into the reactor vessel.

It is primarily to the prevention of such **LOCA'**s and their potential consequences that nuclear reactor safety is addressed. To prevent **LOCA'**s from evolving into serious accidents with offsite releases of radioactive material, reactors must be designed to shut down quickly and reliably when necessary, and redundant cooling systems must be available to remove the heat that remains in the reactor core after the nuclear chain reaction has been shut down. In the event that such systems fail and fuel melting does occur, possibly allowing radioactive materials to breach the reactor vessel, most nuclear power plant designs include a large concrete containment building that would limit the release of radioactive materials to the environment.

This Section reviews the general principles behind nuclear power plant safety and describes the basic systems in a nuclear power plant designed to prevent or mitigate nuclear accidents. Subsequently, a discussion of the range of possible accidents that could occur is provided, the methodology for estimating the risk of severe nuclear power plant accidents is explained, and current estimates of such risk are given. Finally, the effectiveness of regulatory systems and management on safety, including the impact of human factors, are addressed. Most of this discussion focuses **on light water reactors**

(**LWRs**) – comprising **Pressure Water Reactors** (**PWR**) and **boiling water reactors (BWR**) -- which account for about 78% of the nuclear power plants in operation worldwide today (or 86% in terms of net MWe). The other major reactor types currently in use are gas-cooled reactors (9%), heavy water reactors (7%), graphite moderated light water reactors (5%) and liquid metal fast breeder reactors (1%).2 Safety aspects of new, advanced reactor designs are discussed in Section 6.0.

4.2. THERMAL POWER AFTER REACTOR SHUTDOWN

After the nuclear chain reaction ceases, radioactivity remaining in the fuel will generate heat as a result of radioactive decay. Assuming that the reactor had been operating for a substantial period, the power generated immediately after shutdown will be approximate 7% of the level before shutdown. For a 3000 MWth reactor, with 1000 MWe capacity, this implies and initial decay power level of about 200 MWth. Due to the rapid decay of short-lived species, this decay heat level decreases rapidly, but it is this heat that imposes the requirement that, in a light-water reactor, cooling water remain available to prevent damage to the fuel.

4.3. SAFETY PHILOSOPHY: DEFENSE IN DEPTH

The concept of "defense in depth" is the most fundamental principle underlying the safety of today's nuclear reactors. As stated in the International Nuclear Safety Advisory Group's basic safety principles, published in 1988, it centers on having "several levels of protection including successive barriers preventing the release of radioactive material to the environment." Defense in depth includes:

1. avoiding accident "precursors" that could lead to physical damage to the plant and to the various barriers to the release of radioactive material (accident prevention)
2. measures to:
   a. prevent accident precursors from evolving into accidents
   b. protect the public and the environment from harm in the event that accidents do occur and barriers to the release of radioactive material are not completely effective (accident mitigation).

There are five levels of protection under the defense in depth philosophy:

1. Conservative design, quality assurance, surveillance activities and a general safety culture. This combination is intended to ensure that the reactor and various plant components will operate with a high degree of reliability with only a small chance of malfunctioning.
2. Control of operations, including the ability to respond to abnormal, un-anticipated events or to any indication of system failure. Redundant instruments monitor the various operational process variables (such as the temperature of water as it leaves the reactor) and trigger automatic responses, such as shutting down the reactor, when necessary. An example of an abnormal event would be the loss of

off-site power to operate critical safety systems that could be needed in an emergency, which is compensated by having several backup electricity generators at the plant.

3. **Engineered safety features (ESF'**s) that halt the progress of accidents that are considered in the design and, when necessary, mitigate their consequences. The most extreme among the range of accidents considered in the design are termed **design basis accidents,** such as a major break in the primary coolant system that leads to **a LOCA. Severe accidents** that are beyond the design basis have a low probability of occurrence, but backup safety systems offer protection even if an accident progresses beyond the design basis assumptions.

   ESF's work in parallel or as backup to normal operating systems to safeguard essential safety functions: controlling reactor power, cooling the fuel, and confining the radioactivity within the reactor. An example of this is a system which automatically moves the control rods into the core to shut down the reactor in the event of an equipment malfunction in the core. Since the coolant water must still continue to circulate so as to prevent a heat buildup and possible fuel melting, another **ESF** provides emergency core cooling in the event that the primary coolant circuit has also been damaged. The containment structure around the reactor is another **ESF**, serving to prevent or limit the release of any radioactivity that is released from the reactor in an accident situation.

4. **Accident management strategies** to help operators decide quickly on appropriate actions. The aim of these measures is to prevent or limit damage to the core, preserve the integrity of the containment, and maintain the functions of design features such as vents and filters installed in the containment intended to preserve containment integrity in the event of a serious accident.

5. **Emergency planning**, which is intended to mitigate the radiological health consequences should an accidental release of radioactive material occur. Emergency planning includes early notification of an accident, radiation monitoring, decontamination, sheltering and/or evacuation of nearby residents within a specified radius of the plant, and possibly the administration of protective measures. (One such protective measure is distribution of potassium iodide tablets, which would block radioactive iodine that could be emitted in a nuclear accident from accumulating in the thyroid and increasing the risk of thyroid cancer.)

Another way of looking at defense-in-depth is in terms of the various physical barriers present in a nuclear power plant that serve to prevent the release of radioactive material. These physical barriers include the fuel pellets themselves; the fuel cladding, which seals the fuel pellets into the fuel rods; the boundary of the primary coolant system; and, in most reactor designs, the containment structure, which is a hermetically-sealed building designed to confine radioactivity that might otherwise escape because of the failure of other safety barriers.

The 1979 accident at Three Mile Island (TMI), in the United States, which is discussed in detail in Section 5.0, is an example of a major accident in a reactor leading to the breach of several barriers in which a serious release of radioactive material to the environment was prevented because the containment structure maintained its integrity.

4.4. ENGINEERED SAFETY FEATURES

Current light water reactors and most other kinds of nuclear power plants are designed to withstand rare but potentially very serious events. This includes the rupture of a major coolant pipe (which could, under circumstances in which additional safety systems fail, result in the complete loss of coolant from the reactor core) as well as other initiating events such as those identified above.

Protection against such events is provided by engineered safety features. The principal **Engineered Safety Feature (ESF**'s) in **PWR'**s are:

A. the emergency core cooling system

B. the containment building; systems that spray, clean and cool the containment atmosphere;

C. auxiliary feedwater systems which ensure continuous heat removal in the plant's steam generators; and emergency electric power sources.

Similar **ESF**'s are used for **BWR'**s. The function of the emergency core cooling system and the containment building are reviewed in this section. For the sake of brevity, descriptions of these **ESF**'s are provided for only the case of **PWR**'s, which is illustrative of the basic safety principles that also apply to **BWR's** as well as other designs. It should be noted that in **LWR'**s, water serves as both the moderator and the coolant.

In **BWR**'s, the heat generated in the reactor core turns the coolant water directly to steam. In contrast, in **PWR'**s the coolant water that passes through the core is under such high pressure that it cannot boil, but rather transfers its heat in a "steam generator" to a secondary coolant circuit that is maintained at a lower pressure, thus allowing the secondary coolant water to boil into steam.

A. Emergency Core Cooling System: Although early designs of **LWR**'s included systems for core cooling, they did not include the type of emergency cooling systems that could be shown to prevent fuel melting in the event of a loss of coolant accident. This was raised as a major concern in the 1960s by nuclear safety researchers in the US, who felt that such systems must be included in reactor designs to prevent what was labeled "The China Syndrome." This expression takes its name from a concept, initially spoken of in jest, which assumed that if the fuel in a reactor melted down it could burn its way through the bottom of the pressure vessel and concrete substrate and continue burning "all the way to China".

Extensive hearings were held on this subject by the Atomic Energy Commission and its successor agency the **Nuclear Regulatory Commission** (**NRC**). The

critics of nuclear power participated in these hearings (and challenged the adequacy of the requirements for emergency core cooling that were imposed as a result of these hearings). Today the **Emergency Core Cooling System** (**ECCS**) is seen as being critical to preventing the reactor core from overheating in the event of a **LOCA.**

However, **ECCS** adequacy continues to be questioned by certain nuclear power critics. If substantial melting were to take place in the core, large amounts of fuel would be deposited at the bottom of the reactor vessel, and, in the absence of restored cooling, could eventually melt through the reactor vessel. Even if such an extreme state is not reached, high temperatures and breach of the fuel cladding could drive off volatile fission products such as iodine, cesium and noble gases, which could then escape into the environment if there was also a breach of containment.

PWR's usually have three independent **ECCS** subsystems that operate at different system pressures. Each subsystem has multiple backups in terms of both equipment and flow path. If a small break in a **PWR'**s primary coolant circuit occurred, causing a moderate pressure drop, the **ECCS**'s high-pressure injections system would be activated to replenish the primary coolant lost through the break. A larger break in the coolant circuit would cause a rapid large pressure drop and would actuate the **accumulator injection system**. This subsystem ensures that large tanks of water containing boron are available to flood the reactor core. (The element boron is a neutron absorber and would ensure the cessation of fission reactions in the reactor core.) Finally, **the low-pressure injection system** would be actuated if pressure continues to drop below a preset level. This subsystem would continue to pump borated water from a large storage tank into the reactor long after the accumulators are empty, after which the subsystem automatically would switch to pumping borated water from the containment sump (into which excess water would have over-flown). The accumulator injection system is a passive system; it does not require the operation of pumps and valves, which are dependent on a power supply. The high- and low-pressure injection systems, on the other hand, require active operation, i.e. the presence of a power supply.

B. **Containment Systems: Containment systems** are intended to hold essentially all the steam and radioactivity that might be released from the reactor vessel in a **LOCA**, and are provided for all **LWR'**s as well as most other reactor types. The exception is a number of earlier design reactors in the Soviet Union, discussed later in this chapter, which have a less effective containment system. A typical **PWR** containment structure is made of reinforced concrete, over one meter in thickness, with an internal steel liner. The entire primary coolant circuit, including the reactor vessel, is enclosed in the containment structure. The containment structure is capable of withstanding the maximum temperature and pressure that could be expected if all the water in the primary coolant circuit was expelled into

the containment building as steam. Various systems inside the containment building would be available to cool and clean up the containment atmosphere.

One of these involves spraying water on to the steam, or passing the steam over ice beds, to cause it to condense. In addition, sodium hydroxide may be introduced to the containment atmosphere through the containment spray system in order to remove radioactive iodine; similarly, the ice inventory may include chemical substances that would remove fission products from the containment atmosphere. Filters are also used to remove iodine and particulate matter. Radioactive noble gases, krypton and xenon, cannot be removed, but holding them in the containment building for a certain period allows them to decay substantially into non-radioactive species before they are released to the environment.

Unfortunately, when a serious accident occurred at the Chernobyl-4 reactor in the Soviet Union in 1986, the containment provided at the plant was incapable of containing the accident that occurred. A graphite-moderated light water reactor of the Soviet **RBMK** design, Chernobyl-4 was equipped with a concrete structure capable only of partial confinement of radioactive gases. Thus, the structure was unable to effectively contain the accident and massive graphite fire at the reactor. There are approximately 20 **RBMK** units currently operating in the Soviet Union. Details of the **RBMK** reactor and the Chernobyl-4 accident are described in Section 5.0. A number of the Soviet **PWR**s (**WER'**s) do not have containment either. These Soviet PWRs do not comply with the guidelines established by the **International Atomic Energy Agency** (**IAEA)** and the regulatory agencies in Western countries and would not be licensed in most countries in the world. There are, moreover, several such reactors outside the Soviet Union: four in former East Germany (which have been shut down due to safety concerns), four in Bulgaria and two in Czechoslovakia

4.5. TYPE AND SEVERITY OF NUCLEAR ACCIDENTS

A wide range of incidents and accidents can be postulated in nuclear power plants, from minor incidents, such as when a specific operating procedure is not followed or a non-safety related piece of equipment malfunctions, to a variety of serious accident scenarios that are conceivably possible. The **Nuclear Regulatory Commission (NRC)** as well as the **IAEA** and the **Nuclear Energy Agency** of the **Organization for Economic Cooperation and Development** (**OECD/NEA**) have, within the past few years, developed accident severity scales in order to facilitate communications and understanding among nuclear personnel, the media and the general public through the use of a simple classification system.

The French adopted an accident severity scale in April 1988 which includes six progressive categories, with the least important incidents designated level 1 and the most serious accidents being designated level 6. The levels are distinguished by the risk of radioactive discharge outside the installation where an accident has occurred. Lesser

incidents are those in which radioactive discharges are below the allowed annual limits, or in which operating difficulties have occurred that do not impose directly a radiation risk but which may reveal weaknesses that need to be remedied.

An accident severity scale has also been developed and used in Japan. It is graduated into nine levels, from 0 to 8, and assesses event severity in terms of three criteria: effect on the public; effect on personnel; and effect on reactor safety (impact on integrity of "defense in depth," etc.). All reactor events that are reportable to the regulatory authority can be evaluated on this scale, including minor events having no effect on plant safety.

Drawing from the experience in using severity scales in France and Japan and proposals to use such scales in other countries, the **IAEA** and the **NEA** convened a series of meetings of experts and developed the **International Nuclear Event Scale** (**INES**).

The **International Nuclear Event Scale** (**INES**) is a tool to promptly and consistently communicate to the public the safety significance of reported events at nuclear installations. By putting events into proper perspective, the Scale can ease common understanding among the nuclear community, the media and the public. The group was guided in its work by the findings of a series of international meetings held to discuss general principles underlying such a scale. Initially applied for a trial period to classify events at nuclear power plants, 32 countries participated in the trial and international agencies, and user countries monitored progress. The Scale operated successfully and now has been made available for formal adoption by each country.

The **Scale** also has been extended and adapted to enable it to be applied to all nuclear installations associated with the civil nuclear industry and to any events occurring during the transport of radioactive materials to and from those facilities. Events are classified on the Scale at seven levels. Their descriptors and criteria are shown below with examples of the classification of nuclear events which have occurred in the past at nuclear installations. The lower levels (1-3) are termed **incidents**, and the upper levels (4-7) **accidents**. Events which have no safety significance are classified as level 0 / below scale and are termed deviations. Events which have no safety relevance are termed out of scale.

Although the same scale is used for all installations, it is physically impossible for events to occur which involve the release to the environment of considerable quantities of radioactive material at some types of installation. For these installations, the upper levels of the scale would not be applicable. These include research reactors, unirradiated nuclear fuel treatment facilities and waste storage sites. Industrial accidents or other events which are not related to nuclear or radiological operations are not classified and are termed "out of scale". For example, although events associated with a turbine or generator can affect safety-related equipment, faults affecting only the availability of a turbine or generator would be classified as out of scale. Similarly, events such as fires are to be considered out of scale when they do not involve any possible radiological hazard and do not affect the safety layers.

The Scale is not appropriate as the basis for selecting events for feedback of operational experience, as important lessons can often be learnt from events of relatively minor significance. It is not appropriate to use the Scale to compare safety performance among countries. Each country has different arrangements for reporting minor events to the public, and it is difficult to ensure precise international consistency in rating events at the boundary between level 0 and level 1. The statistically small number of such events, with variability from year to year, makes it difficult to provide meaningful international comparisons. Although broadly comparable, nuclear and radiological safety criteria and the terminology used to describe them vary from country to country. The INES has been designed to take account of this fact.

The scale is intended to be more or less logarithmic, so that each successively higher level on the scale should correspond to about a tenfold drop in the number of events. As examples of how past events have been classified in INES, the 1986 Chernobyl accident, which had widespread environmental and health effects, has been classified as Level 7. The 1979 TMI accident severely damaged the reactor core but had very limited offsite consequences and it was classified as Level 5. The Windscale accident, in which there was a significant external release of fission products, has been classified as Level 5.

4.6. PROBABILISTIC RISK ASSESSMENT

Whether consciously or unconsciously, individuals and institutions use probability and risk assessment in everyday decision making. Financial institutions, for example, use probability in decision making to determine whether a borrower will be able to repay a loan. In mathematical terms this can be written as p(R\H) where p is "the probability of' and R is "the loan will be repaid" and H is "the borrower's past financial history, assets, liabilities etc". The symbol \ denotes "given." Risk is a commonly used word that conveys a variety of meanings to different people but is defined in the dictionary as "the possibility of loss or injury." In the example quoted above, risk is also an element in the decision process and for a bank represents the probability of a given loan not being repaid.

Risk is defined mathematically as the product of the probability of an outcome times the consequences of this outcome. Thus by combining probability (or likelihood) assessments with risk analysis, a financial institution can minimize the adverse outcomes from loan non-repayment. The same principles of risk analysis (probability and consequence) can be used to evaluate an individual's estimated lifespan, airline accidents, nuclear and chemical plant accidents, etc. Regardless of the evaluation being performed, the mathematical laws of probability and the utility assessment of risk are the sample.

In recent years, scientists and engineers working on the development and construction of new nuclear reactors have turned increasingly to the use of **probabilistic risk assessment** (**PRA**) as a tool to help estimate the likelihood and consequence of accidents in the facility that could lead to financial losses and personal injury on- and off-site. The first major application of **PRA** to nuclear reactor safety was made by Professor

Norman Rasmussen of the Massachusetts Institute of Technology (MIT) and colleagues who performed the well known Reactor Safety Study. The study pointed out that a major element in the characterization of the radioactive releases associated with potential nuclear power plant accidents is the identification of the accident sequences that can potentially lead to risks to the public.

Current **PRA** analyses of nuclear power plants utilize what is called "event tree methodology." An event tree is a logic method for identifying the various possible outcomes of a given event called the initiating event. (This technique, which is known in business circles as the application of decision trees, is widely used in many business applications where the initiating event is a particular business decision and the various outcomes depend on subsequent decisions.) In nuclear reactor safety the initiating event is generally a system failure and the subsequent events are determined by the characteristics of the reactor system and the engineering.

Use of PRA enables nuclear engineers to identify and, as a consequence, rectify prior to construction any weaknesses in a system which could lead to system failure and possible release of radioactive materials into the biosphere. It also facilitates prioritizing possible safety improvements that could be made in terms of their degree of safety significance. **PRA** also enables an estimate to be made of the probability of a serious accident occurring when a number of reactors are operating over a long period of time. **PRA** also helps demonstrate whether a plant meets "safety goals" which the **NRC** and other national regulatory bodies have developed in recent years, as discussed in the next section.

4.7. SAFETY GOALS

After the **TMI** accident in 1979, the **NRC** decided to develop an explicit policy statement on safety philosophy and the consideration of costs in **NRC** safety decisions. The agency proceeded to develop a policy statement on safety goals for nuclear power plants, issuing interim safety goals in 1983 and final goals in 1986. The final goals provided both qualitative and quantitative goals:

1. Nuclear power plant operation should not impose significant additional risk to an individual's life and health;
2. Societal risks should be comparable to or less than the risks of generating electricity by viable competing technologies;
3. The risk of prompt fatalities to the average individual in the vicinity of a power plant should not exceed 0.1% of all prompt fatality risks from other accidents in the US; and
4. The societal risk of cancer fatalities to the population near a power plant should not exceed 0.1% of all cancer fatality risks from other causes. In addition, **NRC** stated that, as a guideline for implementing these goals, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 10i per reactor-year of operation.8

Subsequently NRC decided that a core damage probability of less than 104 per reactor-year as a subsidiary goal could be useful in evaluating regulations related to accident prevention.

Other countries have since developed safety goals comparable to the **NRC**'s, and the **AIEA** is now developing international guidance on the establishment of national safety goals.

4.8. REGULATION OF NUCLEAR POWER AND ITS EFFECTIVENESS

Regulatory authorities serve on behalf of the government to license, regulate and oversee the safe operation of nuclear facilities and the safe use of nuclear materials. Such organizations are responsible for developing safety standards and regulations; conducting reviews of license applications against those standards and regulations and taking licensing actions; monitoring the operations of licensed facilities to ensure their continued safety; taking necessary enforcement measures where safety levels are not met; conducting safety research programs and providing safety and licensing-related information to the public. Regulatory organizations need to have sufficient resources and technical expertise to carry out their responsibilities, as well as the necessary legal authority and free access to facilities and information.

The approach to fulfilling this regulatory responsibility varies substantially among countries who have licensed nuclear power plants. Some countries -- most notably the United States - - follow a highly prescriptive approach, under which detailed technical regulations have been developed as well as detailed guidance on complying with these regulations. Demonstration of compliance with the regulations is accepted as a demonstration that overall standards of safety will be achieved. Other countries are less prescriptive and require the licensee to demonstrate only that broad safety requirements will be met. In either case, the regulatory authority ultimately requires plants to comply with detailed operating specifications -- or "technical specifications" -- limiting the plant's conditions of operation. Also, in either case, the burden of proof that a proposed facility will not have an adverse impact on public health, safety and the environment falls on the licensee. While regulators should maintain an arm's length from the industry they regulate, experience suggests that regulatory organizations should also work cooperatively with industry rather than in an adversarial manner. The **NRC** has been accused in the past of taking too adversarial an approach towards license applicants and unnecessarily causing cost increases.

At the same time, the **NRC** has been accused of not maintaining sufficient independence from industry. For example, the Union of Concerned Scientists, a leading nuclear power critic in the US, wrote in 1987 that "nuclear power is an inherently dangerous technology requiring the highest standards of care and performance." They fault the NRC for "indifference and shortsightedness [which] have allowed so many generic technical problems to persist for so long," and state that "NRC's primary and instinctive allegiance is still to the industry it regulates... Congress must assume a more assertive oversight role to see that the **NRC** lives up to its safety-first mandate.""

The organizational relationships between nuclear power proponents and regulators differ from country to country. In the United States, the Atomic Energy Commission was responsible for both promotion of nuclear power and regulation thereof until 1974, when the regulatory functions were separated out and given to the new **Nuclear Regulatory Commission. NRC** is organizationally completely separate from its licensees. In contrast, the French **Service Central de Surety des Installations Nucleaires** (**SCSIN**) and its sole reactor licensee **Electricite de France** (**EdF)** are both part of the Ministry of Industry. **SCSIN** is now moving in the direction of independence from **EdF**. However, there is some perception that this move is intended primarily to please the general public and was not necessary to ensure regulatory independence, as well as apprehension that this could lead to power struggles and adversarial relations as in the US. Another example is the UK, where the Nuclear Installations Inspectorate (NII) is part of a larger agency which regulates industry in general, the Health and Safety Executive (HSE), but is completely separate from the industries it regulates (including utilities).

The **International Nuclear Safety Advisory Group's** (**INSAG**) basic safety principles recommend a clear separation between the responsibilities of the regulatory authority and other organizations, so that the regulators retain independence as a safety authority and are protected from undue pressure. **INSAG** also believes this will ensure that safety is the only mission of the regulatory personnel.

With respect to future applications to construct power plants of advanced designs, there is increasing support in the **US** and elsewhere for modifications in the nuclear plant licensing procedure to introduce greater licensing efficiency as well as preserve or possibly improve safety levels. In the US, one of the key proposals is to streamline the licensing process by issuing a combined construction and operating license in a single step rather than the approach followed in the past in which these licenses were issued separately and as the result of separate licensing proceedings.

Standardized designs for certain advanced reactors are now being reviewed by the **NRC**, which will decide whether to certify that these designs are acceptable for referencing in subsequent utility license applications. This pre-approval of designs could make the licensing process more predictable by removing most design questions from the process. Pre-approval of possible power plant sites is also gaining support as it would expedite the licensing schedule.

Regulatory considerations are very relevant to the question of building nuclear power plants in developing countries. Government authorities in developing countries would require adequate resources and capabilities to review and evaluate nuclear power plant license applications and to oversee safety during plant operations. As noted above, there are no hard and fast rules regarding the organization of national regulatory authorities, except that the authority should be organizationally independent of the regulated industry and competent.

4.9. EFFECT OF MANAGEMENT ON SAFETY

A study undertaken by the MIT's Nuclear Engineering Department compared nuclear operating experience in the major nuclear power countries to understand why **US** plants have been consistently outperformed (i.e., in terms of plant availability) by their foreign counterparts.' The study found that managerial reforms are the key to improving US plant performance, rather than changes in the environment within which these plants are operated including US regulatory zeal, diverse plant ownership patterns, and financial regulation by the states, factors which have been widely blamed for the poor performance of US plants.

Industry wide cooperation between utilities, suppliers and regulators has started late in the US and there is still deep distrust between utilities and regulators as well as competition among suppliers, the authors found. They also suggest that utilities that show consistently good results operate with a large degree of managerial involvement in day-to-day activities. Investing in a plant's intellectual resources through training programs and staff exchanges with other organizations can foster an "esprit de corps" that benefits plant operations. Good management can be expected to benefit plant safety as much as plant performance. **INSAG'**s basic safety principles include guidelines with respect to safety culture and responsibility of the operating organization. With respect to safety culture, they state that:

The starting point for the necessary full attention to safety matters is with the senior management of all organizations concerned. Policies are established and implemented which ensure correct practices, with the recognition that their importance lies not just in the practices themselves but also in the environment of safety consciousness which they create...

These matters are especially important for operating organizations and the staff directly engaged in plant operation. For the latter, at all levels, training emphasizes the significance of their individual tasks from the standpoint of basic understanding and knowledge of the plant and the equipment at their command, with special emphasis on the reasons underlying safety limits and the safety consequences of violations.

With respect to responsibility of the operating organization, the **INSAG** principles state that: Once the operating organization accepts possession, it is in complete charge of the plant, with full responsibility and commensurate authority for approved activities in the production of electric power. Since these activities also affect the safety of the plant, the operating organization establishes policy for adherence to safety requirements, establishes procedures for safe control of the plant under all conditions, including maintenance and surveillance, and retains a competent, fit and fully trained staff.

4.10. HUMAN FACTORS

One factor believed to contribute to anxiety about the safety of nuclear power plants is the possibility that accidents might be caused or aggravated by human error. Human error can occur at many stages in the design, manufacture or construction of a nuclear power plant. It can also be crucial in the operation and maintenance of power plants.

Human error has contributed to many past events and was chiefly responsible for the accident at TMI, in which an operator shut down the **ECCS** even though the reactor core was being uncovered, because he had faulty information (design flaws also contributed to the accident). Human error combined with the application of inappropriate operating procedures caused a chain of uncontrollable events which led to the disaster at the Chernobyl Power Station.

Judgments involved in the subject known as "human factors" are extremely complex. For example, the role of the operators of a nuclear power plant is far more than a mechanical one. Plant operators must have knowledge and understanding of the plant which they will be required to apply, in conjunction with the plant's automatic control system, to ensure that the plant operates reliably and safely.

Human factors has attracted a great deal of attention in the nuclear industry since the TMI accident. Improvement in control room designs is one major benefit that has resulted from application of this science to nuclear power plants. However, it is clear that in addition to human factors, good management is a key element of accident prevention. To the extent that accidents caused by human error reflect the shortcomings of the management system, efforts to correct defects in organization, training, or procedures will lead to commensurate gains in plant safety level. It is clearly essential that utility managers at the highest level make a high priority of nuclear safety and allocate sufficient funds for safety-related activities, including human factors.

SECTION FIVE: MAJOR NUCLEAR ACCIDENTS

Although there have been many minor incidents at commercial nuclear energy facilities, there have in fact been only three major accidents to nuclear power plants since the development of nuclear energy began in the late 1940's. The three accidents, which occurred at Windscale in England, Three Mile Island in the USA and Chernobyl in the Soviet Union, were so fundamentally different in kind from each other that a description of each is considered to be worthwhile.

5.1. THE WINDSCALE ACCIDENT

In the early 1950's, the United Kingdom decided to go ahead with the development of nuclear weapons for defense purposes. In order to proceed on this route the UK required a supply of weapons-grade plutonium which it decided to manufacture in a number of weapons-material production "piles" (the original name for nuclear reactors) which it built at Windscale on the Northwest coast of England.

The Windscale Pile (as it is known) was one of the first reactors ever to be built and was, by today's standards, a very primitive type of system. The fuel utilized was natural uranium in the form of metal rods clad in a special alloy made from magnesium and aluminum, chosen because of its low neutron absorption characteristics. The moderator was high purity graphite and the whole system was cooled by air. The reactor ran at a relatively low temperature.

5.1.1. WIGNER STORED ENERGY

At the time these reactors were built very little was known about the effects of bombardment of graphite at low temperatures resulting in the production of so-called "defects" in the internal structure of the graphite due to the carbon atoms being knocked out of their normal positions in the graphite lattice. These displaced atoms are capable of returning to their normal positions again at which time their stored energy (known as Wigner stored energy after the name of Eugene Wigner who discovered it) is released in the form of heat.

Although the phenomenon of Wigner stored energy had been known for some time when the reactors were built, there was very little knowledge about how such stored energy might be released. It was known that stored energy builds up progressively with irradiation and the rate at which it accumulates is temperature dependent, virtually no energy being stored above 400°C, and that a spontaneous release of stored energy could occur as happened in the Windscale No. 1 Pile in 1952, while the pile was shut down, but without any harmful effects.

As a result of this experience procedures were instituted for the controlled release of this stored energy by allowing the chain reaction in the reactor core to commence without coolant airflow thus raising the graphite and uranium temperatures and starting the so-called Wigner energy release in the graphite. Under these conditions the release becomes self-sustaining. Eight such releases of stored energy had taken place by the end of 1956 but it had been found difficult to release energy in all the graphite in the pile and on three occasions a second heating was found necessary.

5.1.2. THE WINDSCALE FIRE

In October 1957 the No. 1 pile was shut down and a Wigner energy release was started. After some hours the nuclear heating was stopped as planned but the temperature of the graphite appeared to the plant operator to be dropping rather than increasing. Consequently the pile operator decided to boost the release with a second nuclear heating. During this second heating a rapid rise in temperature of the uranium cartridges was observed at which time the control rods were again inserted to reduce the power. As a result of this second heating the graphite temperatures rose rapidly, leading to oxidation of the uranium which had been exposed by the overheating.

This gradually led to the failure and combustion of other uranium cartridges and subsequently to combustion of the graphite itself, all of which was exacerbated by the introduction of air into the pile in attempts to cool it. Over the next day a number attempts were made to cool the pile but without effect and eventually it became necessary to couple water hoses to the top of the pile and to flood the affected channels. This technique proved successful and after 24 hours the pile was cold.

5.1.3. HEALTH CONSEQUENCES OF THE ACCIDENT

Examination of the workers revealed that fourteen had received exposures higher than normal during the accident but even the highest exposure was only *50%* above the **ICRP**

safe continuous level. Moreover, the highest level of Iodine-131 measured in the thyroid gland was very small and well below the level at which harm could be done.

The exhaust cooling gases from the Windscale piles were normally fed to the atmosphere through tall stacks equipped with filters to trap radioactive particulate matter. During the fire these filters worked adequately and it was subsequently found that no harmful amounts of plutonium or any other elements had been released with the exception of Iodine-131, a radioactive isotope of iodine. The risk from inhalation of radioactive materials was found to be as insignificant outside the factory as it was inside and no restrictions were placed on the consumption of vegetables, eggs, meat and water in the area as a consequence.

However a problem arose due to the deposition of the radioactive iodine on the grass under the area over which the plume from the chimney passed and which was eaten by the cows from the local farms. Since iodine tends to concentrate in the milk, the potential danger to young children and others drinking milk from cows which had eaten the grass on which the radioactive iodine had deposited becomes apparent. As a precaution, therefore, milk deliveries from twelve milk producers within a two-mile radius of Windscale were stopped for a time until the levels of radioactive iodine had reduced to an acceptably low level. The special Committee from the Medical Research Council which was set up after the fire to investigate the health consequences of the accident concluded: "After examining the various possibilities, we are satisfied that it is in the highest degree unlikely that any harm has been done to the health of anybody, whether a worker in the Windscale plant or a member of the general public".

5.1.4. CAUSE OF THE ACCIDENT

Following the report by the Committee of Inquiry, the Chairman of the Atomic Energy Authority (then Sir Edwin Plowden) wrote in a Memorandum that the cause of the accident was attributable to inadequacies in the instrumentation provided for the operation of the Wigner energy release and to faults of judgement by the operating staff, which themselves were attributable to weakness of organization (the Atomic Energy Authority).

5.1.5. COMMENTARY

This particular accident occurred in a type of reactor which no longer exists since the Windscale piles which were used exclusively for the production of military materials were, in fact, never restarted after the 1957 fire. The gas-cooled, graphite moderated power reactors (Magnox and AGR) in operation in the United Kingdom today run at much higher temperatures than the Windscale piles and do not require the periodic release of Wigner stored energy. Moreover, they utilize carbon dioxide as a coolant, which is inert and would not lead to increased combustion in the event of an overheating incident. The accident itself led to a number of organizational changes in reactor management and in particular to the realization of the need for close liaison between the management of the reactor site and the local interests. These changes have led to the tight controls which are in place today in the United Kingdom.

5.2 . THREE MILE ISLAND ACCIDENT

In March 1979, the No. 2 unit at the Three Mile Island (TMI) nuclear power station in Pennsylvania, USA, suffered a severe core degradation accident, the only one of its kind to happen to any **pressurized water power reactor** (**PWR**) in the world to date. In order to understand what happened during the accident, a brief description of the plant is necessary.

TMI-2 is a pressurized water reactor with three interrelated cooling circuits. Heat generated by the reactor core is transferred to water circulating in the primary circuit which is under high pressure (about 2200 psi) to keep it from boiling. The heat from the primary circuit is transferred to a secondary circuit by means of two steam generators which produce the steam for the steam turbine which drives the electricity generator. After passing through the turbine the steam is condensed back to water by a third circuit which circulates water between the condenser and the cooling towers.

Under normal operating conditions, essentially all radioactivity is contained within the uranium oxide fuel pellets, and the fuel cladding tubes which are made from zirconium alloy that resists corrosion and high temperatures. In the event that fission products escape through the fuel cladding, such as through defects, these are trapped in the primary coolant from which they can be removed in the reactor purification system. However, krypton and xenon do not readily dissolve in water, particularly at high temperatures and collect as a gas above the coolant when the system is depressurized. The core of the reactor is encased in a pressure vessel which is a 36-foot high tank with steel walls about nine inches thick. The reactor pressure vessel and the remainder of the primary coolant system, which includes the pressurizer, steam generators and associated piping, are contained in the reactor (or containment) building. The containment building has steel-lined thick concrete walls and is the final barrier to the outside environment. The auxiliary building is located close to, but external to, the containment building. During the TMI-2 accident, radioactivity was released to the environment when radioactive liquids were pumped from the reactor building to this auxiliary building.

5.2.1. EVENTS CONTRIBUTING TO THE ACCIDENT

On the day of the accident, a malfunction occurred to components that maintain the flow of coolant water to the steam generators in the secondary loop. This resulted in a loss of ability to remove heat from the primary loop with the result that most of the heat generated by the reactor remained in the reactor vessel and primary loop. This caused the coolant water temperature and pressure to increase rapidly which, in turn, caused a relief valve on the pressurizer to open allowing steam and water to discharge to the reactor coolant drain tank located in the basement in accordance with design procedures.

The drain tank is equipped with a pressure-limiting rupture disc. As there was no valve position indicator for the pressure relief valve clearly visible in the control room, the fact that the pressure relief valve was open was not deduced by the operators for more than

two hours during which time water continued to be discharged through the valve into the drain tank. As the reactor pressure continued to fall due to the open pressure relief valve and resultant loss of primary coolant, the high pressure safety injection system (which is part of the emergency core cooling system) began automatic operation as intended.

This system was twice cut off or reduced in flow manually by operators who interpreted instrument readings to indicate that water level in the reactor was adequate. As coolant inventory declined due to continued loss of coolant through the open relief valve and the cutback of the high pressure injection system, a number of flow anomalies developed to which the operators responded in varying ways. During this period, significant fractions of the core became uncovered for extended periods and core damage resulted. Voids were created in the system, preventing natural circulation cooling and interfering with forced circulation, which was finally reestablished in one of the two loops of the system.

So much water and steam were discharged through the relief valve that the storage capacity of the drain tank was quickly exceeded, causing the rupture disc to burst, allowing some 250,000 gallons of radioactive coolant to be discharged into the reactor building sump and basement. Radioactive coolant water in the reactor building sump was then automatically pumped into the sump tank in the auxiliary building which was already about half full. Consequently, much of the water spilled into the auxiliary building, which was not designed to contain radioactive material. This liquid did not contain significant amounts of radioactivity, however, because major fuel damage did not occur until about two hours later.

After fuel damage occurred, radioactive materials were transported through the primary coolant system via the letdown line to the makeup and purification system in the auxiliary building. Because this liquid was a stream of primary coolant directly from the reactor, it contained significant amounts of radioactivity. As a result of liquid leaks in the makeup and purification system, large amounts of radioactive material were released into the auxiliary building. No longer held under pressure, krypton, xenon and other volatile radio-nuclides evolved from the water into the auxiliary building atmosphere.

In one of the least expected and most highly publicized facets of the incident, the upper section of the reactor pressure vessel became occupied by hydrogen formed by reaction of primary coolant water with overheated zircaloy cladding when the core was partially exposed. A portion of this hydrogen escaped into the reactor containment building with the water vented through the pressure relief valve, and this hydrogen ignited at 10 hours after initiation of the incident, resulting in a containment pressure spike of 28 psi. This hydrogen ignition or explosion was unreported for some time to both the **NRC** and the press, but for several days, wide and sensational publicity was given to the presence of the hydrogen "bubble" in the reactor, and to the possibility of its explosion, the risk of which was nonexistent, since no free oxygen could be present in the gas under the conditions in the reactor. Removal of the hydrogen from the system became, in press reporting, one of the most dramatic and risky aspects of the incident, with attention focused on the possibility of explosion in the containment building as hydrogen released

-o the containment atmosphere built up. 'This was avoided by activation of a catalytic recombination which kept the hydrogen concentration below combustible limits.

5.2.2. RADIOACTIVITY RELEASE TO THE ENVIRONMENT

During the accident, approximately 50 percent of the noble gases and particulate cesium, 30 percent of the iodine and small quantities of other fission products normally present in irradiated fuel were released from the damaged fuel into the primary coolant water. Before being released into the environment, the small amount of the airborne radioactivity released to the reactor building was filtered and monitored.

The highly efficient filtration system in the auxiliary and fuel handling buildings was designed to remove more than 99 percent of radioactive cesium, strontium and alpha-emitting radio-nuclides, In addition to mechanical filtration, ventilated air in these buildings was also passed through multiple charcoal filters, which chemically removed 90 to 95 percent of the radioactive iodine.

While offsite radiation levels never exceeded about 35 mrem/hr, and total exposure to any individual in the areas of highest activity are estimated not to have exceeded 80 mrem, compared to a typical background exposure of around 300 mrem/yr per person, the presence of offsite radioactivity and the exaggerated threat of its massive release from a possible breach of the containment due to possible hydrogen explosion, coupled with sensational media reporting, generated intense nationwide concern for over a week.

The prospect of a large-scale evacuation of the surrounding population and the planning for that contingency further aroused public concern. On the third day, evacuation of pregnant women and children under six years of age residing within five miles of the plant site was officially recommended by the governor of Pennsylvania. General evacuation was never ordered (although the possibility of such evacuation was openly discussed by the press and officials at every level, including President Carter) but an estimated 80,000-200,000 residents of the area voluntarily left their homes. Most returned shortly after Governor Thornburgh advised some days later that it was safe to do so.

5.2.3. POTENTIAL HEALTH EFFECTS FROM THE ACCIDENT

It was the release of radioactivity from the plant and the appearance of detectable amounts of radiation beyond the plant boundaries which led to the most serious public and media reactions even though radiation exposure and contamination never reached significant levels from the standpoint of health. Traces of radioactive iodine were detected authorities in some milk samples, but at levels so low that none was ever removed from the market. It has since been estimated that cumulative exposure from the incident could result in one additional cancer death, one added nonfatal cancer, and one additional birth defect over the next 25 years among the two million people within 50 miles of the facility. These two million people are statistically expected to suffer 325,000 cancer deaths from natural causes other than the T'MI accident.

Although there have been many allegations of increased leukemia and other cancers in people living in the area, particularly young people, studies by the Pennsylvania

Department of Health have not revealed incidences greater than normal. These findings have, moreover, been confirmed by a team of independent epidemiologists who have been studying the allegations for the TMI Health Fund and whose results were published in September 1990.

Nevertheless, one of the major concerns emerging during the period of the accident was the psycho-behavioral impact on local residents. The Pennsylvania Department of Health has reported that for some months after the accident many local residents suffered from severe distress.

5.2.4. COMMENTARY

The Three Mile Island accident was caused by a combination of human error and system malfunction. It resulted in the degradation and partial melting of the reactor core and the total loss of the reactor which is still being decontaminated. Nevertheless, in spite of the enormous media attention given to it at the time, the safety features engineered into the system prevented the release of all but trivial amounts of radioactivity into the biosphere.

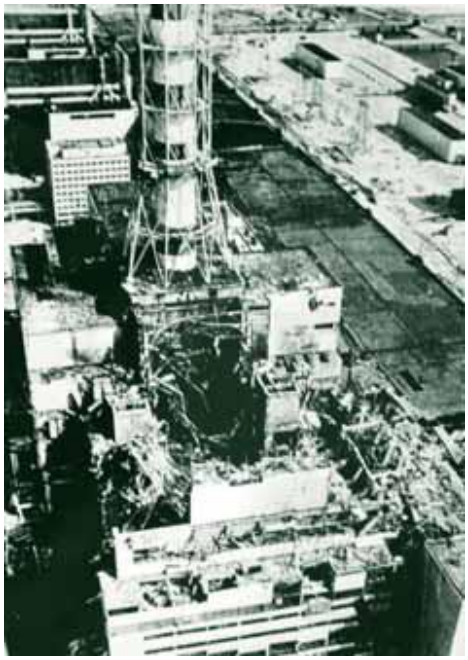
The accident had a big effect on the industry and also on the regulators since it clearly pointed up the defects in operator training and in some of the engineered safety features of the reactor system which had hitherto been considered adequate. As a result, the industry set up it own "watchdog", the **Institute for Nuclear Power Operations** (**INPO**), which is independent of the **Nuclear Regulatory Commission. INPO** oversees the operations of the industry and will take action as and when required to ensure industry compliance with good practice.

For its part, the **NRC** required a number of back-fits to be carried out on existing reactors to ensure that future incidents of this kind have a very low probability of happening. The **NRC** also instituted a program for reactor operator training which is designed to improve the quality of performance of future operators. Most of these improvement have since been implemented by overseas operators of light water reactors as part of the overall safety improvements of reactors all over the world.

5.3. THE CHERNOBYL ACCIDENT

PHOTOGRAPH OF FOUR REACTORS AT CHERNOBYL

CLOSE-UP PHOTOGRAPH OF REACTOR #4 AT CHERNOBYL



On April 26, 1986, the worst accident in the history of commercial nuclear power generation occurred at the Chernobyl Nuclear Power Station some 60 miles north of Kiev in the Ukraine. The accident caused extensive damage to the reactor and the building which housed it; some 31 people died as a result of the fire and explosion, or as a result of receiving lethal radiation doses.

A significant release of fission products occurred, contaminating the land around the station and requiring the evacuation of around 135,000 people from their homes. The radioactive cloud generated by the accident over many days was carried by winds all over Europe and led to restrictions on the consumption of meat and vegetables which became contaminated from it. Although the latent health effects may not be statistically significant when viewed against the normal mortality rate over the next 40 years, nevertheless the accident has had a big impact on public concern about nuclear safety. It is, therefore, desirable to provide a brief description of the reactor and the events which contributed to the accident.

The accident destroyed the Chernobyl 4 reactor, killing 30 operators and firemen within three months and several further deaths later. One person was killed immediately and a second died in hospital soon after as a result of injuries received. Another person is reported to have died at the time from a coronary thrombosis. Acute radiation syndrome (ARS) was originally diagnosed in 237 people on-site and involved with the clean-up and it was later confirmed in 134 cases. Of these, 28 people died as a result of ARS within a few weeks of the accident. Nineteen more subsequently died between 1987 and 2004 but their deaths cannot necessarily be attributed to radiation exposure. Nobody off-site suffered from acute radiation effects although a large proportion of childhood thyroid cancers diagnosed since the accident is likely to be due to intake of radioactive iodine fallout. Furthermore, large areas of Belarus, Ukraine, Russia and beyond were contaminated in varying degrees.

The Chernobyl disaster was a unique event and the only accident in the history of commercial nuclear power where radiation-related fatalities occurred. However, the design of the reactor is unique and the accident is thus of little relevance to the rest of the nuclear industry outside the then Eastern Bloc.

The Chernobyl Power Complex, consisted of four nuclear reactors of the RBMK-1000 design, units 1 and 2 being constructed between 1970 and 1977, while units 3 and 4 of the same design were completed in 1983. Two more RBMK reactors were under construction at the site at the time of the accident. To the southeast of the plant, an artificial lake of some 22 square kilometres, situated beside the river Pripyat, a tributary of the Dniepr, was constructed to provide cooling water for the reactors.

This area of Ukraine is described as Belarussian-type woodland with a low population density. About 3 km away from the reactor, in the new city, Pripyat, there were 49,000 inhabitants. The old town of Chernobyl, which had a population of 12,500, is about 15 km to the southeast of the complex. Within a 30 km radius of the power plant, the total population was between 115,000 and 135,000.

5.3.1. EVENTS LEADING TO THE ACCIDENT

Ironically the immediate cause of the accident was an experiment designed to improve the safety of the plant. The objective of the experiment was to test the turbo-generator's ability to provide in-house power after shutting off its steam supply for the short time needed for the emergency diesels to start and come online, nominally 40 to 50 seconds, and required the reactor to be at about 25% full power.

This test had been attempted twice before in 1982 and 1984, on which occasion it was found that the voltage output decreased faster than desired and the purpose of the test was to verify proper operation of a new voltage regulator design for the generator. In the subsequent inquiry into the causes of the accident, it became clear that the experimental test had been badly planned, that the safety case had been inadequate, and that the operators had departed from laid down operating procedures and had violated several operating rules.

The test procedure itself called for turning off the emergency core cooling system. Operator actions included disconnecting the signal that automatically shuts down the reactor when two turbo-generators. are disconnected, operating the main coolant pumps in a regime where cavitation might occur, turning off various protection system signals, and operating with less than the minimum required number of inserted control rods. Power reduction to the test power level of 700-1000 MWt began but was halted while the operators disconnected one of the two turbo-generators from the reactor. Four main cooling pumps, and two feed-water pumps were connected to the turbo-generator to be run down. The operators also disabled the signal which results in automatic reactor shutdown when both turbo-generators are disconnected. This action was intended to permit rerunning the test if needed - but the test procedure did not call for disabling this emergency system.

In addition, the operators disabled the emergency core cooling system, but this was done in accordance with test procedure. Before power reduction could continue, the grid controller requested the operators to hold power and not continue with the test. In complying with this request a further violation of normal plant operating procedures occurred since continuous operation at power with the emergency core cooling system disabled is a violation.

Following the delay, the operators continued the power descent and disengaged the local automatic power regulation system. A further operator error was made at this point when the operators failed to set the backup automatic controller to its proper "hold power" set-point. This resulted in the operators being unable to control the reactor power which began a rapid unplanned power reduction, falling to as low as 30 MWt before they were able to stabilize power at about 200 MWt.

This unplanned power reduction allowed the build-up of xenon (which is a strong neutron absorber) to a sufficiently high level that it reduced core reactivity which had to be compensated by withdrawal of control rods. Attempts were then made to increase power to the required level of 700-1000 MWt but were unsuccessful due to low core reactivity.

This was made more difficult by the fact that the control rods had been mostly withdrawn to compensate for the buildup of xenon.

Consequently, with the reactor only at 200 MWt, the decision was made to proceed with the test and two of the eight main circulation pumps, which up to then had not been in operation were started up and the flow rate of the water to the core was thereby increased. The result was a reduction in steam formation and a fall in water level in the steam drums which the operators tried to increase by using feed-water pumps. The immediate effect was to reduce core reactivity because of the reduction in steam voids and the operators responded by removing manual control rods from the core.

Conditions were thus produced with a potential for a large increase in steam voids and core reactivity. Nevertheless the experiment was started at which time events were only about one minute away from disaster. At a lower operating regime the **RBMK** reactor is fundamentally unstable due to its design. The reason for this involves the concepts of the positive void coefficient and the positive power coefficient.

At the time of the accident the Chernobyl reactor was being operated at less than 20% full power and thus in the unstable region. Immediately the experiment commenced, steam supply to one of the operating turbo-generators was shut off, which should have automatically caused a shutdown of the reactor. However, the operators had deliberately disabled the protection system to keep the reactor running so that the experiment could be repeated if the first attempt was unsuccessful. The turbo-generator rapidly decelerated and the four main circulating pumps connected to it started to run down. The water in the core started to boil increasing the volume of steam and creating voids in the core.

As a result of the positive void coefficient the power of the reactor started to rise and a positive feedback ensued (i.e. the power increased by itself). Although the operators tried to stop the reactor from "running away" by inserting the control rods as rapidly as possible, it was far too late. The rate of increase of power was such that the power rose in an uncontrolled manner to some 100 times full power in a matter of a few seconds causing severe fuel damage and fuel channel disruption.

A violent steam explosion occurred due to the interaction of water with the molten fuel, and blew off the 1000 tonne reactor cap and ejected burning material into the air, some of which landed on the roof of the joint turbine hall and put the adjacent undamaged reactor at risk. The term "void coefficient" means that if the power from the fuel increases, or the flow of coolant water decreases, or a combination of both, the amount of steam in the fuel channel increases.

This causes the density of the coolant to decrease because of the steam voids which have been created in the water. In most reactor designs, such as the **LWR,** the production of voids causes the number of neutrons to decrease and thus reduce the power. In this case the void coefficient is said to be "negative". In the case of the **RBMK** however, the design of the core is such that it has a "positive void coefficient" so that when the coolant density decreases the number of neutrons increases and thus the

reactor power increases. Although the initial fires were started by the burning debris ejected from the reactor, the main fire was due to the graphite also catching fire. This raging fire acted as a chimney to loft particulates of the fuel and fission products very high into the air.

5.3.2. IMMEDIATE IMPACT OF THE CHERNOBYL ACCIDENT

It is estimated that all of the xenon gas, about half of the iodine and cesium, and at least 5% of the remaining radioactive material in the Chernobyl 4 reactor core (which had 192 tonnes of fuel) was released in the accident. Most of the released material was deposited close by as dust and debris, but the lighter material was carried by wind over the Ukraine, Belarus, Russia and to some extent over Scandinavia and Europe. The casualties included firefighters who attended the initial fires on the roof of the turbine building. All these were put out in a few hours, but radiation doses on the first day were estimated to range up to 20,000 millisieverts (mSv), causing 28 deaths – six of which were firemen – by the end of July 1986. The next task was cleaning up the radioactivity at the site so that the remaining three reactors could be restarted, and the damaged reactor shielded more permanently. About 200,000 people ('liquidators') from all over the Soviet Union were involved in the recovery and clean-up during 1986 and 1987. They received high doses of radiation, averaging around 100 millisieverts. Some 20,000 of them received about 250 mSv and a few received 500 mSv. Later, the number of liquidators swelled to over 600,000 but most of these received only low radiation doses. The highest doses were received by about 1000 emergency workers and on-site personnel during the first day of the accident.

Initial radiation exposure in contaminated areas was due to short-lived iodine-131; later caesium-137 was the main hazard. (Both are fission products dispersed from the reactor core, with half lives of eight days and 30 years, respectively. 1.8 EBq of I-131 and 0.085 EBq of Cs-137 were released.) about five million people lived in areas contaminated (above 37 kBq/m^2 Cs-137) and about 400,000 lived in more contaminated areas of strict control by authorities (above 555 kBq/m^2 Cs-137).

The plant operators' town of Pripyat was evacuated on 27 April (45,000 residents). By 14 May, some 116,000 people that had been living within a 30 kilometers radius had been evacuated and later relocated. About 1000 of these returned unofficially to live within the contaminated zone. Most of those evacuated received radiation doses of less than 50 mSv, although a few received 100 mSv or more. In the years following the accident, a further 220,000 people were resettled into less contaminated areas, and the initial 30 km radius exclusion zone (2800 km^2) was modified and extended to cover 4300 square kilometers. This resettlement was due to application of a criterion of 350 mSv projected lifetime radiation dose, though in fact radiation in most of the affected area (apart from half a square kilometers) fell rapidly so that average doses were less than 50% above normal background of 2.5 mSv/yr.

5.3.3. ENVIRONMENTAL AND HEALTH EFFECTS OF THE ACCIDENT

Several organizations have reported on the impacts of the Chernobyl accident, but all have had problems assessing the significance of their observations because of the lack of reliable public health information before 1986. In 1989, the World Health Organization (WHO) first raised concerns that local medical scientists had incorrectly attributed various biological and health effects to radiation exposure. Following this, the Government of the USSR requested the International Atomic Energy Agency (IAEA) to coordinate an international experts' assessment of accident's radiological, environmental and health consequences in selected towns of the most heavily contaminated areas in Belarus, Russia, and Ukraine. Between March 1990 and June 1991, a total of 50 field missions were conducted by 200 experts from 25 countries (including the USSR), seven organizations, and 11 laboratories. In the absence of pre-1986 data, it compared a control population with those exposed to radiation. Significant health disorders were evident in both control and exposed groups, but, at that stage, Subsequent studies in the Ukraine, Russia and Belarus were based on national registers of over one million people possibly affected by radiation. By 2000, about 4000 cases of thyroid cancer had been diagnosed in exposed children. However, the rapid increase in thyroid cancers detected suggests that some of it at least is an artefact of the screening process. Thyroid cancer is usually not fatal if diagnosed and treated early. In February 2003, the IAEA established the Chernobyl Forum, in cooperation with seven other UN organizations as well as the competent authorities of Belarus, the Russian Federation and Ukraine. In April 2005, the reports prepared by two expert groups – "Environment", coordinated by the IAEA, and "Health", coordinated by WHO – were intensively discussed by the Forum and eventually approved by consensus. The conclusions of this 2005 Chernobyl Forum study (revised version published 2006[i]) are in line with earlier expert studies, notably the UNSCEAR 2000 report which said that "apart from this [thyroid cancer] increase, there is no evidence of a major public health impact attributable to radiation exposure 14 years after the accident. There is no scientific evidence of increases in overall cancer incidence or mortality or in non-malignant disorders that could be related to radiation exposure." As yet there is little evidence of any increase in leukemia, even among clean-up workers where it might be most expected. However, these workers – where high doses may have been received – remain at increased risk of cancer in the long term.

The Chernobyl Forum report says that people in the area have suffered a paralyzing fatalism due to myths and misperceptions about the threat of radiation, which has contributed to a culture of chronic dependency. Some "took on the role of invalids." Mental health coupled with smoking and alcohol abuse is a very much greater problem than radiation, but worst of all at the time was the underlying level of health and nutrition. Apart from the initial 116,000, relocations of people were very traumatic and did little to reduce radiation exposure, which was low anyway. Psycho-social effects among those affected by the accident are similar to those arising from other major disasters such as earthquakes, floods and fires.

According to the most up-to-date estimate of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), the average radiation dose due to the accident received by inhabitants of 'strict radiation control' areas (population 216,000) in the years 1986 to 2005 was 61 mSv (over the 20-year period), and in the 'contaminated' areas (population 6.4 million) it averaged 9 mSv, a minor increase over the dose due to background radiation over the same period (50 mSv).

The numbers of deaths resulting from the accident are covered in the *Report of the Chernobyl Forum Expert Group "Health"*, and are summarized in Chernobyl Accident Appendix 2: Health Impacts. Some exaggerated figures have been published regarding the death toll attributable to the Chernobyl disaster. A publication by the UN Office for the Coordination of Humanitarian Affairs (OCHA) lent support to these. However, the Chairman of UNSCEAR made it clear that "this report is full of unsubstantiated statements that have no support in scientific assessments, and the Chernobyl Forum report also repudiates them

5.3.4. PROGRESSIVE CLOSURE OF THE CHERNOBYL PLANT

In the early 1990s, some US$400 million was spent on improvements to the remaining reactors at Chernobyl, considerably enhancing their safety. Energy shortages necessitated the continued operation of one of them (unit 3) until December 2000. (Unit 2 was shut down after a turbine hall fire in 1991, and unit 1 at the end of 1997.) Almost 6000 people worked at the plant every day, and their radiation dose has been within internationally accepted limits. A small team of scientists works within the wrecked reactor building itself, inside the shelter. Workers and their families now live in a new town, Slavutich, 30 km from the plant. This was built following the evacuation of Pripyat, which was just 3 km away. Ukraine depends upon, and is deeply in debt to, Russia for energy supplies, particularly oil and gas, but also nuclear fuel. Although this dependence is gradually being reduced, continued operation of nuclear power stations, which supply half of total electricity, is now even more important than in 1986.

When it was announced in 1995 that the two operating reactors at Chernobyl would be closed by 2000, a memorandum of understanding was signed by Ukraine and G7 nations to progress this, but its implementation was conspicuously delayed. Alternative generating capacity was needed, either gas-fired, which has ongoing fuel cost and supply implications, or nuclear, by completing gas-fired, which has ongoing fuel cost and supply implications, or nuclear, by completing Khmelnitski unit 2 and Rovno unit 4 ('K2R4') in Ukraine. Construction of these was halted in 1989 but then resumed, and both reactors came on line late in 2004, financed by Ukraine rather than international grants as expected on the basis of Chernobyl's closure.

5.3.5. CHERNOBYL TODAY

Chernobyl unit 4 is now enclosed in a large concrete shelter which was erected quickly to allow continuing operation of the other reactors at the plant. However, the structure is neither strong nor durable. The international Shelter Implementation Plan in the 1990s involved raising money for remedial work including removal of the fuel-containing

materials. Some major work on the shelter was carried out in 1998 and 1999. Some 200 tonnes of highly radioactive material remains deep within it, and this poses an environmental hazard until it is better contained. A New Safe Confinement structure will be built by the end of 2011, and then will be moved into place on rails. It is to be an 18,000 tonne metal arch 105 metres high, 200 metres long and spanning 257 metres, to cover both unit 4 and the hastily-built 1986 structure. The Chernobyl Shelter Fund, set up in 1997, had received €810 million from international donors and projects towards this project and previous work. It and the Nuclear Safety Account, also applied to Chernobyl decommissioning, are managed by the European Bank for Reconstruction and Development (EBRD), which announced a €135 million contribution to the fund in May 2008. The total cost of the new shelter is estimated to be €1.2 billion. Used fuel from units 1 to 3 is stored in each unit's cooling pond, in a small interim spent fuel storage facility pond (ISF-1), and in the reactor of unit 3.

In 1999, a contract was signed for construction of a radioactive waste management facility to store 25,000 used fuel assemblies from units 1-3 and other operational wastes, as well as material from decommissioning units 1-3 (which will be the first RBMK units decommissioned anywhere). The contract included a processing facility, able to cut the RBMK fuel assemblies and to put the material in canisters, which will be filled with inert gas and welded shut. They will then be transported to the dry storage vaults in which the fuel containers would be enclosed for up to 100 years. This facility, treating 2500 fuel assemblies per year, would be the first of its kind for RBMK fuel. However, after significant parts of the storage structures had been built, technical deficiencies in the concept emerged, and the contract was terminated in 2007. The interim spent fuel storage facility (ISF-2) is now planned to be completed by others by mid-2013. In April 2009, Nukem handed over a turnkey waste treatment center for solid radioactive waste (ICSRM, Industrial Complex for Radwaste Management). In May 2010, the State Nuclear Regulatory Committee licensed the commissioning of this facility, where solid low- and intermediate-level wastes accumulated from the power plant operations and the decommissioning of reactor blocks 1 to 3 is conditioned. The wastes are processed in three steps. First, the solid radioactive wastes temporarily stored in bunkers is removed for treatment. In the next step, these wastes, as well as those from decommissioning reactor blocks 1-3, are processed into a form suitable for permanent safe disposal. Low- and intermediate-level wastes are separated into combustible, compactable, and non-compactable categories. These are then subject to incineration, high-force compaction, and cementation respectively. In addition, highly radioactive and long-lived solid waste is sorted out for temporary separate storage. In the third step, the conditioned solid waste materials are transferred to containers suitable for permanent safe storage.

As part of this project, at the end of 2007, Nukem handed over an Engineered Near Surface Disposal Facility for storage of short-lived radioactive waste after prior conditioning. It is 17 km away from the power plant at the Vektor complex within the 30-km zone. The storage area is designed to hold 55,000 m^3 of treated waste which will be

subject to radiological monitoring for 300 years, by then the radioactivity will have decayed to such an extent that monitoring is no longer required.

Another contract has been let for a Liquid Radioactive Waste Treatment Plant, to handle some 35,000 cubic meters of low- and intermediate-level liquid wastes at the site. This will need to be solidified and eventually buried along with solid wastes on site. In January 2008, the Ukraine government announced a four-stage decommissioning plan which incorporates the above waste activities and progresses towards a cleared site.

5.3.6. RESETTLEMENT OF CONTAMINATED AREAS

In the last two decades there has been some resettlement of the areas evacuated in 1986 and subsequently. Recently the main resettlement project has been in Belarus. In July 2010, the Belarus government announced that it had decided to settle back thousands of people in the 'contaminated areas' covered by the Chernobyl fallout, from which 24 years ago they and their forbears were hastily relocated. Compared with the list of contaminated areas in 2005, some 211 villages and hamlets had been reclassified with fewer restrictions on resettlement. The decision by the Belarus Council of Ministers resulted in a new national program over 2011-15 and up to 2020 to alleviate the Chernobyl impact and return the areas to normal use with minimal restrictions. The focus of the project is on the development of economic and industrial potential of the Gomel and Mogilev regions from which 137,000 people were relocated.

The main priority is agriculture and forestry, together with attracting qualified people and housing them. Initial infrastructure requirements will mean the refurbishment of gas, potable water and power supplies, while the use of local wood will be banned. Schools and housing will be provided for specialist workers and their families ahead of wider socio-economic development. Overall, some 21,484 dwellings are slated for connection to gas networks in the period 2011-2015, while about 5600 contaminated or broken down buildings are demolished. Over 1300 kilometers of road will be laid, and ten new sewerage works and 15 pumping stations are planned. The cost of the work was put at BYR 6.6 trillion ($2.2 billion), split fairly evenly across the years 2011 to 2015 inclusive.

The feasibility of agriculture will be examined in areas where the presence of caesium-137 and strontium-90 is low, "to acquire new knowledge in the fields of radiobiology and radioecology in order to clarify the principles of safe life in the contaminated territories." Land found to have too high a concentration of radionuclides will be reforested and managed. A suite of protective measures is to be set up to allow a new forestry industry whose products would meet national and international safety standards. In April 2009, specialists in Belarus stressed that it is safe to eat all foods cultivated in the contaminated territories, though intake of some wild food was restricted.

Protective measures will be put in place for 498 settlements in the contaminated areas where average radiation dose may exceed 1 mSv per year. There are also 1904 villages with annual average effective doses from the pollution between 0.1 mSv and 1 mSv. The goal for these areas is to allow their re-use with minimal restrictions, although already radiation doses there from the caesium are lower than background levels anywhere in

the world. The most affected settlements are to be tackled first, around 2011- 2013, with the rest coming back in around 2014-2015.

5.3.7. LESSONS LEARNED FROM CHERNOBYL

Leaving aside the verdict of history on its role in melting the Soviet 'Iron Curtain', some very tangible practical benefits have resulted from the Chernobyl accident. The main ones concern reactor safety, notably in Eastern Europe. (The US Three Mile Island accident in 1979 had a significant effect on Western reactor design and operating procedures. While that reactor was destroyed, all radioactivity was contained – as designed – and there were no deaths or injuries.) While no-one in the West was under any illusion about the safety of early Soviet reactor designs, some lessons learned have also been applicable to Western plants. Certainly the safety of all Soviet-designed reactors has improved vastly. This is due largely to the development of a culture of safety encouraged by increased collaboration between East and West, and substantial investment in improving the reactors.

Modifications have been made to overcome deficiencies in all the RBMK reactors still operating. In these, originally the nuclear chain reaction and power output could increase if cooling water were lost or turned to steam, in contrast to most Western designs. It was this effect which led to the uncontrolled power surge that led to the destruction of Chernobyl 4 All of the RBMK reactors have now been modified by changes in the control rods, adding neutron absorbers and consequently increasing the fuel enrichment from 1.8 to 2.4% U-235, making them very much more stable at low power. Automatic shut-down mechanisms now operate faster, and other safety mechanisms have been improved. Automated inspection equipment has also been installed. A repetition of the 1986 Chernobyl accident is now virtually impossible, according to a German nuclear safety agency report.

Since 1989, over 1000 nuclear engineers from the former Soviet Union have visited Western nuclear power plants and there have been many reciprocal visits. Over 50 twinning arrangements between East and West nuclear plants have been put in place. Most of this has been under the auspices of the World Association of Nuclear Operators (WANO), a body formed in 1989 which links 130 operators of nuclear power plants in more than 30 countries. Many other international programs were initiated following Chernobyl. The International Atomic Energy Agency (IAEA) safety review projects for each particular type of Soviet reactor are noteworthy, bringing together operators and Western engineers to focus on safety improvements. The Convention on Nuclear Safety adopted in Vienna in June 1994 is another outcome. The Chernobyl Forum report said that some seven million people are now receiving or eligible for benefits as 'Chernobyl victims', which means that resources are not targeting the needy few percent of them. Remedying this presents daunting political problems however.

5.3.8. HEALTH EFFECTS OF THE ACCIDENT

The Soviets have calculated that about 10% of the graphite (about 250 tons) was burned and some 3-4% of the fuel was expelled from the core. The release of

radioactive material from the core did not occur in a single massive event. Only 25% of the materials released escaped during the first day of the accident as a result of the explosion. The rest escaped over a nine-day period as a result of the fire before it was contained. It has been estimated that the proportions of the core inventory deposited at various distance from Chernobyl was as follows:

- On-site 0.3-0.5%
- 0-20 km 1.5-2%
- Beyond 20 km 1.0-1.5%

The immediate health impact was, of course, on the plant personnel and rescue workers, a number of whom received massive doses of radiation from which they died. Others had to be hospitalized for treatment of radiation burns. Outside the immediate area of the reactor accident the doses were too small to cause "acute" radiation effects.

Initially the local population was instructed to remain indoors and to close their windows, but as the levels of radiation began to increase, evacuation commenced and arrangements were made for decontamination of the skin and clothing where necessary. Outside the Soviet Union the doses received from the fallout were large enough, particularly in Western Europe, to cause an appreciable increase above the average natural radiation exposure.

In the first year the estimated increase over natural background was about 20% in the countries of the European Community. Nevertheless, an International Panel of Experts, convened by the Commission of the European Communities to advise on the feasibility of studies on health effects in western Europe from the Chernobyl accident, concluded that the levels of exposure were so low as to preclude any effects being detected in the exposed populations of the EEC.

Although there was no immediate health impact on the local population from the fallout from the accident, the doses received might be expected to result in an increased incidence of radiation-induced cancers in later years. At the present time, however, adverse health effects due to radiation exposure have not been observed. A study carried out by the International Chernobyl Project has pointed out that many of the local clinical investigations of health effects were poorly done and produced confusing , and often contradictory results.

Nevertheless, the International Project stated in its report that "....adverse health effects have not been substantiated by those local studies which were adequately performed or by the studies under the Project". The Project report points out, however, that there were significant non-radiation health disorders in the populations of both survey contaminated and surveyed control settlements studies by the Project, but no health disorders that could be attributed directly to radiation exposure.

However, as the Soviet Union has a population around 275-million and over the next 40 years the number of deaths can be expected to be approximately 30-million, of which some 7.6-million will be from cancer. Consequently, deaths due to Chernobyl, which appear to lie in the range 4,000/38,000, may be impossible to detect with any certainty.

In fact, the International Chernobyl Project states in its recent report that "On the basis of the doses estimated by the Project and currently accepted radiation risk estimates, future increases over the natural incidence of cancers or hereditary effects would be difficult to discern, even with large and well- designed long term epidemiological studies".

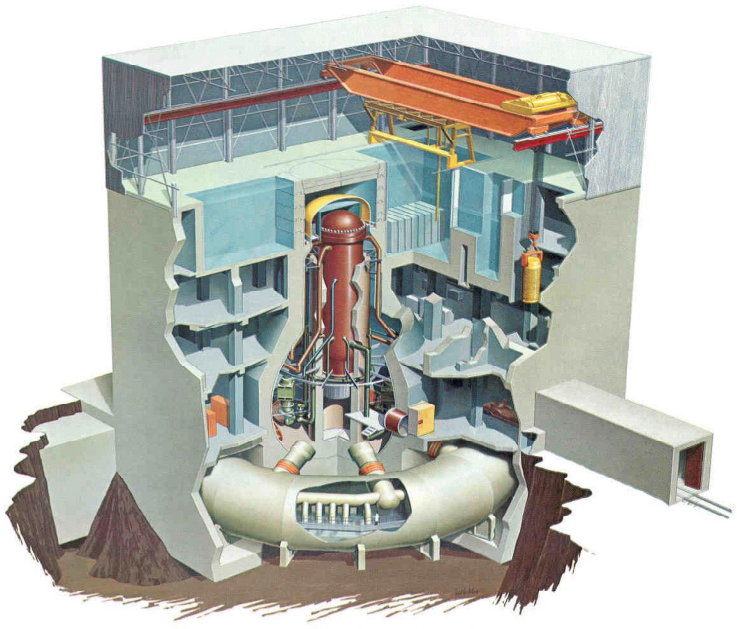
5.3.9. COMMENTARY

A number of factors combined together to bring about this accident. It could not have occurred had the **RBMK** not had certain design features, notably its positive void coefficient at low power operation, or if the design had included safety systems to cope with such a characteristic. Equally the accident could not have occurred had there not been the application of inappropriate operating rules by the operators.

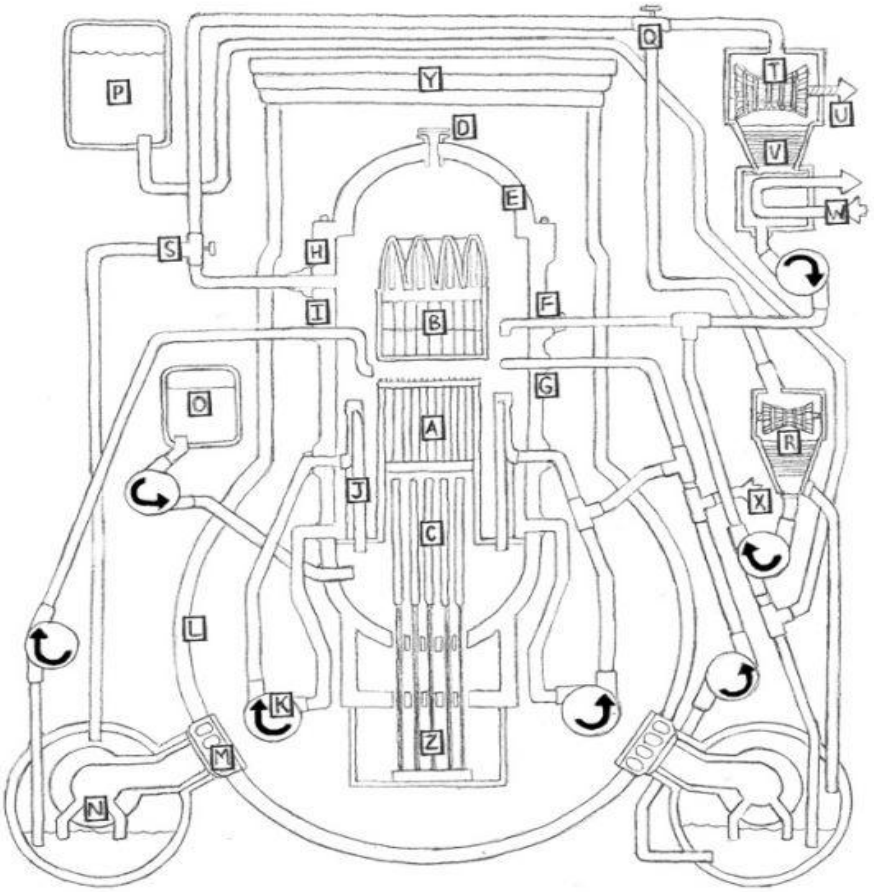
From the standpoint of the developing countries the lessons from the Chernobyl accident are of great importance. In the first instance it stresses the need for ensuring that the reactors being built have adequate safety features, particularly those which prevent an accident from happening in the event of operator error. Such systems will, presumably, be purchased from exporting countries and will have been certified by the regulatory bodies in those countries. More importantly, however, the Chernobyl accident stresses the importance of good operator training and supervision.

5.4. THE EVENT IN THE GE-MARK I REACTOR AT FUKUSHIMA, JAPAN

ISOMETRIC VIEW OF THE GE-MARK I



SCHEMATIC OF GE-MARK I REACTOR AT FUKUSHIMA, JAPAN



Mark I Reactor Components

- **A.** Uranium fuel rods
- **B.** Steam Separator and dryer assemblies
- **C.** Graphite control rods
- **D.** Vent and head spray
- **E.** Reactor Vessel
- **F.** Feedwater inlet
- **G.** Low pressure coolant injection inlet
- **H.** Steam Outlet
- **I.** Core spray inlet
- **J.** Jet pump
- **K.** Recirculation pump
- **L.** Concrete Shell "Drywell"
- **M.** Venting System
- **N.** Suppression pool
- **O.** Boron tank
- **P.** Condensate storage tank

5.4.1. THE NUCLEAR EVENT AT FUKUSHIMA, JAPAN

The earthquake that hit Japan was 5 times more powerful than the worst earthquake the nuclear power plant was designed and built for (the Richter scale works logarithmically; the difference between the 8.2 that the plants were built for and the 8.9 that happened is 5 times, not 0.7). So the first hooray for Japanese engineering, everything held up.

When the earthquake hit with 8.9, the nuclear reactors all went into automatic shutdown. Within seconds after the earthquake started, the control rods had been inserted into the core and nuclear chain reaction of the uranium stopped. When the rods are inserted into the reactor core the turbines are automatically taken off-line. Now, the auxiliary diesel generators need to start and activate the cooling system which has to carry away the residual heat. The residual heat load is about 3% of the heat load under normal operating conditions.

The earthquake destroyed the external power supply of the nuclear reactor. That is one of the most serious accidents for a nuclear power plant, and accordingly, a “plant black out” receives a lot of attention when designing backup systems. The power is needed to keep the coolant pumps working. Since the power plant had been shut down, it cannot produce any electricity by itself any more.

Things were going well for an hour. One set of multiple sets of emergency Diesel power generators kicked in and provided the electricity that was needed. Then the Tsunami came, much bigger than people had expected when building the power plant. The tsunami took out all multiple sets of backup Diesel generators.

When designing a nuclear power plant, engineers follow a philosophy called “Defense of Depth”. That means that you first build everything to withstand the worst catastrophe you can imagine, and then design the plant in such a way that it can still handle one system failure (that you thought could never happen) after the other. A tsunami taking out all backup power in one swift strike is such a scenario. The last line of defense is putting everything into the third containment, that will keep everything, whatever the mess, control rods in our out, core molten or not, inside the reactor.

When the diesel generators were gone, the reactor operators switched to emergency battery power. The batteries were designed as one of the backups to the backups, to provide power for cooling the core for 8 hours. And they did.

Within the 8 hours, another power source had to be found and connected to the power plant. The power grid was down due to the earthquake. The diesel generators were destroyed by the tsunami. So mobile diesel generators were trucked in.

This is where things started to go seriously wrong. The external power generators could not be connected to the power plant (the plugs did not fit). So after the batteries ran out, the residual heat could not be carried away any more.

At this point the plant operators begin to follow emergency procedures that are in place for a “loss of cooling event”. It is again a step along the “Depth of Defense” lines. The power to the cooling systems should never have failed completely, but it did, so they “retreat” to the next line of defense. All of this, however shocking it seems to us, is part of the day-to-day training the operators go through as part of operator training, right through to managing a core meltdown.

It was at this stage that people started to talk about core meltdown. Because at the end of the day, if cooling cannot be restored, the core will eventually melt (after hours or days), and the last line of defense, the core catcher and third containment, would come into play.

But the goal at this stage was to manage the core while it was heating up, and ensure that the first containment (the Zircaloy tubes that contains the nuclear fuel), as well as the second containment remain intact and operational for as long as possible, to give the engineers time to fix the cooling systems.

Because cooling the core is such a big deal, the reactor has a number of cooling systems, each in multiple versions (the reactor water cleanup system, the decay heat removal, the reactor core isolating cooling, the standby liquid cooling system, and the emergency core cooling system). Which one failed when or did not fail is not clear at this point in time.

So imagine a pressure cooker on the stove, heat on low, but on. The operators use whatever cooling system capacity they have to get rid of as much heat as possible, but the pressure starts building up. The priority now is to maintain integrity of the first containment (keep temperature of the fuel rods below 2200°C), as well as the second containment, the pressure cooker. In order to maintain integrity of the pressure cooker (the second containment), the pressure has to be released from time to time. Because the ability to do that in an emergency is so important, the reactor has 11 pressure release valves. The operators now started venting steam from time to time to control the pressure. The temperature at this stage was about 550°C.

This is when the reports about “radiation leakage” starting coming in. The Senior Engineer explained above why venting the steam is theoretically the same as releasing radiation into the environment, but why it was and is not dangerous. The radioactive nitrogen as well as the noble gases do not pose a threat to human health. At some stage during this venting, the explosion occurred. The explosion took place outside of the third containment (our “last line of defense”), and the reactor building. Remember that the reactor building has no function in keeping the radioactivity contained.

It is not entirely clear yet what has happened, but this is the likely scenario: The operators decided to vent the steam from the pressure vessel not directly into the environment, but into the space between the third containment and the reactor building (to give the radioactivity in the steam more time to subside). The problem is that at the high temperatures that the core had reached at this stage, water molecules can “disassociate” into oxygen and hydrogen – an explosive mixture. And it did explode, outside the third containment, damaging the reactor building around.

It was that sort of explosion, but inside the pressure vessel (because it was badly designed and not managed properly by the operators) that lead to the explosion of Chernobyl. This was never a risk at Fukushima. The problem of hydrogen-oxygen formation is one of the most important in the design of a power plant (if you are not Soviet, that is), so the reactor is built and operated in a way it cannot happen inside the containment. It happened outside, which was not intended but a possible scenario and OK, because it did not pose a risk for the containment.

So the pressure was under control, as steam was vented. Now, if you keep boiling your pot, the problem is that the water level will keep falling and falling. The core is covered by several meters of water in order to allow for some time to pass (hours, days) before it gets exposed. Once the rods start to be exposed at the top, the exposed parts will reach the critical temperature of 2200 °C after about 45 minutes. This is when the first containment, the Zircaloy tube, would fail.

And this started to happen. The cooling could not be restored before there was some (very limited, but still) damage to the casing of some of the fuel. The nuclear material itself was still intact, but the surrounding Zircaloy shell had started melting. What happened now is that some of the byproducts of the uranium decay – radioactive Cesium and Iodine – started to mix with the steam. The big problem, uranium, was still under control, because the uranium oxide rods were good until 3000 °C. It is confirmed that a very small amount of Cesium and Iodine was measured in the steam that was released into the atmosphere.

It seems this was the “go signal” for a major plan B. The small amounts of Cesium that were measured told the operators that the first containment on one of the rods somewhere was about to give. The Plan A had been to restore one of the regular cooling systems to the core. Why that failed is unclear. One plausible explanation is that the tsunami also took away / polluted all the clean water needed for the regular cooling systems.

The water used in the cooling system is very clean, demineralized (like distilled) water. The reason to use pure water is the above mentioned activation by the neutrons from the Uranium: Pure water does not get activated much, so stays practically radioactive-free. Dirt or salt in the water will absorb the neutrons quicker, becoming more radioactive. This has no effect whatsoever on the core – it does not care what it is cooled by. But it makes life more difficult for the operators and mechanics when they have to deal with activated (i.e. slightly radioactive) water.

But Plan A had failed – cooling systems down or additional clean water unavailable – so Plan B came into effect. This is what it looks like happened:

In order to prevent a core meltdown, the operators started to use sea water to cool the core. I am not quite sure if they flooded our pressure cooker with it (the second containment), or if they flooded the third containment, immersing the pressure cooker.

The point is that the nuclear fuel has now been cooled down. Because the chain reaction has been stopped a long time ago, there is only very little residual heat being produced now. The large amount of cooling water that has been used is sufficient to take up that heat. Because it is a lot of water, the core does not produce sufficient heat any more to produce any significant pressure. Also, boric acid has been added to the seawater. Boric acid is “liquid control rod”. Whatever decay is still going on, the Boron will capture the neutrons and further speed up the cooling down of the core.

The plant came close to a core meltdown. Here is the worst-case scenario that was avoided: If the seawater could not have been used for treatment, the operators would have continued to vent the water steam to avoid pressure buildup. The third containment would then have been completely sealed to allow the core meltdown to happen without releasing radioactive material. After the meltdown, there would have been a waiting period for the intermediate radioactive materials to decay inside the reactor, and all radioactive particles to settle on a surface inside the containment. The cooling system would have been restored eventually, and the molten core cooled to a manageable

temperature. The containment would have been cleaned up on the inside. Then a messy job of removing the molten core from the containment would have begun, packing the (now solid again) fuel bit by bit into transportation containers to be shipped to processing plants. Depending on the damage, the block of the plant would then either be repaired or dismantled.

5.4.2. COMMENTARY FROM THE PRINCIPAL ENGINEER

- The plant is safe now and will stay safe.
- Japan is looking at an **INES** Level 4 Accident: Nuclear accident with local consequences. That is bad for the company that owns the plant, but not for anyone else.
- Some radiation was released when the pressure vessel was vented. All radioactive isotopes from the activated steam have gone (decayed). A very small amount of Cesium was released, as well as Iodine. If you were sitting on top of the plants' chimney when they were venting, you should probably give up smoking to return to your former life expectancy. The Cesium and Iodine isotopes were carried out to the sea and will never be seen again.
- There was some limited damage to the first containment. That means that some amounts of radioactive Cesium and Iodine will also be released into the cooling water, but no Uranium or other nasty stuff (the Uranium oxide does not "dissolve" in the water). There are facilities for treating the cooling water inside the third containment. The radioactive Cesium and Iodine will be removed there and eventually stored as radioactive waste in terminal storage.
- The seawater used as cooling water will be activated to some degree. Because the control rods are fully inserted, the Uranium chain reaction is not happening. That means the "main" nuclear reaction is not happening, thus not contributing to the activation. The intermediate radioactive materials (Cesium and Iodine) are also almost gone at this stage, because the Uranium decay was stopped a long time ago. This further reduces the activation. The bottom line is that there will be some low level of activation of the seawater, which will also be removed by the treatment facilities.
- The seawater will then be replaced over time with the "normal" cooling water
- The reactor core will then be dismantled and transported to a processing facility, just like during a regular fuel change.
- Fuel rods and the entire plant will be checked for potential damage. This will take about 4-5 years.
- The safety systems on all Japanese plants will be upgraded to withstand a 9.0 earthquake and tsunami (or worse)
- As the Senior Engineer of the **NTPBMR Prototype Project,** I believe the most significant problem will be a prolonged power shortage. 11 of Japan's 55 nuclear

reactors in different plants were shut down and will have to be inspected, directly reducing the nation's nuclear power generating capacity by 20%, with nuclear power accounting for about 30% of the national total power generation capacity. I have not looked into possible consequences for other nuclear plants not directly affected. This will probably be covered by running gas power plants that are usually only used for peak loads to cover some of the base load as well. I am not familiar with Japan 's energy supply chain for oil, gas and coal, and what damage the harbors, refinery, storage and transportation networks have suffered, as well as damage to the national distribution grid. All of these items will increase the cost of energy and lead to higher electricity bill, as well as lead to power shortages during peak demand and reconstruction efforts.

- This all is only part of a much bigger picture. Emergency response has to deal with shelter, drinking water, food and medical care, transportation and communication infrastructure, as well as electricity supply. In a world of lean supply chains, we are looking at some major challenges in all of these areas.

ASCENTRUST LLC
The POWER of ENGINEERING

APPENDIX
INTRODUCTION TO
NUCLEAR POWER AND NTPBMR TECHNOLOGY

SECTION C: Nuclear History and Regulations

ASCENTRUST LLC
The POWER of ENGINEERING

HISTORY OF NUCLEAR POWER DEVELOPMENT
IN THE UNITED STATES
FOR
STAKEHOLDERS AND CONSULTANTS
OF
NUCLEAR TECHNOLOGY PEBBLE BED MODULAR REACTOR
(NTPBMR) PROJECT

This short history of nuclear regulation provides a brief overview of the most significant events which have led to the present situation with respect to the Licensing of Nuclear Power Facilities. Because this is an introduction to the process of licensing we will only be considering the important highlights of the history of the **Atomic Energy Commission (**AEC) and the **Nuclear Regulatory Commission** (NRC)

SECTION ONE: THE FORMATIVE YEARS OF NUCLEAR REGULATION, 1946-62

1.1. THE DAWN OF THE ATOMIC AGE

The use of nuclear energy against the Japanese cities of Hiroshima and Nagasaki in August 1945 ushered in the “Atomic Age”, a new historical epoch, breathlessly promoted in countless news reports, magazine articles, films, and radio broadcasts.

Within a short time after the end of World War II, politicians, journalists, scientists, and business leaders were suggesting that peaceful applications of nuclear power could be as dramatic in their benefits as nuclear weapons were awesome in their destructive power.

Nuclear physicist Alvin M. Weinberg told the **Senate's Special Committee on Atomic Energy** in December 1945: "Atomic power can cure as well as kill. It can fertilize and enrich a region as well as devastate it. Observing that ideas for the civilian uses of atomic energy ranged "from the practical to the fantastic," it cited a few examples: atomic-powered airplanes, rockets, and automobiles, large electrical generating stations, small "home power plants" to provide heat and electricity in individual homes, and tiny atomic generators wired to clothing to keep a person cool in summer and warm in winter.

Developing nuclear energy for civilian purposes, as even the most enthusiastic proponents recognized, would take many years. The government's first priority was to maintain strict control over atomic technology and to exploit it further for military purposes.

The **Atomic Energy Act of 1946**, passed as tensions with the Soviet Union were developing into the cold war, acknowledged in passing the potential peaceful benefits of atomic power. But it emphasized the military aspects of nuclear energy and underscored the need for secrecy, raw materials, and production of new weapons. The 1946 law did not allow for private, commercial application of atomic energy; rather, it created a virtual government monopoly of the technology.

To manage the nation's atomic energy programs, the act established the five-member **Atomic Energy Commission** (**AEC**).

1.2. THE 1954 ATOMIC ENERGY ACT

In 1954, Congress passed new legislation that for the first time permitted the wide use of atomic energy for peaceful purposes. The 1954 **Atomic Energy Act** redefined the atomic energy program by ending the government monopoly on technical data and making the growth of a private commercial nuclear industry an urgent national goal. The measure directed the **Atomic Energy Commission** (AEC):

"To encourage widespread participation in the development and utilization of atomic energy for peaceful purposes."

At the same time, it instructed the agency to prepare regulations that would protect public health and safety from radiation hazards. Thus, the 1954 act assigned the AEC three major roles:

1. To continue its weapons program,
2. To promote the private use of atomic energy for peaceful applications,
3. To protect public health and safety from the hazards of commercial nuclear power.

Those functions were in many respects inseparable and incompatible, especially when combined in a single agency. The competing responsibilities and the precedence that the **AEC** gave to its main objectives: military and promotional duties for the peaceful application of Nuclear Power gradually damaged the agency's credibility on regulatory issues and undermined public confidence in its safety program.

The **AEC**'s regulatory program was most directly affected by the agency's commitment to encouraging the rapid growth of civilian nuclear power. The initial impetus for peaceful atomic development came mostly from considerations other than meeting America's energy demands. In the early 1950s, projections of future energy requirements predicted that atomic power would eventually play an important role in the nation's energy supplies, but they did not suggest an immediate need to construct atomic power reactors.

The prevailing sense of urgency, that led to the 1954 Atomic Energy Act and to the growth of commercial nuclear power derived instead largely from the fear of falling behind other nations in fostering peaceful atomic progress. The strides that Great Britain was making in the field seemed disturbing enough, but the possibility that the Soviet Union might surpass the United States in civilian power development was even more ominous.

AEC commissioner Thomas E. Murray described a "nuclear power race" in a 1953 speech and warned that the "stakes are high." He added: "Once we become fully conscious of the possibility that power hungry countries will gravitate toward the USSR if it wins the nuclear power race, . . . it will be quite clear that this power race is no Everest-climbing, kudos-providing contest." Like Murray, many government officials emphasized that surrendering America's lead in expanding the peaceful applications of atomic energy would deal a severe blow to its international prestige and world scientific dominance.

The eagerness to push for rapid civilian nuclear development was intensified by an impulse to show that atomic technology could serve constructive purposes as well as destructive ones. The assertions made shortly after World War II that atomic energy could provide spectacular advances that would raise living standards throughout the world remained unproven and largely untested. As the nuclear arms race took on more

terrifying proportions with the development of thermonuclear bombs, the desire to demonstrate the benefits of atomic energy became more acute. President Dwight D. Eisenhower, spurred by the detonation of the Soviet Union's first hydrogen device, starkly depicted the horror of nuclear warfare in a widely publicized address to the United Nations in December 1953.

By 1954, a broad political consensus viewed the development of nuclear energy for civilian purposes as a vital goal. The Atomic Energy Act of that year resulted partly from perceptions of the long-range need for new energy sources, but mostly from the immediate commitment to maintain America's world leadership in nuclear technology, enhance its international prestige, and demonstrate the benefits of peaceful atomic energy. It infused the atomic power program with a sense of urgency, and in that atmosphere, the AEC established its developmental and regulatory policies. The 1954 act gave the AEC wide discretion on how to proceed. Despite the general agreement on ultimate objectives, the means by which they should be accomplished soon created sharp differences.

1.3. THE AEC AND THE DEVELOPMENT OF COMMERCIAL NUCLEAR POWER

The **AEC** favored a partnership between government and industry in which private firms would play an integral role in demonstrating and expanding the use of atomic power.

1. The **AEC** was directed toward encouraging development of the uses of atomic energy in the framework of the American free enterprise system.
2. It was the **AEC**'s belief that competitive economic nuclear power would be most quickly achieved by construction and operation of full-scale plants by industry itself.

To accomplish its objectives, the **AEC** announced a "power demonstration reactor program" in January 1955. The agency offered to perform research and development on power reactors in its national laboratories, to subsidize additional research undertaken by industry under fixed-sum contracts, and to waive for seven years the established fuel use charges for the loan of fissionable materials (which the government would continue to own).

For their part, private utilities and vendors would supply the capital for construction of nuclear plants and pay operating expenses other than fuel charges. The purpose of the demonstration program was to stimulate private participation and investment in exploring the technical and economic feasibility of different reactor designs. At that time, no single reactor type had clearly emerged as the most promising of the several that had been proposed.

The **AEC**'s incentives received a mixed response from private industry. For several years, some utility executives had shown a keen interest in investigating the use of nuclear fission for generating electricity. But commercial applications of atomic energy had been thwarted by the severe limitations on access to technical information dictated by the 1946 **Atomic Energy Act**.

In 1953, when the **Joint Committee on Atomic Energy**, created by the 1946 act to carry out congressional oversight of the **AEC**, conducted public hearings on peaceful atomic development, spokesmen for private firms emphasized that industrial progress was possible only if the restrictions on obtaining data were eased. By opening nuclear technology to commercial applications, the **1954 Atomic Energy Act** largely satisfied those complaints.

From the perspective of utility companies, the act offered an opportunity to participate in nuclear development and gain experience in a technology that promised to help meet long-term energy demands. Vendors of reactor components welcomed the prospects of expanding their markets, not only in the United States but also in foreign countries where the need for new sources of power was more immediate.

The enthusiasm of the private utility industry for nuclear power development, however, was tempered by other considerations. Although experiments with AEC-owned reactors had established the technical feasibility of using nuclear fission to produce electricity, many scientific and engineering questions remained to be answered. Despite the financial inducements the **AEC** offered through its power demonstration reactor program, the capital and operating costs of atomic power were certain to be much higher than those of fossil fuel plants, at least in the early stages of development. Across the industry, the prospects of realizing short-term profits from nuclear power were dim.

In addition to financial considerations, recognition of the hazards of the technology intensified industry's reservations about nuclear power. Based on experience with government test reactors and the prevailing faith in the ability of scientists and engineers to solve technological problems, the **AEC** and industry leaders regarded the chances of a disastrous atomic accident as remote. But they did not dismiss the possibility entirely.

Mindful of both the costs and the risks of atomic power, the electric utility industry responded to the 1954 **Atomic Energy Act** and the **AEC**'s demonstration program with restraint. Although many utilities were interested in exploring the potential of nuclear power, few were willing to press ahead rapidly in the face of existing uncertainties. The **AEC** was gratified, and rather surprised, that by August 1955 five power companies--either as individual utilities or as consortiums--had announced plans to build nuclear plants. Two decided to proceed without government assistance and three others submitted proposals for projects under the **AEC'**s power demonstration program.

The **Joint Committee on Atomic Energy** was less impressed with the response of private industry to the 1954 act and the AEC's incentives. The Democratic majority on the committee favored a larger government role in accelerating nuclear development, which conflicted with the AEC's commitment to encouraging maximum private participation. The issue became a major source of contention between the AEC and the Joint Committee, contributing a philosophical dispute to relations that were already strained by political differences and a bitter personal feud between Strauss and Joint Committee chairman Clinton P. Anderson.

In 1956, two Democratic members of the Joint Committee, Representative Chet Holifield and Senator Albert Gore, introduced legislation directing the **AEC** to construct six pilot nuclear plants, each of a different design, in order to "advance the art of generation of electrical energy from nuclear energy at the maximum possible rate." Supporters of the bill contended that the United States was falling behind Great Britain and the Soviet Union in the quest for practical and economical nuclear power. Opponents of the measure denied that the United States had surrendered its lead in atomic technology and insisted that private industry was best able to expedite further development.

1.4. THE AEC'S REGULATORY PROGRAM

The **AEC**'s determination to push nuclear development through a partnership in which private industry played a vital role had a major impact on the agency's regulatory policies. The **AEC'**s fundamental objective in drafting regulations was to ensure that public health and safety were protected without imposing overly burdensome requirements that would impede industrial growth.

Other proponents of nuclear development shared those views. They realized that safety was indispensable to progress; an accident could destroy the fledgling industry or at least set it back many years. At the same time, they worried that regulations that were too restrictive or inflexible would discourage private participation and investment in nuclear technology.

The inherent difficulty the **AEC** faced in distinguishing between essential and excessive regulations was compounded by technical uncertainties and limited operating experience with power reactors. The safety record of the **AEC**'s own experimental reactors engendered confidence that safety problems could be resolved and the possibility of accidents kept to "an acceptable calculated risk."

Experience to that time offered little definitive guidance on some important technical and safety questions, such as the effect of radiation on the properties of reactor materials, the durability of steel and other metals under stress in a reactor, the ways in which water reacted with uranium, thorium, aluminum, and other elements in a reactor, and the measures needed to minimize radiation exposure in the event of a large accident

1.5. THE LICENSING PROCESS

The **AEC**'s regulatory staff, created soon after the passage of the **1954 Atomic Energy Act,** confronted the task of writing regulations and devising licensing procedures rigorous enough to assure safety but flexible enough to allow for new findings and rapid changes in atomic technology.

Within a short time the staff drafted rules and definitions on radiation protection standards, distribution and safeguarding of fissionable materials, and reactor operators' qualifications. It also established procedures for licensing privately-owned reactors.

The 1954 act outlined a two-step procedure for granting licenses. If the **AEC** found the safety analysis submitted by a utility for a proposed reactor to be acceptable, it would issue a construction permit. After construction was completed and the **AEC** determined

that the plant fully met safety requirements, the applicant would receive a license to load fuel and begin operation.

Because of the uncertainties in technical knowledge and the AEC's goal of encouraging different reactor designs, the agency had to judge license applications on a case-by-case basis. The early state of the technology precluded the possibility of formulating universal standards for all aspects of reactor engineering. The regulatory staff reviewed the information that applicants supplied on the suitability of the proposed site, construction specifications, a detailed plan of operation, and safety features.

The proposal received further scrutiny from a panel of outside experts, the **Advisory Committee on Reactor Safeguards** (**ACRS**). The **ACRS**, composed of part-time consultants who were recognized authorities on various aspects of reactor technology, conducted its own independent review of the application. The recommendations of the staff and the **ACRS** went to the commissioners, who made the final decision on whether or not to approve a construction permit or operating license. (Later, the Commission delegated consideration of regulatory staff and **ACRS** judgments to panels drawn from the "**Atomic Safety and Licensing Board**" while retaining final jurisdiction in licensing cases if it chose to review a board ruling).

The **AEC** did not require that a prospective power reactor owner submit finalized technical data on the safety of a facility to receive a construction permit. The agency was willing to grant a conditional permit as long as the application provided "reasonable assurance" that the projected plant could be constructed and operated at the proposed site "without undue risk to the health and safety of the public."

The two-step licensing system enabled the **AEC** to authorize construction of nuclear plants while allowing time to investigate any outstanding safety questions and prescribe modifications in initial plans. Agency officials recognized that the wisdom of permitting construction to proceed without first resolving all potential safety problems was disputable, but they saw no alternatives in light of the existing state of the technology and the commitment to rapid development of atomic power. They were confident that regulatory requirements were adequate to guard against the hazards of nuclear generating systems.

The **AEC** acknowledged, however, that it could not eliminate all risks. C. Rogers McCullough, chairman of the **ACRS**, informed the Joint Committee in 1956 that because of technical uncertainties and limited operating experience, "the determination that the hazard is acceptably low is a matter of competent judgment."

1.6. THE POWER REACTOR DEVELOPMENT COMPANY CONTROVERSY

It soon became apparent how the AEC's judgment on safety issues could be influenced by its ambition to promote the private development of nuclear power. The Commission's actions in granting a construction permit for a commercial fast breeder reactor, despite the reservations of the ACRS, ignited an acrimonious controversy with the Joint Committee and raised questions about the AEC's regulatory program.

In January 1956, the **Power Reactor Development Company** (**PRDC**), a consortium of utilities led by **Detroit Edison**, applied for a permit to build a fast breeder in Lagoona Beach, Michigan, located on Lake Erie within thirty miles of both Detroit and Toledo, Ohio. The **AEC** had already received applications for two privately-financed light-water reactors, but the **PRDC** proposal was the first to come in under the power demonstration program.

The fast breeder reactor that the **PRDC** planned was far more advanced in its technological complexity than light-water models, with which scientists and engineers had greater experience and familiarity. After review of the **PRDC**'s application and discussions with company representatives, the **ACRS** concluded in an internal report to the Commission that "there is insufficient information available at this time to give assurance that the **PRDC** reactor can be operated at this site without public hazard." The **ACRS** also expressed uncertainty that its questions about the reactor's safety could be resolved within the **PRDC'**s proposed schedule for obtaining an operating license. The **ACRS** urged that the **AEC** expand its experimental programs with fast breeders to seek more complete data on the issues the **PRDC** application raised.

The public dispute over the **PRDC** case was triggered by statements of Chairman Strauss and Commissioner Murray in congressional budget hearings. After the **AEC** requested a supplemental appropriation for the civilian power program, the Commissioners were subjected to sharp criticism by Clarence Cannon, chairman of the **House Appropriations Committee**, when they appeared to testify in June 1956 on the need for the expenditures.

Cannon, a strong public power advocate, badgered Strauss about private industry's lack of progress in atomic development and suggested that the PRDC had no "intention of building this reactor at any time in the determinable future." Strauss, anxious to show that industry was making good headway, replied: "They [PRDC] have already spent eight million dollars of their own money to date on this project. I told you they were breaking ground on August 8. I have been invited to attend the ceremony; I intend to do so." Inadvertently, he had revealed that he planned to attend the ground breaking ceremony for a reactor whose construction permit was still being evaluated by the AEC.

During hearings the following day, Commissioner Murray, in an effort to demonstrate the need for research and development funds, disclosed the conclusions of the ACRS on the PRDC application. Murray was so uneasy about the safety implications of the committee's report that he went to see Joint Committee Chairman Anderson and outlined its contents.

Members of the Joint Committee were angered and disturbed by the revelations of Strauss and Murray, not only because of safety concerns but also because the AEC had failed to inform them officially about the reservations of the ACRS. The AEC was obliged by the 1954 Atomic Energy Act to keep the Joint Committee "fully and currently informed" about its activities, and committee members believed that in the case of the ACRS report the agency had failed to carry out its charge.

The Joint Committee immediately requested a copy of the ACRS document. The AEC was reluctant to agree, and after long deliberation, offered to deliver a copy only if the Joint Committee would keep it "administratively confidential." The committee refused to accept the report under those conditions.

The AEC was even less accommodating with the state of Michigan. When Governor G. Mennen Williams, who learned of the ACRS report from Senator Anderson, asked the AEC for a copy, it refused on the grounds that "it would be inappropriate to disclose the contents of internal documents."

Meanwhile, the AEC's regulatory staff was completing its review of the PRDC's application. The staff took a more optimistic view of the safety of the proposed reactor than had the ACRS. Since the company had agreed to perform tests on the questions raised by the committee, the staff recommended that it be granted a construction permit.

On August 2, 1956, the Commission decided to issue the permit by a vote of three to one (Murray was the dissenter). It acknowledged the concerns of the ACRS by inserting the word "conditional" in the construction permit to emphasize that the company would have to settle the uncertainties about safety before receiving an operating license. Commissioner Harold S. Vance summarized the majority's reasoning during discussion of the application. "We are doing something that we ordinarily would not do," he said, "in that we would not ordinarily issue a construction permit unless we were satisfied that reasonable safety requirements had been met." But he added: "It may be some time before reasonable assurance can be obtained. If we were to delay the construction permit until then, it might delay a very important program. If we didn't think that the chances were very good that all these questions would be resolved, we would not issue the permit."

The AEC's decision elicited angry protests from the Joint Committee. Congressman Holifield, citing Strauss's earlier announcement of his plans to attend the groundbreaking ceremonies for the plant, charged that the AEC chairman was acting in a "reckless and arrogant manner." Anderson accused the agency of conducting "star chamber" proceedings and pledged that the Joint Committee would "ascertain the full facts involved in this precipitate action

1.7. THE PRICE ANDERSON ACT

The Joint Committee soon acted to prevent a recurrence of the AEC's conduct in the PRDC case. Anderson ordered the committee staff to prepare a study of the AEC's licensing procedures and regulatory organization, including consideration of whether regulatory and promotional responsibilities should be carried out by separate agencies.

The staff concluded that the creation of separate agencies was inadvisable at the time, principally because of the difficulty of recruiting qualified personnel for purely regulatory functions. It did, however, suggest other reforms in the AEC's regulatory structure and procedures. Anderson implemented his staff's proposals by introducing legislation to establish the ACRS as a statutory body, direct that its reports on licensing cases be

made public, and require public hearings on all reactor applications. The AEC opposed all three measures, but muted its objections because Anderson presented them as amendments to a bill to provide indemnity insurance for reactor owners, which the agency strongly favored.

The AEC regarded indemnity legislation as essential for stimulating private investment in nuclear power, a view that industry spokesmen and the **Joint Committee on Atomic Energy** shared. Since they recognized that the chances of a severe reactor accident could not be reduced to zero, even the most enthusiastic industry proponents of atomic power were reluctant to push ahead without adequate liability insurance.

Private insurance companies would offer up to $60 million of coverage per reactor, an amount that far exceeded what was available to any other industry in the United States. But in the event of a serious accident, it seemed insufficient to pay claims for deaths, injuries, and property damages in areas surrounding the malfunctioning plant. Therefore, industry executives sought a government program to provide additional insurance protection.

H. R. Searing, chairman of the board of Consolidated Edison, declared that although his company would proceed with the construction of its Indian Point plant near New York City it would not load fuel and begin operation unless the insurance question were resolved. General Electric's Francis McCune went even further by telling the Joint Committee in 1957 that if Congress did not enact indemnity legislation, his company would stop work on Commonwealth Edison's Dresden station, then under construction. He suggested that without a government insurance plan, the market for civilian atomic energy would collapse and vendors would withdraw from the field.

Spurred by the industry's concerns, both the AEC and the Joint Committee considered methods by which the government could provide additional liability insurance for reactor owners. Their efforts culminated in legislation introduced by Senator Anderson and Congressman Melvin Price, which proposed that the government underwrite $500 million of insurance beyond the $60 million available from private companies. The AEC initially opposed setting a specific upper limit on the amount because there was no reliable way to estimate the possible damages from a reactor accident. But Anderson, wanting to avoid a "blank check" for industry, rather arbitrarily decided on the $500 million figure. The bill stipulated that Congress could authorize additional payments if necessary and also required that reactor owners contribute funds to the insurance pool as their plants were licensed. With strong support from the AEC and the industry, Congress passed the Price-Anderson bill in August 1957.

In final form, the measure included Anderson's reforms of the AEC's licensing procedure. Although the agency disliked Anderson's amendments, it accepted them to avoid jeopardizing or retarding approval of the indemnity bill. The Price-Anderson Act was a regulatory measure in effect because it provided insurance protection to victims of a nuclear accident, but it was largely promotional in motivation. Industry, the AEC, and

the Joint Committee believed that it would remove a serious obstacle to private atomic development.

1.8. THE GROWTH OF NUCLEAR POWER

The PRDC case and the Price-Anderson Act clearly illustrated the AEC's emphasis on developmental rather than regulatory efforts. The precedence that the AEC gave to promoting the growth of nuclear power resulted from a number of considerations. The **1954 Atomic Energy Act** made it a national goal to encourage the widespread use of atomic energy for peaceful purposes, but private industry was often hesitant to assume the costs and risks of development.

The AEC sought to persuade or induce private interests to invest in nuclear power. This seemed particularly urgent because of the intense pressure the Joint Committee placed on the agency to speed progress and its persistent threat to require the AEC to construct prototype plants if private firms failed to act promptly. One important way that the AEC pursued its objective of private development was to write regulations designed to protect public safety without being overly burdensome to industry.

Safety questions were largely a matter of judgment rather than something concrete or quantifiable, and AEC officials found it easier to assume that such issues had been or would be satisfactorily resolved than to assume that reactors would be built. When it issued a construction permit for the PRDC fast breeder reactor, for example, the Commission's vision of an advanced technology plant that showed the effectiveness of its power demonstration reactor program outweighed the reservations of the ACRS.

Aware of the implications that safety questions posed for the development of the technology, the AEC believed that nuclear science, in due time, would provide the answers to any outstanding problems. In short, the desire for tangible signs of promise was more compelling than first resolving more ethereal safety issues.

The AEC's emphasis on stimulating atomic development did not mean that it was inattentive to safety issues. The regulations that the staff drafted shortly after passage of the **1954 Atomic Energy Act** reflected careful consideration of the best scientific information and judgment available at the time. The AEC recognized and publicly acknowledged the possibility of accidents in such a new and rapidly changing technology; it never offered absolute assurances that accidents would not occur. Nevertheless, it believed that compliance with its regulations would make the chances of a serious accident very small.

The agency did not view its developmental efforts as more important than regulatory policies, but it clearly viewed the need to encourage industrial growth as more immediate.

By 1962, the AEC's efforts to stimulate private participation in nuclear power development had produced some encouraging results. In a report to President Kennedy, the agency proudly pointed out that in the short time since atomic technology had been

opened to private enterprise, six "sizeable" power reactors had begun operation, and two of those had been built without government subsidies.

Despite industry's lingering concerns about the costs of nuclear power relative to fossil fuels, the AEC's developmental and regulatory programs had fostered the initial growth of commercial nuclear power. The agency predicted that by the year 2000 nuclear plants might provide up to fifty percent of the nation's electrical generating capacity. Despite the AEC's claims, the future of the nuclear industry remained precarious. The fourteen reactors in operation or under construction were still far from being commercially competitive or technologically proven, and interest in further development among utilities appeared to be flagging. Both the AEC and Joint Committee were acutely aware of and deeply disturbed about those uncertainties.

To make matters worse from the perspective of nuclear proponents, there were signs of increasing public opposition to, or at least concern about, nuclear power hazards. In the early days of nuclear power development, public attitudes toward the technology were highly favorable, as the few opinion polls on the subject revealed. Press coverage of nuclear power was also overwhelmingly positive. An article in *National Geographic* in 1958, for example, concluded that "abundant energy released from the hearts of atoms promises a vastly different and better tomorrow for all mankind."

In the late 1950s and early 1960s, however, the public became more alert to and anxious about the hazards of radiation, largely as a result of a major controversy over radioactive fallout from nuclear weapons testing. One result was that the public became increasingly troubled about the risks of exposure to radioactivity from many sources, including nuclear power.

1.9. RADIATION PROTECTION

Before World War II, the dangers of radiation were a matter of interest and concern mostly to a relatively small group of scientists and physicians. Within a short time after the discovery of x-rays and natural radioactivity in the 1890s, scientific investigators concluded that exposure to radiation could cause serious health problems, ranging from loss of hair and skin irritations to sterility and cancer.

Ignorance of the hazards of x-rays and radium and use of them for frivolous purposes led to tragic consequences for people who received large doses of radiation. As experience with and experimental data on the effects of radiation gradually accumulated, professionals developed guidelines to protect x-ray technicians and other radiation workers from excessive exposure.

In 1934, a recently formed American committee representing professional societies and x-ray equipment manufacturers recommended for the first time a quantitative "tolerance dose" of radiation, 0.1 roentgen per day of whole-body exposure from external sources. Committee members believed that levels of radiation below the tolerance dose were generally safe and unlikely to cause injury "in the average individual."

The following year, an international radiation protection committee composed of experts from five nations took similar action. Neither body regarded its recommended tolerance dose as definitive because empirical evidence remained fragmentary and inconclusive. They were confident, however, that available information made their proposals reasonable and provided an adequate margin of safety for the relatively small number of individuals exposed to radiation in their jobs.

Then came Hiroshima. The dawn of the atomic age made radiation safety a vastly more complex task for two reasons.

1. Nuclear fission created many radioactive isotopes that did not exist in nature. This meant that instead of considering only x-rays and radium, professionals in the field of radiation protection had to evaluate the hazards of new radioactive substances about which even less was known.
2. The problem of radiation safety extended to significantly larger segments of the population who might be exposed to radiation from the development of new applications of atomic energy.

Radiation protection broadened from a medical issue of limited proportions to a public health question of, potentially at least, major dimensions. As a result of the drastically altered circumstances, scientific authorities reassessed their recommendations on radiation protection. They modified their philosophy of radiological safety by abandoning the concept of "tolerance dose," which assumed that exposure to radiation below the specified limits was generally harmless. Experiments in genetics indicated that reproductive cells were highly susceptible to damage from even small amounts of radiation.

By the early 1940s, most scientists had rejected the idea that exposure to radiation below a certain threshold was inconsequential, at least for genetic effects. The American committee of radiation experts, named the **National Committee on Radiation Protection** (NCRP) in 1946, took action that reflected the consensus of opinion by replacing the terminology of "tolerance dose" with "maximum permissible dose," which it thought better conveyed the principle that no quantity of radiation was certifiably safe. It defined the permissible dose as that which "in the light of present knowledge, is not expected to cause appreciable bodily injury to a person at any time during his lifetime." While acknowledging the possibility of suffering harmful effects from radiation in amounts below the allowable limits, the NCRP emphasized that the permissible dose was based on the belief that "the *probability* of the occurrence of such injuries must be so low that the risk should be readily acceptable to the average individual."

Because of the growth of atomic energy programs and the substantial increase in the number of individuals working with radiation sources, the NCRP decided by 1948 to reduce its recommended occupational exposure limits to fifty percent of the 1934 level. Its international counterpart, named **the International Commission on Radiological Protection** (ICRP) after World War II, adopted the same maximum permissible dose.

The new maximum permissible whole body dose that the NCRP and ICRP recommended was 0.3 roentgens per six-day work week, measured by exposure of the "most critical" tissue in blood-forming organs, gonads, and lens of the eye. Higher limits applied for less sensitive areas of the body. In addition to the levels established for exposure to x-rays or gamma rays, the NCRP and ICRP also issued maximum permissible concentrations in air and water of a list of radioactive isotopes that give off alpha or beta particles, known as "internal emitters."

Alpha and beta particles cannot penetrate into vital human tissue from outside the body, but if they enter the body by consumption of contaminated food or water or by breathing of contaminated air, they can pose a serious health hazard.

The allowable limits established by both groups applied only to radiation workers, but because of the genetic effects of radiation and the possibility that other people could be exposed in an accident or an emergency, each also issued guidelines for larger segments of the population. In view of the greater sensitivity of young persons to radiation, the NCRP recommended that the occupational maximum permissible dose be reduced by a factor of ten for anyone under age eighteen.

The ICRP went further by proposing a limit of one-tenth the occupational level for the general population. Neither committee had any legal authority or official standing, but since their recommendations reflected the findings and opinions of leading experts in the field of radiation protection, they exercised decisive influence on government agencies concerned with radiological safety.

The AEC used the NCRP's occupational limits in its own installations, and after passage of the **1954 Atomic Energy Act,** in its regulations for licensees. The agency's radiation protection regulations, which were first issued for public comment in 1955 and became effective in 1957, followed the NCRP's recommendations for radiation workers and set a permissible dose of one-tenth the occupational level for members of the general population potentially affected by the operations of licensees.

1.10. THE FALLOUT CONTROVERSY

In the immediate postwar period, deliberations over the risks of radiation and permissible exposure levels were confined mostly to scientific circles. Concern about radiation moved from the rarified realms of scientific and medical discourse to the front page as a result of the fallout controversy.

The testing of nuclear weapons in the atmosphere by the United States, the Soviet Union, and Great Britain produced radioactive fallout that spread to populated areas far from the sites of the explosions. The fallout debate made radiation hazards a bitterly contested political issue for the first time.

Scientists disagreed sharply about how serious a risk fallout presented to the population, and the question became a prominent subject in news reports, magazine stories, political campaigns, congressional hearings, and scientific studies. This not only called public attention to the potential health hazards of relatively small amounts of radiation

(as opposed to acute exposure), but also made clear that scientists did not know a great deal about the effects of low-level radiation.

The fallout controversy affected the AEC's regulatory program in two important ways.

1. It led to a tightening of the agency's radiation standards. In response to increasing public concern and the findings of scientific groups, the NCRP and the ICRP both lowered their recommended permissible levels of exposure. They acted to provide a larger margin of safety but emphasized that there was no evidence that the previous levels had been dangerously high. They reduced their limits for occupational exposure to an average of 5 rem per year after age eighteen while continuing to suggest that population levels be restricted to ten percent of occupational levels (0.5 rem per year) for individuals. They added a new stipulation that, for genetic reasons, the average level for large population groups should not exceed one-thirtieth of the occupational limit, or 0.17 rem per year. The AEC promptly adopted the new recommendations as a part of its regulations; it issued them for comments in 1959 and made them effective on January 1, 1961.
2. The fallout debate further influenced the AEC's regulatory program by arousing public anxieties about the health effects of low-level radiation. This was evident, for example, in citizen protests against the dumping of low-level radioactive wastes in ocean waters. The AEC had authorized the dumping of such wastes under prescribed conditions for over a decade, but it became a subject of controversy only after the fallout issue sensitized public opinion to radiation hazards. In a similar manner, the first widespread objections to the construction of proposed nuclear power plants arose in the wake of the fallout debate.

At the end of the first decade that followed passage of the **1954 Atomic Energy Act,** the prospects for rapid nuclear power development were mixed. Impressive strides had been taken, to be sure, but many uncertainties remained. Public support for the technology appeared to be strong but, as Ravenswood and Bodega Bay had shown, it could not be taken for granted.

Beginning in the mid-1960s, however, a variety of considerations fueled an unanticipated boom in the nuclear power industry that resolved some of the unknowns about nuclear progress while raising a host of new questions for the AEC's regulatory staff.

SECTION TWO: THE NUCLEAR POWER DEBATE, 1963-75

2.1. THE GREAT BANDWAGON MARKET

During the late 1950s and early 1960s the use of nuclear power to generate electricity was a novel and developing technology. Since relatively few plants were operating, under construction, or on order, the scope of the AEC's regulatory functions such as reactor siting, licensing, and inspection was still limited.

During the later 1960s, however, the nation's utilities rapidly increased their orders for nuclear power stations, participating in what Philip Sporn, past president of the **American Electric Power Service Corporation**, described in 1967 as the "great bandwagon market." At the same time, the size of plants being built also expanded dramatically. The sudden arrival of commercially competitive nuclear power placed unprecedented demands on the AEC's regulatory staff and raised new safety problems that reactor experts had not considered previously. The surge in reactor orders and the growth in the size of individual reactors also spurred new concerns about the environmental impact of nuclear power and intensified public uneasiness about the safety of the technology.

The bandwagon market was an outgrowth of several developments that enhanced the appeal of nuclear power to utilities in the mid- and late 1960s. One was the intense competition between the two leading vendors of nuclear plants, General Electric and Westinghouse.

In 1963, General Electric made a daring move to increase its reactor sales and to convince utilities that nuclear power had arrived as a safe, reliable, and cost-competitive alternative to fossil fuel. It offered a "turnkey" contract to **Jersey Central Power and Light Company** to build the 515 electrical megawatt Oyster Creek plant near Toms River, New Jersey. For a fixed cost of $66 million, General Electric agreed to supply the entire plant to the utility (the term "turnkey" suggested that the utility would merely have to turn a key to start operating the facility).

The company's bid was successful, winning out not only over Westinghouse but also over manufacturers of coal-fired units. General Electric expected to lose money on the Oyster Creek contract, but hoped that the plant would help to stimulate the market for nuclear power.

The Oyster Creek contract opened the "turnkey era" of commercial nuclear power and came to symbolize the competitive debut of the technology. Glenn T. Seaborg, chairman of the AEC, told President Johnson that it represented an "economic breakthrough" for nuclear electricity.

Westinghouse followed General Electric's lead in offering turnkey contracts for nuclear plants, setting off a fierce corporate battle. The turnkey plants were a financial blow for both companies; their losses ran into the hundreds of millions of dollars before they stopped making turnkey arrangements. One General Electric official commented: "It's going to take a long time to restore to the treasury the demands we put on it to establish ourselves in the nuclear business." But the turnkey contracts fulfilled General Electric's hopes of stirring interest among and orders from utilities. They played a major role in triggering the bandwagon market.

There were other important considerations that convinced a growing number of utilities to buy nuclear plants. The spread of power pooling arrangements among utilities, which

encouraged the construction of larger generating stations by easing fears of excess capacity and over- expansion.

A utility with extra or reserve power could sell it to other companies through interconnections. The desirability and feasibility of using larger individual plants worked to the benefit of nuclear vendors. They emphasized that bigger plants would produce "economies of scale" that would cut capital costs per unit of power and improve efficiency. This helped to overcome a major disadvantage of nuclear power relative to fossil fuel--the heavy capital requirements for building atomic plants. During the late 1960s designs for nuclear facilities leapfrogged from the 500 to the 800 to the 1000 electrical megawatt range even though operating experience was still limited to units in the range of 200 megawatts or less. The practice of "design by extrapolation" had been employed for fossil-fuel units since the early 1950s. Before the mid-1960s this approach appeared to work well, and it was natural that vendors extended it to nuclear units.

In addition to turnkey contracts, system interconnections, and increasing unit size, growing national concern about air pollution in the 1960s made nuclear power more attractive to utilities. Coal plants were major contributors to the deterioration of air quality and were obvious targets for clean-up efforts.

As the campaign to improve the environment gained strength, the electric utility industry became more mindful of the cost of pollution control in fossil-fuel plants. They increasingly viewed nuclear power as a good alternative to paying the expenses of pollution abatement in coal-fired units.

The bandwagon market for nuclear power reached its peak during 1966 and 1967, exceeding, in the words of a General Electric official, "even the most optimistic estimates." In 1965, the year before the reactor boom gathered momentum, nuclear vendors sold four nuclear plants with a total of 17 percent of the capacity that utilities purchased that year. In 1966, by contrast, utilities bought 20 nuclear units that made up 36 percent of the electrical capacity committed. The following year nuclear vendors sold 31 units that represented 49 percent of the capacity ordered. In 1968, the number of reactor orders dropped to 17, but the percentage of the capacity filled with nuclear plants remained high at 47 percent.

The bandwagon market orders were large facilities that far exceeded the size of operating reactors. Between 1963, when the 515 electrical megawatt Oyster Creek reactor was ordered, and 1969, when the plant began operation, the AEC issued 38 construction permits for units that were larger than Oyster Creek. Of those plants, 28 were in the range of 800 to 1100 megawatts. The degree of extrapolation from small plants to mammoth ones was a matter of concern even to some strong nuclear advocates. By the late 1960s, it was apparent that design by extrapolation was not as successful as anticipated earlier.

2.2. BURDENS OF THE BANDWAGON MARKET

The rapid increase in the number of reactor applications and in the size of proposed plants placed enormous burdens on the AEC's regulatory staff. The flood of applications inevitably caused licensing delays because the staff lacked enough qualified professionals. Between 1965 and 1970, the size of the regulatory staff increased by about 50 percent, but its licensing and inspection case load increased by about 600 percent.

The average time required to process a construction permit application stretched from about a year in 1965 to over 18 months by 1970. The growing backlog drew bitter complaints from utilities applying to build plants and from nuclear vendors. One utility executive predicted that if delays became commonplace, "it can safely be asserted that the splendid promise of nuclear power will have had a very short life." Another was even more critical, calling the licensing process "a modern day Spanish Inquisition" carried out by "AEC engineers, scientists, and consultants {who} have no serious economic discipline." The AEC attempted to streamline its licensing procedures but found it impossible to reduce review time or to satisfy the demands of the industry.

The licensing process lengthened not only because of the number of applications that the AEC had to evaluate but also because of the complexity of the proposals it received. The growth in the size of reactors and the practice of design by extrapolation raised many complex safety issues that could not be easily resolved.

The exercise of careful judgment in assessing reactor applications was always critical, but it became even more so as utilities campaigned to build plants closer to populated regions. Although the AEC adopted an informal prohibition against "metropolitan siting" in urban locations (such as the proposed Ravenswood plant in downtown New York), it was more receptive to "suburban siting" fairly close to urban populations.

This reduced the emphasis on one traditional means of protecting the public from the consequences of a nuclear accident--"remote siting." It placed greater dependence on the other general method of shielding the public from the effects of an accident--engineered safeguards (a term later superseded by "engineered safety features") that were built into the plant. Even as the relative importance of engineered safeguards increased in the 1960s, questions arose about their reliability in preventing a massive release of radioactivity to the environment in the event of a severe accident.

2.3. ENGINEERED SAFEGUARDS

The engineered safeguards in nuclear plants differed in design and operation, but they served the same basic functions. A number of systems were placed in reactors to remove heat and reduce excessive pressure if an accident occurred.

They included, for example, passive core sprays and pressure suppression pools, "safety injection" systems that would shoot large volumes of water into the reactor vessel, and combinations of filters, vents, scrubbers, and air circulators that would collect and retain radioactive gases and particles released by an accident.

The final line of defense if the engineered safeguards failed was the containment building, a large, often dome-shaped structure that surrounded the reactor and associated steam-producing equipment as well as the safety systems.

Reactor experts were confident that in almost any situation the engineered safety features built into a plant and the containment structure would protect the public from the effects of an accident. But they were troubled by the possibility that a chain of events could conceivably take place that would bypass or override the safety systems, and in the worst case, breach containment. "No one is in a position to demonstrate that a reactor accident with consequent escape of fission products to the environment will never happen," Clifford K. Beck, the AEC's deputy director of regulation, told the Joint Committee in 1967. "No one really expects such an accident, but no one is in a position to say with full certainty that it will not occur."

The AEC strived to reduce the likelihood of an accident to a minimum. It based its decisions on the safety of reactor designs and plant applications on operating experience, engineering judgment, and experiments with test reactors. Experience with the first commercial reactors had been encouraging; it had provided a great deal of information that was useful in understanding reactor science. But it was of limited application to the newer and larger reactors that utilities were building by the late 1960s.

The rapid growth in reactor design placed a premium on the careful use of engineering judgment. In order to decrease the chances of a major accident that could threaten public health, the AEC required multiple back-up equipment and redundancies in safety designs. It also employed conservative assumptions about the ways in which an accident might damage or incapacitate safety systems in its evaluation of reactor proposals.

The regulatory staff sought to gain as much experimental data as possible to enrich its knowledge and inform its collective engineering judgment. This was especially vital in light of the many unanswered questions about reactor behavior.

The AEC had sponsored hundreds of small-scale experiments since the early 1950s that had yielded key information about a variety of reactor safety problems. But they provided little guidance on the issue of greatest concern to the AEC and the ACRS by the late 1960s--**a core meltdown caused by a loss-of-coolant accident.**

Reactor experts had long recognized that a core melt was a plausible, if unlikely, occurrence. A massive loss of coolant could happen, for example, if a large pipe that fed cooling water to the core broke. If the plant's emergency cooling system also failed, the build-up of "decay heat" (which resulted from continuing radioactive decay after the reactor shut down) could cause the core to melt.

In older and smaller reactors, the experts were confident that even under the worst conditions--an accident in which the loss of coolant melted the core and it, in turn, melted through the pressure vessel that held the core--the containment structure would prevent a massive release of radioactivity to the environment. As proposed plants

increased significantly in size, however, they began to worry that a core melt could lead to a breach of containment. This became their primary focus partly because of the greater decay heat the larger plants would produce and partly because nuclear vendors did not add to the size of containment buildings in corresponding proportions to the size of reactors.

2.4. THE PROBLEM OF CORE MELTDOWN

The greatest source of concern about a loss-of-coolant accident in large reactors was that the molten fuel would melt through not only the pressure vessel but also through the thick layer of concrete at the foundation of the containment building. The intensely radioactive fuel would then continue on its downward path into the ground. This scenario became known as the "**China syndrome**," because the melted core would presumably be heading through the earth toward China.

Other possible dangers of a core meltdown were that the molten fuel would breach containment by reacting with water to cause a steam explosion or by releasing elements that could combine to cause a chemical explosion. The precise effects of a large core melt were uncertain, but it was clear that the results of spewing radioactivity into the atmosphere could be disastrous.

The **Advisory Committee for Reactor Safety** (ACRS) and the regulatory staff regarded the chances of such an accident as low; they believed that it would occur only if the **emergency core cooling system** (ECCS), made up of redundant equipment that would rapidly feed water into the core, failed to function properly. But they acknowledged the possibility that the ECCS might not work as designed. Without containment as a fail-safe final line of defense against any conceivable accident, they sought other means to provide safeguards against the China syndrome.

2.5. THE EMERGENCY CORE COOLING CONTROVERSY

At the prodding of the ACRS, which first sounded the alarm about the China syndrome, the AEC established a special task force to look into the problem of core melting in 1966. The committee, chaired by William K. Ergen, a reactor safety expert and former ACRS member from **Oak Ridge National Laboratory,** submitted its findings to the AEC in October 1967.

The report offered assurances about the improbability of a core meltdown and the reliability of emergency core cooling designs, but it also acknowledged that a **loss-of-coolant accident** (LOCA) could cause a breach of containment if the **Emergency Core Cooling System** (ECCS) failed to perform.

Containment could no longer be regarded as an inviolable barrier to the escape of radioactivity. This represented a milestone in the evolution of reactor regulation. In effect, it imposed a modified approach to reactor safety. Previously, the AEC had viewed the containment building as the final independent line of defense against the release of radiation; even if a serious accident took place the damage it caused would be restricted to the plant.

Once it became apparent that under some circumstances the containment building might not hold, however, the key to protecting the public from a large release of radiation was to prevent accidents severe enough to threaten containment. And this depended heavily on a properly designed and functioning ECCS.

The problem facing the AEC regulatory staff was that experimental work and experience with emergency cooling was very limited. Finding a way to test and to provide empirical support for the reliability of emergency cooling became the central concern of the AEC's safety research program.

Plans had been underway since the early 1960s to build an experimental reactor, known as the **Loss-of-Fluid-Tests** (LOFT) facility, at the AEC's reactor testing station in Idaho. Its purpose was to provide data about the effects of a loss of coolant accident. For a variety of reasons, including weak management of the test program, a change of design, and reduced funding, progress on the LOFT reactor and the preliminary tests that were essential for its success were chronically delayed.

Despite the complaints of the ACRS and the regulatory staff, the AEC diverted money from LOFT and other safety research projects on existing light-water reactor design to work in the development of fast-breeder reactors.

A proven fast breeder was an urgent objective for the AEC and the Joint Committee; Seaborg described it as "a priority national goal" that could assure "an essentially unlimited energy supply, free from problems of fuel resources and atmospheric contamination."

To the consternation of the AEC, experiments run at the Idaho test site in late 1970 and early 1971 suggested that the ECCS in light-water reactors might not work as designed. As a part of the preliminary experiments that were used to design the LOFT reactor, researchers ran a series of "semiscale" tests on a core that was only nine inches long (compared with l44 inches on a power reactor). The experiments were run by heating a simulated core electrically, allowing the cooling water to escape, and then injecting the emergency coolant. To the surprise of the investigators, the high steam pressure that was created in the vessel by the loss of coolant blocked the flow of water from the ECCS. Without even reaching the core, about 90 percent of the emergency coolant flowed out of the same break that had caused the loss of coolant in the first place.

In many ways the semiscale experiments were not accurate simulations of designs or conditions in power reactors. Not only the size, scale, and design but also the channels that directed the flow of coolant in the test model were markedly different than those in an actual reactor.

Nevertheless, the results of the tests were disquieting. They introduced a new element of uncertainty into assessing the performance of ECCS. The outcome of the tests had not been anticipated and called into question the analytical methods used to predict what

would happen in a loss-of-coolant accident. The results were hardly conclusive but their implications for the effectiveness of ECCS were troubling.

The semiscale tests caught the AEC unprepared and uncertain of how to respond. Harold Price, the director of regulation, directed a special task force he had recently formed to focus on the ECCS question and to draft a "white paper" within a month. Seaborg, for the first time, called the **Office of Management and Budget** to plead for more funds for safety research on light-water reactors.

While waiting for the task force to finish its work, the AEC tried to keep information about the semiscale tests from getting out to the public, even to the extent of withholding information about them from the Joint Committee. The results of the tests came at a very awkward time for the AEC. It was under renewed pressure from utilities facing power shortages and from the Joint Committee to streamline the licensing process and eliminate excessive delays. At the same time, Seaborg was appealing--successfully--to President Nixon for support of the breeder reactor, and controversy over the semiscale tests and reactor safety could undermine White House backing for the program. By the spring of 1971, nuclear critics were expressing opposition to the licensing of several proposed reactors, and news of the semiscale experiments seemed likely to spur their efforts.

For those reasons, the AEC sought to resolve the ECCS issue as promptly and quietly as possible. It wanted to settle the uncertainties about safety without arousing a public debate that could place hurdles in the way of the bandwagon market. Even before the task force that Price established completed its study of the ECCS problem, the Commission decided to publish "interim acceptance criteria" for emergency cooling systems that licensees would have to meet.

It imposed a series of requirements that it believed would ensure that the ECCS in a plant would prevent a core melt after a loss-of-coolant accident. The AEC did not prescribe methods of meeting the interim criteria, but in effect, it mandated that manufacturers and utilities set an upper limit on the amount of heat generated by reactors. In some cases, this would force utilities to reduce the peak operating temperatures (and hence, the power) of their plants. Price told a press conference on June 19, 1971 that although the AEC thought it impossible "to guarantee absolute safety," he was "confident that these criteria will assure that the emergency core cooling systems will perform adequately to protect the temperature of the core from getting out of hand."

The interim ECCS criteria failed to achieve the AEC's objectives. News about the semiscale experiments triggered complaints about the AEC's handling of the issue even from friendly observers. It also prompted calls from nuclear critics for a licensing moratorium and a shutdown of the eleven plants then operating. Criticism expressed by the **Union of Concerned Scientists** (UCS), an organization established in 1969 to protest misuse of technology that had recently turned its attention to nuclear power, received wide publicity.

The UCS took a considerably less sanguine view of ECCS reliability than that of the AEC. It sharply questioned the adequacy of the interim criteria, charging, among other things, that they were "operationally vague and meaningless." Scientists at the AEC's national laboratories, without endorsing the alarmist language that the UCS used, shared some of the same reservations.

As a result of the uncertainties about ECCS and the interim criteria, the AEC decided to hold public hearings that it hoped would help resolve the technical issues. It wanted to prevent the ECCS question from becoming a major impediment to the licensing of individual plants.

The AEC insisted that its critics had exaggerated the severity of the ECCS problem. The regulatory staff viewed the results of the failed semiscale tests as serious but believed that the technical issues the experiments raised would be resolved within a short time. It did not regard the tests as indications that existing designs were fundamentally flawed and it emphasized the conservative engineering judgment it applied in evaluating plant applications. But the ECCS controversy damaged the AEC's credibility and played into the hands of its critics.

Instead of frankly acknowledging the potential significance of the ECCS problem and taking time to fully evaluate the technical uncertainties, the AEC acted hastily to prevent the issue from undermining public confidence in reactor safety or causing licensing delays. This gave credence to the allegations of its critics that it was so determined to promote nuclear power and develop the breeder reactor that it was inattentive to safety concerns.

2.6. NUCLEAR POWER AND THE ENVIRONMENT

By the time that the ECCS issue hit the headlines, other questions about the environmental effects of nuclear power had eroded public support for the technology. The problem of industrial pollution and the deteriorating quality of the natural environment took on growing urgency as a public policy issue during the 1960s.

The increasing public and political concern with environmental protection, occurring at the same time that demand for electricity was doubling every ten years or so, placed utilities in a quandary. As an article in *Fortune* magazine put it: "Americans do not seem willing to let the utilities continue devouring...ever increasing quantities of water, air, and land. And yet clearly they also are not willing to contemplate doing without all the electricity they want. These two wishes are incompatible. That is the dilemma faced by the utilities.

Utilities increasingly viewed nuclear power as the answer to that dilemma. It promised the means to meet demand for power without causing air pollution, and environmental concerns were a major spur to the growth of the great bandwagon market.

Environmentalists recognized the benefits of nuclear power compared to fossil fuel, but they were more equivocal in their attitudes toward the technology than were industry representatives. Their ambivalence was perhaps best summarized by the statement of a

leading environmental spokesman in 1967: "I think most conservationists may welcome the coming of nuclear plants, though we are sure they have their own parameters of difficulty."

Officials of the AEC actively promoted the idea that nuclear power provided the answer to both the environmental crisis and the energy crisis. Seaborg was especially outspoken on this point. Although he acknowledged that nuclear power had some adverse impact on the environment, he insisted that its effects were much less harmful than those of fossil fuel. In comparison with coal, he once declared, "there can be no doubt that nuclear power comes out looking like Mr . Clean."

2.7. THERMAL POLLUTION

The view of nuclear power as beneficial to the environment relative to conventional fuels was undermined in the late 1960s by a major controversy over the effects of waste heat from nuclear plants on water quality, widely known as "thermal pollution."

Thermal pollution resulted from cooling the steam that drove the turbines to produce electricity in either a fossil fuel or nuclear plant. The steam was condensed by the circulation of large amounts of water, and in the process the cooling water was heated, usually by 10 to 20 degrees fahrenheit, before being returned to the body of water from which it came. This problem was not unique to nuclear plants but it was more acute in them, largely because fossil plants used steam heat more efficiently than nuclear ones. The problem of thermal pollution created more anxiety than previously during the 1960s because of the growing number of plants, the larger size of those plants, and the increasing inclination of utilities to order nuclear units.

Thermal pollution caused concern because it was potentially harmful to many species of fish. It could also disrupt the ecological balance in rivers and streams, allowing plants to thrive that made water look, taste, and smell unpleasant. Technical solutions to deal with thermal pollution were available, but they required extra costs in the construction and operation of steam-electric plants. Cooling towers of different designs or cooling ponds, for example, would greatly alleviate the release of waste heat to the source body of water. Utilities resisted adding cooling apparatus to the plants they planned to build, however, because of the expense and an appreciable loss of generating capacity.

Advocates of stronger federal action to protect the environment in the news media, Congress, and state and federal agencies urged the AEC to require its licensees to guard against the effects of thermal pollution. The AEC refused on the grounds that it lacked the statutory authority to impose regulations on hazards other than radiation. It argued that the 1954 Atomic Energy Act restricted its regulatory jurisdiction to radiological dangers, a view the Department of Justice and federal courts upheld.

This did not placate the AEC's critics, who accused it of ignoring a serious problem that nuclear plants exacerbated. Several members of Congress introduced legislation to grant the AEC authority over thermal pollution but the agency opposed those measures unless fossil fuel plants had to meet the same conditions. The AEC feared that nuclear

power would be placed at a competitive disadvantage if plant owners had to provide cooling equipment that was not required in fossil-burning facilities. The AEC came under increasing criticism for its position. The most prominent attack appeared in a *Sports Illustrated* article in January 1969. It assailed the AEC for failing to regulate against thermal pollution and attributed its inaction to a fear of the "financial investment that power companies would have to make...to stop nuclear plants from frying fish or cooking waterways wholesale." The article was a distorted and exaggerated presentation, but it contributed to a growing perception that instead of being a solution to the dilemma of producing electricity without causing serious environmental damage, nuclear power was a part of the problem.

Eventually the controversy over thermal pollution died out. One reason was that Congress passed legislation that gave the AEC authority to regulate against thermal pollution and that applied to most fossil fuel plants as well. A more important reason was that utilities increasingly took action to curb the consequences of discharging waste heat.

Although they initially resisted the calls for cooling equipment, they soon found that the costs of responding to litigation, enduring postponements in the construction or operation of new plants, or suffering a loss of public esteem were less tolerable than those of building cooling towers or ponds. By 1971, most nuclear plants being built or planned for inland waterways (where the problem was most acute) included cooling systems.

The legacy of the thermal pollution debate lingered on. It undermined confidence in the AEC and wakened public doubts about the environmental impact of nuclear power. It played a vital role in transforming the ambivalence that environmentalists had demonstrated toward the technology into strong and vocal opposition. As a result of the thermal pollution issue, the AEC and the nuclear industry frequently found themselves included among the ranks of enemies of the environment.

2.8. THE RADIATION DEBATE

The thermal pollution question was the first but not the only debate over the effects of nuclear power that aroused widespread public concern in the late 1960s and early 1970s. A major controversy that arose over the effects of low-level radiation from the routine operation of nuclear plants also fed fears about the expanding use of the technology.

Drawing on the recommendations of the **National Committee on Radiation Protection**, the AEC had established limits for public exposure to radiation from nuclear plants of 0.5 rem per year for individuals. To determine the allowable release of radioactive effluents from a plant, it assumed that a person stood outdoors at the boundary of the facility 24 hours a day, 365 days a year. Licensees generally met the requirements easily. In 1968, for example, releases from most plants measured less than three percent of the permissible levels for liquid effluents and less than one percent for gaseous effluents.

The conservative assumptions of the AEC and the performance of operating plants did not prevent criticism of the AEC's radiation standards. A number of observers suggested that, in light of the uncertainties about the effects of low-level radiation, the AEC's regulations were insufficiently rigorous and should be substantially revised.

This first emerged as a widely-publicized issue when the state of Minnesota, responding to questions raised by environmentalists, stipulated in May l969 that a plant under construction must restrict its radioactive effluents to a level of about three percent of that allowed by the AEC.

The adequacy of the AEC's radiation standards became even more contentious in the fall of 1969, when two prominent scientists, John W. Gofman and Arthur R. Tamplin, suggested that if everyone in the United States received the permissible population dose of radiation, it would cause 17,000 (later revised to 32,000) additional cases of cancer annually. Gofman and Tamplin worked at **Livermore National Laboratory**, funded by the AEC, and their position as insiders gave their claims special credibility. They initially proposed that the AEC lower its limits by a factor of ten and later urged that it require zero releases of radioactivity.

Gofman and Tamplin not only argued that the existing standards of the AEC and other radiation-protection organizations were inadequate but also challenged the prevailing consensus that the benefits of nuclear power were worth the risks. Gofman was especially harsh in his analysis; he insisted that in its radiation protection regulations, "the AEC is stating that there is a risk and their hope that the benefits outweigh the number of deaths." He added: "This is legalized murder, the only question is how many murders."

The AEC denied Gofman's and Tamplin's assertions on the grounds that they extrapolated from high doses to estimate the hazards of low-level exposure, and that, furthermore, it was impossible for the entire nation to receive the levels of radiation that applied at plant boundaries. Most authorities in the field of radiation protection agreed with the AEC that the risks of effluents from nuclear power were far smaller than Gofman and Tamplin maintained.

Nevertheless, in an effort to provide an extra measure of protection, reassure the public, and undercut the appeal of its critics, in June 1971 the AEC issued for public comment new "design objectives" for nuclear plants that would, in effect, reduce the permissible levels of effluents by a factor of about one hundred. This action elicited protests from industry representatives and from radiation-protection professionals, but it did not impress many critics, who expressed doubt that the AEC would enforce the new guidelines. The controversy focused public attention, once again, on the effects of low-level radiation, but it did little to clarify a complex and ambiguous issue.

2.9. NEPA AND CALVERT CLIFFS

In addition to the objections that its positions on thermal pollution and radiation standards stirred, the AEC provoked sharp criticism for its response to the **National**

Environmental Policy Act (NEPA). The law, passed by Congress in December 1969 and signed by President Nixon on January 1, 1970, required federal agencies to consider the environmental impact of their activities. The measure was in many ways vague and confusing and it gave federal agencies broad discretion in deciding how to carry out its mandate.

The AEC acted promptly to comply with NEPA, but its procedures for doing so brought protests from environmentalists. The agency took a narrow view of its responsibilities under NEPA.

In a proposed regulation that it issued in December 1970, it included, for the first time, non-radiological issues in its regulatory jurisdiction. But it also stipulated that it intended to rely on the environmental assessments of other federal and state agencies (rather than conducting its own), it agreed to consider environmental issues in licensing board hearings only if raised by a party to the proceeding, and it postponed any review of NEPA issues in licensing cases until March 1971.

The AEC declined to take an expansive view of its responsibilities under NEPA for several reasons. One was the conviction that the routine operation of nuclear plants was not a serious threat to the environment and, indeed, was beneficial compared to burning fossil fuel. The major products of nuclear power generation that affected the environment, radiation releases and thermal discharges, were covered by other legislation.

Implementation of NEPA might divert the AEC's limited human resources from tasks that were more central to its mission. The regulatory staff was "all but overwhelmed" by the flood of reactor applications and did not relish the idea of having to spend large amounts of time on environmental reviews. Most importantly, the AEC feared that weighing environmental issues other than radiation and thermal releases would cause unwarranted delays in licensing plants. The time required for evaluating applications was already increasing and the AEC worried that NEPA could force a "quantum leap" in the length of the process. It sought to strike a balance between environmental concerns and the need for electrical power in framing its regulations.

Environmentalists complained that the AEC had failed to fulfill the purposes of NEPA and took the agency to federal court over the application of AEC's regulations to the Calvert Cliffs nuclear units, then under construction on the Chesapeake Bay in rural Maryland. On July 23, 1971, the United States Court of Appeals for the District of Columbia handed down a ruling that was a crushing defeat for the AEC. The court sternly rebuked the agency in its most widely-quoted statement: "We believe that the Commission's crabbed interpretation of NEPA makes a mockery of the Act." The Calvert Cliffs decision was, in the words of *Nucleonics Week,* a "stunning body blow" to the AEC and the nuclear industry.

The Calvert Cliffs decision was another in a series of setbacks for the AEC and nuclear power. It was apparent by the summer of 1971 that public distrust of the AEC was

growing and support for nuclear power was declining. The cumulative effect of controversies over ECCS, thermal pollution, radiation standards, NEPA, and other issues eroded public confidence in the AEC's commitment to safety and raised doubts about the benefits of nuclear power.

Antinuclear activists capitalized on growing uneasiness about the health and environmental effects of the technology. Some of the critics were well-informed and responsible in their arguments, but others were one-sided and inaccurate. Attempts by nuclear proponents to correct a plethora of misleading and exaggerated stories, advertisements, speeches, and other presentations inevitably failed to win as much attention or produce the same effect.

To make matters worse for the AEC, it suffered from the general disillusionment with the government, established institutions, and science that prevailed by the late 1960s, largely as a result of the Vietnam war. One college student summarized the situation after listening to a debate between Victor Bond, a radiation expert from Brookhaven National Laboratory, and a vocal AEC critic: "Dr. Bond sounds good but we can't believe him. He works for the government."

2.10. SCHLESINGER'S RESPONSE TO CALVERT CLIFFS

By the summer of 1971, the AEC was an embattled agency, largely though not exclusively because of regulatory issues. Seaborg, after serving as chairman for ten years, resigned his post in July 1971 and Nixon appointed James R. Schlesinger, assistant director of the **Office of Management and Budget**, to take his place.

Schlesinger was determined to make the AEC more responsive to environmental concerns and to improve it's tarnished public image. As an important first step in those efforts, he and William O. Doub, who took a seat on the Commission at the same time that Schlesinger assumed the chairmanship, concluded that the AEC should not appeal the Calvert Cliffs ruling, and, after considering the alternatives, their colleagues agreed. The AEC announced its decision on August 26, 1971.

The AEC's response to the Calvert Cliffs decision brought a storm of protests from utilities who feared long delays in the licensing of plants that were nearly ready for operation. Schlesinger explained the AEC's new position in a speech he delivered to a meeting of industry groups in Bal Harbour, Florida on October 20, 1971.

He told his audience that although the long-term outlook for nuclear power appeared "bullish," the pace of development depended on two variables: "first, the provision of a safe, reliable product; second, achievement of public confidence in that product." Schlesinger declared that the AEC's policy of promoting and protecting the industry had been justified to help nuclear power get started, but since the industry was "rapidly approaching mature growth," the AEC must redefine its responsibilities. "You should not expect the AEC," he announced, "to fight the industry's political, social, and commercial battles." Rather, he added, the agency's role was "primarily to perform as a referee serving the public interest." The message of Schlesinger's speech was unprecedented; it

proclaimed a sharp break with the AEC's history and a new direction in the agency's approach to its regulatory duties.

Schlesinger's efforts to narrow the divisions between nuclear proponents and critics and to recover the AEC's regulatory credibility produced, at best, mixed results. Many environmentalists were pleased with the AEC's acceptance of the Calvert Cliffs ruling and with Schlesinger's Bal Harbour speech.

Their guarded optimism about Schlesinger's attitudes was perhaps best summarized by the title of an article about him in *National Wildlife* magazine: "There's a *Bird Watcher* Running the Atomic Energy Commission." But major differences between the AEC and environmentalists remained; many of the same issues that had aroused concern before Schlesinger's arrival continued to generate controversy.

2.11. THE ECCS HEARINGS

One of those issues was the reliability of emergency core cooling systems. In light of the objections to the interim acceptance criteria for ECCS that the AEC had published in June 1971, the agency decided to hold a rulemaking hearing on the issue that would apply to all licensing cases. It hoped that this would avoid repeating the same procedures and deliberating over the same questions in case- by-case hearings and that generic hearings would provide a means to resolve issues common to all plants.

The ECCS hearings got underway in early 1972 and stretched into 135 days over a period of a year and a half. When they ended, the transcripts of the proceedings filled more than 22,000 pages. The ECCS hearings led to a final rule that made some small but important revisions in the interim criteria. They also produced acrimonious testimony and front-page headlines that often reflected unfavorably on the AEC's safety programs and that further damaged its credibility.

2.12. RADIACTIVE WASTE DISPOSAL

Another issue that undermined confidence in the AEC in the early 1970s was its approach to high-level radioactive waste disposal. The growth of the nuclear power industry made the safe disposal of intensely radioactive spent fuel rods and other waste materials an increasingly urgent matter.

The AEC had investigated means of dealing with reactor wastes for years, but had not found a solution to the problem. As early as 1957, a scientific consensus had concluded that deep underground salt beds were the best repositories for long-lived and highly radioactive wastes. In 1970, in response to increasing expressions of concern about the lack of a policy for high-level waste disposal from scientific authorities, members of Congress, and the press, the AEC announced that it would develop a permanent repository for nuclear wastes in an abandoned salt mine near Lyons, Kansas.

It aired its plans without conducting thorough geologic and hydrologic investigations, and the suitability of the site was soon challenged by the state geologist of Kansas and other scientists. The uncertainties about the site generated a bitter dispute between the AEC on the one side and members of Congress and state officials from Kansas on the other.

It ended in 1972 in great embarrassment for the AEC when the reservations of those who opposed the Lyons location proved to be well-founded.

In addition to debates over ECCS and high-level waste disposal, questions over reactor design and safety, quality assurance, the probability of a major reactor accident, and other issues fueled the controversy over nuclear power. The number of contested hearings for plant licenses steadily grew. The ongoing controversy frustrated Schlesinger's hopes of increasing public confidence in the AEC and of defusing the conflicts between opposing views.

By highlighting the issues on which the AEC's performance was suspect, it also obscured the requirements that the regulatory staff imposed over the protests and against the wishes of the nuclear industry, the high standards that it demanded in the design and construction of nuclear plants, and the conservative assumptions that it applied in evaluating plant applications and formulating radiation- protection regulations.

2.13. THE END OF THE AEC

As the nuclear power debate continued, the AEC came under increasing attacks for its dual responsibilities for developing and regulating the technology. This became a major argument that nuclear critics cited in their indictments of the AEC; it was, said one, "like letting the fox guard the henhouse."

The question of creating separate agencies to promote and to regulate the civilian uses of nuclear energy had arisen within a short time after passage of the **1954 Atomic Energy Act**, but in the early stages of nuclear development it had seemed premature and unwarranted. It gained greater support as both the industry and antinuclear sentiment grew, and it took on greater urgency after the Arab oil embargo and the energy crisis of 1973-74. One of President Nixon's responses to the energy crisis was to ask Congress to create a new agency that could focus on, and presumably speed up, the licensing of nuclear plants. After much debate, Congress divided the AEC into the **Energy Research and Development Administration** and the **Nuclear Regulatory Commission** in legislation it passed in 1974. **The Energy Reorganization Act,** coupled with the **1954 Atomic Energy Act**, constituted the statutory basis for the NRC. The new agency inherited a mixed legacy from its predecessor, marked both by 20 years of conscientious regulation and by unresolved safety questions, substantial antinuclear activism, and growing public doubts about nuclear power.

SECTION THREE: THE NUCLEAR REGULATORY COMMISSION

3.1. THE MANDATE OF THE NRC

The Nuclear Regulatory Commission began its operations as a separate agency in January 1975. In many ways, it carried on the legacy inherited from the AEC. It performed the same licensing and rulemaking functions that the regulatory staff had discharged for two decades. It also assumed some new administrative and regulatory duties.

The NRC, unlike the AEC's regulatory staff, was the final arbiter of regulatory issues; its judgment on safety questions was less susceptible to being overridden by developmental priorities. This did not mean that the NRC acted without regard to industry concerns or that its officials always agreed on policy matters, but it did mean that the agency's statutory mandate was clearly focused on ensuring the safety of nuclear power.

The NRC devoted a great deal of attention during its first few months to organizational tasks. At the same time it carried out a variety of regulatory responsibilities. It continued to review plant applications and to issue construction permits and operating licenses for new units. The NRC deliberated over a number of pressing problems shortly after its establishment.

One issue that received particular notice, both within and outside of the NRC, was the safeguarding of nuclear materials. The term "safeguards" applied to the prevention of theft, loss, or diversion of nuclear fuel or other materials or the sabotage of nuclear plants. This question took on greatly increased importance and visibility in the early 1970s because of growing apprehension about the activities and intentions of terrorist groups. There was a wave of terrorist bombings, assassinations, hijackings, and murders at that time, perhaps the most shocking of which was the murder of Israeli athletes at the 1972 Olympics.

The increase in such attacks around the world raised new concerns that terrorists would be able to build an atomic bomb, which was underscored by the well-publicized warnings of some nuclear experts that making a bomb was not terribly difficult for anyone who obtained the necessary materials. As a result, the AEC, and after its abolition, the NRC, substantially strengthened regulatory requirements for the transportation of nuclear materials and for nuclear plant security.

The NRC also devoted considerable attention to the export of nuclear materials to foreign countries. The United States was by far the leading supplier of nuclear fuel and other materials for the production of nuclear power abroad, and the NRC exercised important responsibilities for ensuring that nuclear exports did not encourage the proliferation of nuclear weapons or make them available to terrorists.

3.2. THE BROWNS FERRY FIRE

Despite the prominence of safeguards problems, the central issue for the NRC at the time of its creation remained reactor safety. There were two events in the early months of the NRC's existence that commanded the particular attention of the agency and the public.

1. The first was a major fire at TVA's Browns Ferry nuclear plants near Decatur, Alabama in March 1975. In the process of looking for air leaks in an area containing trays of electrical cables that operated the plants' control room and safety systems, a technician set off the fire. He used a lighted candle to conduct the search, and the open flame ignited the insulation around the cables. The fire raged for over seven

hours and nearly disabled the safety equipment of one of the two affected units. The accident was a blow to the public image of nuclear power and the recently-established NRC. It focused new attention on preventing fires from threatening plant safety and on the possibility of "common-mode failures," in which a single cause could initiate a chain of events that incapacitated even redundant safety features.

2. The second source of unusually extensive discussion and considerable controversy shortly after the NRC began operations was the publication of the final version of the "**Reactor Safety Study**" that the AEC had commissioned in 1972. The purpose of the study was to estimate the probability of a severe reactor accident, an issue that the AEC had never found a satisfactory means of addressing. To direct the study the AEC had recruited Norman C. Rasmussen, a professor of nuclear engineering at MIT. Rasmussen, assisted by AEC staff members, applied new methodologies and sophisticated "fault-tree" analyses to project the likelihood of a serious nuclear accident. The final Rasmussen report, released in October 1975, concluded that in comparison to other risks, including fires, explosions, toxic chemicals, dam failures, airplane crashes, earthquakes, tornadoes, and hurricanes, those from nuclear power were very small.

3.3. THE REACTOR SAFETY STUDY

The Rasmussen report, while hailed as a pioneering effort that enlightened a complex subject, also drew criticism from both inside and outside the NRC. Some authorities suggested that the study failed to account for the many paths that could lead to major accidents. Others complained that the data in the report did not support its executive summary's conclusions about the relative risks of nuclear power. After considering the arguments on both sides of the issue, the Commission in January 1979 issued a policy statement that withdrew its full endorsement of the study's executive summary.

3.4. THREE MILE ISLAND

Within a short time, discussion of severe nuclear accidents ceased to be strictly a matter of theoretical projections. On March 28, 1979, an accident at Unit 2 of the Three Mile Island nuclear station near Harrisburg, Pennsylvania made the issue starkly and alarmingly real. As a result of a series of mechanical failures and human errors, the accident (researchers later determined) uncovered the reactor's core and melted about half of it.

The immediate cause of the accident was a pressure relief valve that stuck open and allowed large volumes of reactor coolant to escape. The reactor operators misread the signs of a **loss-of-coolant accident** and, for several hours, failed to take action to cool the core. Although the plant's emergency cooling systems began to work according to design, the operating crew decided to reduce the flow from them to a trickle. By the time that the nature of the accident was recognized and the core was flooded with coolant, the reactor had suffered irreparable damage.

The credibility of the nuclear industry and the NRC fared almost as badly. Uncertainty about the causes of the problem, confusion about how to deal with it, conflicting

information from government and industry experts, and contradictory appraisals about the level of danger in the days following the accident often made the authorities appear inept, deceptive, or both.

Press accounts fed public fears and fostered a deepening perception of a technology that was out of control. Walter Cronkite told television viewers that as a result of the accident, "the danger faced by man for tampering with natural forces, a theme from the myths of Prometheus to the story of Frankenstein, moved closer to fact from fancy." Newspapers ran headlines warning, for example of a "RACE WITH NUCLEAR DISASTER" and "RISK OF MELTDOWN." Long after the technological dangers had subsided, the psychological effects of the TMI accident lingered on.

In some ways, the TMI accident produced reassuring, or at least encouraging, information for reactor experts about the design and operation of the safety systems in a large nuclear plant. Despite the substantial degree of core melting that occurred, containment was not breached. From all indications, the amount of radioactivity released into the environment as a result of the accident was very low. One estimate suggested that of 66 million curies of iodine-131 in the reactor at the time of the accident, only 14 or 15 curies escaped. Further, the emergency core cooling systems worked effectively once plant operators allowed them to run according to design.

Those findings were overshadowed by the unsettling disclosures of TMI. It focused attention on possible causes of accidents that the AEC/NRC and the nuclear industry had not considered extensively. Their working assumption had been that the most likely cause of a **loss-of-coolant accident** was a break in a large pipe that fed coolant to the core. But the destruction of the core at TMI had resulted not from a large pipe break but from a relatively minor mechanical failure that operator errors had drastically compounded.

Perhaps the most distressing revelation of TMI was that an accident so severe could occur at all. Neither the AEC/NRC or the industry had ever claimed that a major reactor accident was impossible, despite multiple and redundant safety features built into nuclear plants. But they had regarded it as highly unlikely, to the point of being nearly incredible.

The TMI accident demonstrated graphically that serious consequences could arise form unanticipated events. This enhanced the credibility of nuclear critics who had argued for years that no facility as complex as a nuclear plant could be made fool-proof. Public opinion polls taken after TMI showed a significant erosion in support for nuclear power. One survey found for the first time that the number of respondents who opposed building more nuclear units exceeded those who favored new plants. At the same time, the polls indicated that the public did not want to abandon nuclear power or close existing plants.

3.5. THE NRC'S RESPONSE TO THREE MILE ISLAND

The NRC responded to TMI by re-examining the adequacy of its safety requirements and imposing new regulations to correct deficiencies. It placed much greater emphasis

on "human factors" in plant performance in an effort to avoid a repeat of the operator errors that had exacerbated the accident. The agency developed new requirements for operator training, testing and licensing, and for shift scheduling and overtime.

In cooperation with industry groups, it promoted the increased use of reactor simulators and the careful assessment of control rooms and instrumentation. In addition, the agency expanded its resident inspector program to station at least two of its inspectors at each plant site.

The NRC devoted greater attention to other problems that had received limited consideration before TMI. They included the possible effects of small failures that could lead to major consequences, such as happened at Three Mile Island. The agency sponsored a series of studies on the ways in which "small breaks and transients" could threaten plant safety.

A second area on which the NRC focused was the evaluation of operational data from licensees. It established a new **Office for Analysis and Evaluation of Operational Data** to systematically review information from and the performance of operating plants. This action reflected the belated recognition that malfunctions similar to those at TMI had occurred at other plants, but the information had never been assimilated or disseminated.

The NRC undertook other initiatives as a result of TMI. It decided to survey radiation protection procedures at operating plants in order to assess their adequacy and to look for ways to improve existing regulations. It expanded research programs on problems that TMI had highlighted, including fuel damage, fission-product release, and hydrogen generation and control.

In light of the confusion and uncertainty over evacuation of the areas surrounding TMI during the accident, the NRC also sought to upgrade emergency preparedness and planning. Those and other steps it took in the wake of the accident were intended to reduce the likelihood of a major accident, and, in the event one occurred, to enhance the ability of the NRC, the utility, and the public to cope with it.

While the NRC was still deliberating over and revising its requirements in the aftermath of TMI, another event shook the industry and further undercut public support for nuclear power. This time, the NRC was a distant though interested observer rather than a direct participant.

3.6. CHERNOBYL

On April 26, 1986, unit 4 of the nuclear power station at Chernobyl in the Ukraine, a satellite of the then USSR underwent a violent explosion that destroyed the reactor and blew the top off it. The explosion and subsequent fire in the graphite core spewed massive amounts of radioactivity into the environment.

The accident occurred during a test in which operators had turned off the plant's safety systems and then lost control of the reactivity in the reactor. Without emergency cooling or a containment building to stop or at least slow the escape of radiation, the areas around the plant quickly became seriously contaminated and a radioactive plume spread far into other parts of the Soviet Union and Europe. Although the radiation did not pose a threat to the United States, one measure of its intensity in the Soviet Union was that levels of iodine-131 around the Chernobyl reactor were three times as high after the incident than they were after the TMI accident.

The design of the Chernobyl reactor was entirely different than that of U.S. plants, and the series of operator blunders that led to the accident defied belief. Supporters of nuclear power emphasized that a Chernobyl-type accident could not occur in commercial plants in the United States (or other nations) and that American reactors featured safety systems and containment to prevent the release of radioactivity. But nuclear critics pointed to Chernobyl as the prime example of the hazards of nuclear power. A representative of the Union of Concerned Scientists remarked: "The accident at Chernobyl makes it clear. Nuclear power is inherently dangerous."

A popular slogan that quickly appeared on the placards of European environmentalists was: CHERNOBYL IS EVERYWHERE. The Chernobyl tragedy was a major setback to the hopes of nuclear proponents to win public support for the technology and to spur orders for new reactors. U.S. utilities had not ordered any new plants since 1978 and the number of cancellations of planned units was growing. "We're in trouble," conceded a spokesman for the Atomic Industrial Forum. "If the calls I have received from people in the industry are a good indication, they are all very worried."

SECTION FOUR: LICENSING NEW PLANTS

4.1. EFFECTS ON LICENSING FROM CHERNOBYL

The Chernobyl accident added a new source of concern to long- standing controversies over the licensing of several reactors in the United States. In the aftermath of Three Mile Island, the NRC had suspended the granting of operating licenses for plants that were in the pipeline.

The "licensing pause" for fuel loading and low-power testing ended in February 1980. In August 1980 the NRC issued the first full-power operating license (to North Anna-2 in Virginia) since TMI. In the following nine years it granted full-power licenses to over forty other reactors, most of which had received construction permits in the mid-1970s. In 1985 it authorized the undamaged Three Mile Island Unit 1, which had been shut down for refueling at the time of the TMI-2 accident, to resume operation.

4.2. EMERGENCY PLANNING

Although many of the licensing actions aroused little opposition, others triggered major controversies. The two licensing cases that precipitated what were perhaps the most bitter, protracted, and widely publicized debates were Seabrook in New Hampshire and

Shoreham on Long Island, New York. The key, though hardly the sole, issue in both cases was emergency planning.

The Three Mile Island accident had vividly demonstrated the deficiencies in existing procedures for coping with an off-site nuclear emergency. The lack of effective preparation had produced confusion, uncertainty, and panic among members of the public faced with the prospect of exposure to radiation releases from the plant. After the accident, the NRC, prodded by Congress to improve emergency planning, adopted a rule that required each nuclear utility to come up with a plan for evacuating the population within a ten mile radius of its plant(s) in the event of a reactor accident. The rule applied to plants in operation and under construction. It called for plant owners to work with state and local police, fire, and civil defense authorities to put together an emergency plan that would be tested and evaluated by the NRC and the **Federal Emergency Management Agency** (FEMA). The NRC expected cooperation between federal, state and local government officials to upgrade emergency plans and provide better protection for the public if a serious nuclear accident occurred.

The NRC did not, however, anticipate that state and local governments would try to prevent the operation of nuclear plants by refusing to participate in emergency preparations. That was precisely what the states of New York and Massachusetts sought to do in the cases of Shoreham and Seabrook.

In New York, Governor Mario M. Cuomo and other state officials claimed that it would be impossible to evacuate Long Island if Shoreham suffered a major accident. Although plant proponents pointed out that emergency plans did not require the evacuation of all of Long Island if a serious accident occurred, the state refused to join in emergency planning procedures or drills. The NRC granted Shoreham a low-power operating license, but the state and utility, Long Island Lighting, eventually reached a settlement in which the company agreed not to operate the plant in return for concessions from the state.

A similar issue arose at Seabrook, though the outcome was different. The plant is located in the state of New Hampshire, but the ten mile emergency planning zone extended across the state line into Massachusetts. By the time that construction of the plant was completed, Massachusetts governor Michael S. Dukakis, largely as a result of Chernobyl, had decided that he would not cooperate with emergency planning efforts for Seabrook.

New Hampshire officials worked with federal agencies to prepare an emergency plan, but Massachusetts, arguing that crowded beaches near the Seabrook plant could not be evacuated in the event of an accident, refused. As a result of the positions of New York regarding Shoreham and Massachusetts regarding Seabrook, in 1988 the NRC adopted a "realism rule," which was grounded on the premise that in an actual emergency state and local governments would make every effort to protect public health and safety. Therefore, in cases in which state and/or local officials declined to participate in

emergency planning, the NRC and FEMA would review and evaluate plans developed by the utility.

On that basis, the NRC issued an operating license for the Seabrook plant. The arguments that raged over emergency planning and other issues at Shoreham and Seabrook attracted a great deal of attention, spawned heated controversy, and raised anew an old question of the relative authority of federal, state, and local governments in licensing and regulating nuclear plants.

4.3. ONE-STEP LICENSING

The lengthy and laborious licensing procedures that applicants had to undergo in the cases of Shoreham (which had received a construction permit in 1973), Seabrook (which had received a construction permit in 1977), and other reactors stirred new interest in simplifying and streamlining the regulatory process. It seemed apparent that the complexity of the licensing process was a major deterrent to utilities who might consider building nuclear plants.

By the late 1980s, the nuclear option looked more appealing to some observers, including some environmentalists, because of growing concern about the consequences of burning fossil fuel, especially acid rain and global warming. Furthermore, nuclear vendors were advancing new designs for plants that greatly reduced the chances of TMI-type and other severe accidents.

One way that the NRC proposed to facilitate licensing procedures was to replace the traditional two-step process with a one-step system. This would ease the burden on applicants, but it raised a vitally important question: what level of detail would the NRC require in applications for advanced plants in order to satisfy its concerns about their safety? The agency had never required the detailed technical information in construction permit proposals that it expected in operating license applications, but in a one-step licensing process it was unclear how much data would be needed to evaluate and certify safety designs.

After long discussions that reflected differing views among commissioners, staff, and nuclear vendors, the NRC reached a decision on what constituted an "essentially complete design." It established a "graded approach" in which the level of detail that an applicant would be required to submit varied according to the system's, structure's, or component's relationship to plant safety. The objective of the NRC's action was to ensure safety while providing flexibility for the development of new designs.

4.4. RADIATION STANDARDS

While the NRC was deliberating over a number of new regulatory procedures and problems, it was also reviewing some old issues. The most prominent of those questions was radiation standards. The NRC had begun work on revising its radiation protection regulations in the aftermath of Three Mile Island. Although the AEC had issued "design objectives" that in effect reduced the permissible levels of radioactive effluents from nuclear plants in the 1970s, the basic regulations for occupational and population

exposure had remained unchanged since 1961 (an average of 5 rem per year for radiation workers and 0.5 rem annually for individuals in the general population).

Based upon new recommendations of the **National Committee on Radiation Protection** (NCRP) and the **International Committee on Radiation Protection** (ICRP) and upon new research findings, the NRC tightened its regulations in several regards, the most prominent of which was to restrict population exposure to 100 (rather than 500) millirem per year.

Despite new scientific information and epidemiological studies, the health effects of low-level radiation remained a source of uncertainty and controversy. Some studies provided results that were very reassuring about the hazards of radiation emissions from nuclear plants. A major survey conducted by the **National Cancer Institute**, for example, found no increased risk of cancer in 107 counties of the United States located near 62 nuclear power plants. But other evidence was more disquieting, such as a cluster of cancer cases near the Pilgrim reactor in Massachusetts and a high incidence of leukemia in children around the Sellafield reprocessing plant in Britain.

4.5. BELOW REGULATORY CONCERN

None of the studies on the effects of low-level radiation was, or claimed, to be, definitive. The subject continued to be a source of interest to and debate among scientists. It also continued to be a source of considerable anxiety to the public. The most graphic evidence of public apprehension about radiation was the reaction to the NRC's announcement of a new policy on radiation levels that were "**Below Regulatory Concern**" (BRC).

n June 1990, the NRC published a policy statement outlining its plans to establish rules and procedures by which small quantities of low-level radioactive materials could be largely exempted from regulatory controls. The agency proposed that if radioactive materials did not expose individuals to more than 1 millirem per year or a population group to more than 1000 person-rem per year, they could be eligible for the exemption from full-scale regulatory control. This would not be granted automatically; the NRC would consider requests for exemptions for sites that met the dose criteria through its rulemaking or licensing processes. It intended that the BRC policy would apply to consumer products, landfills, and other sources of very low levels of radiation. The NRC explained that the BRC policy would enable it to devote more time and resources to major regulatory issues and thereby better protect public health and safety.

The NRC's announcement of its intentions on BRC was greeted with a firestorm of protest from the public, Congress, the news media, and antinuclear activists. Some critics suggested that the agency was defaulting on its responsibility for public health and that BRC would allow the nuclear industry to discard dangerously radioactive wastes in public trash dumps. It was, alleged one antinuclear group, "a trade-off of people's lives in favor of the financial interests of the nuclear industry." In public meetings that the NRC held to explain BRC, aroused citizens called repeatedly for the resignation of the commissioners or their indictment under criminal charges. Eventually, the Commission

decided to defer any action on the BRC issue. The outcry over BRC underscored the difficulty of even attempting to sponsor a calm and reasoned discussion on the subject of radiation hazards.

The uproar over BRC was one of several indications of how the regulatory environment had changed since the passage of the **1954 Atomic Energy Act** made possible the development of nuclear power for electrical generation. A public that had welcomed the growth of nuclear power in the 1950s had become skeptical of the technology and suspicious of those responsible for its safety.

Nuclear plants had become larger, more complicated, and more costly to build. The longest running nuclear plant until its closure in 1992, Yankee Rowe in Massachusetts, had a capacity of 175 electrical megawatts and was constructed for about $39 million. By comparison, for example, Seabrook had a capacity of 1150 electrical megawatts and cost over $6 billion to build.

The length and complexity of the licensing process had grown commensurately. The owners of Yankee Rowe applied for a construction permit in 1956 and received an operating license in 1960 without a murmur of protest. Seabrook's owners applied for a construction permit in 1973 and received an operating license in 1990 after long legal proceedings and many angry demonstrations. The contrasts between Yankee Rowe and Seabrook were results of a series of inter-related technological, administrative, and political developments that shaped the history of nuclear regulation.

SECTION FIVE: NEW ISSUES, NEW APPROACHES

The focus of the NRC's activities gradually shifted away from licensing requirements for new plants to overseeing the safety of operating plants. Since it received no applications for construction permits after 1978 and had completed work on most operating license applications a decade later, it devoted much less attention and fewer resources to its licensing responsibilities. During the first half of the 1980s, the NRC's deliberations and policy decisions were in large measure a response to Three Mile Island. By the latter part of the decade, however, the agency was addressing a wide range of new questions relating to the safety of the about 100 plants in operation. Not surprisingly, the issues it considered often raised difficult and divisive questions for which there were no ready answers.

5.1. PLANT MAINTENANCE

One of the first and most important issues that the NRC tackled as it turned its attention to the regulation of operating nuclear plants was maintenance. It estimated in 1985 that more than 35 percent of the "abnormal occurrences" that it had reported to Congress over the previous ten years were directly attributable to maintenance deficiencies.

Many of the problems arose from human errors, such as failing to follow procedures, installing equipment incorrectly, or using the wrong parts to make repairs. The need for improvements in maintenance was underscored when an incident at the Davis-Besse plant in Ohio resulted in the loss of all feedwater in 1985. Failures in feedwater pumps,

including auxiliary pumps that had not been tested or maintained, caused what could have produced a major accident.

The nuclear industry was well aware of shortcomings in maintenance programs and took steps to make improvements. The NRC applauded the efforts of the industry but insisted that licenses still "had a long way to go in the maintenance area." Therefore, in June 1988 the Commission directed the NRC staff to draft a maintenance rule as a matter of "HIGHEST priority."

In June 1991, despite industry objections that a rule was not necessary, the Commission voted to issue a regulation that required adequate maintenance programs of all commercial nuclear plants. It acknowledged the substantial improvements that many licensees had made, but it concluded that an industry-wide regulation was still necessary. The NRC worked with the industry to establish procedures for monitoring the effectiveness of maintenance programs.

5.2. DECOMMISIONING

Another key issue that the NRC considered was the decommissioning of plants, the final step of the life cycle for operating facilities. Between 1947 and 1975, a total of 50 nuclear plants, including five small experimental power reactors, were decommissioned. In the late 1970s, this experience gave the NRC confidence that decommissioning of nuclear plants would not present major problems when their licenses expired. In response to an investigation by the U.S. **General Accounting Office**, congressional hearings, and a petition from environmental organizations, however, the NRC took a closer look at the subject.

In 1984, the staff reported to the Commission that existing regulations covered decommissioning in a "limited, vague, or inappropriate way and are not fully adequate." As a result, the NRC drafted a rule that required licensees to specify how they planned to ensure that sufficient funding was available to clean up the sites on which their plants were located and to make certain that radiation levels at decommissioned sites were low enough to allow the land to be used for other purposes. After soliciting public comments and making modest revisions in the draft, the NRC published a final rule in 1988.

The decommissioning rule was much more comprehensive than earlier NRC regulations but it did not resolve all of the issues that arose on the subject. Within a short time after the rule became final, the agency faced an unprecedented and unanticipated question--what to do about funding for "prematurely shut down reactors." Three plants, including Shoreham, closed well before their operating licenses expired, which raised questions about how to pay for costs of decommissioning reactors that had not operated long enough to accumulate adequate funding.

This issue was underscored by the costs of decommissioning the Yankee Rowe plant, which ran much higher than projected. While the NRC wrestled with this question, it also deliberated over the level of radiation that should be permitted at the sites of

decommissioned plants. This issue generated opposing views and sometimes sharp differences between the NRC and the **Environmental Protection Agency.**

5.3. LICENSE RENEWAL

As decommissioning issues were debated, the NRC devoted considerable attention and resources to the question of license renewal. While some utilities were closing reactors long before their 40-year operating licenses expired, others were weighing the possibility of extending the lives of plants beyond 40 years.

The 40-year licensing period for nuclear plants was a rather arbitrary compromise written into the **1954 Atomic Energy Act** that was not based on technical grounds or operating experience. In the late 1970s, industry groups closely examined the issue of plant life extension for the first time. The Electric Power Research Institute, for example, concluded that reconditioning of old plants offered potentially major benefits, but it cautioned that the benefits depended on financial considerations as well as on technical assessments, environmental issues, and projections of power availability. Those uncertainties were compounded by industry's concern that the NRC was not prepared to address the issues surrounding license renewal promptly and knowledgeably.

n 1985, the NRC, prodded by Chairman Nunzio J. Palladino, undertook a careful analysis of license renewal. The agency had sponsored research on the critical question of the safety effects of plant aging for years, but many technical questions remained to be answered. License renewal also raised complex legal and policy issues. The NRC staff cited the "central regulatory question" that plant life extension presented: "What is an adequate licensing basis for renewing the operating license of a nuclear power plant?"

The NRC deliberated over this issue and its corollaries for several years. Eventually, it decided that the maximum length of an extended license would be 20 years. It also concluded that using the existing regulatory requirements governing a plant would offer reasonable assurance of adequate protection if its license were renewed, provided that the "current licensing basis" was modified to account for age-related safety issues.

In 1991, the Commission approved a regulation on the technical requirements for license renewal. After considering ways to evaluate the environmental consequences of license renewal, the NRC elected to develop a generic environmental impact statement that covered effects that were common to all or most nuclear plants. In April 1998, Baltimore Gas and Electric became the first utility to apply for license renewal for its Calvert Cliffs plants on the Chesapeake Bay. Duke Energy Corporation followed suit in July 1998 when it sought license extensions for its Oconee nuclear units in South Carolina.

5.4. RISK ASSESSMENT AND NUCLEAR SAFETY

As the NRC considered its policies on license renewal, representatives of the nuclear industry expressed concern that the costs and uncertainties of the regulatory process would negate the potential advantages of plant life extension. This was consistent with

strong industry criticism of the NRC's regulations or the ways in which they were implemented.

A report prepared for an industry group, for example, concluded in 1994 that the NRC's policies and practices represented a "serious threat to America's nuclear energy resource" by distracting plant management, undermining public trust in nuclear power, and "pricing nuclear power out of the competitive energy marketplace." Industry protests about regulatory burdens were nothing new, of course, but they had taken on increased urgency and intensity by the early part of the 1990s.

Industry officials complained that NRC regulations were in many cases intrusive, excessive, and potentially counterproductive. They particularly objected to the agency's numerical ratings of plant performance, which they found to be arbitrary and inconsistent. In September 1998, the Commission indefinitely suspended the "Systematic Assessment of Licensee Performance" program, which the agency had created in the wake of the Three Mile Island accident to evaluate and score management practices in several different categories of plant operation. In June 1999, it began a pilot program to test methods of providing more consistent and predictable plant evaluations.

As a part of its reexamination of the regulatory process, the NRC evaluated the role of risk assessment and performance indicators. The benefits of risk assessment had been debated since the Rasmussen report without making a major impact on the formulation or enforcement of the NRC's rules.

Nuclear industry representatives complained that the NRC relied too heavily on "prescriptive" regulations. They urged the agency to place greater emphasis on non-prescriptive performance-based assessments that would recognize the significant improvements that industry had achieved since Three Mile Island. This would allow licensees greater leeway to determine how to accomplish regulatory goals and presumably cut costs without sacrificing safety.

In 1991, the Commission instructed the agency staff to investigate the feasibility of using more performance-based regulations that focused on a "result to be obtained, rather than prescribing to the licensee how the objective is to be obtained." This initiative received strong support from Ivan Selin, chairman of the NRC from 1992 to 1995, from his successor, Shirley Ann Jackson, chairman from 1995 to 1999, and from their colleagues on the Commission.

The effective employment of performance-based regulation was closely tied to informed analyses of risk. In 1995, the Commission unanimously approved a policy statement that encouraged the application of probabilistic risk assessment "as an extension and enhancement of traditional regulation."

The agency believed that risk analysis would enable it to "focus on those regulated activities that pose the greatest risk to the public" and to ease "unnecessary burdens on licensees." The industry and the NRC agreed on this general objective, but many

uncertainties about how to apply the concept of risk assessment in practice had not been resolved. The industry was concerned that the NRC gave unwarranted emphasis to the redundant "defense-in-depth" approach that had been applied since the earliest days of the nuclear power industry.

Those concerns were magnified in 1997 when the Commission voted to require a containment spray system in a new Westinghouse plant design even though risk assessments indicated that the design was "safe enough" without the spray system. Despite this affirmation of the importance of defense-in-depth, the NRC continued to search for ways to use probabilistic risk assessment to improve the regulatory process.

5.5. THE MILLSTONE CONTROVERSY

Although risk-informed regulation offered many potential benefits for evaluating the technical performance of nuclear plants, it was not a reliable way to detect safety issues that could generate acute public concern. In that regard, it was not necessarily a useful means of building public confidence in nuclear power technology or the NRC.

This was amply demonstrated when a series of problems arose at the Millstone nuclear station, which included three plants located on the northern side of Long Island Sound in Connecticut. The safety issues at Millstone required attention, but they were not so serious that risk analysis was likely to identify them as priority matters. As Commissioner Nils J. Diaz commented in 1997, of the many issues raised about Millstone, "only a handful appear to have been safety-significant." Nevertheless, the failures at Millstone created a great deal of controversy and a barrage of criticism of the NRC.

The uproar over Millstone began in the early 1990s when several plant employees claimed that they were harassed, intimidated, and/or dismissed from their jobs by the owner of the plants, Northeast Utilities, for calling attention to safety problems and violations of NRC regulations. The NRC investigated the concerns raised by the "whistle-blowers" and determined that the safety issues they raised were not of major significance and had been corrected. But the agency also concluded that the utility had harassed employees and assessed it a fine of $100,000, the maximum amount allowed by law. This did not satisfy the dissidents at Millstone and elsewhere, who insisted that the NRC was neither prompt nor firm in dealing with the issues they cited or in protecting them from retaliation by their employers. As a result of the complaints from Millstone and other plants, the agency reexamined and eventually tightened its policies in order to provide better protection to whistle-blowers who contacted it about safety issues.

Meanwhile, new revelations at Millstone generated increasing NRC scrutiny. It also commanded growing media attention, much of which was sharply critical of the NRC. In 1993 and again in 1994 the NRC fined Northeast Utilities for procedural violations that the agency viewed as serious lapses in the management of the Millstone units. The utility pledged to improve its performance and "to resolve issues raised by its employees."

Nevertheless, another issue raised by company employees soon triggered new reservations about safety at Millstone and the effectiveness of the NRC's enforcement policies. In this case, the whistle-blowers objected to the company's practice of placing the entire nuclear core into the spent fuel pool at Millstone Unit 1 during refueling operations. The plant's "final safety analysis report," which provided the basis for its operating license, specified that only one-third of the spent fuel rods would be moved into the pool. But Millstone-1 had performed "full-core off-loads" for years as an "emergency" procedure with the knowledge of the NRC. Finally, after employees questioned the practice, Northeast Utilities applied for a license amendment that expressly permitted full-core off-loading, and in November 1995 the NRC granted its approval.

By that time, the utility and the NRC were the subjects of extensive and unflattering coverage in the local media. In March 1996, the criticism reached a new level of visibility when Time magazine ran a cover story on the whistle-blowers who had "caught the Nuclear Regulatory Commission at a dangerous game." It suggested that an accident in a spent fuel pool posed the hazard of "releasing massive amounts of radiation and rendering hundreds of square miles uninhabitable." It charged that the NRC "may be more concerned with propping up an embattled, economically straitened industry than with ensuring public safety." NRC chairman Jackson conceded that the Time article demonstrated that "not all aspects of nuclear regulation or nuclear operations in certain places are as they should be," but she strongly denied the implication that "the Millstone situation borders on an impending TMI- or Chernobyl-type disaster."

Amid the growing criticism, the NRC conducted its own reviews to identify and correct errors that the Millstone experience brought to light. An internal task force reported in September 1996 that the "safety significance of Millstone's refueling practices was low." Nevertheless, it recommended a series of procedural, informational, and management improvements.

The agency also undertook a careful study of a frequently-used provision in its regulations that allowed licensees to make changes in their plants without NRC permission under certain conditions. In 1999, after considerable debate over the threshold for permitting such changes, the Commission approved revisions designed to clarify the rule and provide guidance on when NRC consent was necessary within a risk-informed framework.

While the NRC examined its own regulations and procedures, it conducted an expanding probe of the Millstone plants. In May 1996 the NRC's inspector general faulted the agency for failing to recognize the problems at Millstone and impose corrective actions much earlier. When the NRC's investigations, along with those conducted by the utility, turned up hundreds of performance and procedural deficiencies, the agency took the unusual step of stipulating that the three plants, all of which had been shut down, would not be allowed to restart without a formal vote of the Commission. Eventually, after the utility made management changes, took a series of steps to address its shortcomings,

and decided to permanently close Millstone-1, the Commission authorized the restart of units 2 (in 1999) and 3 (in 1998). The series of problems at Millstone underscored the general difficulties that the NRC had encountered with plants that did not perform up to standards and did not correct their deficiencies promptly or effectively. The Commission devoted a great deal of energy to dealing with the many aspects of encouraging or forcing improvements in plants that did not fully meet its requirements.

5.6. NUCLEAR MATERIALS SAFETY AND SAFEGUARDS

Although reactor safety issues captured a lion's share of public notice, the NRC also devoted substantial resources to a variety of complex matters in the area of nuclear materials safety and safeguards. The protection of nuclear materials from theft or diversion remained a major agency concern, though it did not command the level of public attention it had received in the 1970s.

In cooperation and sometimes in conflict with other government agencies, the NRC evaluated the safety problems involved in building and operating repositories for high-level and low-level radioactive waste. Despite federal legislation that attempted to provide the means for establishing permanent waste sites and the efforts of federal and state officials, scientists, engineers, and other professionals, the disposal of radioactive wastes remained a source of intense public concern and bitter political controversy.

The NRC also considered its role in regulating certain medical uses of radioactive materials. Although it exercised only limited responsibilities in the field of "radiation medicine," it sought to ensure that patients received the proper doses of radiation from procedures under its regulatory authority. Its rules elicited protests from medical practitioners and organizations who complained about regulatory overkill that intruded into physician-patient relationships.

The issues surrounding the regulation of nuclear materials, the problems at Millstone, and the use of risk assessment underscored patterns in the history of nuclear regulation over a period of four decades. The nuclear industry and materials licensees often asserted that regulatory requirements were too burdensome, too inflexible, and too strict. Nuclear critics, on the other hand, frequently lamented that regulatory requirements were too lax, too sympathetic to industry concerns, and too inattentive to public safety. The NRC, and the AEC before it, attempted to find a proper balance between essential and excessive regulation, but this was a difficult and uncertain task that usually elicited complaints from one side or all sides of regulatory issues. The NRC sought to separate valid criticisms from those that were exaggerated or ill-informed, but this process won few plaudits from its different (and frequently competing) constituencies. "The bane of the regulator," a senior agency official remarked in 1998, "is to feel unloved." The ongoing effort to promote the safe use of nuclear materials and the safe operation of nuclear power plants without imposing undue burdens on licensees ensured that nuclear regulation would remain a complex and controversial public policy issue.

ASCENTRUST LLC
The POWER of ENGINEERING

APPENDIX
INTRODUCTION TO
NUCLEAR POWER AND NTPBMR TECHNOLOGY

SECTION D: Decommissioning

This white paper deals with the process involved in the decommissioning process of a **Nuclear Power Plant** after the term of its useful life.

PART ONE: INTRODUCTION

1.1 LICENSING

The **Nuclear Regulatory Commission** (**NRC**) sets requirements for the safe operation of commercial nuclear power reactors, licenses the construction and operation of the reactors, and inspects them to assure they are operating safely within the agency's regulations. According to the **NRC**, there are 104 operating nuclear power reactors at 65 sites. These plants use nuclear energy to generate electricity and generate approximately 22% of electricity in the United States. From the 104 operating plants, there are approximately 48 licensees, 4 reactor vendors, and 80 different nuclear power plant designs.

Commercial nuclear power plants are licensed by the **NRC** for a 40-year operating period with possible renewal of the license for an extended period of operation of up to 20 additional years. The last new license granted by the **NRC** was issued in 1978 and there are currently no new licensing requests. Further, the **NRC** is not expecting any new applications in the near future. However, in 1998 the **NRC** began to receive applications for extensions to the operation licenses after the 40 years of operational licensing. The first applications were for two plants, Calvert Cliffs and Oconee. These two plants have applied for and received a 20-year license renewal. Further, the **NRC** and other regulatory authorities are encouraging the extension of the production lifetime of existing nuclear power plants in order to prolong the useful life of these production facilities. Unless license extensions are granted, all current licenses will expire by 2035. It should also be noted that the NRC may issue an order to a licensee to suspend or permanently cease operations if the licensee fails to operate the facility in accordance with the terms of the license.

1.2 NUCLEAR VALUE CHAIN

Due to low and steady variable costs, nuclear power plants provide long-term stability of total costs. This allows nuclear power plants to offer forward sales that capture a market premium which can be much more valuable than the margin from low current production costs. In particular, when the electricity industry is considered as a business, the economic value of nuclear power plants can be defined in stages.

The **Nuclear Energy Institute** ("**NEI**") refers to this analysis of economic value as the "nuclear value chain." The nuclear value chain includes:

1.2.1 LOW PRODUCTION COST

The going forward cost of electricity from a nuclear power plant is clearly competitive when compared to the market clearing price of electricity in the day-ahead market. However, nuclear units have significantly more value than simply the price they receive for electricity in the wholesale market.

1.2.2 IMPROVED PERFORMANCE

The industry can continue to achieve improved performance through increased rates, shorter refueling outages, higher fuel burn-ups, and better management of operational costs.

1.2.3 FUTURE PRICE STABILITY

Nuclear facilities can leverage its high degree of future price stability by selling at a premium to large users an assured source of electricity supply at a known price. For example, presently some users in California are willing to pay this premium to protect themselves against the damaging effects of price volatility in the day-ahead market.

1.2.4 SITE VALUE

Nuclear power plants have significant additional site value, such as switchyards, access to the power grid, ingress and egress, and spare cooling capacity. In many cases, nuclear power sites were planned for more units than were built, providing room to build additional non-nuclear generation. Such diverse generation would enable a single site to execute forward sales in the bilateral contract market and participate in the day-ahead market, in particular selling highly profitable 10-minute spinning reserve capacity.

1.2.5 CLEAN AIR COMPLIANCE VALUE

The substantial emissions avoided by the use of nuclear energy reduce the compliance obligation and associated costs for affected fossil-fueled power plants, including capital outlays to bring fossil-fueled plants into compliance.

Accordingly, based on the many advantages of nuclear power plants shown in the above nuclear value chain, the number of new nuclear power plants built, as well as the sale of existing plants, will increase.

PART TWO: DECOMMISSIONING

2.1 DECOMMISSIONING

When nuclear facilities are shut down permanently, they enter a decommission process which will lead to the release of the site for unrestricted uses. Specifically, decommissioning a nuclear power plant can be defined as the cessation of operations and the withdrawal of the facility from service, followed by its transformation into an out-of-service state and eventually, its complete

removal. Decommissioning activities are intended to place the nuclear facility in a condition that provides for the health and safety of the general public and the environment.

Decommission begins when operations at a nuclear power plant are terminated. In most cases, the nuclear fuel, the mobile radioactive materials in the process systems, and the radioactive waste produced during normal operations are removed as soon as the plant ceases to operate. Certain equipment can also be removed and discarded.

If the entire facility were to be dismantled immediately, however, the decommissioning workers would be exposed to higher levels of radiation than if the dismantlement were to be accomplished in several steps. Therefore, decommissioning activities have been divided into three stages. Each of these stages can be defined by two characteristics: the physical state of the plant and its equipment, and the surveillance needed to maintain that physical state.

2.1.1 STAGE ONE DECOMMISSIOONING

Stage one decommissioning entails removing the spent fuel from the reactor, draining the liquid systems, disconnecting the operating systems, blocking and sealing the mechanical openings such as valves and plugs, and controlling the atmosphere inside the containment building. The facility is kept under surveillance, access is limited and routine inspections are carried out to assure that the plant remains in a safe condition.

2.1.2 STAGE TWO DECOMMISSIONING

Stage two decommissioning requires all equipment and buildings which can be easily dismantled to be removed or decontaminated and made available for other uses, leaving only the reactor core structure and its extensive shielding. The containment building and the ventilation system may be modified or removed if they are no longer needed for safety reasons, or they may be decontaminated to allow access for other purposes. Other buildings and equipment which are not radioactive may be converted for new purposes as well. Surveillance during Stage 2 is reduced, but it is desirable to continue periodic spot checks of the buildings as well as surveillance of the surrounding environment.

2.1.3 STAGE THREE DECOMMISSIONING

requires that, unless the site, buildings or equipment are to be re-used for other nuclear purposes, all materials with radioactivity levels exceeding those closely equivalent to the natural radiation environment will be removed and the site released without restrictions or further surveillance.

These three stages may be carried out by rapidly progressing from one stage to the next or carried out over a prolonged period lasting as long as 100 years or more. Although most facilities intend to complete all three stages, a facility could remain at Stage 1 or Stage 2 for a relatively long period of time, or decommissioning could proceed directly from Stage 1 to Stage 3.

According to the **NRC**, however, decommissioning must be completed within 60 years of permanent cessation of operations. In contrast, conservation groups such as the Sierra Club lobby for a 30 to 50 year completion time-frame. However, some decommissioning tasks cannot begin immediately after plant cessation. For example, current dry storage cask designs are licensed for spent fuel with a core discharge decay time averaging approximately five years or longer. Therefore, decommissioning operations for the plant's "fuel building" cannot be expected to begin prior to five years after the cessation of plant operations.

One open question regarding **NRC** licensing relates to possible deregulation of the nuclear power industry. Deregulation may cause some **NRC** licensees to cease being an "electric utility", as defined in NRC regulations. If this occurs, the **NRC** will require the licensees to meet more stringent decommissioning funding assurance requirements that apply to non-electric utilities. Further, **NRC** is considering revising its financial and decommissioning funding assurance requirements.

2.2 ACCEPTABLE DECOMMISSIONING ALTERNATIVES

Decommissioning involves three different alternatives:

A. **DECON,**
B. **SAFSTOR,**
C. **ENTOMB**.

2.2.1 DECON

Under **DECON** (immediate dismantlement), shortly after the nuclear facility closes, equipment, structures, and portions of the facility containing radioactive contaminants are removed or decontaminated to a level that permits release of the property and termination of the **NRC** license. Note that the required work force during **DECON** is one-third to one-tenth the required number of people employed during normal operations. As is evident, the work force and associated costs are high.

2.2.2 SAFSTOR

Under **SAFSTOR,** often called "delayed **DECON**," a nuclear facility is maintained and monitored in a condition that allows the radioactivity to decay; afterwards, the nuclear facility is dismantled. For example, if a new plant is built next to an

existing plant, then this will enable the existing plant to go into **SAFSTOR** upon license expiration. The personnel that operate the new plant will be able to look over the **SAFSTOR** plant without incurring significant costs. Therefore, decommissioning the plant after **SAFSTOR** will lower the cost of decommissioning. It follows that if new nuclear power plants are ever built, it would be likely that they would be built next to existing facilities. This may allow the older facilities to be placed into SAFSTOR at little cost.

2.2.3 ENTOMB

Under **ENTOMB,** radioactive contaminants are encased in a structurally sound material such as concrete and appropriately maintained and monitored until the radioactivity decays to a level permitting release of the property. **ENTOMB** is not presently allowed by **NRC** regulations but is under consideration as a possible option.

A licensee may also choose to adopt a combination of the first two alternatives in which some portions of the facility are dismantled or decontaminated while other parts of the facility are left in **SAFSTOR**. The decision may be based on factors besides radioactive decay such as availability of waste disposal sites. However, most facilities will use either immediate **DECON** or a **DECON** after some period of **SAFSTOR**.

As stated, under NRC regulations, decommissioning must be completed within 60 years. A time beyond that will be considered only when necessary to protect public health and safety in accordance with NRC regulations.

2.3 ACTUAL DECOMMISSIONING EXPERIENCE

As of January 2015, there have only been five plants that have completed the **DECON** process, three nuclear power plants, and two **Department of Energy** ("**DOE"**) plants. Further, six nuclear power plants are now in various stages of dismantlement and decontamination and eleven nuclear power reactors are currently in long term storage (**SAFSTOR**).

2.4 DECOMMISSIONING COST ESTIMATES

The total cost of decommissioning is dependent on the sequence and timing of the various stages one through three, described above. Deferment of a stage tends to reduce its cost, due to decreasing radioactivity, but this may be offset by increased storage and surveillance costs.

Decommissioning contributes less than 5% to total electricity generation costs. In the United States, many utilities have revised their cost projections downwards in the light of experience, and estimates now average $325 to $500 million per reactor and up.

2.5 FINANCING OF DECOMMISSIONING COSTS

Financing methods vary; however, the most common methods are:

2.5.1 PREPAYMENT

Money is deposited in a separate account to cover decommissioning costs even before the plant begins operation. This may be done in a number of ways but the funds cannot be withdrawn other than for decommissioning purposes.

2.5.2 EXTERNAL SINKING FUND

(Nuclear Power Levy): A fund is built up over the years from a percentage of the electricity rates charged to consumers. Proceeds are placed in a trust fund outside the utility's control. This method is the main method in the United States, where sufficient funds are set aside during the reactor's operating lifetime to cover the cost of decommissioning.

2.5.3 SURETY FUND, LETTER OF CREDIT, INSURANCE

Purchased by the utility to guarantee that decommissioning costs will be covered even if the utility defaults

In the United States, utilities generally collect 0.1 to 0.2 cents per kW-hour to fund decommissioning. They must then report regularly to the **NRC** on the status of their decommissioning funds. The fund presently has $22.5 billion of the total estimated cost of decommissioning all U.S. nuclear power plants, leaving a liability of about $9.5 billion to be covered over the operating lives of 104 active reactors.

Further, in accordance with **NRC** regulations, decommissioning cost estimates are required at five different periods, which are:

A. At the time of **NRC** licensing,

B. Five years before anticipated shutdown,

C. With a **Post-Shutdown Decommissioning Activities Report** (**PSDAR**) submittal,

D. Two years following shutdown (this is the first time that the cost estimate has to be site specific, prior to this the facility could use estimates from similar sites as their basis),

E. Two years preceding the anticipated termination of the license.

Note that decommissioning costs do not include the cost of removal and disposal of spent fuel or of non-radioactive structures and materials beyond that necessary to terminate the license.

2.6 NUCLEAR DECOMMISSIONING TRUSTS

As should be appreciated, nuclear facilities have extraordinary costs at the end of their lives. By **NRC** regulation, these costs must be collected and managed during the life of the facility, creating several valuation issues. As the term of its license ends, a nuclear facility will be decommissioned and radioactive portions safely removed or contained. As stated, typical decommissioning costs for nuclear facilities approach $500 million dollars per reactor, based on **NRC** minimum facility funding for a large nuclear unit. Although funding depends on unit size, and other factors, these current dollar estimates for decommissioning costs and the future cost could be triple this estimate, or more, by the end of a typical full life of these facilities.

By regulation, the dollars collected for decommissioning are periodically deposited into an externally managed investment fund or trust (external sinking fund), discussed above, and kept separate from an owner's other assets. The objective is to accrue an amount that is sufficient to pay for decommissioning costs as of the termination date of the facility.

Two types of trust funds can be used to accrue amounts for decommissioning:

2.6.1 A QUALIFIED TRUST FUND

A qualified trust fund is provided special timing considerations and tax benefits. Internal Revenue Code Section 468A allows for the establishment of qualified trust funds. Under the qualified trust funds, contributions to these funds are immediately deductible in computing taxable income. Although any revenue that may be received specifically for decommissioning is included in taxable income, the contributions to a qualified trust fund are immediately deductible as an offset. The net effect is that no taxable income will be recognized until expenditures are actually incurred for decommission, at which time actual decommissioning costs are treated as deductible expenses. This tax method has the advantage of recognizing revenues during the same future tax-period that the expense will be incurred.

2.6.2 A NON-QUALIFIED TRUST FUND

The non-qualified trust fund receives no special tax treatment. In contrast, contributions to non-qualified trust funds are treated as income during the tax period earned and therefore are not immediately deductible. Thus, while amounts collected from customers are included in taxable income, the contribution to a non-qualified trust does not offer a current tax deduction. Consequently, non-qualified funds collected from customers need to include a "gross up" for taxes, to allow sufficient after-tax amounts to fund the trusts.

The income earned by the funds is also subject to different tax rates. Qualified funds are subjected to a 20% federal tax rate. The non-qualified funds are taxed at the federal tax rate, which currently is typically 35%. Although it is advantageous to maximize contributions to a qualified find, the amount that is allowed for deposit into a qualified trust fund is restricted by rules governed by the state regulatory commission and the Internal Revenue Service.

Any amounts withdrawn from a qualified fund are taxable during the tax period of the withdrawal. For the non-qualified find, however, there is no taxable income recognition on withdrawal from the funds because no tax deduction had been allowed on the original contribution. A tax deduction for the actual decommissioning costs expended is taken for both types of funds during the tax period of the expenditure. Therefore, to the extent that money withdrawn from a qualified fund is used to meet decommissioning expenses, there will be an equal offset between revenue and expenses for the tax period.

Equity allocations in qualified trusts and non-qualified trusts continue to grow with target allocations of 55% for both trusts. Currently, the qualified trusts equity allocation is 48% of assets, while the non-qualified equity allocation is 56% of assets. NISA expects 1999 total contributions to qualified decommissioning trusts to be $1,074 million, and $356 million for non-qualified trusts. In addition, 58% of the investor owned nuclear decommissioning trusts are subject to state income taxes. The median state tax rate is 7.8%, where the maximum is 12.8% and the minimum is 2.0%.

A study conducted by **NISA** indicated the following factors that contribute to the uncertainty of funding of the decommissioning liability, according to owners. These factors are (ranked by degree of uncertainty): Waste Disposal Cost Inflation Regulatory Environment Asset Returns Early Decommissioning Deregulation Labor Cost Inflation Energy Cost Inflation Method of Decommissioning Nuclear Decommissioning Inflation. NRC licensees are required to annually adjust the amount of decommissioning funding assurance based on inflation estimations.

As part of the CPI, Labor and Energy costs are naturally correlated therewith. Although, low level waste may not be correlated with the CPI, based upon the above formula, the annual low-level waste inflation would need to be 27% in order to have the total decommissioning inflation be 6.0% above CPI.

The costs for low-level waste disposal are determined by market conditions of the demand for the disposal of low-level waste and the supply capacity of facilities that can accept the low-level waste.

Historical escalation of low-level waste has been higher than CPI escalation; however, for the following reasons, this may not be the case in the future: 1. The Federal government has stated that it is the individual state's responsibility to dispose of the low-level radioactive waste. The states have formed eleven compacts to date where the states within each compact will work together to decide upon, where to develop new disposal facilities that could be used for all of the states within that compact. There will be economic pressure on the states to develop their own disposal sites if the low-level disposal costs continue to escalate, or as the existing facilities reach their waste capacity. 2. Rapidly increasing fees for disposal of low-level waste have spawned the creation of a niche market for firms specializing in the management of low-level waste. Since these firms are controlling the low level waste disposal of several companies, they are in a better position to negotiate disposal fees. These firms also specialize in volume reduction or waste treatment so that the waste could be disposed of in solid waste landfills. 3. Efficiencies in decommissioning should be studied as more and more nuclear power plants go through decommissioning.

2.7 DECOMMISSIONING FINANCIAL ASSURANCE REQUIREMENTS

An NRC licensee may take credit for projected earnings on the prepaid decommissioning trust funds using a 2% annual real rate of return from the time of future funds' collection through the projected decommissioning period. This includes the periods of safe storage, final dismantlement, and license termination, if the licensee's rate-setting authority does not authorize the use of another rate. However, actual earnings of existing funds may be used to calculate future find needs.

2.8 INSURANCE REQUIREMENTS

Any surety method or insurance used to provide financial assurance must be open-ended, or if written for a specific term, must be renewed automatically. The exception is if ninety days or more preceding the renewal date, the issuer notifies the Commission, the beneficiary, and the licensee of its intent to not renew. The surety or insurance must also provide that the full amount be paid to the beneficiary, automatically preceding the expiration date without proof of forfeiture, if the licensee fails to provide a replacement acceptable to the Commission within thirty days after receipt of notification of cancellation. In addition, the surety or insurance must be payable to a trust established for decommissioning costs, and the trustee and trust must be acceptable to the Commission. The surety method or insurance must remain in effect until the commission has terminated the license.

2.9 ACCEPTABLE PAYMENTS FOR DECOMMISSIONING

The NRC licensee is permitted to use 3% of the generic amount of decommissioning funds, even while the facility is operating for engineering design, work package preparation, and licensing activities. After submitting the certification of permanent cessation of operations and the certification that the fuel has been removed from the reactor vessel, the licensee may use an additional 20% of the funds for any legitimate decommissioning activities. However, the licensee is prohibited from using the remaining 77% of the generic decommissioning funds until a site specific cost estimate is submitted to the NRC.

Further, the licensee must not perform any decommissioning activity that results in there no longer being reasonable assurance that adequate funds will be available for decommissioning.

2.10 DISPOSAL OF HIGH-LEVEL AND LOW-LEVEL RADIOACTIVE WASTE

During decommissioning, both high-level and low-level radioactive waste must be disposed of properly. High-level radioactive wastes are:

A. Irradiated (spent) reactor fuel;

B. Liquid waste resulting from the operation of the first-cycle solvent-extraction system, and the concentrated wastes from subsequent extraction cycles in a facility for reprocessing irradiated fuel;

C. Solids into which such liquid wastes have been converted.

The **DOE** became responsible for the permanent disposal capacity for spent fuel and other high-level nuclear wastes in the Nuclear Waste Policy Act of 1982. The **DOE** was suppose to be able to accept waste in 1998; however, the DOE is still investigating possible sites. Presently, Yucca Mountain in Nevada is being developed as a disposal facility; however, it is not likely that this site will be available prior to 2015.

Although the **DOE** is responsible for the disposal of the spent fuel, the licensees are incurring significant costs in the construction and monitoring of the **ISFSI** (**Independent Spent Fuel Storage Installation**) which is required since the **DOE** is not ready to accept the spent fuel. This has created a tremendous amount of litigation where the licensees are suing the **DOE**. It is expected that this litigation may go on for several years.

Low-level waste is any radioactive waste that is not classified as high-level waste. As stated above, there are currently only three active licensed disposal facilities of low-level radioactive waste.

2.11 U.S. PRICE-ANDERSON ACT

The **Price Anderson Act** provides coverage for "any legal liability" arising from a "nuclear incident" with three specific exclusions:

A. Worker's compensation claims for persons employed at the site in connection with the activity,

B. Claims arising out of an act of war,

C. Damage to property at the site used in connection with the activities of the licensee.

This last exclusion implies that the **Price Anderson Act** does not cover decommissioning costs. Federal Statutes require reactor operators to maintain primary financial protection equal to the maximum amount of liability insurance available from private insurance sources at reasonable terms. See **10 C.F.R. .sctn. 50.54(w).**

The Act provides a three layered system of financial protection and indemnity agreements. In the first tier, licensees are required to provide proof of financial assurance protection in an amount equal to the maximum liability insurance available from private sources, currently $200 million. The second tier provides for a retrospective premium payment mechanism, whereby the industry would share liability for any damage resulting from a nuclear incident, currently $9.5 billion. In the event of such an incident, each commercial reactor licensee would be assessed a prorated share of damages up to the statutory maximum of $83.9 million per reactor per incident, but are limited to no more than $10 million annually per reactor per incident. In the third tier, the indemnity is guaranteed by the U.S. government.

2.12 PROPERTY INSURANCE

To meet the requirements of **10 C.F.R. .sctn. 50.54(w),** nuclear power plant licensees need to purchase the maximum coverage available. Currently, there are two levels of property insurance that provide coverage of post-accident stabilization and decontamination costs, "primary" and "excess" coverages. For example, both American Nuclear Insurers ("ANI") and Nuclear Electric Insurance Limited ("NEIL") offer primary property coverage up to a limit of $500 million. ANI offers excess coverage in the amount of $600 million, and NEIL offers excess coverage in the amount of $2.25 billion. The combined amount of coverage available is at least $1.1 billion, and potentially as much as $3.85 billion in property insurance.

ASCENTRUST LLC
The POWER of ENGINEERING

APPENDIX
INTRODUCTION TO
NUCLEAR POWER AND NTPBMR TECHNOLOGY

SECTION E: Proliferation

SECTION ONE: EXECUTIVE SUMMARY

The changing nuclear landscape and the integrated nature of the world's nuclear industry strengthen the case for a concerted effort by industry and government to develop jointly a new set of understandings of what the future nuclear proliferation dangers are, and to work closely together in the design and implementation of measures to prevent such proliferation.

Industry and governments have generally considered the issue of nuclear nonproliferation a political and security matter for government. Industry's view - broadly shared by most governments, but contested by some aspects of civil society - is that the nuclear power industry has no direct responsibility for nuclear weapons proliferation. Industry feels it is already highly controlled and regulated, and that abuses are largely the consequences of actions by rogue states and associated networks determined to develop a nuclear weapons program.

Yet sensitive nuclear technology, including technology ostensibly for peaceful purposes, has found its way into nuclear weapons programs since the 70s (into India, Pakistan, North Korea Iraq and now Iranian nuclear weapons programs to name a few) and industry was involved in many cases. These are significant examples of where equipment and material designated for peaceful purposes can, even inadvertently, be misused for non-peaceful purposes. States have in the past responded to these events, and to the failure of the international community to detect in a timely manner the weapons programs in a number of states, by taking remedial action such as the establishment of export control mechanisms in the Nuclear Suppliers Group (NSG), starting in 1978, the conclusion of an Additional Protocol to states' safeguards agreement with the IAEA in 1997 to assist with early detection of undeclared activities and, more recently, counter-proliferation actions through the Proliferation Security Initiative (with around 90 participants) and actions pursuant to April 2004 Security Council Resolution 1540. Membership of and adherence to these measures, guidelines and actions are not universal and many are still voluntary.

The global nuclear landscape is continually changing and never more so than in recent years. This is likely to put an even greater strain on efforts to contain nuclear proliferation. Climate change and rapidly increasing global energy needs have dramatically increased the attractiveness of nuclear energy as a known provider of base load power with a very low carbon footprint. We now face what some have called a 'nuclear renaissance', or a 'second nuclear age'. The challenge for the world is to ensure that this renaissance continues to be managed safely and securely at a time when nuclear proliferation pressures are on the increase. Importantly, we know that sensitive nuclear technologies, to which all NPT members believe they have a treaty-given right, can be diverted to non-peaceful use with relative ease by determined proliferators and that we need to find ways of better controlling them, perhaps through multilateral mechanisms. This will have an impact on industry interests and actions.

The ICNND's brief is broad, strategic and seeks to shape the future international nuclear order. Its aim is to add value to the sum total of efforts currently underway all over the

world to manage the large and growing nuclear challenges we face. Examining the role of industry in this context is one area where the ICNND will be able to add value. Under 'Peaceful Uses of Nuclear Energy', the ICNND lists four objectives for further examination by the Commission:

- *Establishment of a global understanding that ensuring 3S (safeguards, safety and security) are indispensable for peaceful uses of nuclear energy.*
- *Development of internationally agreed arrangements for effective control of sensitive nuclear technology (enrichment and reprocessing)*
- *Development of mechanisms for ensuring long-term supply of nuclear fuel and fuel management services so that states will not feel compelled to develop national fuel-cycle capabilities.*
- *Development by the nuclear industry of a comprehensive 'Code of Conduct' ranging from responsible uranium supply to support for the development of proliferation-resistant fuel cycle technologies*

There is an opportunity for industry to become a more active partner with governments to shape the world's nuclear future, to get on the front foot and to take a more proactive and less defensive approach.

Much of the world's nuclear industry is multinational, with significant public/private cross-ownership where commercial interests, nonproliferation interests and national strategic interests can overlap or collide. Yet governments have tended to manage proliferation as a political issue with virtually no industry involvement other than an expectation that it comply with directives which themselves can be difficult to follow or implement. Industry surveys in the US have shown that industry assesses it own performance in meeting export controls requirements as less than perfect. At the same time, some governments and states have found it convenient to ignore or tolerate proliferation where it suited their strategic or security positions.

Governments rarely include industry representatives in proliferation information exchanges or policy discussions in groups such as the Nuclear Suppliers Group (NSG) except through the occasional outreach activity. Yet industry is at the front line of the development and spread of dual-use nuclear technology and has the capacity to prevent, limit or place conditions upon the spread of that technology, as well as report it, and to influence the type of nuclear technology that is developed in the future.

Industry should be an active partner with Governments in the drafting of regulations and treaties that affect their activities, to ensure that they create a level playing field for all industry players and make operational sense to encourage compliance.

However optimistic the outlook is for the future of nuclear energy (and the global financial crisis may slow things in the short to medium term), the fact remains that in the eyes of the public, it remains a high-risk industry where a major incident can have disastrous consequences. Governments also consider that the rise in nuclear power

worldwide does increase the risk of proliferation, even if they understand that value of nuclear energy as a provider of energy with a low carbon footprint. An aggressive growth program risks accentuating those fears. We know that in many parts of the world, the public remains opposed to the introduction of nuclear energy, even if attitudes are starting to shift.

Being politically more proactive does not mean that industry has to abandon its evidence-based approach to risk. It can, however, help industry in its ambition to strengthen and sustain public confidence, both in the reliability of nuclear technology and in the people and institutions responsible for its use.

Moreover, governments have under active consideration the development of new rules of the game which may have real impact on the development of the industry, most notable among them proposals: to multilateralize the nuclear fuel cycle; to limit the spread of sensitive nuclear technologies; and to change NSG rules to insist that countries not exercise the right to develop sensitive technology as a condition of supply, as well as making the adoption of the Additional Protocol a mandatory condition of supply.

In the nuclear industry, commercial interests are tightly woven into national interests, especially when it comes to the right to develop sensitive nuclear technologies such as enrichment. The controversial two-tier system enshrined in the Nuclear Nonproliferation Treaty (NPT) between nuclear weapon states and non-nuclear weapon states spills over into the peaceful uses domain. Initiatives to limit the possession and use of sensitive nuclear technologies to those who have them now for good nonproliferation reasons is currently opposed by emerging nuclear powers which will not, on understandable equity grounds, accept the perpetuation of a two-tier system in the nuclear power industry.

There is no chance these states will even consider foregoing the right to develop sensitive aspects of the nuclear fuel cycle, or see them centralized or regionalized under multinational control, in the absence of a solid commitment from the nuclear armed states to achieving a world without nuclear weapons.

In this context, a global call for disarmament might also become the business of industry. It may be worth exploring whether industry is prepared to make a public commitment to the goals of disarmament and nonproliferation as a sign of good faith and in the interests of the future bona fides of the business as well as a contribution to dismantling the two-tier system.

The world's chemical industry certainly understood (eventually) the advantage of demonstrating to shareholders and to the public its commitment to chemical disarmament and nonproliferation, especially in light of its inadvertent contribution to Iraq's chemical weapons program. The industry understood that if it was going to be regulated intensively and obtrusively, there were distinct advantages to industry being an active collaborator in ensuring that its business did not contribute to chemical proliferation, while at the same time having a direct say in how commercial confidentiality could be preserved through the Chemical Weapons Treaty. The 1989 **Government-**

Industry Conference Against Chemical Weapons provided a useful vehicle to publicly set the basis for a successful government-industry partnership for this purpose.

Without the chemical industry's active support and collaboration, that treaty could most probably not have come into existence. It is, however, the case that **GICCW** took place in the context of the emerging global consensus among states that chemical weapons should be abolished altogether. No such consensus exists for nuclear weapons, other than the aspiration to general and complete disarmament in the NPT. Yet the global, integrated nature of the nuclear business, its very close connection to government and a changing nuclear policy landscape, including the renewed push towards progress in nuclear disarmament, argue strongly in favor of more regular government-industry collaboration, including through joint monitoring, reporting and enforcement of the rules and export controls. A jointly negotiated declaration as to how that could be done would add a new dimension to the global nuclear conversation.

Initial signs are that some industry players see opportunities and advantages to becoming more engaged in the global nonproliferation agenda. An increasingly globally integrated industry needs to take a global view and be more globally engaged. The CEO of AREVA has agreed to become a member of the ICNND's Advisory Board. Members of industry are now active participants in second-track discussions about the future role of nuclear industry in a growing nuclear power market. The 2008 WNA policy documents and its Charter of Ethics and Principles of Uranium Stewardship spell out clearly industry responsibilities in ensuring 3S (safeguards, safety and security) are indispensable for peaceful uses of nuclear energy. The Australian Uranium Association has begun to advocate for best practice in support of nonproliferation and its uranium stewardship principles support broader engagement to bring that about.

CONCLUSION

The engagement of industry as a whole will require intense diplomatic effort and will have to be managed adroitly. Large commercial interests are at stake and if there are to be additional standards, they will need to be universally applied.

More information is needed about industry's views on these matters, and, given the very close relationship between much of the world's established nuclear industry and government, government views are also important.

What is set in train today, given the complexity of the issues and the relatively long lead time involved in the nuclear business, will play out over the decades to come.

The paper takes the long view and argues for a more concerted partnership approach between governments and industry which will set benchmarks for the joint management of this enterprise over time.

This paper examines the opportunities and constraints relating to intensified government-industry cooperation in light of increased global interests in nuclear energy, while mitigating the attendant risks.

Key industry players and the current state of the nuclear industry are identified. The paper then examines the 'nuclear renaissance' and its implications for additional proliferation risks. It looks at the case for greater nuclear industry engagement in nonproliferation, incentives and disciplines, as well as possible disincentives for industry to take a more active role in nonproliferation and in the limitation of the pursuit of sensitive technologies which give rise to proliferation concerns.

Some preliminary thoughts are advanced as to how such a process might yield more concerted and regular government-industry collaboration in a way which is effective, sustainable and can generate the confidence of government, the public and industry. This includes a brief exploration of the value of a Code of Conduct or other arrangements for the effective management of future nuclear proliferation risks

Finally, the paper will examine the possibility of convening a nuclear government–industry conference or summit similar to the Australian sponsored government–industry conference against Chemical Weapons in 1989, which would discuss these issues, perhaps agree on a joint strategy for intensified collaboration in nonproliferation, or at least make recommendations in this regard.

SECTION TWO: THE NUCLEAR INDUSTRY

2.1 SCOPE OF ACTIVITIES

Nuclear power industry activities can be broadly divided into fuel cycle activities, reactor activities and support activities. Fuel cycle activities include uranium mining and milling to produce ore concentrates, conversion of uranium ore concentrates into uranium hexafluoride or uranium dioxide, uranium enrichment, fuel fabrication, spent fuel reprocessing and nuclear waste management, and the design and construction of fuel cycle facilities. Reactor activities include reactor design and construction, reactor operation, maintenance and decommissioning. Both reactor and fuel cycle services rely upon a number of support activities, including consulting, legal services, parts manufacturing, fuel transportation and fuel supply brokers, research and development (R&D) institutions (government, enterprise or university-based) and industry bodies.

The industry activities of most proliferation interest are the fuel cycle activities, and reactor design, which determines the physical and isotopic nature of the irradiated fuel.

2.2 KEY PLAYERS

The nuclear industry is dominated by three companies that engage in fuel cycle, reactor and support activities. The French company AREVA holds the largest market share in the global nuclear market, (25-30%) and is developing reactors in a joint venture with Mitsubishi Heavy Industries, followed by General Electric-Hitachi, and Westinghouse (77% owned by Toshiba). Russia's Atomstroyexport and the China National Nuclear Corporation are positioning themselves to challenge the market dominance of these

three Western-Japanese nuclear companies in turnkey reactor sales. Six companies operate commercial enrichment facilities, the China National Nuclear Corporation (2); Eurodif (1); Rosatom (4); Japan Nuclear Fuel Limited (1); Urenco (3) and the United States Enrichment Corporation (1). Three additional multinational enrichment facilities are being planned for construction in the United States. The only two commercial reprocessing plants are operated by AREVA (La Hague, France) and Sellafield Ltd (Sellafield, UK). The eight largest uranium ore producers were responsible for approximately 85% of global production in 2014, and include Cameco, Rio Tinto, Areva, Kazatomprom, Rosatom, BHP Billiton, Navoi, Uranium One and General Atomics. The largest reactor operator in the industry is Electricité de France (59 reactors).

2.3 PRESENT CAPACITY

There are approximately 439 operational nuclear reactors globally. These Nuclear reactors account for 16% of world electricity production, and 57% of global nuclear generating capacity is situated in the United States, France and Japan. 34 new reactors are under construction in China (7); Russia (7); India (6); South Korea (3); Canada (2); Slovakia (2); Japan (2); Argentina (1); France (1); Finland (1); Iran (1) and Pakistan (1). The first two Generation III+ reactors, both European Pressurized Reactor (EPR) designs, are presently under construction at Flammaville, France and Olkiluoto, Finland.

SECTION THREE: DRIVERS OF EXPANSION IN NUCLEAR ENERGY

3.1 INCREASING ENERGY DEMAND

Global population growth, and economic growth in developing countries, resulting in higher per capita energy consumption, are projected to increase primary energy demand by a factor of approximately 2.5 by 2050 if present policies remain unchanged, and electricity demand by a factor of 1.8 to 3.7. In most countries that produce nuclear energy, present generating capacity will also need to be renewed, including the 342 reactors (of 439 globally) currently aged 20 years or older. Increased demand for fresh water will also increase demand for desalination plants that are increasingly likely to be powered by nuclear energy. If present nuclear power capacity and the share of nuclear in the total energy production mix are to be maintained, then more power reactors will be needed.

3.2 CLIMATE CHANGE

Nuclear power is the only mature base load electricity production method that does not burn fossil fuels and is a relatively low emitter of greenhouse gases, making it an attractive alternative to fossil fuels to governments seeking to reduce their carbon emissions. Yet the contribution that nuclear power can make to mitigating climate change is limited by the long lead times required to bring nuclear power plants online, the present lack of capacity to respond to a rapid increase in demand for nuclear power, and the fact that electricity accounts for only 27% of greenhouse gas emissions. Nuclear power will still have to compete against renewable energy in the future as those technologies mature.

3.3 RISING FOSSIL FUEL PRICES

Large increases and volatility in fossil fuel prices in recent years make nuclear energy a more attractive option because fuel prices account for a relatively small proportion of the total cost of nuclear power generation (the majority of the cost being the plant itself), protecting electricity production costs from fluctuations in uranium prices to an extent unparalleled with coal, natural gas or oil.

3.4 ECONOMICS OF NUCLEAR POWER

Increases in the price of fossil fuels and the anticipated pricing of carbon emissions in many Western countries are expected to improve the economics of nuclear power vis-à-vis other base load power generation options. Where nuclear power remains at a disadvantage is in the sizeable construction costs incurred prior to the reactor producing electricity for sale, as well as reactor decommissioning and waste disposal costs. The slow in construction of nuclear reactors in the West in recent years means that cost estimates are uncertain, and the viability of nuclear power constructed without government assistance in deregulated electricity markets is questionable, especially in light of the recent credit crunch.

3.5 ENERGY SECURITY

Concerns about reliability of oil and natural gas supplies in recent decades stem from rising prices and fears of political interference in supply. Governments have considered including or increasing the share of nuclear power in their energy generation mix in order to reduce dependence on fossil fuels. Major uranium producers such as Canada and Australia are viewed as reliable energy suppliers due to their stable domestic political environments. This strategy should not, however, be understood as seeking energy 'independence'. Given the internationalized nature of the nuclear fuel cycle, the process is rather one of diversification. Further, France and Japan have not been able to reduce their dependence upon imported oil by expanding nuclear energy production, as oil constitutes a very small part of total electricity generation in those two countries.

SECTION FOUR: DETRACTORS TO THE USE OF NUCLEAR POWER

4.1 PUBLIC OPPOSITION TO NUCLEAR POWER

The public aversion to nuclear power that peaked during the 1990s is diminishing. The World Nuclear Association attributes this to the impeccable safety record of the nuclear industry after the Three-Mile Island and Chernobyl accidents, the fact that the health effects of Chernobyl were less severe than expected, and community acceptance of nuclear waste repositories. Nevertheless, nuclear phase-out plans or de facto moratoriums on nuclear build are in place in Switzerland, Sweden, Spain, Germany and Belgium, often in response to public aversion to nuclear power.

Even in Japan, where nuclear power is well established, the public remains wary of its dangers, especially following significant incidents at nuclear power plants such as the earthquake damage in July 2007 to parts of the Kashiwazaki-Kariwa Nuclear Power Plant.

That said, the potential of nuclear power to combat climate change may be a decisive factor in changing public attitudes to nuclear power. A poll taken in Europe in 2008 reports a decline in European hostility to nuclear power, as that hostility has yielded to the more pressing concern of global warming. According to that poll, 44% of people in the European Union support nuclear energy, up from 37% in 2005, and 45% oppose it, down from 55% four years ago. In Australia, where a majority of Australians remains opposed to nuclear energy, recent polling suggest that Australians are increasingly attuned to the argument that nuclear energy needs to be part of the future energy mix.

4.2 MANUFACTURING CAPACITY CONSTRAINTS ON EXPANSION

The capacity of the global nuclear industry is the major constraint upon a rapid expansion in nuclear energy. Supply bottlenecks in human resources, heavy forgings and other reactor parts are likely to worsen as demand increases. Other key components such as reactor cooling pumps, diesel generators, and control and instrumentation equipment have long lead times, requiring up to six years to procure and manufacture. Personnel qualified to design, construct and operate nuclear facilities are increasingly difficult to employ as present employees approach retiring age, and a decreasing number of university degrees are awarded in nuclear relevant fields. Governments and intergovernmental nuclear agencies have put in place measures to encourage students to enter the nuclear field and support nuclear R&D, however the maintenance of power reactor skills and competence has been largely left to industry.

The OECD Nuclear Energy Agency estimates that, based upon historical experience in the 1980s and the expansion in global industrial capacity since, nuclear industry capacity may feasibly increase to meet projected demand, as additional capacity would not be required until after 2020, from bringing 10 reactors online per year up to 2020, to 40-50 per year in the 2030s and 50-60 in the 2040s. These figures suggest that the long lead times for nuclear projects will allow industry sufficient time to rebuild and expand capacity such that construction schedules and reactor safety are not compromised in the coming nuclear renaissance. The disadvantage of long lead times is that they limit the contribution that nuclear energy may make to reducing carbon emissions.

SECTION FIVE: ASSESSING PROLIFERATION RISK EXPANSION

5.1 PROLIFERATION RISK IN HIGH BUILD ENVIRONMENT

The proliferation risk of the second nuclear age is determined by three principal factors: whether the expansion takes place in existing nuclear power states or new nuclear power states; the geostrategic contexts of countries acquiring nuclear technology for the first time; and the nature of the nuclear technology acquired.

Eighty per cent of the expansion in nuclear power is forecast in countries already using nuclear power. Newly-minted nuclear countries are likely to account for only 5% of global nuclear capacity by 2020. China, Russia and India will account for the largest increases in new nuclear generating capacity by 2020, though the United States, France and Japan will retain their dominant position, producing 50% of global generating capacity. The non-nuclear power countries which have planned or approved nuclear power generation are Vietnam, Turkey, Iran, Indonesia, Belarus and the United Arab Emirates

(UAE), although in Indonesia popular opposition may yet prevent plans going ahead. Countries without a present nuclear power capacity which have proposed or intend to use nuclear power are Thailand, Bangladesh, Bahrain, Egypt, Ghana, Georgia, Israel, Jordan, Kazakhstan, Kuwait, Libya, Malaysia, Namibia, Nigeria, Oman, the Philippines, Qatar, Saudi Arabia, Uganda, Venezuela and Yemen.

The states seeking nuclear power for the first time are concentrated in Africa, the Middle East and Southeast Asia. All are zones of varying degrees of domestic political instability. The Middle East is strategically unstable and directly affected by the Iranian enrichment program. While Southeast Asian countries are not directly in the line of North Korean nuclear threats, their security would nonetheless be affected by a deteriorating East Asian strategic environment were Pyongyang's nuclear ambitions to be unchecked. In all three regions, states have genuine reasons for wanting to develop nuclear power, including growing energy demand and the desire to preserve fossil fuels for export, and in many cases had been interested in acquiring nuclear power prior to the Iranian and North Korean proliferation crises. Significantly, Vietnam and Indonesia have signaled their intent not to develop an enrichment capacity, as have Bahrain and the UAE.

While no other state with recent nuclear energy ambitions has expressed intent to develop enrichment or reprocessing capabilities, Egypt has refused to rule out its acquisition of such technologies on equity grounds. Such attitudes do not allay suspicions that the renewed interest in nuclear power in the Middle East is at least in part a hedging strategy in response to Iran's nuclear program. Some analysts have expressed concern over the proliferation risks posed by the lack of regulatory competence in the region. No new plans for enrichment or reprocessing have been advanced in Africa or South East Asia, though fuel preparation may become economically viable as more plants come online in the region.

Important factors in the realization of these nuclear energy ambitions are whether these states will be able to pay for their nuclear energy plans and whether they can develop and finance the necessary regulatory and technical bases to realize them safely. They do suggest fertile ground for increased assistance from established nuclear powers and industry to help them develop competence in regulation and effective export controls. In the present economic climate, the ability to finance these costly projects, however, is far from assured.

Nuclear reactors themselves, in particular the standard light water reactors (LWRs), are not considered a high proliferation risk because the isotopic content of the spent fuel and the difficulty of separating plutonium from the spent fuel assembly mean that they are not effective producers of fissile material. No additional states currently have plans to construct commercial enrichment plants, though Argentina, Brazil and South Africa have the capacity and so far insist on the right to do so in future. No state currently has firm plans to construct a commercial reprocessing plant. Renewed US support for reprocessing as a method of dealing with the waste disposal problems has led to R&D cooperation with South Korea on pyro-processing techniques, a reprocessing technique

that present research shows to be more 'proliferation safe' than the PUREX process presently used, but is by no means 'proliferation resistant'.

Nuclear energy ambitions among states without an existing nuclear power capability are not of direct proliferation concern, especially if sensitive technologies are not pursued.

However, views on whether an increase in the number of power reactors around the world poses an increase in nuclear proliferation dangers differ. John Ritch, who was President Clinton's Ambassador to the IAEA in the 1990s and is current Director General of the World Nuclear Association (WNA), is not convinced that even a tenfold increase in power reactors in the would have a significant impact on nuclear proliferation. He believes that by far the greatest problem is rogues states determined to develop a nuclear weapons program and their number has not significantly increased in the last 10-15 years. This is in contrast with the views in the 2008 report of the International Security Advisory board of the U.S. Department of State that 'the rise in nuclear power worldwide, and particularly within Third World countries, inevitably increases the risks of proliferation.

There is always a risk that the establishment of even the most basic nuclear infrastructure and expertise can presage later pursuit of a full nuclear fuel cycle. At the very least, it gives such countries that option. Under cover of their rights to develop such technology, the examples of Iran and the DPRK have presented great challenges to the international community in managing future nuclear ambitions by new states under the current international rules, which have not deterred a determined proliferating state.

5.2 MITIGATING THE PROLIFERATION RISK

Three strategies suggest themselves to policymakers and industry to mitigate the proliferation risks of the second nuclear age: technical solutions, commercial solutions and political solutions. Technical solutions would include making fissile material more technically difficult to produce, and include the development of nuclear reactors that produce less or no fissile material and/or make any fissile material more difficult to extract. Commercial solutions might include replacing turnkey reactor sales contracts to build-own-operate contracts, or inserting minimum nonproliferation requirement provisions into supply contracts. Political solutions would include placing the nuclear fuel cycle under multilateral control and restricting supply to those states with an Additional Protocol in place with the IAEA. Industry is a necessary partner in all three approaches.

SECTION SIX: INDUSTRY AND NON-PROLIFERATION

6.1 THE CASE FOR INDUSTRY INVOLVEMENT IN NON-PROLIFERATION

Industry generally abides by the international nonproliferation regime, in most instances cooperating with national safeguards obligations, physical protection of nuclear materials and export controls. Industry must, in partnership with government and the IAEA, manage the unique threats of nuclear accident, nuclear terrorism and nuclear proliferation, all of which have significant public policy implications. The industry does not actively promote nonproliferation, though it actively manages and mitigates the

threats of nuclear accident and, to a lesser degree, nuclear terrorism, at both the industry-wide and company level.

The lack of active industry engagement in nonproliferation advocacy does not necessarily increase the risk of proliferation. That said, a more active partnership on nonproliferation may well be needed for the future, as the world's nuclear industry grows, and where, as a consequence of growing demand, governments look to tighten the nonproliferation regime.

The nuclear industry, fairly or unfairly, continues to suffer public image problems and must be, like Caesar's wife 'above suspicion'. The slightest misstep is likely to have far graver consequences for the industry than for other industries utilizing different sources for energy production. The Chernobyl, Three Mile Island and Fukushima incidents and their effect on the acceptability of nuclear power were dramatic enough. (If another country were to acquire nuclear weapons using technology sold by a particular company, its corporate image and the image of the industry as a whole would be tarnished. Were a nuclear weapon detonated, either by a state or non-state actor, then the nuclear power industry would come under massive public and governmental pressure to demonstrate that it posed a zero proliferation risk.) This could badly damage the industry's prospects and perhaps even its survival. So industry has a strong impetus to support nonproliferation.

The proposition that an active nonproliferation stance by industry could be a public confidence-building tool and even a commercial imperative rather than primarily a box to tick should at least be tested.

6.2 OBSTACLES AND INCENTIVES TO INDUSTRY INVOLVEMENT

Depending on one's perspective, obstacles and incentives for greater industry involvement in nonproliferation are two sides of the same coin, given the high degree of public/private and cross-border ownership in the industry. It is not so easy to determine where private interests end and public interest starts. As many nuclear companies are wholly or partially government-owned, or enjoy close links with government, this should augur well for tighter government-industry cooperation in nonproliferation, provided there is a commitment to this by both government and industry.

That said, a strong perception persists within the nuclear industry that nonproliferation is a government responsibility and is adequately managed by governments. Governments tend to see proliferation as a political issue which is not the domain of industry. On the other hand companies are concerned that recognizing a link between their activities and weapons proliferation could tarnish their corporate image and damage business. There is also a common perception that the nuclear industry is already overregulated and does not require any additional regulatory burden to address proliferation. In particular, industry is concerned by additional costs that may be incurred in actively preventing proliferation.

On the face of it, the disincentives for industry to get ahead of government are wide-ranging, from loss of profits, to corporate image concerns, to loss of competitiveness

within the industry. Companies fear that if they tighten their conditions of sale to prevent proliferation, they will be undercut by less scrupulous suppliers seeking to improve their market share, resulting in a loss of competitiveness and profits. They are also wary of the effect nonproliferation cooperation would have on their image of independence from government. Companies are rightly concerned that information sharing resulting from any increased cooperation with government raises the issue of the protection of proprietary information. Companies offering products or services that are more proliferation prone than others on the market will suffer a loss of sales and profits and may go out of business if they act in furtherance of nonproliferation.

The nonproliferation rules and treaties are drafted by government and governments are responsible for ensuring they are implemented through domestic legislation. Governments rarely include industry representatives in proliferation information exchanges or policy discussions in groups such as the Nuclear Suppliers Group (NSG), except through the occasional outreach activity. Another obstacle arises from differing levels of support for the nonproliferation regime among governments and divergent attitudes towards the acquisition of enrichment and reprocessing technology by states not already possessing them. It may be the case that were industry to be more supportive of the nonproliferation regime than governments, corporate interests and national interests might also diverge.

The close relationship between government and the nuclear industry does not guarantee that nonproliferation commitments will take precedence. Nuclear cooperation agreements continue to be pursued between advanced nuclear states and countries in all regions of the world, apparently without real concern about the possible proliferation dangers that such assistance might give rise to. There are instances where governments have not acted in the interests of nonproliferation first, and where they have been swayed by the commercial interests of their nuclear industry or by overriding strategic and security concerns.

Industry surveys in the US have shown that industry assesses its own performance in meeting export controls requirements as less than perfect.

What this suggests is the need for a concerted effort by industry and government to develop jointly a new set of understandings of what the future proliferation dangers are as well as a demonstrable commitment to nonproliferation, which can also be as good for business as they are for security. To be really effective, this probably needs to be at the global level.

Active industry support and engagement will be necessary if major changes are made to the international market structure in order to make it more proliferation safe, for example in placing enrichment facilities under multilateral control. Given the high costs of fuel cycle activities, it has been suggested that finding economies of scale through a multinational approach could fulfill the dual role of keeping costs down while helping support nonproliferation policies. For example, companies and states might consider becoming shareholders in multi-nationally-owned modern centrifuge facilities, using

leased centrifuge machines under 'black box' conditions as an alternative to investing in their own smaller, high-cost enrichment facilities. Such facilities would, of course, need to be accessible to nations yet to develop their own fuel cycle facilities, maybe even as joint plant operators as well as consumers. These facilities would need to develop appropriate rules for the supply of nuclear fuel which not only supports nonproliferation but effectively guarantees security of supply free of capricious political interference.

Industry is also at the front line of the development and spread of dual-use nuclear technology and has the capacity to prevent, limit or place conditions upon the spread of that technology, as well as report it, and to influence the type of nuclear technology that is developed in the future. Industry reporting of sales could assist the IAEA in assessing the completeness of member-state declarations.

Large nuclear companies can exert considerable pressure upon their national governments in their nuclear policy choices. Therefore an industry which makes nonproliferation a priority may also help reinforce the nonproliferation commitments of government. Making a commitment to nonproliferation part of the corporate brand might in fact deliver practical benefits for companies, helping to cultivate better relationships with regulators and nonproliferation advocates, and dispel the poor image created by the anti-nuclear lobby. Of course there are limits to the pressure that even larger nuclear companies can exercise when they are publicly owned and where broader national security and strategic concerns come into play.

Industry-wide initiatives to stem proliferation would require a harmonization of business practices, ensuring that no company was disadvantaged for being more proactive on proliferation and thereby discouraging the first mover. More generally, industry should be an active partner with governments in the drafting of regulations and treaties that affect their activities, to ensure that they make operational sense and to encourage compliance.

6.3 REQUIREMENTS TO ENGAGE INDUSTRY

The nuclear industry currently cooperates with governments to fulfill their nonproliferation obligations, abiding by export controls and their safeguards inspection and reporting requirements. Industry has been effectively engaged in Generation IV reactor activities under the Advanced Fuel Cycle Initiative of GNEP, to develop proliferation safe reactor designs with US government R&D funding. Beyond their obligations and R&D cooperation, the industry contribution to nonproliferation is minimal, and advances in nuclear safety and security have little to offer by way of precedent, as they have primarily engaged nuclear operators. Nonproliferation values are, however, contained in the WNA Charter of Ethics and Principles of Uranium Stewardship.

6.4 WHAT CAN INDUSTRY DO?

Not enough is known about how far industry is prepared to go in taking a more prominent stand on nonproliferation. Members of the Australian Uranium Association (AUA) have shown an interest in encouraging industry to become more prominent and confident advocates in favour of nonproliferation because they think, by and large,

industry has a good story to tell and because of their commitment to uranium stewardship principles.

We have listed here some general considerations and ideas for designing initiatives to further engage industry on nonproliferation. This includes examining the pros and cons of an industry-wide Code of Conduct and a government-industry conference which might help set the tone for the future management of the 'second nuclear age'.

More information is needed on how industry would respond to these ideas, or indeed other ideas for how such increased engagement might be effected.

Such information could be obtained in numerous ways. One way would be a survey which could be put to a selection of key industry representatives on which a future government-industry dialogue might be based.

The Commission could consider asking the ICNND Secretariat or one of the participating research centers to design and send a survey to the key industry players and perhaps also to IAEA member states to assist in shaping the ICNND's own recommendations about greater industry engagement in nonproliferation.

6.5 WHOM TO ENGAGE

The answer to this question will depend upon the type of initiative and desired outcome of any industry engagement. There are two broad options for whom to engage: key companies supplying sensitive nuclear technology, or as many nuclear industry companies as possible.

Targeting the suppliers of sensitive nuclear technology would engage those whose conduct will bear most directly upon the future of the nonproliferation regime. Stemming the expansion of enrichment and reprocessing capabilities and/or multilateralising those existing facilities will require the direct cooperation of those companies. Companies to be engaged on this matter should include Areva, Rosatom, Urenco, Eurodif, China National Nuclear Corporation, Japan Nuclear Fuel Limited, Westinghouse, GE Electric, Silex, Industreas Nucleares do Brasil, BNFL, Cameco, the Pakistan Atomic Energy Corporation, Nuclear Fuel Complex (India), the Nuclear Energy Corporation of South Africa and any other company with control of enrichment or reprocessing facilities and technology.

Another possibility is to engage as many nuclear industry companies as possible, whether they engage in fuel cycle, reactor or support activities, in order to create an industry-wide norm and momentum in favor of nonproliferation. This broader strategy would ensure that smaller companies were as committed to nonproliferation as the market leaders, and that industry leader commitment to nonproliferation would not be undermined by other companies who had not been similarly engaged by government. It also addresses the concern that a large number of nuclear activities have the potential to contribute to a weapons capability by building up the necessary infrastructure and expertise.

Whether done sequentially or at the same time, targeting the key industry players in sensitive nuclear technology and engaging the wider industry will be necessary to achieve an industry-wide commitment to nonproliferation that is also capable of delivering practical results. It is likely that the companies dealing with sensitive nuclear technology will not commit to nonproliferation without an assurance that the rest of the industry will support rather than undercut them, while the industry as a whole is unlikely to commit to nonproliferation without the leadership of the major companies.

This might be a role for an industry peak body such as the WNA, which could begin by engaging the suppliers of sensitive nuclear technology with the intention of associating a commitment to nonproliferation with leadership of the industry, as well as encouraging smaller companies to comply with new standards of appropriate industry behavior.

6.6 OUTCOMES

Two types of outcomes may result from industry engagement – symbolic outcomes, in which industry declares its support for preventing proliferation as an exercise in public diplomacy, and practical outcomes, in which companies take active measures in order to prevent proliferation. Both types of outcomes are desirable and mutually reinforcing. A symbolic commitment would raise awareness of the role of industry in facilitating or preventing proliferation and provide a standard against which industry could be held responsible for the proliferation implications of its conduct, while generating a positive public image for the industry.

Practical outcomes could fill some of the gaps in the nonproliferation regime and contribute to the overall strengthening of the regime. Examples include:

- Industry collaboration in the establishment of multilateral fuel cycle services;
- Making minimum nonproliferation standards a condition of supply of nuclear technology written into contracts e.g. requiring that states purchasing nuclear reactors have an Additional Protocol (or equivalent safeguards agreement) in place with the IAEA;
- Reporting suspicious procurement efforts to national authorities or the IAEA;
- Disclosing sales information to assist the IAEA in verifying the completeness of state nuclear declarations;
- Developing technologies with a lower proliferation risk and ceasing sales of products that pose an unacceptable proliferation risk;
- Government-industry-IAEA consultation in the drafting of any new regulations, treaties or protocols, or in updating existing instruments to ensure that they are as effective as possible;
- Mechanisms for sharing nonproliferation best practices, and for enforcing compliance with such measures;

- Assisting states with fledgling nuclear power programs to develop or strengthen their competence in regulation, safety and effective export controls.

These outcomes may be achieved through industry self-regulation or through cooperative action between government and industry. They are more likely to be successful if done collaboratively with government and agencies like the IAEA.

SECTION SEVEN: ETHICS, NUCLEAR INDUSTRY AND PROLIFERATION

7.1 CODE OF CONDUCT

Industry Codes of Conduct are a form of business self-regulation and may be divided into three main types of codes, those with an aspirational purpose (a code of ethics), an educational or advisory purpose (a code of conduct) or a restrictive purpose (an enforceable code of practice). The Biological Weapons Convention experience with codes of conduct is not entirely applicable to the nuclear industry, yet the Biological Weapons Convention experience highlights some important questions that any nuclear industry code of conduct would need to respond to, including the need to clearly define the purpose, audience and function of the code of conduct.

The WNA has a series of principles and codes it has developed over time, the latest iteration of which was published in January 2008 called the 'New WNA Policy'. These polices include the WNA Charter of Ethics, the WNA Principles of Uranium Stewardship and Principles for Managing Radiation, Health and Safety, Waste and the Environment and the International Council of Mining and Metals (ICMM) Sustainable Development Principles. These policies have, according to the WNA, been developed by industry leaders with the support of the full WNA membership and key organizations such as the IAEA and the ICMM. According to the WNA, these codes 'hold the status of a policy and ethical declaration by the full WNA membership, which encompass most of the wide range of enterprises that comprise the global nuclear industry-from uranium miners, to equipment suppliers, service providers, and generators of electricity.'

With around 180 members, the WNA represents 90% of worldwide uranium production and of nuclear power generation. The WNA has pledged to obtain, from all relevant enterprises, formal commitment to a Code of Practice that translates its principles into worldwide industry performance; to conduct periodic audits, peer reviews and public information activities. The WNA does not have a mandate to enforce any of the provisions of its code of practice and ethics. These codes and principles are ultimately enforceable through national legislation and regulation in accordance with a number of international treaties and statutes covering the range of peaceful nuclear activities.

7.2 SELF-REGULATION

In the nuclear industry there are also two examples of more elaborated processes for sharing and disseminating best practice and information, one of which is still under development:

The World Association of Nuclear Operators (WANO), formed in May 1989 in response to the Chernobyl accident to improve safety standards at nuclear power plants worldwide

such that a repeat accident would never occur, provides a forum for the exchange of operating experience in a 'culture of openness' amongst various nuclear operators. WANO conducts voluntary peer reviews of nuclear safety at another member's plant and provides a report on that plant based on safety criteria and quantitative performance indicators, in addition to providing workshops and seminars and technical support and exchange.

The effectiveness of WANO is attributed to the fact that 'the nuclear industry perceived them as its own ideas, operating to serve the industry's own interest. These organizations also had direct access to the utility CEOs, who could bring powerful peer pressure to bear on any CEO whose utility was lagging behind.'

The recently-established World Institute for Nuclear Security (WINS), a joint initiative of NTI and the Institute of Nuclear Materials Management, intends to bring together representatives from government, industry, academia and think tanks in an effort to share best practices on nuclear security, in a similar model to WANO.

WANO (and possibly WINS also, in the future) provides an example of how industry initiatives to improve the safety record of nuclear operators have surpassed the minimum safety standards imposed by national legislation and have facilitated more uniform safety standards internationally. A commitment to nuclear safety is a very common corporate social responsibility principle for companies operating nuclear reactors. The sharing of best practices, performance indicators and peer reviews are mechanisms that could be transposed into the nonproliferation arena, as WINS is attempting to do for nuclear security.

7.3 GLOBAL NON-PROLIFERATION CODE

Nonproliferation involves a broad and complex network of treaties, rules, and actions in multiple locations using open-source and classified information. It does not seem to lend itself to location-specific safety and security codes of conduct. The question must therefore be asked whether any kind of code of conduct can add anything to practical nonproliferation efforts. It might be said that any code of conduct is no replacement for rules and regulations pursuant to international treaty obligations and export control legislation and that in fact it may be inimical to nonproliferation to settle for a ‘Code of Conduct’ in lieu of legal obligations.

While codes of conduct may be implemented at many different levels (company, national, regional, universal), a nuclear industry code of conduct would also be ineffective if it were not universal in application. As noted, codes of ethics relating to nonproliferation already exist in the industry and have probably done little to deter those entities determined to sell equipment to sensitive locations, as the information about Chinese sales of dual-use equipment to Iran would suggest.

A new code of conduct would be either advisory or enforceable, and draw lessons from the lack of success of present codes of conduct. The difficulty of devising international industrial enforcement mechanisms (especially if customer/government complicity is

involved) suggests that an advisory code of conduct would be the most achievable for the nuclear industry at present.

An industry code of conduct would need to be drafted by industry representatives in consultation with government regulators, nonproliferation experts and representatives of intergovernmental nuclear agencies to ensure that all stakeholders are included in the drafting stage and are more likely to accept the finished product. The process of drafting a code of conduct could be beneficial in raising awareness of issues and facilitating debate about appropriate conduct, and should entail provisions for review and revision in order to 'keep the conversation going'. A provisional code of conduct could be drafted by a ginger group of companies, or a working group of WNA members, which would then seek the input of governments, international agencies and other companies. The draft code of conduct could then be presented to an industry conference, or at a special conference called for the purpose of improving industry contribution to nuclear nonproliferation (this need not necessarily be the government-industry conference discussed below). The code of conduct would be implemented by companies, but could benefit from government and industry body promotion.

This could be a lengthy process, and seen to be duplicating obligations which government and industry already have under existing legal regimes. It could be difficult to enforce. However, we were encouraged to hear from some industry representatives that they viewed the process itself as part of the answer, as it would highlight engagement which, properly managed, would eventually lead to the right result.

An interim step would be to encourage nuclear industry companies to include a commitment to nonproliferation in their corporate social responsibility statements, alongside commitments to sustainable development, nuclear safety and security.

SECTION EIGHT: GOVERNMENT-INDUSTRY CONFERENCE

8.1 GOVERNMENT-INDUSTRY CONFERENCE

8.1.1 GOVERNMENT-INDUSTRY CONFERENCE AGAINST CHEMICAL WEAPONS

The idea of a government-industry conference is drawn from the experience of the Chemical Weapons Convention, in which a Government-Industry Conference against Chemical Weapons in 1989 played an important role in the conclusion of the treaty in 1993 after more than twenty years of negotiation. GICCW was the culmination of four to five years of intensive consultation and diplomatic activity between government and the chemical industry. It started with the establishment of the Australia Group in 1985 when Australia brought together representatives of industrial nations which exported certain relevant chemicals to ensure that their industries were not associated with the production of chemical weapons. Australia also launched in 1988 an Asia-Pacific regional initiative to work cooperatively with neighboring countries to prevent chemical proliferation in the Asia-Pacific region. As leader of the Australia Group, Australian officials had also started to engage with government and leading chemical industry representatives in capitals on how best to advance the objective of preventing the spread of chemical weapons, while not impeding the legitimate activities of the civil

chemical industry and protecting their commercial interests. In the absence of a peak body for the chemical industry, Australian diplomats worked with key industry players, such as Hoechst, Bayer and Monsanto, to form a spearhead group to bring other industry players on board.

Strong support from one of the two principal chemical weapons possessors, the United States, provided important political impetus. Then US Secretary of State James Baker and then Australian Foreign Minister Gareth Evans had discussed how to build on the momentum of the Paris Conference on Chemical Weapons in January 1989. What followed was a joint announcement on 7 March 1989 that Australia would host a government-industry conference. Baker made the announcement in Vienna at a meeting of foreign ministers of countries participating in the talks on Conventional Forces in Europe.

Political backing was also important from the members of the Conference on Disarmament (CD) who were negotiating the Chemical Weapons Treaty. They needed to be assured that the government-industry conference was not an attempt to open up a second negotiating forum. In an address to the CD in June 1989, then Australian Foreign Minister Gareth Evans sought to provide such assurances.

By the time GICCW took place in Canberra in September 1989, most of the essential groundwork for a joint approach had been laid. At the conclusion of the conference, chemical industry representatives released a statement, '(1) express[ing] their willingness to work for an early conclusion of a global chemical weapons ban; (2) oppos[ing] misuse of industrial products for the dangerous proliferation of CW; (3) commit[ting] industry to continue its dialogue with governments on ways to implement a CW convention; and (4) accept[ing] a self-policing role.'

The world's chemical industry certainly understood (eventually) the advantage of demonstrating to shareholders and to the public its commitment to chemical disarmament and nonproliferation, especially in light of its inadvertent contribution to Iraq's chemical weapons program. The industry knew that if it was going to be regulated intensively and obtrusively, there were distinct advantages to industry being an active collaborator in ensuring that their business did not contribute to chemical weapons proliferation, while at the same time having a direct say in how commercial confidentiality could be preserved through the Chemical Weapons Treaty. The 1989 Government-Industry Conference Against Chemical Weapons provided a useful vehicle to publicly set the basis for a successful government-industry partnership for this purpose.

Through the conference and by participating as advisers to the negotiations at the CD in Geneva, industry developed confidence in the process which helped developed a level playing field with an equal impact on all companies, while ensuring that commercial and technological confidentiality was maintained.

The situation of the chemical industry differs from that of the nuclear industry as the Nuclear Nonproliferation Treaty (NPT) is already in existence, political support for complete nuclear disarmament is not as strong as it was for chemical weapons other

than the aspiration to general and complete disarmament in the NPT and inspections of nuclear facilities are already in place, which was not the case for chemical weapons.

GICCW took place in the context of the emerging global consensus that chemical weapons should be abolished altogether. However, without the chemical industry's active support and collaboration, that treaty could not have come into existence.

8.1.2 THE BIOLOGICAL WEAPONS CONVENTION

Efforts to engage industry in the negotiation of a Verification Protocol to the Biological Weapons Convention were not successful. There was neither the political nor the business support among the key players for this. Diplomats who participated in these negotiations report that it was widely perceived that the reason the US withdrew from the Protocol negotiations in 2001, causing them to come to a halt, was because of pressure from Pharma, their peak pharmaceutical/biotechnology body. Similar views were voiced by European pharmaceutical enterprises. Their main concern was a perception that the CWC regime was too intrusive and thus highly threatening to commercial confidentiality. There is also strong scepticism about the verifiability of BW proliferation.

8.1.3 MOVE TOWARDS A GOVERNMENT-INDUSTRY CONFERENCE AGAINST NUCLEAR PROLIFERATION

The global, integrated nature of the nuclear business, it's very close connection to government and a changing nuclear policy landscape, including the renewed push towards progress in nuclear disarmament, argue strongly in favor of more regular government-industry collaboration, including through joint monitoring, reporting, and enforcement of the rules and export controls. A jointly negotiated declaration as to how that would be done would add a new dimension to the global nuclear conversation.

As for the prospect of a conference modelled on the GICCW, a similar intensive diplomatic effort would be required in preparation for any nuclear industry-government conference, and the effort would require an agent with strong government backing. In a similar fashion to the GICCW, a ginger group comprising some of the key companies outlined earlier in this section should be formed, and a conference should include as many industry players as possible.

8.2 POSSIBLE OUTCOME OF A GOVERNMENT-INDUSTRY CONFERENCE

Outcomes of the conference could include a declaration of the type resulting from GICCW, approval of a Code of Conduct or code of practice, or scheduling of regular government-industry consultation meetings on the margins of NSG meetings, and/or in parallel with the NPT Preparatory Conferences or Review Conferences, industry body conferences or IAEA meetings.

New, groundbreaking announcements are also possible, given that supplier governments have under active consideration the development of new rules of the game which may have real impact on the development of the industry. These include proposals to multi-lateralize the nuclear fuel cycle; to limit the spread of sensitive nuclear technologies; to change NSG rules to insist that countries not exercise the right to

develop sensitive technology as a condition of supply, as well as making the adoption of the Additional Protocol a mandatory condition of supply.

This might be overly ambitious, given how tightly commercial interests are woven into national interests, especially when it comes to the right to develop sensitive nuclear technologies such as enrichment. It must be remembered that the controversial 'two-tier' system enshrined in the NPT between nuclear weapon states and non-nuclear weapon states could also spill over into the peaceful uses domain. Initiatives to limit the possession and use of sensitive nuclear technologies to those who already have them now-albeit for good nonproliferation reasons-is opposed by emerging nuclear industry powers who will not accept the perpetuation of a two-tier system in the nuclear power industry. There is also virtually no chance states will even consider foregoing the right to develop sensitive aspects of the nuclear fuel cycle, or see them centralized or regionalized under multinational control, in the absence of a solid commitment from the nuclear armed states to achieving a world without nuclear weapons.

In this context, a global call for disarmament might also become the business of industry. It may be worth exploring whether industry is prepared to make a public commitment to the goals of both disarmament and nonproliferation as a sign of good faith, in the interests of the future bona fides of the business and as a contribution to dismantling the two-tier system.

Being politically more proactive does not mean that industry has to abandon its evidence-based approach to risk. The international community will need to be confident that growth in nuclear energy will be managed responsibly. Being proactive can help industry in its ambition to 'strengthen and sustain public confidence, both in the reliability of nuclear technology and in the people and institutions responsible for its use.'

With this in mind, there may be a case for involving global stakeholders from civil society in the global conversation, something a government-industry conference might include in its final declaration.

8.2.1 PARTICIPANTS, LOCATIONS AND TIMING OF CONFERENCE

Logistics, timing and location of such a conference are details that can be elaborated on in the event the ICNND supports the idea. The chemical industry is much larger than the nuclear industry, so it should not be difficult to put together a representative group for the nuclear industry. From government, a good starting point would be members of the Board of Governors of the IAEA and all states with plans to establish nuclear power in the foreseeable future. The conference could be hosted by one of the co-chairs' countries (Australia or Japan); or in a host nation with a major interest in the future development of nuclear energy or in an established international location such as Vienna. Whichever country hosts it would need to work closely with a supportive industry body. As to timing, given the organizational challenges and the need to canvass widely industry and government views before holding a conference, it might be best to hold the meeting after the May 2010 NPT Review Conference.

8.2.2 PROSPECTS

The changing nuclear landscape and the integrated nature of the world's nuclear industry strengthen the case for a concerted effort by industry and government to develop jointly a new set of understandings of future nuclear proliferation dangers, and to work closely together in the design and implementation of measures to prevent such proliferation.

Initial signs are that some industry players see opportunities and advantages to becoming more engaged in the global nonproliferation agenda. An increasingly globally integrated industry needs to take a global view and be more globally engaged. The CEO of AREVA has agreed to become a member of the ICNND's Advisory Board. Members of industry are now active participants in second-track discussions about the future role of nuclear industry in a growing nuclear power market. The industry is represented by the WNA and could be engaged as an active partner. The 2008 WNA policy documents, and its Charter of Ethics and Principles of Uranium Stewardship spell out industry responsibilities to ensure the 3S (safeguards, safety and security) are indispensable for peaceful uses of nuclear energy.

The engagement of industry as a whole will require intense diplomatic effort and will have to be managed adroitly. Large commercial and national security interests are at stake and if there are to be additional standards, they will need to be universally applied.

More information is needed about industry's views on these matters, and, given the very close relationship between much of the world's established nuclear industry and government, government views are also critical.

As a first step, a smaller industry group could be engaged to conduct the initial consultations with industry in collaboration with a supportive government. The Australian Uranium Association might be a candidate for such a role, given its strong public support for the principles of uranium stewardship.

SECTION NINE: RECOMMENDATIONS

The ICNND agree to the following steps to be reported on ahead of its June Moscow meeting:

9.1 Commission an industry-wide survey to gauge industry attitudes to nonproliferation threats and industry's future role (a draft list of elements for a survey is at Annex B). This could also include industry trade associations and professional associations of nuclear industry employees which might be keen to encourage industry to increase its commitment to nonproliferation and disarmament.

Invite one of the designated research centers to conduct a survey on its behalf

9.2 Commission further research into the need for an additional industry Code of Conduct or other effective arrangements, based on an assessment of current codes and activities in the nuclear domain.

Invite one of the designated research centers to conduct this research.

9.3 Meet with a selection of industry representatives in Moscow in June 2009 to gauge views on codes of conduct and a government-industry conference in 2010. Discuss other steps for government-industry partnership in managing the 'second nuclear age' with minimal proliferation risks.

9.4 Designate a national industry association and an interested government to act as a ginger group to canvass support for a government-industry conference and to design an agenda for that conference, using the 1989 Government-Industry Conference Against Chemical Weapons as a model.

APPENDIX
INTRODUCTION TO
NUCLEAR POWER AND NTPBMR TECHNOLOGY

SECTION F: History of Gas-Cooled Reactors

ASCENTRUS
The POWER of ENGINEERING

INTRODUCTION TO THE
NUCLEAR TECHNOLOGY PEBBLE BED MODULAR REACTOR
FOR
STAKEHOLDERS AND CONSULTANTS

ASCENTRUS
The POWER of ENGINEERING

TABLE OF CONTENTS

SECTION ZERO: PREAMBLE

This document started off its life as a white paper which was part of a large group of introductory papers prepared by the Senior Engineer of the **Nuclear Technology Pebble Bed Modular Reactor (NTPBMR) Project.** This document will attempt to present an integrated view on the safety characteristic of the **NTPBMR** Nuclear Power Plant and make the case for the inclusion of the **NTPBMR** in the energy mix for the Twenty-first century and beyond. This document also includes a fairly detailed history of gas-cooled reactors.

This document is a compilation of a series of introductory essay undertaken by the Senior Engineer covering many aspects of the production of electrical power with the use of Nuclear Power. The objective of these white papers was to provide a knowledge basis consisting of factual, up-to-date information on aspects of nuclear power that are needed in order to make the case for the safety of the existing Nuclear Power production fleet in the U.S. and overseas. The method used was a systematic and selective collection, condensation, and presentation of existing information. This document was written to provide background information to interested third parties and our strategic partners in the Nuclear industry who are not familiar with our technology. This document was undertaken to familiarize the staff of our stakeholders with the historical background on the Gas-Cooled Reactors.

Section One of this document introduces us to the **Nuclear Technology Pebble Bed Modular Reactor** and outlines its safety characteristics.

Section two outlines the history of the use of Nuclear Energy in the United States with particular emphasis on the regulatory aspects which formed the basis of the existing licensing procedure.

Section Three sets out the case for the use of Nuclear Energy in the mitigation of the **CO_2** global warming problem. There is a broad international consensus within the reactor-safety community concerning the key elements that are necessary in the design and operation of a nuclear power reactor to achieve a very high level of safety. Section three then discusses the key elements required and how the pebble bed nuclear reactor provides the technology with the best resume.

Section Four provides a historical review of gas-cooled reactors. This includes the history of the following gas-cooled technologies:

- Magnox series of Reactors
- Uranium Natural Graphite Gaz
- The Advance Gas Reactor
- High-temperature Gas-cooled Reactor
- Peach Bottom
- Fort St. Vrain

- Dragon
- Pebble Bed Modular Reactors

SECTION ONE: INTRODUCTION

The global demand for electrical energy is increasing at an exponential rate and the major portion of this increase in demand is being met through the combustion of fossil fuels, mainly coal and natural gas. Both of these fuels emit carbon dioxide (**CO_2**) during the combustion process and while these plants are cheaper to construct the environmental impact, due to the emissions of carbon dioxide, a potent green-house gas, is having a deleterious effect on our atmosphere. Increased public awareness of the climate changes induced by the use of fossil fuels for the production of electrical power have focused fresh attention on nuclear power as a viable alternative for the production of electrical power for the twenty first century and beyond.

No new nuclear power plants have been licensed, in the U.S., since 1978 and the construction of the last plant to be licensed in the U.S. was completed in late 1995. This shortfall in the construction of new production facilities generating power by the use of nuclear energy is due primarily to the negative press and a change in the regulatory climate created by two **Loss of Coolant Accidents (LOCA**). The first significant **LOCA** Nuclear event occurred at Three Mile Island, in the Northern seaboard of the United States and the other occurred at Chernobyl, a City in the Ukraine which was then in the Soviet Bloc.

These two nuclear accidents provided the initial impetus for the **Nuclear Regulatory Commission (NRC)** to change its view on the inherent safety of the Nuclear power production program and to adopt a more adversarial posture with regards to the standards and enforcement of regulations pertaining to the licensing, construction and operations of Nuclear Power facilities.

The use of nuclear power in the production of electricity has a direct benefit to the environment in terms of air quality. Nuclear power presently accounts for approximately 20% of the electrical energy produced in the United States, and approximately 65% of the non-carbon emitting production, Nuclear power annually avoids the emission of 175 million tons of carbon dioxide into the atmosphere. This decrease in the emission of carbon dioxide, a so called greenhouse gas, by the Nuclear Power Industry has even attracted global interest in funding for our project through the use of European and Canadian carbon credits.

The Nuclear Technology Pebble Bed Modular Reactor (**NTPBMR**) offers a permanent solution for the licensing of the existing nuclear power plants. The **NTPBMR** is the only nuclear technology which can be prototyped and licensed to be used to replace the production capacity of an existing nuclear power plant without increasing the diameter of its evacuation zone. They are gas-cooled, small, modular, inherently safe and use a demonstrated nuclear technology. Each individual **NTPBMR** reactor modules will be engineered and licensed as a process with all the major systems and sub-

systems of the power plant fabricated in an off-site manufacturing facility. In addition each process involving a system or a sub-system will be manufactured under a set of code standards registered with the **NRC,** International Standards Organization (**ISO)**, International Atomic Energy Agency (**IAEA**) and other code compliance organizations such as American Society for Testing and Materials (**ASTM**).

The important characteristics of **NTPBMR** are:

- A Nuclear reactor using gas as the core coolant will eliminate completely the types of problem which occurred at Three Mile Island and Chernobyl, in their water-cooled nuclear reactor.
- Advances in gas turbine technologies will allow us to use helium as the coolant. Helium is an ideal cooling agent for a nuclear reactor since it is completely inert chemically, within the temperature ranges involved in a nuclear reactor vessel it remains in a single phase and its neutron absorption cross-sections are quite low.
- The inert nature of Helium will allow the filtration system of the **Nuclear Helium Gas Supply System (NHGSS)** to extract nearly 100% or radioactive fission products from the coolant. The **NHGSS** with filtration will reduce the radioactivity level in the turbine room by three orders of magnitude over existing water-cooled reactors.
- The low radioactivity level in the turbine will ensure that an insignificant amount of radiation will be added to the cooling $\mathbf{CO_2}$ which will return to our thermal heat sink. The thermal heat sink will be the atmosphere. The increase in radiation levels to the atmosphere will not be measurable over the background.
- The fuel element is a completely ceramic pebble containing low enriched Uranium Oxide ($\mathbf{UO_2}$) as fuel.
- The reactor core contains approximately 360,000 uranium fueled pebbles about the size of tennis balls. Each pebble contains about 9 grams of low enriched Uranium Oxide ($\mathbf{UO_2}$) in 10,000 to 15,000 (depending on the design) tiny grains of sand-like micro-sphere coated particles each with its own a hard silicon carbide shell.
- The particle fuel consists of a spherical kernel of fissile or fertile fuel material encapsulated in multiple coating layers. The multiple coating layers form a miniature, highly corrosion resistant pressure vessel and an essentially impermeable barrier to release of gaseous and metallic fission products. This capability has been demonstrated at temperatures in excess of those that are predicted to be achieved under worst-case accident conditions in the **NTPBMR.**
- The micro-spheres are tri-coated with pyrolytic carbon and silicon carbide. The pyrolytic carbon layer absorbs the fission fragments and the Silicon Carbide coating retains these fission fragments and radioactive gasses within the micro-sphere. These micro-spheres are embedded in a graphite matrix material.
- The Uranium Oxide ($\mathbf{UO_2}$) fuel has a melting temperature of approximately 2800^{o}C

while the ceramic coating does not have a melting point and begins to degrade approximately at 2100°C, and the degradation of the ceramic shell in the 50 or so hours required to empty the reactor would require temperatures in excess of 4000°C.

- Another unique feature of pebble bed reactors is the online refueling capability in which the pebbles are re-circulated with checks on integrity and consumption of uranium. This system allows new fuel to be inserted during operation and used or damaged fuel to be discharged and stored on site for the life of the plant.
- The online refueling capability allows for the extraction of all the nuclear fuel in the event of a **LOCA.** Extraction of all the nuclear elements in the core will mitigate the melting of the fuel pebbles.
- The online refueling capability allows for the replacement of the fuel elements with pure graphite balls, in the event of an extraction of the fuel elements due to a **LOCA.** Insertion of these graphite balls into the core of the reactor will completely eliminate the possibility of core damage due to a thermal excursions following a loss of coolant.
- The moderating environment of the **NTPBMR** is nuclear graphite. The **Nuclear Reactor Vessel (NRV)** will house several hundred tons of Nuclear Graphite. The Nuclear graphite has high thermal mass and will allow for passive cooling of the reactor core in the loss of coolant event.
- The **Emergency Core Fire Suppression System (ECFSS)** is liquefied carbon dioxide. The carbon dioxide fire suppression system will mitigate the risk of a graphite fire of the type which occurred at Windscale, in England, in the early days of the English gas-cooled Magnox program. The carbon dioxide will also act as a passive emergency core cooling system to extract heat from the core.
- The comparatively small size and the lack of complexity in the design of a pebble-bed reactor adds to their economic feasibility. Each power module will produce approximately 110 megawatts (electric).
- The simplicity of design of our power plant is dramatic. These units will have only two dozen major plant subsystems which we believe can all be plant manufactured, licensed separately and moved to the proposed nuclear site. .

The **NTPBMR** modules are designed to produce 110MWe each. To place this in context a 100Mwe generator would produce the electricity consumed by 30,000 average homes. A single **NTPBMR** module would consist typically of a single main building, covering an area of approximately 13,000 square feet(130 x 100 feet).. The height of the building would be approximately 120 feet, the majority of the structure will be below ground level. The part of the building that would be visible above ground is equivalent to a four story building.

Next to the containment area there would be a unit control room, a high voltage switch yard, and a cooling tower for inland facilities. More than one **NTPBMR** module can be

located on an existing licensed site. These reactors can be built next to each other and thus creating an energy park. It is possible for an **NTPBMR** energy park to be made up of as many as 10 modules which share a common control center and switching to the power grid. Ten **NTPBMR** modules will have a production capability of 1,100 MWe.

SECTION TWO: INTRODUCTION TO THE NUCLEAR POWER REGULATION

In order to be able to appreciate the market which has been created for the **Nuclear Technology Pebble Bed Modular Reactor (NTPBMR)** in the nuclear mix of the U.S. we must take a historical detour and review the main events which brought us from the beginning of the atomic age to the dawn of the twenty-first century.

2.1. PROLOGUE TO THE ATOMIC AGE.

- Ionizing radiation was discovered by Wilhelm Rontgen in 1895, by passing an electric current through an evacuated glass tube and producing continuous X-rays.
- In 1896 Henri Becquerel found that pitchblende (an ore containing radium and uranium) caused a photographic plate to darken. He went on to demonstrate that this was due to beta radiation (electrons) and alpha particles (helium nuclei) being emitted.
- Villard found a third type of radiation from pitchblende: gamma rays, which were much the same as X-rays.
- Then in 1896 Pierre and Marie Curie gave the name 'radioactivity' to this phenomenon, and in 1898 isolated polonium and radium from the pitchblende.
- Radium was later used in medical treatment. In 1898 Samuel Prescott showed that radiation destroyed bacteria in food.
- In 1902 Ernest Rutherford showed that radioactivity as a spontaneous event emitting an alpha or beta particle from the nucleus created a different element. He went on to develop a fuller understanding of atoms and in 1919 he fired alpha particles from a radium source into nitrogen and found that nuclear rearrangement was occurring, with formation of oxygen.
- Niels Bohr was another scientist who advanced our understanding of the atom and the way electrons were arranged around its nucleus through to the 1940s.
- By 1911 Frederick Soddy discovered that naturally-radioactive elements had a number of different isotopes (radio-nuclides), with the same chemistry.
- Also in 1911, George de Hevesy showed that such radio-nuclides were invaluable as tracers, because minute amounts could readily be detected with simple instruments
- In 1932 James Chadwick discovered the neutron.

- Also in 1932 Cockcroft and Walton produced nuclear transformations by bombarding atoms with accelerated protons, then in 1934 Irene Curie and Frederic Joliot found that some such transformations created artificial radio-nuclides.
- The next year Enrico Fermi found that a much greater variety of artificial radio-nuclides could be formed when neutrons were used instead of protons. Fermi continued his experiments, mostly producing heavier elements from his targets, but also, with uranium, some much lighter ones.
- At the end of 1938 Otto Hahn and Fritz Strassman in Berlin showed that the new lighter elements were barium and others which were about half the mass of uranium, thereby demonstrating that atomic fission had occurred.
- Lise Meitner and her nephew Otto Frisch, working under Niels Bohr, then explained this by suggesting that the neutron was captured by the nucleus, causing severe vibration leading to the nucleus splitting into two not quite equal parts. They calculated the energy release from this fission as about 200 million electron volts.
- Frisch then confirmed this figure experimentally in January 1939. This was the first experimental confirmation of Albert Einstein's paper putting forward the equivalence between mass and energy, which had been published in 1905.

2.2 HARNESSING NUCLEAR FISSION

These 1939 developments sparked activity in many laboratories.

- Hahn and Strassman showed that fission not only released a lot of energy but that it also released additional neutrons which could cause fission in other uranium nuclei and possibly a self-sustaining chain reaction leading to an enormous release of energy. This suggestion was soon confirmed experimentally by Joliot and his co-workers in Paris, and Leo Szilard working with Fermi in New York.
- Bohr soon proposed that fission was much more likely to occur in the uranium-235 isotope than in U-238 and that fission would occur more effectively with slow-moving neutrons than with fast neutrons, the latter point being confirmed by Szilard and Fermi, who proposed using a 'moderator' to slow down the emitted neutrons.
- Bohr and Wheeler extended these ideas into what became the classical analysis of the fission process, and their paper was published only two days before war broke out in 1939.
- Another important factor was that U-235 was then known to comprise only 0.7% of natural uranium, with the other 99.3% being U-238, with similar chemical properties. Hence the separation of the two to obtain pure U-235 would be difficult and would require the use of their very slightly different physical properties. This increase in the proportion of the U-235 isotope became known as 'enrichment'.

- The remaining piece of the fission/atomic bomb concept was provided in 1939 by Francis Perrin who introduced the concept of the critical mass of uranium required to produce a self-sustaining release of energy. His theories were extended by Rudolf Peierls at Birmingham University and the resulting calculations were of considerable importance in the development of the atomic bomb. Perrin's group in Paris continued their studies and demonstrated that a chain reaction could be sustained in a uranium-water mixture (the water being used to slow down the neutrons) provided external neutrons were injected into the system.
- They also demonstrated the idea of introducing neutron-absorbing material to limit the multiplication of neutrons and thus control the nuclear reaction (which is the basis for the operation of a nuclear power station).
- In late 1939, Leó Szilárd, (a Hungarian physicist who conceived the nuclear chain reaction in 1933 and patented the idea of a nuclear reactor with Enrico Fermi), wrote the letter for Albert Einstein's signature that resulted in the Manhattan Project, and the atomic bomb.
- Albert Einstein wrote a letter to President Roosevelt expressing his concerns that Nazi Germany may be trying to develop nuclear weapons.

2.3 THE MANHATTAN PROJECT

The Manhattan Project was the codename for a project conducted during World War II to develop the first atomic bombs. The project was led by the United States, and included participation from the United Kingdom and Canada. Formally designated as the Manhattan Engineer District (MED) (sometimes referred to as the Manhattan District) it refers specifically to the period of the project from 1942–1946 under the control of the U.S. Army Corps of Engineers and General Leslie R. Groves. The scientific research was directed by American physicist J. Robert Oppenheimer.

The Manhattan Project, which began as a small research program that year, eventually employed more than 130,000 people and cost nearly US $2 billion ($22 billion in present day value). It resulted in the creation of several research and production sites whose construction and operations were secret.

Project research took place at more than 30 sites, including universities across the United States, Canada, and the United Kingdom. The three primary research and production sites of the project were:

- the plutonium-production facility at what is now the Hanford Site in eastern Washington state
- the uranium-enrichment facilities at Oak Ridge, Tennessee
- the weapons research and design laboratory now known as Los Alamos National Laboratory

The MED maintained control over U.S. atomic weapons production until the formation of the Atomic Energy Commission in January 1947.

2.4. THE DAWN OF THE ATOMIC AGE

The use of nuclear energy against the Japanese cities of Hiroshima and Nagasaki in August 1945 ushered in the “Atomic Age”. Within a short time after the end of World War II nuclear power was being hailed as the premier energy production method for the twentieth century. Developing nuclear energy for civilian purposes, as even the most enthusiastic proponents recognized, would take many years. The government's first priority was to maintain strict control over atomic technology and to exploit it further for military purposes.

2.4.1 THE ATOMIC ENERGY ACT OF 1946

The **Atomic Energy Act of 1946**, passed as tensions with the Soviet Union were developing into the cold war, acknowledged in passing the potential peaceful benefits of atomic power. But it emphasized the military aspects of nuclear energy and underscored the need for secrecy, raw materials, and production of new weapons. The 1946 law did not allow for private, commercial application of atomic energy; rather, it created a virtual government monopoly of the technology. To manage the nation's atomic energy programs, the act established the five-member **Atomic Energy Commission** (**AEC**).

2.4.2 THE 1954 ATOMIC ENERGY ACT

Congress passed new legislation that for the first time permitted the wide use of atomic energy for peaceful purposes. The 1954 **Atomic Energy Act** redefined the atomic energy program by ending the government monopoly on technical data and making the growth of a private commercial nuclear industry an urgent national goal. The measure directed the **Atomic Energy Commission** (AEC):

> **"To encourage widespread participation in the development and utilization of atomic energy for peaceful purposes."**

At the same time, it instructed the agency to prepare regulations that would protect public health and safety from radiation hazards. Thus, the 1954 act assigned the AEC three major roles:

A. To continue its weapons program,

B. To promote the private use of atomic energy for peaceful applications,

C. To protect public health and safety from the hazards of commercial nuclear power.

In the early 1950s, projections of future energy requirements predicted that atomic power would eventually play an important role in the nation's energy supplies, but they did not suggest an immediate need to construct atomic power reactors.

By 1954, a broad political consensus viewed the development of nuclear energy for civilian purposes as a vital goal. The Atomic Energy Act of that year resulted partly from perceptions of the long-range need for new energy sources, but mostly from the immediate commitment to maintain America's world leadership in nuclear technology,

enhance its international prestige, and demonstrate the benefits of peaceful atomic energy.

2.5. THE AEC AND THE DEVELOPMENT OF NUCLEAR POWER

The **AEC** favored a partnership between government and industry in which private firms would play an integral role in demonstrating and expanding the use of atomic power.

A. The **AEC** was directed toward encouraging development of the uses of atomic energy within the framework of the American free enterprise system.

B. It was the **AEC**'s belief that competitive economic nuclear power would be most quickly achieved by construction and operation of full-scale plants by industry itself.

To accomplish its objectives, the **AEC** announced a "power demonstration reactor program" in January 1955. The agency offered to perform research and development on power reactors in its national laboratories, to subsidize additional research undertaken by industry under fixed-sum contracts, and to waive for seven years the established fuel use charges for the loan of fissionable materials (which the government would continue to own).

For their part, private utilities and vendors would supply the capital for construction of nuclear plants and pay operating expenses other than fuel charges. The purpose of the demonstration program was to stimulate private participation and investment in exploring the technical and economic feasibility of different reactor designs. At that time, no single reactor type had clearly emerged as the most promising of the several that had been proposed.

2.5.1 THE AEC'S REGULATORY PROGRAM

The **AEC**'s determination to push nuclear development through a partnership in which private industry played a vital role had a major impact on the agency's regulatory policies. The **AEC'**s fundamental objective in drafting regulations was to ensure that public health and safety were protected without imposing overly burdensome requirements that would impede industrial growth.

Other proponents of nuclear development shared those views. They realized that safety was indispensable to progress; an accident could destroy the fledgling industry or at least set it back many years. At the same time, they worried that regulations that were too restrictive or inflexible would discourage private participation and investment in nuclear technology.

2.5.2 THE LICENSING PROCESS

The **AEC**'s regulatory staff, created soon after the passage of the **1954 Atomic Energy Act,** confronted the task of writing regulations and devising licensing procedures rigorous enough to assure safety but flexible enough to allow for new findings and rapid changes in atomic technology.

Within a short time the staff drafted rules and definitions on radiation protection standards, distribution and safeguarding of fissionable materials, and reactor operators' qualifications. It also established procedures for licensing privately-owned reactors.

2.6 THE PRICE ANDERSON ACT

The AEC regarded indemnity legislation as essential for stimulating private investment in nuclear power, a view that industry spokesmen and the **Joint Committee on Atomic Energy** shared. Since they recognized that the chances of a severe reactor accident could not be reduced to zero, even the most enthusiastic industry proponents of atomic power were reluctant to push ahead without adequate liability insurance.

Private insurance companies would offer up to $60 million of coverage per reactor, an amount that far exceeded what was available to any other industry in the United States. But in the event of a serious accident, it seemed insufficient to pay claims for deaths, injuries, and property damages in areas surrounding the malfunctioning plant. Therefore, industry executives sought a government program to provide additional insurance protection.

2.7 THE GROWTH OF NUCLEAR POWER

The 1954 Atomic Energy Act made it a national goal to encourage the widespread use of atomic energy for peaceful purposes, but private industry was often hesitant to assume the costs and risks of development.

The AEC's emphasis on stimulating atomic development did not mean that it was inattentive to safety issues. The AEC recognized and publicly acknowledged the possibility of accidents in such a new and rapidly changing technology; it never offered absolute assurances that accidents would not occur. Nevertheless, it believed that compliance with its regulations would make the chances of a serious accident very small.

2.8 THE GREAT BANDWAGON MARKET

The bandwagon market was an outgrowth of several developments that enhanced the appeal of nuclear power to utilities in the mid- and late 1960s. One was the intense competition between the two leading vendors of nuclear plants, General Electric and Westinghouse.

In 1963, General Electric made a daring move to increase its reactor sales and to convince utilities that nuclear power had arrived as a safe, reliable, and cost-competitive alternative to fossil fuel. It offered a "turnkey" contract to **Jersey Central Power and Light Company** to build the 515 electrical megawatt Oyster Creek plant near Toms River, New Jersey. For a fixed cost of $66 million, General Electric agreed to supply the entire plant to the utility (the term "turnkey" suggested that the utility would merely have to turn a key to start operating the facility).

Westinghouse followed General Electric's lead in offering turnkey contracts for nuclear plants, setting off a fierce corporate battle. The turnkey plants were a financial blow for both companies.

2.9 NUCLEAR POWER AND THE ENVIRONMENT

The problem of industrial pollution and the deteriorating quality of the natural environment took on growing urgency as a public policy issue during the 1960s. Utilities increasingly viewed nuclear power as the answer to that dilemma. It promised the means to meet demand for power without causing air pollution, and environmental concerns were a major spur to the growth of the great bandwagon market.

Environmentalists recognized the benefits of nuclear power compared to fossil fuel, but they were more equivocal in their attitudes toward the technology than were industry representatives. Their ambivalence was perhaps best summarized by the statement of a leading environmental spokesman in 1967: "I think most conservationists may welcome the coming of nuclear plants, though we are sure they have their own parameters of difficulty."

2.10 NEPA AND CALVERT CLIFFS

In addition to the objections that its positions on thermal pollution and radiation standards stirred, the AEC provoked sharp criticism for its response to the National Environmental Policy Act (NEPA). The law, passed by Congress in December 1969 and signed by President Nixon on January 1, 1970, required federal agencies to consider the environmental impact of their activities. The AEC acted promptly to comply with NEPA, but its procedures for doing so brought protests from environmentalists. The agency took a narrow view of its responsibilities under NEPA.

The Calvert Cliffs decision was another in a series of setbacks for nuclear power. It was apparent by the summer of 1971 that public distrust of the AEC was growing and support for nuclear power was declining. The cumulative effect of controversies over ECCS, thermal pollution, radiation standards, NEPA, and other issues eroded public confidence in the AEC's commitment to safety and raised doubts about the benefits of nuclear power.

By the summer of 1971, the AEC was an embattled agency, largely though not exclusively because of regulatory issues. Seaborg, after serving as chairman for ten years, resigned his post in July 1971 and Nixon appointed James R. Schlesinger, assistant director of the **Office of Management and Budget**, to take his place.

2.11 THE END OF THE AEC

One of President Nixon's responses to the energy crisis brought on by the Oil Embargo, was to ask Congress to create a new agency that could focus on, and presumably speed up, the licensing of nuclear plants. After much debate, Congress divided the AEC into the **Energy Research and Development Administration** and the **Nuclear Regulatory Commission** in legislation it passed in 1974. **The Energy Reorganization Act,** coupled with the **1954 Atomic Energy Act**, constituted the statutory basis for the NRC.

2.12 THE MANDATE OF THE NRC

The Nuclear Regulatory Commission began its operations as a separate agency in January 1975. In many ways, it carried on the legacy inherited from the AEC. It performed the same licensing and rulemaking functions that the regulatory staff had discharged for two decades. It also assumed some new administrative and regulatory duties.

The NRC, unlike the AEC's regulatory staff, was the final arbiter of regulatory issues; its judgment on safety questions was less susceptible to being overridden by developmental priorities. This did not mean that the NRC acted without regard to industry concerns or that its officials always agreed on policy matters, but it did mean that the agency's statutory mandate was clearly focused on ensuring the safety of nuclear power.

2.13 THE REACTOR SAFETY STUDY

The Rasmussen report, while hailed as a pioneering effort that enlightened a complex subject, also drew criticism from both inside and outside the NRC. Some authorities suggested that the study failed to account for the many paths that could lead to major accidents. Others complained that the data in the report did not support its executive summary's conclusions about the relative risks of nuclear power. After considering the arguments on both sides of the issue, the Commission in January 1979 issued a policy statement that withdrew its full endorsement of the study's executive summary.

2.14 THE MOVIE “The China Syndrome”

The China Syndrome is a 1979 American thriller film that tells the story of a reporter and cameraman who discover safety coverups at a nuclear power plant. It stars Jane Fonda, Jack Lemmon, Michael Douglas, Scott Brady, James Hampton, Peter Donat, Richard Herd, and Wilford Brimley. The movie was written by Mike Gray, T.S. Cook and James Bridges. It was directed by Bridges.

The title refers to the concept that if an American nuclear plant melts down, the core will melt through the Earth until it reaches China. China is a metaphor, as the opposite side of the globe from the USA is actually the Indian Ocean.

The film was released on March 16, 1979, just twelve days before the real-life events at Three Mile Island, Pennsylvania. The Three Mile Island accident helped propel *The China Syndrome* into a blockbuster.

2.15 THREE MILE ISLAND

Within a short time, discussion of severe nuclear accidents ceased to be strictly a matter of theoretical projections. On March 28, 1979, an accident at Unit 2 of the Three Mile Island nuclear station near Harrisburg, Pennsylvania made the issue starkly and alarmingly real. As a result of a series of mechanical failures and human errors, the accident (researchers later determined) uncovered the reactor's core and melted about half of it.

The immediate cause of the accident was a pressure relief valve that stuck open and allowed large volumes of reactor coolant to escape. The reactor operators misread the

signs of a **loss-of-coolant accident** and, for several hours, failed to take action to cool the core. Although the plant's emergency cooling systems began to work according to design, the operating crew decided to reduce the flow from them to a trickle. By the time that the nature of the accident was recognized and the core was flooded with coolant, the reactor had suffered irreparable damage.

The credibility of the nuclear industry and the NRC fared almost as badly. Uncertainty about the causes of the problem, confusion about how to deal with it, conflicting information from government and industry experts, and contradictory appraisals about the level of danger in the days following the accident often made the authorities appear inept, deceptive, or both.

In some ways, the TMI accident produced reassuring, or at least encouraging, information for reactor experts about the design and operation of the safety systems in a large nuclear plant. Despite the substantial degree of core melting that occurred, containment was not breached. From all indications, the amount of radioactivity released into the environment as a result of the accident was very low. One estimate suggested that of 66 million curies of iodine-131 in the reactor at the time of the accident, only 14 or 15 curies escaped.

The NRC responded to TMI by re-examining the adequacy of its safety requirements and imposing new regulations to correct deficiencies. It placed much greater emphasis on "human factors" in plant performance in an effort to avoid a repeat of the operator errors that had exacerbated the accident. The agency developed new requirements for operator training, testing and licensing, and for shift scheduling and overtime.

In cooperation with industry groups, it promoted the increased use of reactor simulators and the careful assessment of control rooms and instrumentation. In addition, the agency expanded its resident inspector program to station at least two of its inspectors at each plant site.

The NRC devoted greater attention to other problems that had received limited consideration before TMI. They included the possible effects of small failures that could lead to major consequences, such as happened at Three Mile Island. The agency sponsored a series of studies on the ways in which "small breaks and transients" could threaten plant safety.

A second area on which the NRC focused was the evaluation of operational data from licensees. It established a new **Office for Analysis and Evaluation of Operational Data** to systematically review information from and the performance of operating plants. This action reflected the belated recognition that malfunctions similar to those at TMI had occurred at other plants, but the information had never been assimilated or disseminated.

The NRC undertook other initiatives as a result of TMI. It decided to survey radiation protection procedures at operating plants in order to assess their adequacy and to look for ways to improve existing regulations. It expanded research programs on problems

that TMI had highlighted, including fuel damage, fission-product release, and hydrogen generation and control.

In light of the confusion and uncertainty over evacuation of the areas surrounding TMI during the accident, the NRC also sought to upgrade emergency preparedness and planning. Those and other steps it took in the wake of the accident were intended to reduce the likelihood of a major accident, and, in the event one occurred, to enhance the ability of the NRC, the utility, and the public to cope with it.

While the NRC was still deliberating over and revising its requirements in the aftermath of TMI, another event shook the industry and further undercut public support for nuclear power. This time, the NRC was a distant though interested observer rather than a direct participant.

2.16 CHERNOBYL

On April 26, 1986, unit 4 of the nuclear power station at Chernobyl in the Ukraine, a satellite of the then USSR underwent a violent explosion that destroyed the reactor and blew the top off it. The explosion and subsequent fire in the graphite core spewed massive amounts of radioactivity into the environment.

The accident occurred during a test in which operators had turned off the plant's safety systems and then lost control of the reactivity in the reactor. Without emergency cooling or a containment building to stop or at least slow the escape of radiation, the areas around the plant quickly became seriously contaminated and a radioactive plume spread far into other parts of the Soviet Union and Europe. Although the radiation did not pose a threat to the United States, one measure of its intensity in the Soviet Union was that levels of iodine-131 around the Chernobyl reactor were three times as high after the incident than they were after the TMI accident.

The design of the Chernobyl reactor was entirely different than that of U.S. plants, and the series of operator blunders that led to the accident defied belief. Supporters of nuclear power emphasized that a Chernobyl-type accident could not occur in commercial plants in the United States (or other nations) and that American reactors featured safety systems and containment to prevent the release of radioactivity. But nuclear critics pointed to Chernobyl as the prime example of the hazards of nuclear power. A representative of the Union of Concerned Scientists remarked: "The accident at Chernobyl makes it clear. Nuclear power is inherently dangerous."

A popular slogan that quickly appeared on the placards of European environmentalists was: CHERNOBYL IS EVERYWHERE. The Chernobyl tragedy was a major setback to the hopes of nuclear proponents to win public support for the technology and to spur orders for new reactors. U.S. utilities had not ordered any new plants since 1978 and the number of cancellations of planned units was growing. "We're in trouble," conceded a spokesman for the Atomic Industrial Forum. "If the calls I have received from people in the industry are a good indication, they are all very worried."

2.17 EFFECTS ON LICENSING FROM CHERNOBYL

The Chernobyl accident added a new source of concern to long-standing controversies over the licensing of several reactors in the United States. In the aftermath of Three Mile Island, the NRC had suspended the granting of operating licenses for plants that were in the pipeline.

The "licensing pause" for fuel loading and low-power testing ended in February 1980. In August 1980 the NRC issued the first full-power operating license (to North Anna-2 in Virginia) since TMI. In the following nine years it granted full-power licenses to over forty other reactors, most of which had received construction permits in the mid-1970s.

2.18 EMERGENCY PLANNING

The Three Mile Island accident had vividly demonstrated the deficiencies in existing procedures for coping with an off-site nuclear emergency. The lack of effective preparation had produced confusion, uncertainty, and panic among members of the public faced with the prospect of exposure to radiation releases from the plant. After the accident, the NRC, prodded by Congress to improve emergency planning, adopted a rule that required each nuclear utility to come up with a plan for evacuating the population within a ten mile radius of its plant(s) in the event of a reactor accident. The rule applied to plants in operation and under construction. It called for plant owners to work with state and local police, fire, and civil defense authorities to put together an emergency plan that would be tested and evaluated by the NRC and the **Federal Emergency Management Agency** (FEMA)

2.19 ONE-STEP LICENSING

The lengthy and laborious licensing procedures that applicants had to undergo in the cases of Shoreham (which had received a construction permit in 1973), Seabrook (which had received a construction permit in 1977), and other reactors stirred new interest in simplifying and streamlining the regulatory process. It seemed apparent that the complexity of the licensing process was a major deterrent to utilities who might consider building nuclear plants.

By the late 1980s, the nuclear option looked more appealing to some observers, including some environmentalists, because of growing concern about the consequences of burning fossil fuel, especially acid rain and global warming. Furthermore, nuclear vendors were advancing new designs for plants that greatly reduced the chances of TMI-type and other severe accidents.

One way that the NRC proposed to facilitate licensing procedures was to replace the traditional two-step process with a one-step system. This would ease the burden on applicants, but it raised a vitally important question: what level of detail would the NRC require in applications for advanced plants in order to satisfy its concerns about their safety? The agency had never required the detailed technical information in construction permit proposals that it expected in operating license applications, but in a one-step licensing process it was unclear how much data would be needed to evaluate and certify safety designs.

After long discussions that reflected differing views among commissioners, staff, and nuclear vendors, the NRC reached a decision on what constituted an "essentially complete design." It established a "graded approach" in which the level of detail that an applicant would be required to submit varied according to the system's, structure's, or component's relationship to plant safety. The objective of the NRC's action was to ensure safety while providing flexibility for the development of new designs.

2.20 LICENSE RENEWAL

As decommissioning issues were debated, the NRC devoted considerable attention and resources to the question of license renewal. While some utilities were closing reactors long before their 40-year operating licenses expired, others were weighing the possibility of extending the lives of plants beyond 40 years.

The 40-year licensing period for nuclear plants was a rather arbitrary compromise written into the **1954 Atomic Energy Act** that was not based on technical grounds or operating experience. In the late 1970s, industry groups closely examined the issue of plant life extension for the first time. The Electric Power Research Institute, for example, concluded that reconditioning of old plants offered potentially major benefits, but it cautioned that the benefits depended on financial considerations as well as on technical assessments, environmental issues, and projections of power availability. Those uncertainties were compounded by industry's concern that the NRC was not prepared to address the issues surrounding license renewal promptly and knowledgeably.

In 1985, the NRC, prodded by Chairman Nunzio J. Palladino, undertook a careful analysis of license renewal. The agency had sponsored research on the critical question of the safety effects of plant aging for years, but many technical questions remained to be answered. License renewal also raised complex legal and policy issues. The NRC staff cited the "central regulatory question" that plant life extension presented: "What is an adequate licensing basis for renewing the operating license of a nuclear power plant?"

The NRC deliberated over this issue and its corollaries for several years. Eventually, it decided that the maximum length of an extended license would be 20 years. It also concluded that using the existing regulatory requirements governing a plant would offer reasonable assurance of adequate protection if its license were renewed, provided that the "current licensing basis" was modified to account for age-related safety issues.

In 1991, the Commission approved a regulation on the technical requirements for license renewal. After considering ways to evaluate the environmental consequences of license renewal, the NRC elected to develop a generic environmental impact statement that covered effects that were common to all or most nuclear plants. In April 1998, Baltimore Gas and Electric became the first utility to apply for license renewal for its Calvert Cliffs plants on the Chesapeake Bay. Duke Energy Corporation followed suit in July 1998 when it sought license extensions for its Oconee nuclear units in South Carolina.

SECTION THREE; MAKING THE CASE FOR THE USE NUCLEAR POWER

Over the next 50 years, unless patterns change dramatically, energy production and use will contribute to global warming through large scale greenhouse gas emissions, mainly carbon dioxide (**CO_2**) from the combustion of fossil fuels in the production of electrical energy. Nuclear power is the only option which provides a credible path to the de-coupling of energy production from the combustion of fossil fuels. This section of the document analyzes what will be required to retain nuclear power as a significant option for reducing greenhouse gas emissions and meeting growing needs for electricity supply. Our analysis is guided by a global growth scenario that would expand current worldwide nuclear generating capacity almost threefold, to 1000 billion watts, by the year 2050. Such a deployment would avoid 1.8 billion tons of carbon emissions annually from coal plants, about 25% of the increment in carbon emissions otherwise expected in a business-as-usual scenario.

For a large expansion of nuclear power to succeed, four critical problems must be overcome: Safety, waste, proliferation and cost. The **NTPBMR Project** addresses each of these concerns in depth.

3.1 SAFETY

Modern reactor designs can achieve a very low risk of serious accidents, but “best practices” in construction and operation are essential. The most pressing safety issues which we are going to be concerned with in this presentation is the issue of plant safety with respect to a **Loss of Coolant Accident (LOCA).**

There is a broad international consensus within the reactor-safety community concerning the key elements that are necessary in the design and operation of a nuclear power reactor to achieve a very high level of safety. While the presence of these key elements generally should provide a high level of safety, the absence of one or more of them is always a cause for concern.

In this document the phrase “a high level of safety” means a very low probability of an accident that might cause death or injury to offsite populations due to radioactivity, or might cause important contamination of offsite land and property. It also implies that the risk to onsite workers and the risk of damage to the facility itself are of acceptably low probability, because the elements needed to achieve these low risks are compatible with the elements needed to protect offsite populations and property.

Before discussing the advantages to the **NTPBMR** Technology and describing the major elements used in the **NTPBMR** to accomplish major increases in reactor safety, it is important to describe in broad terms the safety-engineering challenge. Simply stated, for a reactor to be acceptably safe it is necessary to assure under all potential transient conditions. These safety functions will be elaborated in detail under the Section for **Defense in Depth.**

A. That the nuclear chain reaction can be shut down and maintained in a shutdown condition (known as the “reactivity control” function)

B. that the thermal energy (heat) in the reactor, both heat present at the onset of the transient and heat generated by the continuing radioactive decay processes in the core, does not cause melting of the Nuclear fuel elements.

C. that the thermal energy (heat) in the reactor, both heat present at the onset of the transient and heat generated by the continuing radioactive decay processes in the core, is removed to a safe ultimate heat sink.

If all three of these safety items can be accomplished in a **LOCA** accident, the radioactivity within the reactor can be contained. While other crucial functions, such as the containment function and the emergency-protective-action function, need to be accomplished as back-ups in case these vital functions fail, the most important aspects of preventing harm from the radioactivity are the functions of reactivity control and heat removal.

Different reactor designs accomplish these vital safety functions in different ways. It is broadly accepted in the Reactor Design and Nuclear Communities that a design is safer, to the extent that each of these functions is accomplished by relying more on physical principles and passive features and less on active equipment and human intervention. This does not mean that a reactor design relying mainly on active equipment and human intervention cannot be made acceptably safe, but it does mean that there is a broadly accepted hierarchy in which designs incorporating physical principles and passive features to accomplish the vital safety features generally are preferred.

A number of quite new reactor designs among which the top candidate for Generation IV is the **PBMR** type of reactor technology of which the **NTPBMR** surely qualifies, are now under active development.

The key elements that are necessary in the design and operation of a nuclear power reactor to achieve a very high level of safety are the following:

A. A strong base of both scientific and engineering knowledge to support each aspect of the reactor-safety program.

B. A reactor design that accounts for all important potential accident scenarios by employing systems and operational features that reduce the probability of each such scenario, or reduce its potential consequences (or both) to acceptable levels; and the ability to analyze that design well enough to provide high assurance that the above is achieved.

C. A reactor design that utilizes established codes and standards and incorporates adequate margins to assure acceptable performance in light of the uncertainties in knowledge.

D. A reactor design that incorporates a defense-in-depth safety philosophy to maintain multiple barriers, including both physical and procedural barriers as appropriate.

E. A reactor design that uses a philosophy of redundant and diverse safety systems to assure highly reliable performance during all potential accident scenarios.

F. A reactor design that incorporates technical specifications that conservatively define, control, and circumscribe a safe operating envelope.

G. An adequate basis, in experiment, theory, and testing, to support the design specifications and the safety analyses used for safety assurance.

H. The use of quality materials, quality manufacturing of equipment, and quality construction and maintenance practices.

I. An operating philosophy that embodies a profound respect for the possible dangers inherent in reactor operations.

J. A staff of qualified operating and maintenance personnel, supported by a management committed to a strong organizational safety culture, and also supported by a strong engineering capability.

K. An ability to analyze the safety achieved by the operation, in terms of both realistic probabilistic analyses and conservative engineering analyses of the as-built- as-operated facility; and an ability to use the information from such analyses to maintain and enhance safety.

L. Emergency plans that adequately protect offsite populations.

M. An operational safety culture that is both comprehensive and managed properly, and that incorporates an effective self-assessment and corrective-action program.

N. A system that derives safety insights from operating experience and from analyses performed both within the reactor organization itself and elsewhere around the world, and that applies these insights effectively.

O. A strong management organization with both the resources and the motivation to maintain all of the above.

P. An arrangement that has access to a continuing program of nuclear safety research, and that utilizes the insights derived from that research for safety improvement.

Q. An independent regulatory authority that is responsible to the government and the public for overseeing safety, and for taking corrective or enforcement actions as necessary.

Over the past thirty years, international operating experience has demonstrated the importance of high-quality engineering of the facility and high-quality human performance. In the latter arena, operator qualifications and training must be supplemented by operating procedures for both normal and abnormal/emergency situations, and by procedures for accident mitigation.

3.2 WASTE

Geological disposal is technically feasible but execution is yet to be demonstrated or certain. A convincing case has not been made that the long-term waste management benefits of advanced, closed fuel cycles involving reprocessing of spent fuel are outweighed by the short-term risks and costs. Improvement in the open, once through fuel cycle may offer waste management benefits as large as those claimed for the more expensive closed fuel cycles.

Nuclear power has unresolved challenges in long-term management of radioactive wastes. The United States and other countries have yet to implement final disposition of spent fuel or high level radioactive waste streams created at various stages of the nuclear fuel cycle. Since these radioactive wastes present some danger to present and future generations, the public and its elected representatives, as well as prospective investors in nuclear power plants, properly expect continuing and substantial progress towards solution to the waste disposal problem. Successful operation of the planned disposal facility at Yucca Mountain would ease, but not solve, the waste issue for the U.S. and other countries if nuclear power expands substantially.

This document will not address the issues of nuclear waste, proliferation or the fuel cycle which will be implemented for waste disposal and extraction of nuclear materials from existing fuel rods which are stored on the existing power plant sites across the United States. The creation of the disposal of Plutonium will also be addressed through the use of mixed-oxide fuels for the **NTPBMR Project.** These issues will be addressed in other documents of this series.

3.3 PROLIFERATION

The current international safeguards regime is inadequate to meet the security challenges of the expanded nuclear deployment contemplated in the global growth scenario. The reprocessing system now used in Europe, Japan, and Russia that involves separation and recycling of plutonium presents unwarranted proliferation risks.

Proliferation: nuclear power entails potential security risks, notably the possible misuse of commercial or associated nuclear facilities and operations to acquire technology or materials as a precursor to the acquisition of a nuclear weapons capability. Fuel cycles that involve the chemical reprocessing of spent fuel to separate weapons-usable plutonium and uranium enrichment technologies are of special concern, especially as nuclear power spreads around the world;

A section of this document will address the issue of proliferation with the disposal of nuclear waste and the fuel cycle which will be implemented for waste disposal and extraction of nuclear materials from existing fuel rods which are stored on the existing power plant sites across the United States. The creation of the disposal of Plutonium will also be addressed through the use of mixed-oxide fuels for the **NTPBMR Project.**

3.4 COST

In deregulated markets, nuclear power is not now cost competitive with coal and natural gas. However, plausible reductions by industry in capital cost, operation and maintenance costs, and construction time could reduce the gap. Carbon emission credits, if enacted by government, can give nuclear power a cost advantage. A large segment of this document will address the cost control mechanism which will be implemented for the **NTPBMR Project.**

SECTION FOUR: HISTORY OF GAS COOLED NUCLEAR TECHNOLOGIES

4.1 INTRODUCTION

The roots of the **NTPBMR** nuclear technology are traceable to the dawn of the production of electricity with nuclear fission. Of the early power system developed globally one of the largest single programs for civilian nuclear power was the British development and construction of a series of natural-metallic-uranium-fueled-graphite-moderated and **CO_2** cooled reactor power plants. These types of reactors were referred to as **Magnox** reactors. The name **magnox** comes from the alloy used to clad the fuel rods inside the reactor. **Magnox** is an alloy—mainly of magnesium with small amounts of aluminum and other metals—used in cladding unenriched uranium metal fuel with a non-oxidizing covering to contain fission products in nuclear reactors. **Magnox** is short for **Mag**nesium **n**on-**ox**idizing. This material also has the advantage of a low neutron capture cross section

Daniels was director of the Metallurgical Laboratory of the Manhattan Project and, after the war, became concerned to limit or stop the nuclear arms race. In that regard, he became a Board Member of the Bulletin of the Atomic Scientists. In 1947 Daniels conceived the pebble bed reactor, in which helium rises through fissioning uranium oxide or carbide pebbles and cools them by carrying away heat for power production.

4.2 HISTORY OF GAS-COOLED REACTORS

4.2.1 CALDER HALL

Calder Hall was the world's first nuclear power station to deliver electricity in commercial quantities (although the 5 MW "semi-experimental" reactor at Obninsk in the Soviet Union was connected to the public supply in 1954). The design was codenamed **PIPPA** (**Pressurized Pile Producing Power and Plutonium**) by the **United Kingdom Atomic Energy Authority (UKAEA)** to denote the plant's dual commercial and military role. Construction started in 1953. Calder Hall had four Magnox reactors capable of generating 50 MWe of power each. The reactors were supplied by the UKAEA and the turbines by C.A. Parsons & Company. First connection to the grid was on 27 August 1956, and the plant was officially opened by Queen Elizabeth II on 17 October 1956. When the station closed on 31 March 2003, the first reactor had been in use for nearly 47 years. However, in its early life, it was primarily used to produce weapons-grade plutonium, with two fuel loads per year, and electricity production as a secondary purpose. From 1964 it was mainly used on commercial fuel cycles, but it was not until April 1995 that the UK Government announced that all production of plutonium for

weapons purposes had ceased. This type of reactor technology was and is referred to as the **Magnox Reactor.**

4.2.2 MAGNOX

Magnox is a now obsolete type of nuclear power reactor which was designed and is still in use in the United Kingdom, and was exported to other countries, both as a power plant, and, when operated accordingly, as a producer of plutonium for nuclear weapons.

The name **magnox** comes from the alloy used to clad the fuel rods inside the reactor. Magnox reactors are pressurized, carbon dioxide cooled, graphite moderated reactors using natural uranium (i.e. unenriched) as fuel and magnox alloy as fuel cladding. Boron-steel control rods were used. The design was continuously refined, and very few units are identical. Early reactors have steel pressure vessels, while later units (Oldbury and Wylfa) are of reinforced concrete; some are cylindrical in design, but most are spherical. Working pressure varies from 6.9 to 19.35 bar (1 bar = 100 kPa (kilopascals) = 1,000,000 dynes per square centimeter= 0.987 atm (atmospheres) = 14.5038 psi) for the steel pressure vessels, and the two reinforced concrete designs operated at 24.8 and 27 bar

On-load refueling was considered to be an economically essential part of the design for the civilian Magnox power stations, to maximize power station availability by eliminating refueling downtime. This was particularly important for Magnox as the unenriched fuel had a low burnup, requiring more frequent changes of fuel than enriched uranium reactors. However the complicated refueling equipment proved to be less reliable than the reactor systems, and perhaps not advantageous overall.

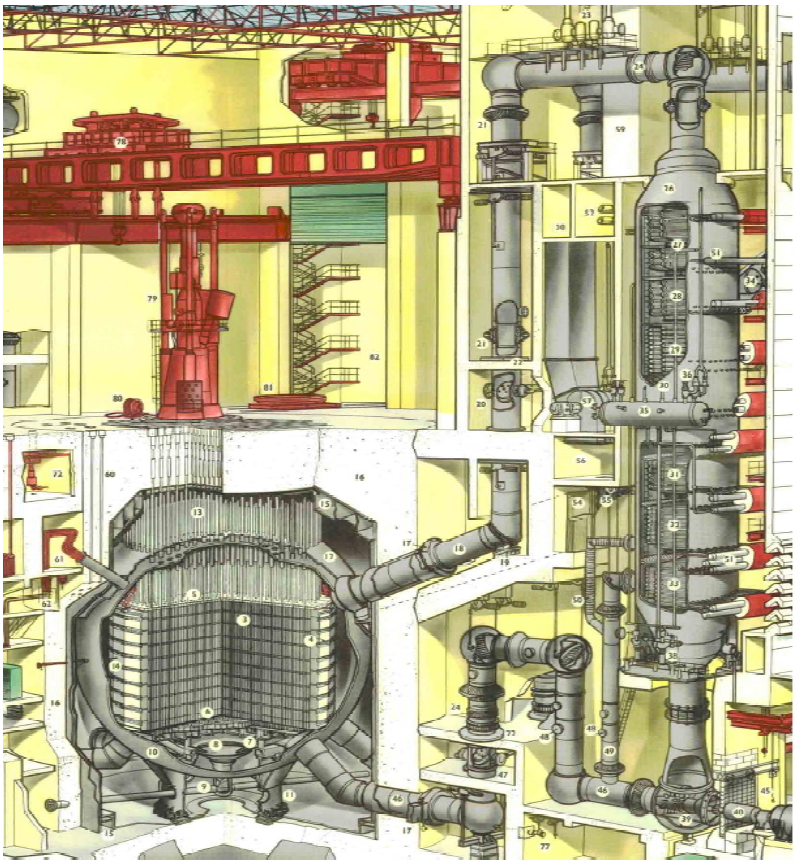
The first Magnox reactors at Calder Hall were designed principally to produce plutonium for nuclear weapons. The production of plutonium from uranium by irradiation in a pile generates large quantities of heat which must be disposed of, and so generating steam from this heat, which could be used in a turbine to generate electricity.

GRAPHIC PRESENTATION OF THE MAGNOX REACTOR

ASCENTRUS
The POWER of ENGINEERING

ARRIAL VIEW OF CALDER HALL MAGNOX REACTOR SITE

THREE DIMENSIONAL VIEW OF REACTOR BUILDING



Although Sir John Cockcroft had advised the government that electricity generated by nuclear power would be more expensive than that from coal, the government decided that nuclear power stations as alternatives to coal fired power stations would be useful to reduce the bargaining power of the coal miners unions, and so decided to go ahead.

A government statement to the House of Commons in 1963 stated that nuclear generation was more than twice as expensive as coal. The "plutonium credit" which assigned a value to the plutonium produced was used, initially secretly, to improve the economic case, although the operators of the power stations were never paid this credit. Once removed from the reactor the used fuel elements are stored in cooling ponds (with the exception of Wylfa which has dry stores) where the decay heat is transferred to the pond water, and then removed by the pond water circulation, cooling and filtration system. The fact that fuel elements can only be stored for a limited period in water before the Magnox cladding deteriorates, and must therefore inevitably be reprocessed, added to the costs of the Magnox program.

The Magnox reactors were considered at the time to have a considerable degree of inherent safety because of their simple design, low power density, and gas coolant. Because of this they were not provided with secondary containment features. A safety design principle at the time was that of the "maximum credible accident", and the assumption was made that if the plant were designed to withstand that, then all other

lesser but similar events would be encompassed. Loss of coolant accidents (at least those considered in the design) would not cause large-scale fuel failure as the Magnox cladding would retain the bulk of the radioactive material, assuming the reactor was rapidly shutdown (a SCRAM), because the decay heat could be removed by natural circulation of air.

As the coolant is already a gas, explosive pressure buildup from boiling is not a risk, as happened in the catastrophic steam explosion at the Chernobyl accident. Failure of the reactor shutdown system to rapidly shutdown the reactor, or failure of natural circulation, was not considered in the design. In 1967 Chapelcross experienced a fuel melt due to restricted gas flow in an individual channel and, although this was dealt with by the station crew without major incident, this event had not been designed or planned for, and the radioactivity released was greater than anticipated during the station design.

In the older steel pressure vessel design, boilers and gas ducting are outside the concrete biological shield. Consequently this design emits a significant amount of direct gamma and neutron radiation, termed direct "shine", from the reactors. For example the most exposed members of the public living near Dungeness Magnox reactor in 2002 received 0.56 mSv, over half the **International Commission on Radiological Protection (ICRP)** recommended maximum radiation dose limit for the public, from direct "shine" alone. The doses from the Oldbury and Wylfa reactors, which have concrete pressure vessels which encapsulate the complete gas circuit, are much lower.

Magnox is also the name of an alloy—mainly of magnesium with small amounts of aluminum and other metals—used in cladding unenriched uranium metal fuel with a non-oxidizing covering to contain fission products. **Magnox** is short for **Mag**nesium **n**on-**ox**idizing. This material has the advantage of a low neutron capture cross-section, but has two major disadvantages: it limits the maximum temperature, and hence the thermal efficiency, of the plant. It reacts with water, preventing long-term storage of spent fuel under water.

Magnox fuel incorporated cooling fins to provide maximum heat transfer despite low operating temperatures, making it expensive to produce. While the use of uranium metal rather than oxide made reprocessing more straightforward and therefore cheaper, the need to reprocess fuel a short time after removal from the reactor meant that the fission product hazard was severe. Expensive remote handling facilities were required to address this danger.

4.2.3 URANIUM NATURAL GRAPHITE GAZ

The **UNGG** (*Uranium Naturel Graphite Gaz*) is an obsolete design of nuclear power reactor developed by France. It was graphite moderated, cooled by carbon dioxide, and fueled with natural uranium metal.

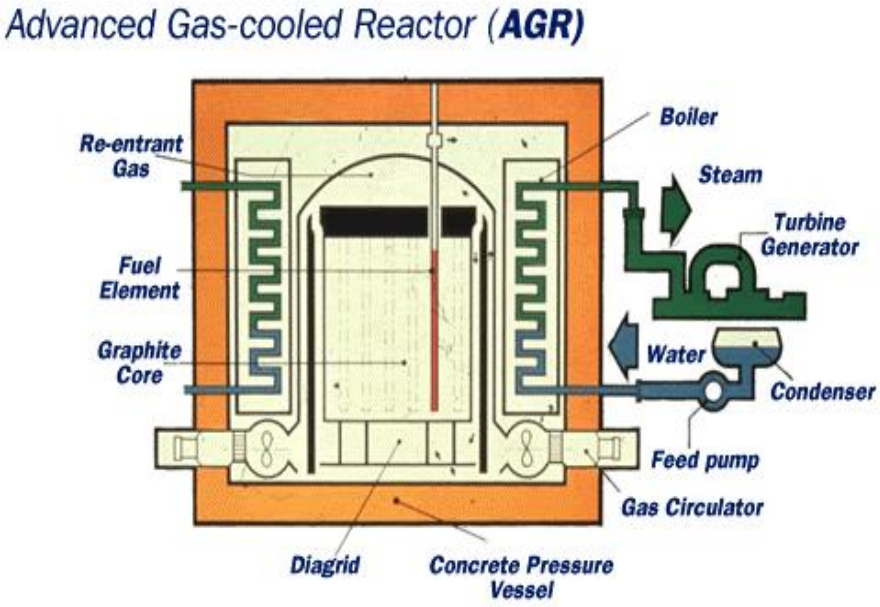
It was developed independently of and in parallel to the British Magnox design, and to meet similar requirements. The main difference between the two designs is that **UNGG** used a horizontal fuel rod orientation, rather than the vertical orientation used in the

Magnox reactor. The fuel cladding material was magnesium-zirconium alloy in the UNGG, as opposed to magnesium-aluminium in Magnox. Both claddings react with water, making short-term reprocessing of the fuel essential, and requiring heavily shielded facilities for this. The **UNGG** and the Magnox are the two main types of **Gas Cooled Reactors** (**GCR**). A **UNGG** reactor is often referred to simply as a **GCR** in English documents, or sometimes loosely as a *Magnox*. The first generation of French nuclear power stations were UNGGs, as was Vandellos unit 1 in Spain. Of ten units built, all are now shut down.

4.2.4 ADVANCED GAS COOLED REACTOR

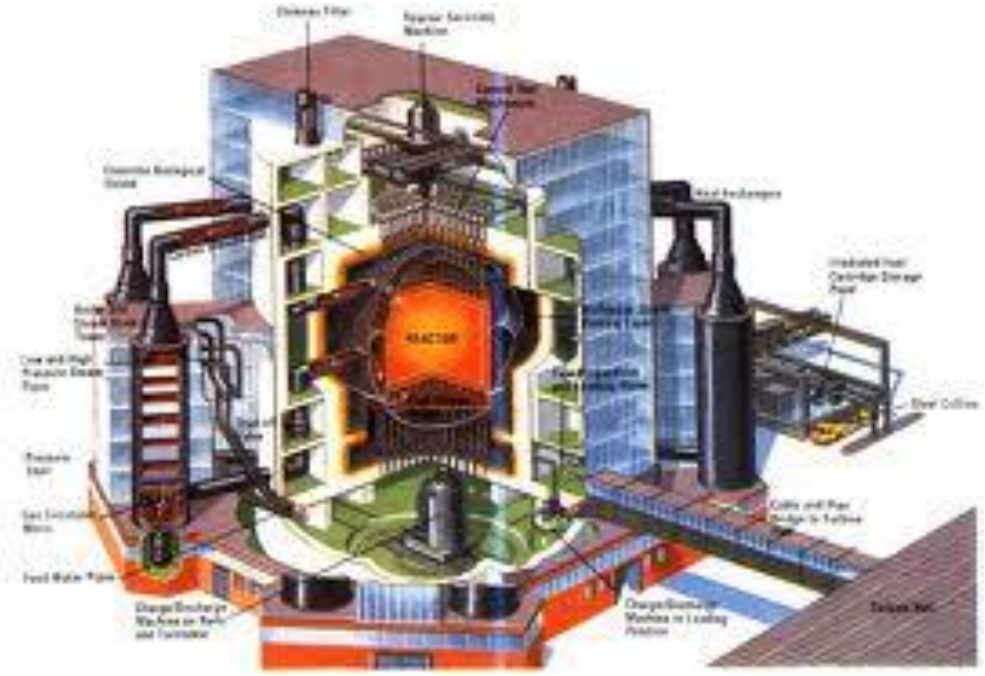
The accepted term for all of these first-generation, carbon dioxide-cooled, graphite-moderated reactors, including the **Magnox** and **UNGG**, is **GCR** for **Gas Cooled Reactor**. The Magnox was replaced in the British power station program by the advanced gas-cooled reactor or **AGR,** which was derived from it.

The **advanced gas-cooled reactors (AGR)** are the second generation of British gas-cooled reactors, using graphite as the neutron moderator and carbon dioxide as coolant. The **AGR** was developed from the **Magnox** reactor, operating at a higher gas temperature for improved thermal efficiency, requiring stainless steel fuel cladding to withstand the higher temperature. Because the stainless steel fuel cladding has a higher neutron capture cross section than Magnox fuel cans, enriched uranium fuel is needed, with the benefit of higher "burn ups" of 18,000 MWt-days per tonne of fuel, requiring less frequent refueling. The first prototype **AGR** became operational in 1962 but the first commercial **AGR** did not come on line until 1976.



AIRIAL VIEW OF AGR NUCLEAR POWER PLANT

THREE DIMENSIONAL RENDERING OF AGR NUCLEAR REACTOR



All **AGR** power stations are configured with two reactors in a single building. Each reactor has a design thermal power output of 1,500 MWt driving a 660 MWe turbine-alternator set. Because of operational restrictions, the various **AGR** stations produce outputs in the range 555 MWe to 625 MWe.

The design of the **AGR** was such that the final steam conditions at the boiler stop valve were identical to that of conventional coal fired power stations, thus the same design of

turbo-generator plant could be used. The mean temperature of the hot coolant leaving the reactor core was designed to be 648°C. In order to obtain these high temperatures, yet ensure useful graphite core life (graphite oxidizes readily in CO_2 at high temperature) a re-entrant flow of coolant at the lower boiler outlet temperature of 278°C is utilized to cool the graphite, ensuring that the graphite core temperatures do not vary too much from those seen in a Magnox station. The superheater outlet temperature and pressure were designed to be 2,485 psia and 543°C.

The fuel is uranium dioxide pellets, enriched to 2.5-3.5%, in stainless steel tubes. The original design concept of the **AGR** was to use a beryllium based cladding. When this proved unsuitable, the enrichment level of the fuel was raised to allow for the higher neutron capture losses of stainless steel cladding. This significantly increased the cost of the power produced by an **AGR.** The carbon dioxide coolant circulates through the core, reaching 640°C (1,184°F)and a pressure of around 40 bar (580 psi), and then passes through boiler (steam generator) assemblies outside the core but still within the steel lined, reinforced concrete pressure vessel. Control rods penetrate the graphite moderator and a secondary system involves injecting nitrogen into the coolant to hold the reactor down. A tertiary shutdown system which operates by injecting boron balls into the reactor has been proposed 'as retrofit to satisfy the Nuclear Installations Inspectorate's concerns about core integrity and core restraint integrity'.

The **AGR** was designed to have a high thermal efficiency (electricity generated/heat generated ratio) of about 41%, which is better than modern pressurized water reactors which have a typical thermal efficiency of 34%. This is due to the higher coolant outlet temperature of about 640 °C (1,184°F) practical with gas cooling, compared to about 325 °C (617°F) for PWRs. However the reactor core has to be larger for the same power output, and the fuel burnup ratio at discharge is lower so the fuel is used less efficiently, countering the thermal efficiency advantage.

Like the **Magnox**, **CANDU** and **RBMK** reactors, and in contrast to the light water reactors, **AGR**s are designed to be refuelled without being shut down first. This on-load refuelling was an important part of the economic case for choosing the **AGR** over other reactor types, and in 1965 allowed the **CEGB** and the government to claim that the **AGR** would produce electricity cheaper than the best coal fired power stations. However fuel assembly vibration problems arose during on-load refuelling at full power, so in 1988 full power refuelling was suspended until the mid-1990s, when further trials led to a fuel rod becoming stuck in a reactor core. Only refuelling at part load or when shut down is now undertaken at **AGR**s.

Note: The **Central Electricity Generating Board (CEGB)** was the cornerstone of the British electricity industry for almost 40 years; from 1957, to privatization in the 1990s.

The AGR was intended to be a superior British alternative to American light water reactor designs. It was promoted as a development of the operationally (if not economically) successful Magnox design, and was chosen from a plethora of competing British alternatives:

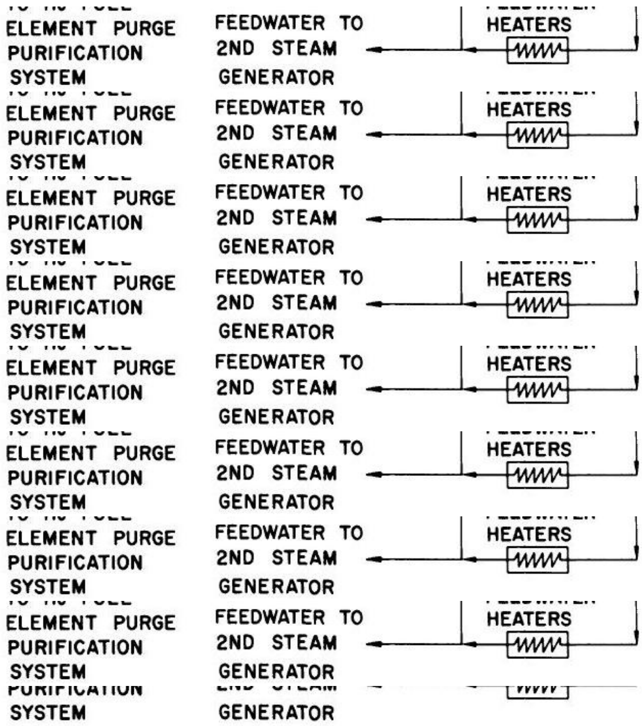
- the helium cooled High Temperature Reactor (**HTR**),
- the Steam Generating Heavy Water Reactor (**SGHWR**) and the Fast Breeder Reactor (**FBR**)
- the American light water pressurized and boiling water reactors (PWR and BWR)
- Canadian CANDU designs.

The **CEGB** conducted a detailed economic appraisal of the competing designs and concluded that the **AGR** proposed for Dungeness B would generate the cheapest electricity, cheaper than any of the rival designs and the best coal fired stations.

There were great hopes for the **AGR** design. An ambitious construction program of five twin reactor stations, Dungeness B, Hinckley Point B, Hunterston B, Hartlepool and Heysham was quickly rolled out, and export orders were eagerly anticipated. However, the AGR design proved to be over complex and difficult to construct on site. Notoriously bad labor relations at the time added to the problems. The lead station, Dungeness B was ordered in 1965 with a target completion date of 1970. After problems with nearly every aspect of the reactor design it finally began generating electricity in 1983. The follow on stations all experienced similar problems and delays. The financing cost of the capital expended, and the cost of providing replacement electricity during the delays, were enormous, totally invalidating the pre-construction economic case.

4.2.5 HIGH TEMPERATURE GAS-COOLED REACTOR (HTGR)

The **HTGR** design was first proposed by the Staff of the Power Pile Division of the Clinton Laboratories (known now as Oak Ridge National Laboratory) in 1947. Professor Dr. Rudolf Schulten in Germany also played a role in development during the 1950s. The Peach Bottom reactor in the United States was the first **HTGR** to produce electricity, and did so very successfully, with operation from 1966 through 1974 as a technology demonstrator. Fort St. Vrain Generating Station was one example of this design that operated as an **HTGR** from 1979 to 1989; though the reactor was beset by some problems which led to its decommissioning due to economic factors, it served as proof of the HTGR concept in the United States (though no new commercial **HTGR**s have been developed there since). HTGRs have also existed in the United Kingdom (the Dragon reactor) and Germany (**AVR** and **THTR**-300), and currently exist in Japan (the **HTTR** using prismatic fuel with 30 MW_{th} of capacity) and China (the HTR-10, a pebble-bed design with 10 MW_e of generation). Two full-scale pebble-bed HTGRs, each with 100 - 195 MW_e of electrical production capacity are under construction in China at the present as of November 2009, and are promoted in several countries by reactor designers. More recently, this reactor design type has been substantially updated and is now proposed in a form known as the Very High Temperature Reactor in the United States.

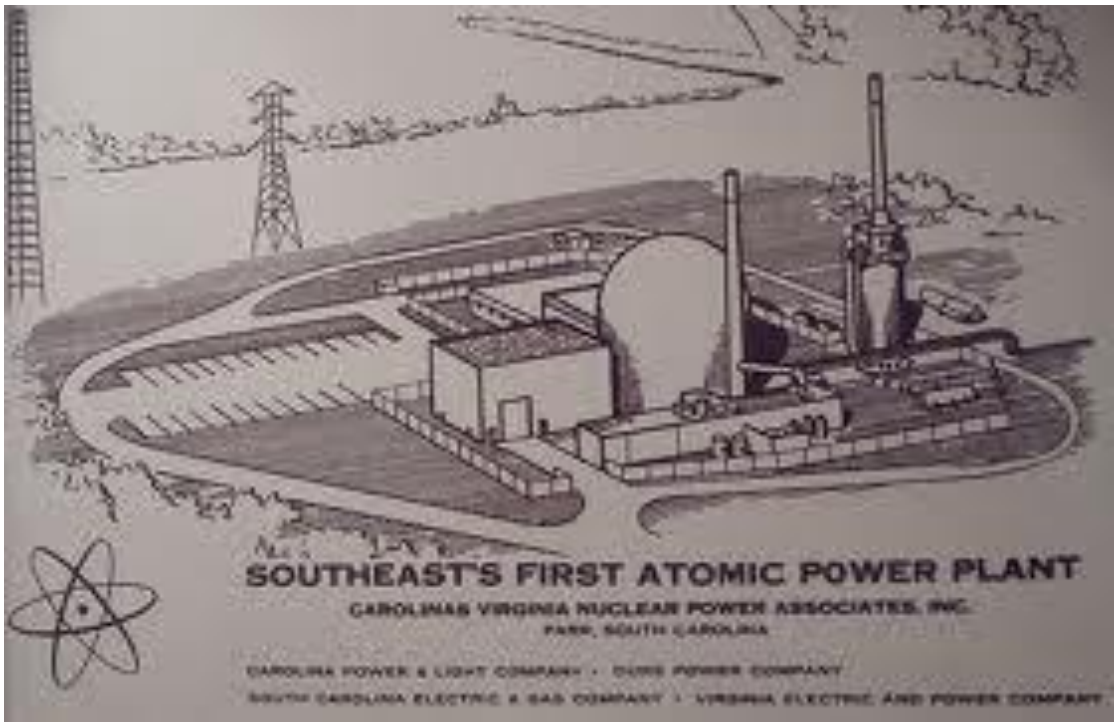


4.2.6 PEACH BOTTOM ATOMIC POWER STATION

The decommissioned Peach Bottom nuclear power plant, is located 50 miles (80 km) southeast of Harrisburg in Peach Bottom Township, York County, Pennsylvania, on the Susquehanna River. The Philadelphia Electric Company (later shortened first to PECO Energy and later to just PECO) became one of the pioneers in the commercial nuclear industry when it ordered Peach Bottom 1 in 1958. The U.S.'s first nuclear power plant (the Shippingport Reactor) had gone on line a year earlier. Peach Bottom Unit 1 was an experimental helium-cooled, graphite-moderated reactor. It operated from 1966 to 1974.

4.2.6.1 INTRODUCTION

Peach Bottom Unit I, owned and operated by the Philadelphia Electric Company, was the first prototype high temperature, gas cooled reactor in the United States. In February 1962, the Atomic Energy Commission issued a construction permit for the Peach Bottom Atomic Power Station, and work at the plant site was begun. This followed the announced intention by Philadelphia Electric in 1960 to build and demonstrate the General Atomic concept of a nuclear power plant capable of producing steam at high temperature and pressure. Philadelphia Electric Company and High Temperature Reactor Development Associates, Inc. and Bechtel Corporation had entered into a Memorandum of Understanding on November 17, 1958, for the design and construction of a high temperature gas cooled graphite moderated prototype nuclear power plant at Peach Bottom. Philadelphia Electric Company and a group of investor-owned utilities throughout the country organized into High Temperature Reactor Development Associates, Inc.



In November 1958, and proposed to develop, design, build, and operate a high performance helium cooled nuclear power plant. This proposal was in response to an invitation issued to industry by the Atomic Energy Commission for the development of a prototype I-ITGR. The proposal requested AEC assistance in support of research and development costs as offered in the Commission's invitation. In August 1959, the Atomic Energy Commission signed contracts with Philadelphia Electric Company and General Atomic for the development, construction, and operation of the Peach Bottom Atomic Power Station.

4.2.6.2 PRE-OPERATIONAL TESTING AND STARTUP

Demonstration testing was started in 1964. In February 1965, during the initial hot run, a construction fire in the containment caused severe cable damage. The fire, believed to have been caused by hot metal from an acetylene torch during construction work, broke out in one of the cable penetrations inside the containment building. The fire was confined to burning insulation of cables and wiring in a relatively small area. No radioactive material was involved, and there were no injuries to personnel. Dense black smoke from the burning cable insulation filled the containment building and left a heavy deposit of soot throughout the structure.

After a delay to repair the fire damage, the hot run tests were resumed and upon completion of the tests, an inspection of the steam generator tube sheet area was made. During this inspection, several leaks were found in the stainless steel tubes of the super-heater section. Further investigation disclosed that these leaks had resulted from

chloride stress corrosion. A decision was made to re-tube the super-heater section with Incoloy tubes. This re-tubing was deferred until after the fuel loading.

Fuel loading was started in February 1966, and one month later, on March 3, initial criticality was achieved. Fuel loading was completed and low power physics tests were performed on the reactor core. The results of these tests showed good agreement with expected results and in May the plant was shut down for the steam generator re-tubing and repair work. Re-tubing of the super-heater section was successfully completed in December 1966. No tube leaks developed in the steam generators throughout the operating lifetime of the plant, indicating the correctness of the design decision to re-tube the super-heater with Incoloy tubes.

4.2.6.3 INITIAL OPERATION

The plant was started in late December and tests were run at various power levels up to and including full power. Difficulties encountered with high steam generator shell temperatures resulted in a slight reduction of operating steam conditions. The steam generator shell is cooled by cold helium return flow. Insufficient flow in the bottom head area caused shell temperatures to reach their maximum at about 60% rated power. Additional baffles were installed in the steam generator interiors to direct coolant flow to the bottom head and an external bottom head cooling unit was added to each steam generator.

Reactor power was increased to 100%, but operating steam temperature and pressure were reduced slightly as a result of the limiting steam generator shell temperatures. Upon completion of the test program, the plant went into commercial operation in June 1967.

Peach Bottom operated at full power for 168 full-power days until January 1968, when the plant was shut down for a scheduled maintenance and fuel surveillance program. Prior to this shutdown, a sudden increase in the primary system activity indicated a failed fuel element or cladding rupture had occurred. This was verified during the shutdown and a new fuel element was substituted for the failed one. The failed fuel element was returned to General Atomic for inspection to determine the cause of failure. The plant was restarted and during the power run to 300 full-power days, the primary system activity continued to increase. The plant was shut down again in late 1968 for maintenance and the surveillance program. Upon investigation, 11 additional failed fuel elements were found and these were removed from the core. New fuel was inserted and the plant was returned to power in early 1969 and operated until the 450 full-power days shutdown.

During this period of operation, the primary system activity level continued to increase, indicating additional fuel damage or cladding rupture was occurring. The plant was operated at reduced power level to minimize the thermal effects on fuel. When the plant was shut down on October 3, 1969, it was found that 78 additional fuel elements had failed. These fuel elements were removed from the core. At this time, a decision was

made to refuel the entire core, replacing the remaining Core 1 fuel elements with an advanced type fuel element.

4.2.6.4 FUEL IMPROVEMENT

The Core 1 fuel elements contained fuel compacts consisting of uranium and thorium in the form of carbides uniformly dispersed as particles in a graphite matrix. Each particle was pyrolytically coated with a dense carbon. This coating protects the fuel material from oxidation reactions during fabrication and serves to increase the retention time of fission products during reactor operation. Fast neutron-induced dimensional changes and damage due to fission product recoils resulted in cracking and distortion of the coatings on the fuel particles. The broken coatings caused the compacts to distort and swell. The radial expansion resulting from the swelling caused the fuel element graphite sleeve containing the compacts to crack. The Core 2 fuel compacts contained particles with coatings which consisted of an inner, low density pyrolytic carbon buffer coating around the carbide fuel kernel and an outer isotropic coating. The improved two-layer Buffer-Isotropic (BISO) coatings exhibited excellent irraditation stability in contrast to the poor stability of the single layer coatings used in Core 1.

The plant was returned to power on July 13, 1970, with the new Core 2 fuel elements. A test program was completed to verify the performance of this core and the results were in good agreement with expected data. Core 2 was operated successfully for its entire design lifetime of 900 equivalent full-power days. The gaseous activity in the primary helium system averaged 0.5 curies during power operation with Core 2.

4.2.6.5 RESEARCH AND DEVELOPMENT PROGRAM

A post-startup research and development program was conducted on Peach Bottom throughout the reactor lifetime. This program, which included core component performance, fission product release and plate-out, circulating activity, coolant chemistry, and other important features of reactor operation, was conducted by General Atomic and Oak Ridge National Laboratory. Two diffusion probes were installed in the No. 1 main helium loop to sample the main loop helium and thus acquire *information on* plate-out activity in the primary system. The program included post-irradiation examination of test fuel elements to verify operating characteristics. Throughout Peach Bottom operation, excellent agreement was found between predicted and actual core physics characteristics, thus verifying the methods used.

Peach Bottom operated very successfully and produced superheated steam at 1000F and 1450 psig, with an overall station availability, excluding planned shutdowns for R and D programs, of 88% and a gross thermal efficiency of 37%. Over 1.2 million net electrical megawatt hours were produced for the Philadelphia Electric Company grid over a lifetime of 1349 equivalent full-power days.

The reactor control system functioned exceptionally well and the plant was operated in a load-following manner during the majority of its lifetime, demonstrating the ability of the HTGR to function in this manner. The performance of the 40 MWe Peach Bottom plant

verified the design philosophy of a helium-cooled nuclear power plant and provided valuable technical data for application to larger HTGR plants.

4.2.6.6 DECOMMISSIONING

On October 31, 1974, Peach Bottom was shut down for decommissioning. The decision to decommission the Peach Bottom HTGR was based on a study of the benefits to be derived from further operation beyond depletion of Core 2 relative to the investment necessary to satisfy the AEC's (now NRC) requirement for a full-term license.

The study indicated that the benefits to be derived from continued operation were not sufficient to justify the large expenses that would be incurred in satisfying the requirements for a *permanent* license. The plant was decommissioned in accordance with the Decommissioning Plan and Safety Analysis Report submitted to the Nuclear Regulatory Commission on August 29, 1974, and amended in May 1975. The plan calls for removal of all radioactivity outside of an Exclusion Area fence within which will remain the *containment* building and the fuel storage pool. This plan was approved by the NRC and an amendment to the Provisional License was issued on July 14, 1975, which allows Philadelphia Electric Company to possess, but not operate, the reactor.

Following the plant shutdown in October 1974, all 804 fuel ele*ments* were removed from the reactor and placed in the spent fuel pool. De-fueling of the reactor was completed in June 1975, and on June 24, 1975 shipping of the fuel to Aerojet Nuclear Company (now EG&G) in Scoville, Idaho, commenced. A total of 44 fuel shipments were made by truck utilizing two PB-1 fuel shipping casks. In addition to the normal fuel shipments, 27 fuel shipments were made in the single-element Hallam ~'~ fuel shipping cask. These shipments were made in support of the Peach Bottom Post-irradiation Experimental Program conducted by General .Atomic Company. Fuel shipping was completed in February 1977. The spent fuel pool was drained and the pool wafer was processed through the rad-waste system prior to release.

In addition to de-fueling and fuel shipping, the helium purification system delay beds were degassed and the primary helium coolant *inventory* was processed through the Liquid Nitrogen Trapping System and released via the plant stack. The helium in the primary system has been replaced with dry nitrogen. The component removal phase (fission product delay beds containing charcoal and dust collectors) began in January 1976. Removal of the fission product delay beds and other activities specified by the Decommissioning Plan were completed during July 1976.

All contaminated components removed were shipped to the licensed burial sites at Morehead, Kentucky, and Barnwell, South Carolina. In addition, 200 gallons of tritrated liquid rad-waste and 300 gallons of contaminated oil were solidified and and shipped to the burial grounds. Additional shipments of contaminated components, trash, charcoal absorbers and miscellaneous piping generated by the decommissioning activities were made to the burial grounds.

From July 1976 to July 1977, the decommissioning activities were suspended pending completion of fuel shipping and the Unit Z outage. All decommissioning activities were completed by January 1978. Final activities included lay-up of the containment building, removal of the radioactive waste disposal system, decontamination as required and the erection of the exclusion area fence to restrict access to the containment and fuel pool buildings. Eleven pipes located within the liquid rad-waste area could not be decontaminated to acceptable levels for unrestricted use. Following final decontamination efforts on these li pipes, all exposed pipe ends were seal-welded and all pipe ends under floor level were plugged and covered with concrete to the original floor level. The exclusion area was extended to include the liquid rad-waste area along with the containment and spent fuel buildings.

The balance of plant equipment, including the low pressure heat cycle, turbine generator and auxiliary equipment have been removed from service and laid up to maintain their salvage value. This equipment will be used either at another facility or sold. Peach Bottom Unit 1 decommissioning also offered a unique opportunity to conduct end-of-life research and surveillance in an HTGR. In March 1975, implementation of the Peach Bottom End-of-Life Program, cosponsored by ERDA and EPRI, was initiated. The prime objective of this program is to validate specific HTGR design codes and predictions by comparison of actual to predicted physics, thermal, fission products and materials behavior in Peach Bottom.

A total of 107 samples of primary circuit ducting and steam generator tubing were removed. These samples were packaged in special inert containers for shipment to General Atomic. Evaluations at General Atomic are continuing, but indicate excellent performance of the steam generator and other materials, together with close correlation of observed and predicted fission product plate-out distributions.

4.2.6.7SUMMARY AND CONCLUSIONS

In conclusion, the Peach Bottom Unit 1 Demonstration Plant was a highly successful venture. It proved that the concept of a helium cooled graphite moderated power reactor was sound and workable. The reactor ran exceptionally well, and through two core lives, proved its reliability as a source of electrical energy. Through the development, design and operation of the nuclear steam supply system, the technology was carried out cooperatively by the reactor developer and manufacturer, the engineering construction contractor and the electric utility, functioning as an integrated team. The success of the project is a tribute to the many talented people who participated in its many facets.

The operation of Peach Bottom Unit 1 demonstrated a new concept in nuclear fuel and fission product control system. The coated particle-graphite matrix fuel element was developed and refined through operating experience, and, in its final version, proved conclusively that it has the capability of being used successfully in commercial plants, based on this design.

The fission product trapping system was a new concept based on new technology, which operated very successfully. Throughout its life, the reactor performed predictably,

as well as reliably. The ability to predict its static and dynamic nuclear characteristics from its first criticality is a tribute to the diligent development work and skill applied by its designers. It would be hard to picture a more successful development and demonstration of a complex scientific system than the Peach Bottom Unit 1 experience.

4.2.7 FORT SAINT VRAIN

4.2.7.1 INTRODUCTION

The Fort Saint Vrain Generating Station is now a natural gas powered electricity generating facility located near the town of Platteville in northern Colorado. It currently has a capacity of just over 1000MW and is owned and operated by Xcel Energy, the successor to the plant's founder, the Public Service Company of Colorado. It went online in this form in 1996.

The facility was built originally as a nuclear power plant. It operated as a nuclear generating power plant from 1977 until 1992. Originally, Fort Saint Vrain Generating Station was built as Colorado's first (and only) nuclear power plant and operated as such from 1977 until 1992. It was one of two **High Temperature Gas Cooled Reactor (HTGR)** power reactors in the United States. The primary coolant was helium which transferred heat to a water based secondary coolant system through steam generators. The reactor fuel was a combination of fissile uranium and fertile thorium microspheres dispersed within a prismatic graphite matrix. The reactor had a power output of 330MWe (842 MWth).

This unique, gas-cooled nuclear power plant was proposed in March 1965 and the application was filed with the **Atomic Energy Commission** (Now called the **NRC)** in October 1966. Construction began in 1968. The building was unique for U.S. commercial reactors; it had a rectangular shape instead of the usual cylindrical domed buildings housing other reactors; this was due to the fact it was merely a steel-frame confinement (as the **HTGR** design is extremely safe, no steel-reinforced, pre-stressed concrete containment building is truly necessary, though the reactor was somewhat contained within a pre-stressed concrete reactor pressure vessel (**PCRV**)). The construction cost reached $200 million, or approximately $0.60/installed watt.

PHOTOGRAPH OF FORT SAINT VRAIN POWER STATION

Initial testing began in 1972 and the first commercial power was distributed in December 1976. Overall, as a first of a kind facility, the plant was technically successful, especially towards the very end of its commercial life, but was a commercial disappointment, due to the numerous teething problems encountered by the advanced technology in use that was arguably not ready for the production environment into which it was deployed, at least not initially.

4.2.7.2 UNIQUE FEATURES OF THE DESIGN

1. The **Fort St. Vrain** (**FSV**) **HTGR** was substantially more efficient than modern light water reactors, reaching a thermal efficiency of 39-40%, excellent for a steam-cycle power plant.
2. **FSV** could easily load follow rather than solely generate base-load power at full capacity all the time.
3. The reactor was extremely fuel efficient, as well, with a maximum burnup of 90,000 MW days thermal (compare to LWRs with burnups of 10,000 - 40,000 MW days thermal). The reason for this is that the core was designed to fertilize the thorium pellets within the fuel with neutrons and then burn the bred fissiles through normal neutronic processes without requiring its removal from the core.
4. Like all **HTGR**s, **FSV** had a design that precluded the possibility of major core damage or radioactive releases in such a quantity that could seriously threaten public safety. (The **NRC** recognized this and allowed operation with much smaller zones compared to **LWR**s.)
5. It was also notable that plant personnel received negligible exposure to ionizing flux during the course of operations.

6. The **PCRV** reflected an innovative **RPV** that had the potential to be substantially less costly than the metallic **RPV**s then in service, which were made of expensive nickel-manganese superalloys (e.g. Inconel, Hastelloy, and Monel) in the case of **PWR**s or surgical grade stainless steel 316L in the case of BWRs.
7. The fuel, by omitting Zircalloy sheathing (allowed due to the inert, non-aqueous core) was made far less expensive.

4.2.7.3 ANALYSIS OF FORT SAINT VRAIN NUCLEAR POWER PLANT

FSV worked, and once debugged, it worked well for a first of a kind facility, demonstrating a promising new concept for the future. However, the problems that occurred leading to its debugging led to its early demise. Many issues occurred early in the operational experience of the **Fort St. Vrain HTGR**. Though these issues did not threaten the safety of the station, utility personnel, or the general public (due to the high level of safety of the design) the problems that did occur put considerable stress upon the personnel, equipment, and facilities present at **FSV**, and made continued operation appear to not be economical to the plant's owner.
Most of the past issues had been resolved at considerable expense and the plant was beginning to perform at a decent level when an economic downturn and the past history of the plant caused the owner to shut it down prior to the end of its design lifetime. There were 3 major categories of issues:

1. Water infiltration & corrosion issues
2. Electrical system issues
3. General facility issues.

The root cause of a large part of the problems with Fort St. Vrain was one piece of equipment, in particular: the helium circulator. Due to the small molecular size of helium, exceedingly close tolerances were needed to ensure that helium did not exfiltrate through the circulator while in use, and moving surfaces, in particular, were hard-pressed to provide the kind of seal required to keep the helium coolant in. Thus a water-lubricated bearing design was used to provide an adequate solution to the potential issue of helium exfiltration. Unfortunately, in satisfactorily preventing helium exfiltration, the designers caused another issue: water infiltration. Circulator bearings had a timed water injection system in the event of circulator trip. The designers of the circulator thus used the pressure of a fluid to counteract the pressure of other fluids. The designers of the circulator, however, had not fully appreciated the transient variations that could occur in the pressure of either fluid, especially the pressure of the bearing water. As such, when bearing water was injected into the circulators, problems could occur if steam or helium pressure which opposed the pressure of the bearing water was not within expected parameters. For instance, the steam pressure could vary considerably due to changes in circulator speed, water flow through the steam generator, stop valve closure or throttle valve actuation, or, in the case of the helium pressure, this could vary based on the level of reactor power generation and core pressurization or depressurization.

Thus, during certain plant evolutions, the bearing water infiltrated into the **PCRV** due to variable pressures of plant fluids.

FSV did have a gas cleanup train that could rapidly remove certain contaminants from the helium, but was limited in volume and was not greatly effective in removing water vapor from the gas within the **PCRV**, and, in fact, could be prevented from working by water vapor icing the chillers within the gas cleanup train, and so, when the reactor descended from power and cooled, the water condensed upon equipment within the **PCRV**; neither the **PCRV** nor the equipment thereof was designed to resist the effects of water-induced corrosion.

FSV's gas cleanup train was driven around regulatory concerns pertaining to theoretical core graphite-water interactions at high temperatures and pressures - which did not occur - the core being high-grade graphite, which did not possess the micro-porous structure of lower grade graphites providing sufficient surface area for substantial chemical reactions. It must be noted that even though the core proper was not reactive, there was some erosion of low-grade ex-core graphite support blocks due to water-gas shift processes, but the core's graphite was not subject to these, and the slight erosion detected did not substantially impact operations, absorb all the infiltrated water or evolved steam, or induce major gas cleanup considerations. Instead, the vast majority of entrained steam and water vapor in the coolant failed to react as the regulators foresaw, and thus, condensed water vapor began corroding in-core and ex-core instrumentation.

Thus, water entered the sealed volume of the **PCRV** and caused havoc with numerous operations-critical systems. Though safety was assured to a substantial level by the design, numerous severe operability problems emerged quickly. Control rod drives rusted, and consequently rapid shutdowns failed when called upon to function. The reserve shutdown system, consisting of borated graphite spheres to be released into the core in the event of an **Anticipated Transient Without Scram** (**ATWS)**, was unavailable at times due to water leaching of the boron and the subsequent unscheduled, impromptu reconfiguration of the graphite spheres into graphite sausage-shaped cylinders due to boric acid precipitation, not contemplated within the design.

Steel tendons within the **PCRV** were found to be corroded due to precipitation of chloride and not to specification upon routine surveillance. Steam generator leaks due to corrosion of the steam generators also occurred (probably due to the original water infiltration problems), adding volumes of figurative fuel to the figurative fire. Flecks of corroded steel even got into the coolant itself and lodged themselves in critical parts of critical machinery, such as control rod drives.

Further, the gas cleanup train's chiller units became iced due to the deposition of water vapor on to their supercold surfaces, rendering them ineffective at times when they were most needed. Some of the blame for the corrosion debacle has to be laid on the regulators, who maintained a consistent improper regulatory focus on chemical reactions involving steam with the high-grade core graphite, as this was the area that drove design of the gas cleanup train; it was foreseeable that the memorandums from Rockville,

Maryland regarding this obviously consumed countless man-hours and drove the designers to distraction on peripheral issues whose occurrence was physically infeasible.

Some of the blame for the corrosion debacle has to be laid on the owner of **FSV**, whose staff failed to respond to moisture alarms that had been going off for months in critical parts of the plant, instead assuming that the moisture alarms were defective. (Licensee staff sent to remove the "defective" moisture alarms for "repair" discovered that the moisture alarms were not defective, for when they removed the "defective" alarms, they got sprayed with a large volume of water.) Still, a large part of the blame must be laid on the designers of the plant themselves, who should have been able to foresee that large scale water infiltration was possible with the complex, buggy circulator design; who should have been able to foresee that the cleanup train should have reserve capacity for steam and water extraction; who should have been able to foresee that since this was not present, that major corrosion of in-core instrumentation and systems could occur and severely degrade the performance and systems of the total plant.

Further, though the literature does not suggest what sorts of motivations or concerns drove the designers of the circulators to choose such a high-complexity, low-tolerance, leak-prone design, this was the major cause of the major plant problems; the designers themselves admitted this, stating: "The **FSV** circulators have 'met all design specifications', however, the bearings, seals, and support systems for the water-lubricated bearing have caused many problems. Further, the circulators employed a steam turbine drive that adds complexity to system operations. These *unique design features* (emphasis added) resulted in water ingress to the core, the primary reason for poor plant availability. The plant electrical system was challenged on numerous occasions and these challenges proved costly to the plant. Transformers had faults and backup generators failed to generate when called upon.

Side channel issues also occurred during operation of backup generating systems, preventing them from generating power. The plant safety was not substantially imperiled; still, these were costly situations that needed to be resolved. Failure of backup power also led to some of the moisture infiltration problems, by disrupting the logic of the bearing water injection systems and/or the helium circulator trip logic. Interestingly, failures of transformers and consequent failure of backup power occurred on at least one occasion due to moisture infiltration into electric cables and subsequent ground faulting when the plant was at low power to remove water from previous moisture infiltration issues.

It is believed that this electrical fault led to further moisture infiltration. General facilities issues Contractor personnel did not help matters related to efficient and safe operations in several cases; in one incident, contractors broke hydraulic units, allowing hydraulic fluid to spray over reactor control cables, and then attempted to weld above the control cables; welding slag fell onto the material used to contain the hydraulic fluid, and ignited it, along with the fluid on the control cables. After a fire involving the cables for 5

minutes, 16 essential control cables were damaged, and the contractors failed to inform plant personnel of the situation for several hours while the reactor was in operation. Contractors were noted to have used welding apparatuses in a manifestly unsafe fashion, disregarding safety protocols. Further, on one occasion, contractors using improperly grounded welding apparatuses tripped neutron protection circuits leading to a nuisance trip of the entire plant.

Operational improvement and closure Due to the water-induced corrosion problems and electrical problems, plant shutdowns were common. As a result, the plant became regarded by Public Service Company of Colorado as a money sink that was too costly to operate on a commercial basis. Thus, in 1989, it was placed under the threat of closure within a year, though not before showing an increase in performance from 1987 - 1989, signaling that many of the "bugs", so to speak, had been worked out of the system. But it was too late; during 1989, the discovery of a critical part of the reactor that had been corroded years before and had to be replaced was the last straw that broke the camel's back. This led to the plant not continuing in operation as the replacement cost of the critical part was too high. The decommissioning and removal of the fuel was completed by 1992.

Analysis Unlike Peach Bottom, Dragon, AVR, HTTR, and HTR-10, all of which represent successful tests proving the principle of the **HTGR** technology, Fort St. Vrain was arguably doomed by the engineering mistake to use a first of a kind engineered, high-complexity steam turbine helium circulator with multiple fluid bearings instead of a simple, commercial off the shelf, low-complexity electric motor-based helium circulator, such as the electrical coolant circulators successfully used for decades in the UK's **AGCR,** which have stood and do stand the test of time in a similar, yet far more chemically hostile environment than that of a helium-cooled reactor core. Lessons learned at Fort St. Vrain have led more recent reactor designs of the **HTGR** type to adopt different strategies to confront issues that occurred there.

For instance, more recent **HTGR** designs have tended to avoid large per-unit cores (in favor of more compact modular units), tended to avoid concrete reactor pressure vessels (in favor of proven carbon or alloy steel reactor pressure vessels), and tended to avoid steam cycles without an intermediate non-water based circuit between the core and the steam generators. Still others, such as the Adams Atomic Engine (using nitrogen), the Romawa Nereus (using helium), and General Atomics **GT-MHR** (using helium) have favored simplification of the high-temperature gas-cooled reactor concept as much as possible, down to practically a reactor and a gas turbine linked together with the reactor using a right-sized, inherently safe core with no water used in the plant design. The GT-MHR, however, is large enough that it has a system for residual heat removal using convected air. The engineering mistakes made and lessons learned at Fort Saint Vrain delayed the HTGR – a very safe, affordable, highly adaptable, efficient, scalable, and perhaps immensely important nuclear technology - by decades due to the technology's buggy performance in this commercial test.

4.2.8 DRAGON

Dragon was a high temperature gas cooled reactor at Winfrith in England operated by **UKAEA**. Its purpose was to test fuel and materials for the European high temperature reactor programme, and was built and managed as an **OECD/NEA** international project.

- **Organization for Economic Co-operation and Development** (**OECD)**,
- The **Nuclear Energy Agency** is an intergovernmental multinational agency that is organized under the Organization for Economic Co-operation and Development. Originally formed on 1 February 1958 with the name **European Nuclear Energy Agency (ENEA)** (the United States participated as an Associate Member), the name was changed on 20 April 1972 to its current name after Japan became a member.

Dragon used helium gas as coolant and coated particle fuel. The Dragon complex consists of the reactor and various buildings constructed in the late 1950s/early 1960s to support operations. The reactor is presently in care and maintenance. Construction started in 1959 and was completed in 1962. Operation started in 1965 with a power output of 20MW. The reactor operated until 1976 and was partially decommissioned in 2005.

4.2.8.1 INTRODUCTION

The Dragon Reactor Experiment in Winfrith/UK was a materials test facility for the early HTR projects like the German THTR pebble-bed reactor and the American Fort St.Vrain prismatic NPP. It was built and managed as OECD/NEA International Joint Undertaking. The DRAGON reactor operated successfully between 1964 and 1975, irradiating an increasing variety of experimental and prototype coated particle fuel as well as testing technological components and structural materials.

4.2.8.2 MAIN CHARACTERISTICS AND DESIGN OF THE PRIMARY CIRCUIT

The main characteristics of the primary circuit were:

- Thermal power 20 MW
- Coolant Helium
- Coolant pressure 20 bar
- Reactor inlet temperature 350°C
- Reactor outlet temperature 750°C
- Coolant flow rate 34,500 kg/h
- Max. temperature at fuel element surface 1000°C
- Average heat flux at fuel element surface 24W/cm²
- Thermal power removed by natural convection 950kW

The design and construction of its primary heat removal system was governed by the following requirements:

- The helium coolant in the primary circuit had to be conducted on a flow path that avoided any contact of the pressure envelope with gas from the hot leg.

- The reactor inlet temperature should on one hand be low enough to prevent extreme creep of the material, but on the other hand should be high enough to limit radiation damage to a minimum.
- In case of a total loss of electrical power to the circulators and pumps the decay heat of the core had to be removed by natural convection. This requirement led to the design with six rising parallel coolant branches, which at their top carried the primary heat exchangers and the circulators.
- In order to accommodate the complete primary circuit within the bioshield, the heat exchangers had to be as compact as possible; as these primary heat exchangers also contained moving parts (by-pass valve) , they should also remain accessible for maintenance.
- In the event of a leak or a pipe rupture within the heat exchanger, radioactivity must not leave the containment; the amount of water entering the core had to be minimized to prevent a major chemical reaction with the red-hot graphite.

4.2.8.3 LAYOUT OF THE PRIMARY CIRCUIT

The reactor pressure vessel is a long, bottle-shaped steel construction divided vertically into two parts by the main shield plug. The upper part contains the fuel handling area with the charge machine, the main entry valve and the control rod drives. This part of the vessel was under full reactor pressure, but kept at 130°C during reactor operation by cool helium from the coolant purification system.

The reactor and reflector rest on the core bedplate in the lower (and wider) part of the pressure vessel. The hot reactor outlet plenum is constituted by the space between the main shield plug and the top of the core. In order to permit sufficient natural convection for the removal of decay heat in case of total loss of electrical power to circulators and secondary pumps, the six primary heat exchanger and gas circulators were placed on rising parallel coolant ducts.

4.2.8.4 MAIN CHARACTERISTICS

The heat produced by the 37 prismatic fuel elements is transmitted to a helium flow of 34, 500 kg/h; in normal operation the average core outlet temperature is 750 °C. This hot coolant is divided among the six coolant loops and brought to the six primary heat exchangers through the insulated inner conduct. At the full power of 20 MWth the outlet temperature of the heat exchangers is 300 °C; the helium enters the one-stage main centrifugal circulator, which thrusts the helium flow back to the pressure vessel through the concentric outer duct of the coolant branch. Due to the compression and the heat transferred from the inner duct, the helium enters the annular inlet plenum in the pressure vessel at 350°C. The returning helium is then redistributed to cool the absorber rods, the shield plug, the reflector and the core bedplate, all of which are exposed to a strong neutron flux. The helium enters the core at 370°C and flows upward between the fuel rods to the outlet plenum.

4.2.8.5 THE PRIMARY HEAT EXCHANGERS/ THE HEAT REMOVAL SYSTEM

In the six primary heat exchangers the thermal energy was transferred from the helium on the shell side to water inside the tube bundle. In order to prevent massive injection of water onto the hot graphite core in case of a tube failure, the pressure of the water in the secondary was only 15.8 bar, whereas the helium pressure was 20 bars. This may not be representative of a power reactor, but Dragon was not intended to drive a steam turbine.

The water enters the primary heat exchangers at 200°C and leaves it with a steam content of 16,4 wt%. There are six completely separate secondary water loops each with its own secondary heat exchanger/ condenser. The latter are also equipped with emergency cooling tube bundles. The main tertiary circuit operated at a pressure of 13 bar ; inlet – and outlet temperatures at the secondary heat exchanger are 50°C and 187°C respectively. In normal operation a bank of dry fin-fan coolers outside the containment cooled the tertiary water. In the case of a complete power failure, the decay heat of the core was evacuated by natural convection to the secondary loops and from there equally by natural convection via the emergency cooling tube bundles in the tertiary heat exchanger to a series of air-cooled shut-down coolers on outside of the container wall.

4.2.8.6 THE MAIN PRIMARY CIRCULATORS

The circulator was a single stage centrifugal type blower driven by a 3-phase induction motor of about 75 kW fed by a variable frequency alternator from a Ward-Leonard set giving continuous variation of speed over the range of 1 300rpm to 12 000 rpm. The rotor shaft carried the impeller at one end and a gas lubricated thrust bearing at the other. This shaft was slightly off vertical to compensate for thrust and it was supported by two radial gas bearings.

The complete unit including the motor is sealed into the primary circuit. No moving parts intersect the circulator casing which is water cooled on the outside in the region of the driving motor. The principal dimensions of the circulator are:

- Rotor shaft diameter: 100 mm
- Impeller overall diameter: 400 mm
- Distance between centres of bearings: 576.6mm

The circulator casing is fitted with thermal insulation around the concentric gas inlet and outlet ducts as well as around the impeller. Due to the water cooling the temperature at the bearings and in the electric motor does not exceed 95°C, although the impeller operates at 350°C.

Gas bearings were adopted for the circulators since they provide a simple means of making the primary circuit leak tight. These gas bearings have certain peculiar characteristics, which had to be studied extensively and required special precautions:

The hydrodynamic lubrication at operating pressure requires a minimum speed of rotation before it will support the weight of the circulator. Therefore the circulators were not routinely run below 1 300 rpm.

Dry friction with oxygen-free helium in the bearings could lead to damage; this situation is avoided by pressing helium as jacking gas (hydrostatic lubrication) during starting and stopping. As a long the time would be taken for the circulator to stop spinning, electric braking through the motor is applied, once the shaft speed is below 1,100rpm primarily to economize on jacking gas.

4.2.8.7 TEMPERATURE CONTROL OF THE PRIMARY CIRCUIT

A fundamental design objective of the pressure vessel and external parts of the primary circuit was to assure that the pressure-barriers were not in contact with coolant from the hot leg of the circuit. During operation helium returning from the heat exchangers at 300–350 °C cooled down all external parts of the heat exchanger branches, the heat exchanger casings and the reactor pressure vessel under the shield plug. The returning coolant stream also kept the moderators, the absorber rods, the core shroud and the core bedplate from overheating. The helium coolant flowed from the six branches to the core inlet plenum at the bottom of the RPV. The stream of relatively cold (50 - 100°C) helium (7,8 g/s) returning from the coolant purification circuit flowed into the upper part of the pressure vessel. This relatively small flow of helium cooled the upper vessel and shield plug itself before continuing downward through an annular section along the pressure vessel wall.

Due to the combined effect of cooling the inside wall with helium and transferring heat to water-cooled thermal shield all around the lower part, the temperatures in the vessel and the external walls of ducts and casings do not exceed 350°C during normal operation In the case of a complete power failure the reactor would trip and only the latent heat and the decay heat would have to be evacuated. The natural convection flow would amount to about 3% of the full circulator flow.

4.2.8.8 PRIMARY COOLANT PURIFICATION

When the helium purification plant for Dragon was designed, a relatively high release of fission products from fuel elements was anticipated. The individual purging of fuel elements was therefore required in order to prevent FP build-up and plate-out in the primary circuit. The invention and development of the coated particle by the Dragon Project resulted in release rates amounting to a small fraction of the originally forecast activity. Normally the purification by-pass stream would be drawn from the purge lines at the bottom of each fuel element. This feature provided an individual sampling facility for each fuel element, which gave reliable data about the rate and nature of fission product release for the various fuel element designs and operating conditions. The purification flow from the purges (total 7.8g/s) turned over the helium inventory approximately every 6 hours. All components except the control valves of this system were located in a series of shielded vaults inside the main containment. Starting from the purged fuel elements, the main purge gas stream is collected in a manifold under the core support grid and routed via a pre-cooler to a set of water-cooled, charcoal filled fission product delay beds. The pre-cooler reduced the helium temperature from 350°C to 100°C; the delay beds removed the decay heat (max 70kW) of short lived fission products and their daughters and cooled the helium to about 35°C. Each of the five delay beds contains

approximately 3 m³ of charcoal in a series of “U”-tubes surrounded by cooling water. Four of the five delay beds were normally used in parallel, the fifth acting as spare. They were designed to delay xenon for 200 h and krypton for 15 h, thus eliminating plate-out of longer-lived daughter isotopes. Downstream of the delay beds, the gas passed to the (chemical) purification plant. Three basically identical trains of purification plant were provided in parallel, permitting several duties of coolant cleanup to be performed simultaneously, while one is undergoing regeneration. The function of the purification system was to remove both the chemical (H_2, H_2O, CO, CO_2, CH_4, N_2) and radioactive impurities, leaving only extremely pure, inactive helium to be returned to the primary coolant circuit. As the precision of the release measurement for the experiments depended on the activity background level, the designers and operators aimed at complete decontamination of the gas returning from the purification plant. A coolant analysis during routine operation measured the following impurities: (H_2, H_2O, CO, CO_2, CH_4,N_2)

Each purification train consisted of a high temperature section or hot plant followed by a low temperature section. In the hot plant, carbon monoxide and hydrogen were oxidized by a bed of CuO at about 350°C, while any oxygen present were removed by pure copper at the end of the bed. In the low temperature section the gas was progressively cooled down to –180 °C in a freezer heat exchanger, on which H_2O and CO_2 plate out as solid deposits. A small cold delay bed, in which charcoal – filled thimbles were surrounded and cooled by boiling liquid nitrogen, then removed all remaining radioactivity except 85-Kr. Impurities leaving the cold delay bed were reabsorbed on 300 liters of charcoal in a cold absorption trap (this latter trap generated no decay heat; it could therefore be kept cool by the gas from the preceding krypton delay bed). The purified gas passed through a tube coil immersed in the boiling liquid nitrogen of the cold delay bed and was then used to cool down the incoming gas in the freezer heat exchanger. This raised the purified helium stream to room temperature before it was returned to the reactor by small gas bearing centrifugal circulators running at 24,000 rpm.

4.2.8.9 HELIUM MANAGEMENT

The helium inventory of the Dragon primary coolant system was about 355kg of which, during operation 175 kg were kept in reserve stores and the dump tanks. Of the other half, 68 kg were flowing in the main heat removal circuit, the rest slowly moved through the Fission Product Removal Plant (36kg), the Helium Purification Plant and the Transfer chamber at the top of the Reactor Pressure Vessel. The heat sink contained the remaining 10kg.

For re-fuelling, the reactor was shut down and the coolant from the primary circuit was pumped back into the storage and dump tanks before the vessel was opened to air at atmospheric pressure. Helium losses of 20-30 kg were registered at each re-fuelling shutdown due to this practice. The other main reason for helium losses was the frequent sampling of helium for the monitoring of experiments, which did not exceed 1kg per

operating day. Maintenance work and “spillage” during insertion and operation of experimental probes and general leakage were considered “unaccountable losses”; these on average amounted to 0.2 kg/day or 0.12% of the circulating inventory. In 1974 however these unaccounted losses rose to 2kg/day. After months of searching, the leakage was traced to small-bore stainless steel piping in the Hot Purification System. The leaks, almost invisible pores and crevices in otherwise healthy lengths of pipe were caused by chloride corrosion: all leaks occurred exactly at sections marked with color-coded PVC tape during the commissioning phase. During normal operation some of these tubes reached at temperatures between 80°C and 120°C. At this temperature the tape decomposed slowly, leaving gaseous HCl trapped on the tube surface. Once the 200 tape markings were removed and all suspect piping replaced, the average unaccounted helium losses dropped to 0.2 kg/day.

4.2.9.10 DESIGN ACCIDENTS

The rupture of the primary circuit by a complete separation of one of the main coolant branches was postulated to verify the effectiveness of the containment. In this **Maximum Credible Accident** (MCA) the usual assumptions concerning release of radioactivity from the core were applied: 100% of gas, 100% of volatile FP (fission products) and 25 % of all other FP are released from the core to the containment. In this unlikely case the dose rate at the perimeter fence (about 100m from the containment center) would be 150 mr/h. The release of helium from the circuit and the direct transport of heat from the core to the gas mixture in the containment would result in an increase of the containment pressure of 0,16 bar (2.3psi) at a gas temperature of 85°C.

If a tube failure in the one of the heat exchangers were to release the entire capacity of one secondary circuit, 320kg of water might react with the graphite core, the water gas ($CO + H_2$) resulting from a complete reaction would raise the pressure in the containment by 0,625 bar, but would not breach the containment. The really dangerous case would arise, if the water gas on escape from the primary circuit could ignite. After combustion, the remaining gas mixture would have reached a temperature of 575 °C and a pressure of 2.2 bar. It could however be shown, that even under conservative assumptions there could not be sufficient water gas in the mixture to ignite.

The case of a complete loss of electrical power to all circulators and pumps the primary, secondary and tertiary circuits were designed to operate by natural convection. As the reactor would immediately scram, only the latent heat and the decay heat from the core had to be removed. When the helium purification plant for Dragon was designed, a relatively high release of fission products from fuel elements was anticipated. The individual purging of fuel elements was therefore required in order to prevent FP build-up and plate-out in the primary circuit. The invention and development of the coated particle by the Dragon Project resulted in release rates amounting to a small fraction of the originally forecast activity. Normally the purification by-pass stream would be drawn from the purge lines at the bottom of each fuel element. This feature provided an individual sampling facility for each fuel element, which gave reliable data about the rate and

nature of fission product release for the various fuel element designs and operating conditions. The purification flow from the purges (total 7.8g/s) turned over the helium inventory approximately every 6 hours. All components except the control valves of this system were located in a series of shielded vaults inside the main containment. Starting from the purged fuel elements, the main purge gas stream is collected in a manifold under the core support grid and routed via a pre-cooler to a set of water-cooled, charcoal filled fission product delay beds.

The pre-cooler reduced the helium temperature from 350°C to 100°C; the delay beds removed the decay heat (max 70kW) of short lived fission products and their daughters and cooled the helium to about 35°C. Each of the five delay beds contains approximately 3 m³ of charcoal in a series of "U"-tubes surrounded by cooling water. Four of the five delay beds were normally used in parallel, the fifth acting as spare. They were designed to delay xenon for 200 h and krypton for 15 h, thus eliminating plate-out of longer-lived daughter isotopes.

Downstream of the delay beds, the gas passed to the (chemical) purification plant. Three basically identical trains of purification plant were provided in parallel, permitting several duties of coolant cleanup to be performed simultaneously, while one is undergoing regeneration. The function of the purification system was to remove both the chemical (H_2, H_2O, CO, CO_2, CH_4, N_2) and radioactive impurities, leaving only extremely pure, inactive helium to be returned to the primary coolant circuit. As the precision of the release measurement for the experiments depended on the activity background level, the designers and operators aimed at complete decontamination of the gas returning from the purification plant.

The design of the pressure vessel and the six heat exchanger/circulator branches permitted a safe and reliable heat removal under all conditions. The fission product removal and purification system kept the helium in the circuit extremely clean without releasing active waste. Among the postulated accidents the simultaneous breach of the primary circuit and a primary heat exchanger tube was considered the most dangerous, as it resulted in the discharge of inflammable water gas into the containment. The concentration of CO and H_2 in the containment however would not exceed the limit of flammability.

The mechanical properties of the construction materials were not affected by irradiation or high temperature and the only one minor case of corrosion attack was recorded. The Dragon Reactor Experiment has shown, that international cooperation can be very effective and has proven, that the melt-down proof ceramic core, the coated particle fuel, the negative temperature coefficient, the non –corrosive coolant and the ability of natural convection decay heat removal of the HTR concept result in a clean and inherently safe reactor.

4.3 THE HIGH TEMPERATURE TEST REACTOR (HTTR)

The **HTTR** is a graphite-moderated gas-cooled research reactor in Oarai, Ibaraki, Japan operated by the Japan Atomic Energy Agency. It uses long hexagonal fuel assemblies, unlike the competing pebble bed reactor designs. HTTR first reached its full design power of 30 MW (thermal) in 1999. Other tests have shown that the core can reach temperatures sufficient for hydrogen production. JAERI (Japan Atomic Energy Research Institute) has undertaken the study of an original design concept of gas turbine high temperature reactor, the GTHTR300 (Gas Turbine High Temperature Reactor 300). The general concept of this study is development of a greatly simplified design that leads to substantially reduced technical and cost requirements for earlier technology deployment. Newly proposed design features on reactor core and gas turbine units enable the GTHTR300 to be an efficient and economically competitive reactor in 2010s. In the first phase of the core design, the four-year operation without a refueling was achieved by efficient use of replaceable burnable poisons under the condition that excess reactivity does not exceed 2.5% A k/k during the operation. The value of 2.5% is conservatively determined based on prevention of severe core damage in an all control withdrawal accident.

4.3.1 INTRODUCTION

Development of the high-temperature gas-cooled reactor (HTGR) has revived in many countries after a major setback in Germany that had led the HTGR development since 1960s. Especially in South Africa, the commercial reactor project, the PBMR (Pebble Bed Modular Reactor) was being developed for many years. Also, various practical design works have been performed in the Netherlands, Russia/USA and China.

In Japan, development of the HTGR technology has been conducted for over 20 years, and the HTTR (High Temperature Engineering Test Reactor) with outlet gas temperature of 950°C and thermal power of 30MW was constructed at Oarai research establishment in JAERI. Its first criticality was attained in 1998 and its full power operation was carried out in the first half of 2001. Operational data for establishing and upgrading the HTGR technology basis has been accumulated ever since.

In parallel to the development and successful operation of the HTTR, JAERI has undertaken the study of an original design concept of gas turbine high temperature reactor, the GTHTR300. The GTHTR300 fully based on the HTTR experience and design studies on future generation HTGR in Japan is expected to be the effective energy source in 2020s. A detailed investigation on core and plant design will be carried out from the pure economical point of view, and redundancy will be eliminated. In addition, basic technologies concerning a gas-turbine system will be established in various R&D.

Prior to the detailed study, a core and plant design concept of the GTHTR300 was proposed and roughly evaluated. The GTHTR300 consisting of a block type core has several advantages such as high thermal power and low pressure drop in the core comparing with those of a pebble bed type core. There are the important factors to improve plant efficiency and economy. On the other hand, a major drawback of the

block-type core is that it takes at least a few months to reload all fuels using a off-load refueling machine. In contrast, on-load refueling is available in the pebble bed type core although a refueling mechanism is more complicated than that of the block type core. In the new core design, the possibility of a refueling interval extending the maximum six years was investigated by installing **replaceable burnable poisons** (RBP) around the core. Due to the RBP, plant availability of the GTHTR has the possibility to become more than 90%, and excess reactivity is kept lower than 2.5% A k~ during the operation. A burn-up control by the RBP is a big advantage from the standpoint of safety, maintenance, safe guard, amount of waste disposals as well as economy.

Regarding the plant design, non-intercooled cycle was adopted for its simplicity, easy maintenance and cost advantage. In the plant layout, outlet helium gas from a compressor cools the Reactor Pressure Vessel (RPV) to keep its temperature lower than the temperature limit of existing low cost steel SA533. In previous designs, low cost steel ~ cannot be available because outlet helium gas of 550°C from a recuperator flows upward along the inner surface of the RPV. The SA533 steel has large cost advantage against new high temperature resistance material such as 9Cr-lMo steel. These original design features are described in detail below as well as the design philosophy of the GTHTR300.

4.3.2 DESIGN PHILOSOPHY OF GTHTR300

Simplicity and economy are the primary concerns. That does not mean that safety is the second concern. Higher level of safety than a current Light Water Reactor (LWR), the so-called severe accident free, is also our goal. A target for electricity cost is 4 Yen/kWh that is by almost two yen cheaper than that of a typical LWR in Japan. In order to reduce the capital, fuel and operation and maintenance (O&M) cost, innovative designs are adopted for the GTHTR300. The followings are major items.

1. The highest power not to lose inherent safety
2. 2. High bum up (average 120GWd/ton), long refueling cycle (the maximum six years) and low amount of waste disposals
3. Air cooling spent fuel storage
4. Conventional steel vessel for reactor pressure vessel
5. Non-intercooled cycle
6. Stand alone gas turbine generator
7. Horizontal gas turbine generator installation User requirement for the GTHTR300 was determined with the help of utilities such as Tokyo Electric Power Company (TEPCO) and The Japan Atomic Power Company (JAPCO), and industries such as Mitsubishi Heavy Industries (MHI) and Fuji Electric (FE). Table 1 shows the user requirement for the GTHTR300.

The major specifications of the GTHTR300 shown in Table 2 were determined to meet the user requirement.

User requirement for the GTHTR300

- Safety goal: Radioactive nuclides release shall be prevented by complete passive systems
- Site condition: Meet the site evaluation requirement for a current LWR Additionally, site evacuation shall not be necessary. replacement of LWR site or new site the same as that of a next generation LWR.
- Seismic condition
- Fuel cycle: fuel burn-up more than 100GWd/ton, High amount of weapon grade Pu shall not be produced
- Nuclear proliferation free
- Radiation protection
- Radioactive waste disposal
- Power level
- Life time
- Availability
- Inspection
- Inspection period
- Economy

Major specifications of the GTHTR300

- Reactor power 600MWth/unit × 4 units
- Outlet gas temperature 850°C
- Efficiency 45.4%
- Reactor core Block type
- Fuel cycle LEU once through cycle
- type UO_2 coated fuel particle
- enrichment less than 20%
- bumup average 120GWd/ton
- refueling once/6 years
- Safety system No active emergency cooling
- Turbomachinery Horizontal orientation
- Reactor pressure vessel Mn-Mo steel
- Radioactive nuclide retention Confinement

4.3.3 SAFETY OF GTHTR300:

4.3.3.1 DEFENCE-IN-DEPTH AND SEVERE ACCIDENT FREE

Defense-in-Depth is a basic philosophy for the GTHTR300 as well as a LWR. Various layers of requirement are used to develop a high level of safety. However, there are major differences between the GTHTR300 and the LWR philosophy. The LWR uses highly reliable, redundant and diverse passive or active safety layers. On the other hand, the GTHTR300 safety shall be kept due to inherent safety characteristics and potentially safe components. Severe accidents are defined as any conditions beyond design-base accidents, causing core damages with fission product releases to the environment,

although all severe accident sequences are very low in probability. The new safety philosophy is to avoid most accidents, and to achieve a probability of severe accidents at least two orders lower than current reactors. Even in the worst event, fuel temperature exceeding its failure limit and excessive fuel oxidation by air ingress can be avoided because of inherent safety features and the passive decay heat removal system. Demonstrable safety

Nearly full-scale worst accident simulation tests can be carried out to obtain licensing before commercial operations because safety assessment by analysis is not usually enough to convince the public and the regulators of trusting this safety concept. In current reactors no accident simulation tests are carded out before commercial operations although inspection and performance tests in normal condition are conducted. On the other hand, safety demonstration by accident simulation tests can, and shall be, requisite to obtain licensing in the GTHTR300.

4.3.3.2 MECHANISTIC SOURCE TERM

Mechanistic source term is used to estimate radionuclide releases for plant siting evaluation instead of non-mechanistic source term based on AEC document TID-14844, Japanese LWR or HTTR safety evaluation guideline. Initial failure of coated fuel particles in the HTIR in manufacturing is only 8×10^{-5} and no apparent failure was found in continuous irradiation tests up to 6.5% FIMA at Japan Material Testing Reactor (JMTR). In addition to these tests, data for long term integrity of the fuel, lift off and plate out behavior will be accumulated in the HTTR operation. Up to date, the 20 MW operation of the HTTR was already finished, and no evidence of additional failure was found. These are the reason why the mechanistic source term can be used for the plant siting evaluation. No containment vessel When the above mentioned mechanistic source term is used for the plant siting evaluation, the effective dose equivalent to whole body in the worst event can meet the dose guideline without the containment vessel. No containment vessel is necessary in the GTHTR300 due to salient fuel performance.

No need for offsite emergency evacuation and no damage on all offsite assets Offsite emergency evacuation is not necessary in the worst event selected for the plant siting evaluation. Furthermore, all offsite assets are kept intact and ensured.

4.3.3.3 PSA AND EVENT SELECTIONS FOR THE SAFETY EVALUATION

Full-scale PSA will not be forced for designing the GTHTR300. It will be used for the selection of postulated events. The categorization of the events to be evaluated is the followings, abnormal conditions " Anticipated operational occurrences (AOO)" and the events beyond AOO "Design basis accidents (DBA)". In addition to these traditional categories, the postulated severest event shall be evaluated among all beyond design basis accident (BDBA). An event with complete loss of forced coolant (depressurization accident) and simultaneous withdrawal of all control rods is selected among all very unlikely events such as anticipated transients without scram (ATWS), station blackout and multiple operator errors. The same worst event is used for the plant site evaluation.

4.3.4 PLANT DESIGN OF GTHTR300

4.3.4.1 CONVENTIONAL STEEL VESSEL

A key technology simplification made in the GTHTR 300 is a reactor pressure vessel (RPV) that can be fabricated of code certified, low-cost steel SA 533/SA508. This is made possible by a newly-conceived plant flow scheme in which coolant is circulated through reactor via a pair of leveled coaxial cross duct, thereby exposing the RPV, and all other vessels to temperature below the design limit and at which the behavior of the materials under irradiation is well understood. Use of SA533/SA508 should also drive down the cost per vessel by a large margin. As a result, despite the three vessel count, the overall vessel system is estimated to still cost less than potential vessel system designs of less vessel count but having to use high-temperature, more costly steels such 21/4Cr-Mo and 9Cr-IMo-V. The latter has yet been certified for reactor vessel construction. The large saving in material and ease fabrication transportation and erection of the vessels of greatly reduced size contribute to the cost advantage of the present vessel system design.

4.3.4.2 PLANT LAYOUT DESIGN

The plant design consists of three subsystem modules, the 600MWt prismatic core reactor, the stand-alone GTG and a heat exchanger, ITX. The three modules are contained in individual vessels and allocated to separate silos. Partitioning the large into properly sized subsystems and arranging them separately proves essential to effectuating and modular construction and maintenance. In the present design the three modules can be factory built and site erected essentially in parallel, with all necessary piping connected when construction is completed. The maintenance can be greatly simplified. The ability to not only construct but also service the plant in truly modular manner will positively impact the economy of the technology.

4.3.4.3 NON-INTERCOOLED DESIGN

The non-intercooled cycle loses plant total efficiency about 2% comparing with the intercooled cycle. However, it saves the capital cost almost 5% based on 92 DOE evaluation. When the capital cost is normally assumed to be two thirds of the electricity cost, the non-intercooled cycle gains approximately 3.3% for the electricity cost. The cost estimation has relatively high margin of errors, and it may be dangerous to evaluate the electricity cost based on roughly determined design specifications. However, if the intercooler is adopted, all water related problems happened in Fort. St. Vrain in USA and AVR in Germany would presumably happen in the GTHTR300. In the HTTR operation, nitrogen gas used for pressurizing water was stagnated in an air cooler, and deteriorated thermal performance of the air cooler. Besides the operational complexity, a sophisticated inspecting machine is necessary since heat transfer tubes composing the primary boundary in the intercooler shall be inspected according to a Japanese regulation. That makes the O&M cost higher. Considering all effects, the non-intercooled design superior to the intercooled design.

4.3.5 REACTOR CORE DESIGN OF GTHTR300

In the severe accident free concept, the maximum fuel temperature during a depressurization accident characterized by a primary pipe rupture plus possible reactivity insertion shall be lower than the temperature limit of 1600°C. Even in this worst accident, the reactivity insertion shall be limited to lower than 2.5% A k& when withdrawal of all control rods is postulated as the worst accident. On the other hand, in the case that this design limitation for the excess reactivity is adopted, the reactor cannot be operated without refueling because of lack of the excess reactivity. On-load refueling is not possible because it needs an innovative refueling machine available at the high temperature condition. A six-year operation without refueling needs the excess reactivity of 45% A k/k at the beginning of the core. The previously proposed way in which burnable poisons (BP) are installed in fuel blocks reduced the excess reactivity at the beginning of the core to 11% A k/k. However, it is still more than four times higher than the limit excess reactivity of 2.5 % A k/k. The peak excess reactivity during the operation is limited but kept relatively high enough to operate the reactor for the maximum six years. To meet this inconsistent requirement, the newly proposed way uses **replaceable burnable poisons (RBP)** around the reactor core and its replacement every two years. The excess reactivity recovers by the replacement of the burnable poison every two years and does not exceed the upper limit since the proper amount of burnable poisons are still placed in the core. This concept was proved by the calculation. The CITATION was used to evaluate the excess reactivity. The white circles show the calculated reactivity considering the replacement of the RBP. This analysis proved that the excess reactivity is maintained the reasonable level for operation but not so high for four years. The four-year operation with the reactivity of less than 2.5% A k& was achieved by the RBP. When the single rod withdrawal is postulated as the worst accident, the six-year operation is possible. However, the RBP is not efficient considering neutron economy, and the four-year operation is not enough for a economical perspective. In the detailed evaluation to be performed next year, the other way not fully dependent on the RBP will be investigated for reducing the initial and remaining enrichment.

4.3.6 DEVELOPMENT ITEMS AND SCHEDULE FOR THE GTHTR300

The detailed design phase started in FY-2001 and it took more than three years. The final economical target for the GTHTR300 is 4 Yen/kWh. Every innovative and sophisticated idea will be accepted, and every redundancy will be eliminated. As for the R&D, the development of a compressor and magnetic bearing for the gas turbine system started in FY-2001. These are key components for this plant and still have technical uncertainties such as aero dynamics in the compressor and control performance of the magnetic bearing.

Check & Review on the design by a special board consisting of members from utilities, universities, industries and the other national research laboratories will be performed every three or four months so that our design can meet the requirement from private sectors especially utilities. In addition to above development in Japan, a control and operational performance test using 1/3 scale whole gas turbine system is scheduled to

be carried out in accordance with the JEARI-MINATOM cooperation. A gas turbine test section will be connected the existing high temperature helium gas loop at OKBM in Russia.

4.4 THE PEBBLE BED MODULAR REACTOR

The **modular pebble bed reactor** (**PBR**) concept originated in Germany in the early 1950's and development of the technology began in earnest in 1956.

4.4.1 BACKGROUND

In the early 1950's, Dr. **Rudolf Schulten** - professor at RWTH Aachen University in Germany, was the inventor of the pebble bed reactor design. This reactor design called for the fabrication of silicon carbide-coated uranium granules into hard, billiard-ball-like spheres to be used as fuel for a new high temperature, helium-cooled type of nuclear reactor. The idea took root and in due course a 46 MW_th (megawatt thermal) experimental pebble bed reactor (the Arbeitsgemeinschaft Versuchsreaktor, or **AVR**) was built at the Jülich Research Centre in Jülich, West Germany. It operated successfully for 21 years but was shut down because the pebble fuel testing program came to a halt.

4.4.2 THE AVR

A 15 MW_e demonstration reactor, Arbeitsgemeinschaft Versuchsreaktor (**AVR**—roughly translated to *working-group research reactor* or *working-group experimental reactor*), was built at the Jülich Research Centre in Jülich, West Germany. The goal was to gain operational experience with a high-temperature gas-cooled reactor. The unit's first criticality was on August 26, 1966. The facility ran successfully for 21 years, and was decommissioned on December 1, 1988, in the wake of the Chernobyl disaster.

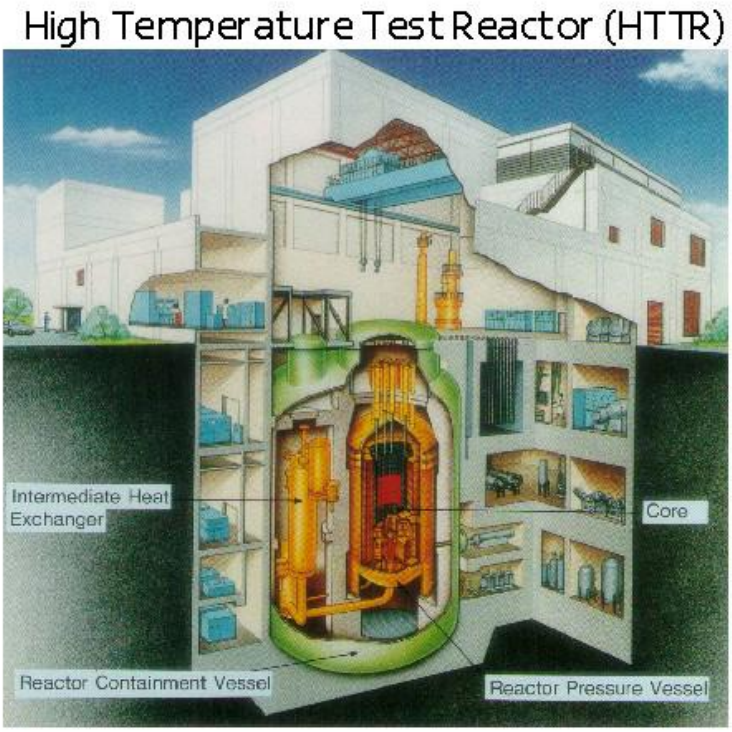
The **AVR** was originally designed to breed ^{233}Uranium from ^{232}Thorium. ^{232}Thorium is about 400 times as abundant in the Earth's crust as ^{235}Uranium, and an effective thorium breeder reactor is therefore considered valuable technology. However, the fuel design of the **AVR** contained the fuel so well that the transmuted fuels were uneconomic to extract—it was cheaper to simply use natural uranium isotopes.

The AVR used helium coolant. Helium has a low neutron cross-section. Since few neutrons are absorbed, the coolant remains less radioactive. In fact, it is practical to route the primary coolant directly to power generation turbines. Even though the power generation used primary coolant, it is reported that the **AVR** exposed its personnel to less than 1/5 as much radiation as a typical light water reactor.

4.4.3 THE THORIUM HIGH-TEMPERATURE NUCLEAR REACTOR

The **THTR-300** was a thorium high-temperature nuclear reactor rated at 300 MW electric (**THTR-300**). The German state of North Rhine Westphalia, in the Federal Republic of Germany, and Hochtemperatur-Kernkraftwerk GmbH (HKG) financed the THTR-300's construction. Operations started on the plant in Hamm-Uentrop, Germany in 1983, and it was shut down September 1, 1989. The THTR was synchronized to the grid for the first time in 1985 and started full power operation in February 1987 Whereas the AVR

was an experimental pebble bed high-temperature reactor (HTR) used to develop the pebble fuel the THTR-300 served as a prototype HTR to use the BISO pebble fuel.



The electrical generation part of the **THTR-300** was finished late due to ever-newer editions and licensing procedures. It was constructed in Hamm-Uentrop from 1970 to 1983 by Hochtemperatur-Kernkraftwerk GmbH (HKG) . Dr. Heinz Riesenhuber, Federal Secretary of Research at that time, inaugurated it, and it first went critical on September 13, 1983. It started generating electricity on April 9, 1985, however it did not receive permission from the atomic legal authorizing agency to feed electricity to the grid until November 16, 1985.

4.4.3.1 DESIGN CHARACTERISTICS:

- The **THTR-300** was a high-temperature reactor with a pebble bed core, consisting of approximately 670,000 spherical fuel compacts each 6 cm in diameter with Uranium-235 and Thorium-232 fuel kernels surounded with a graphite matix.
- The moderator and and core walls consisting of Nuclear graphite.
- The coolant in the reactor vessel was helium.
- The pressure vessel that contained the pebbles was pre-stressed concrete (This was the first time this had been used instead of a steel pressure vessel).

- The THTR-300's power conversion system was similar to the Fort St. Vrain reactor in the USA, in that the reactor coolant transferred the reactor core's heat to water.
- The thermal output of the core was 750 megawatts,
- The energy tranferred from the fuel pebbles to the helium coolant was then tranferrred, via a heat exchanger to water which then was used to generate electricity via a Rankine cycle.
- Because this system used a Rankine cycle, water could occasionally ingress into the helium circuit.
- The electric conversion system produced 307 megawatts of electricity.
- The waste heat from the THTR-300 was exhausted using a dry cooling tower.

On September 1, 1989 the THTR-300 was deactivated due to its cost and increased public scrutiny following both the Chernobyl accident and the THTR-300 fuel pellet event of May 4, 1985, in which a fuel pellet became lodged in a fuel feed pipe to the core. On October 10, 1991, the 180 meter high dry cooling tower, which at one time was the highest cooling tower in the world, was explosively dismantled and from October 22, 1993 to April 1995 the remaining plant was decommissioned.

From 1985 to 1989 the THTR-300 registered 16410 operation hours and generated 2891000 MWh according to a full-load working time of 423 days. By 1982 it was planned by a group of firms to proceed with construction of a HTR-500, the successor of the THTR-300, but up-rated to a thermal output of 1250 megawatts and an electrical output of 500 megawatts.

4.4.4 THE SOUTH AFRICAN PBMR

The historical process leading to South Africa's taking up the High Temperature Reactor technology which was originally developed in Germany, makes for fascinating reading. A key turning-point in the development of the **HTR** was the German government's decision in 1990 to discontinue all work on its **HTR**. This decision came just months after the basic **HTR** modular reactor design, which provided the take-off-point for the later **PBMR** development, had been officially licensed by Germany's Nuclear Safety Commission. The inventor of the **HTR**, Prof. Rudolf Schulten, died suddenly in April 1995, just two weeks after having signed a crucial agreement with South Africa for the transfer of the **HTR** technology.

In 1995 Eskom (The South African Power Company) initiated a concept design and costing exercise with local and overseas contractors. The results of the exercise over the next two years confirmed the basic validity of the **PBMR** parameters.

During 1997 and 1998 Eskom undertook extensive reviews (internal and external) of the project and began discussions with potential local and overseas partners. It was found

that the **PBMR** would be a cost effective option. Consequently, in 1998, the Eskom Council formally launched the PBMR as a priority project.

The South Africans emphasized the uniqueness of the safety features of the **PBMR,** underscoring the difference between so-called "passive" safety incorporated into the latest-generation light water reactor designs of the European EPR and the Westinghouse AP-1000 on the one side, and the "inherent safety" of the **PBMR** on the other. A crucial difference is that in the PBMR a meltdown of the reactor core is not only extremely improbable—as in the EPR and AP-1000—but literally *impossible.*

The basis for the safety case of the South African **PBMR** is the fuel element designed and fabricated by **NuKem** (A German manufacturing company). The fuel element is a completely ceramic pebble containing low enriched Uranium Oxide (**UO_2**) as fuel. The reactor core contains approximately 360,000 uranium fueled pebbles about the size of tennis balls. Each pebble contains about 9 grams of low enriched Uranium Oxide (**UO_2**) in 10,000 to 15,000 (depending on the design) tiny grains of sand-like micro-sphere coated particles each with its own a hard silicon carbide shell. The particle fuel consists of a spherical kernel of fissile or fertile fuel material encapsulated in multiple coating layers. The multiple coating layers form a miniature, highly corrosion resistant pressure vessel and an essentially impermeable barrier to release of gaseous and metallic fission products.

The **NuKem** design for the spherical fuel elements, based on encapsuling tiny particles of fissile fuel in high-temperature ceramic coatings, which is key to the inherent safety features of the PBMR, also provides an unrivaled packaging system for nuclear waste. The ceramic materials employed, remain stable and corrosion-proof for millions of years. In the context of the reactor fuel, the ceramic encapsulation prevents significant release of radioactive substances up to temperatures of 1,800° F or more, far above the maximum temperatures attained in the reactor, even in the "worst-case" accident scenarios.

Among other additional advantages of the PBMR design, is the stable dynamic behavior of the reactor, which is linked to its strongly negative-temperature coefficient. This means, that when the reactor temperature increases beyond a certain point, the efficiency of the fission reactions decreases rapidly, leading to the chain reaction "shutting off" by itself. This not only excludes the possibility of a dangerous runaway chain reaction, with overheating and other negative effects, but also means that the reactor's power output can be regulated essentially by the rate of cooling that the cooling system provides. The faster we cool it, the more power the reactor supplies. And the less we cool it, the less heat the reactor produces, as the fission reactions slow down automatically

The South African **PBMR** Project was cancelled in the first part of the twenty-first century due mainly because of lack of funding.

4.4.5 THE CHINESE HTR-10

In the middle 90's the Chinese bought the licensing rights for the German **AVR,** including a full set of plans and specifications. The design and construction of a 10 MWt high-temperature gas-cooled demonstration reactor (HTR-10), having fuel particles compacted with graphite moderator into 60mm diameter spherical balls (pebble bed) was then commissioned in 2000 by the Institute of Nuclear Energy Technology (INET) at Tsinghua University near Beijing. It reached full power in 2003 and has an outlet temperature of 700-950°C and may be used as a source of process heat for heavy oil recovery or coal gasification. It is similar to both the German **AVR** and also to the South African PBMR since they were both planned and built using the same plans and specifications. It was subject to a test of its intrinsic safety in September 2004 when as an experiment it was shut down with no cooling. Fuel temperature reached less than 1600°C and there was no failure.

Initially the HTR-10 has been coupled to a steam turbine power generation unit, but second phase plans are for it to operate at 950°C and drive a gas turbine, as well as enabling R&D in heat application technologies. This phase will involve an international partnership with Korea Atomic Energy Research Institute (KAERI), focused particularly on hydrogen production.

4.4.6 THE CHINESE HTR-PM

A key R&D project is the demonstration Shidaowan HTR-PM of 200 MWe (two reactor modules, each of 250 MWt) which is being built at Shidaowan in Shandong province, driving a single steam turbine at about 40% thermal efficiency. The size was reduced to 250 MWt from earlier 458 MWt modules in order to retain the same core configuration as the prototype HTR-10 and avoid moving to an annular design like South Africa's PBMR.

China Huaneng Group, one of China's major generators, is the lead organization in the consortium with China Nuclear Engineering & Construction Group (CNEC) and Tsinghua University's INET, which is the R&D leader. Chinergy (a 50-50 joint venture of INET and CNEC) is the main contractor for the nuclear island. Projected cost is US$ 430 million, with the aim for later units being US$ 1500/kWe. The licensing process is under way with NNSA and construction is likely to start early in 2009 with completion expected in 2013.

In March 2005 an agreement between PBMR of South Africa and Chinergy of Beijing was announced. PBMR Pty Ltd is has been taking forward the HTR concept (based on earlier German work) since 1993 and is ready to build a 125 MWe demonstration plant. Chinergy Co. is drawing on the small operating HTR-10 research reactor at Tsinghua University which is the basis of their 100 MWe HTR-PM demonstration module which also derives from the earlier German development.

Both PBMR and HTR-PM are planned for operation about 2013. The new agreement is for cooperation on the demonstration projects and subsequent commercialization, since both parties believe that the inherently safe pebble bed technology built in relatively small units will eventually displace the more complex light water reactors.

4.4.7 THE NUCLEAR TECHNOLOGY PEBBLE BED MODULAR REACTOR

Mr. Joseph Fournier, Senior Design Engineer of **Land and Sea Enterprises, Inc**. and more recently with **AscenTrust, LLC.,** has been associated with the Nuclear Technology of the Pebble Bed Modular Nuclear Reactor since 1969-1972 when Mr. Fournier was a graduate student at the University of Alberta, in Edmonton, and was involved in theoretical research in laser-plasma interactions. The basis of this research was the hope of providing a heating mechanism to increase the temperature of a plasma to the temperature required for fusion.

While on this research program Mr. Fournier chanced to read several papers from Germany which outlined the then new technology which is the subject matter of this historical document. The Germans were proposing the use of spherical pebbles completely encapsulated in graphit and containing micro-spherical pebble of silicon carbide coated, deuterium oxide fuel elements as the primary and secondary containment for the fission fragments which were being produced in the creation of electrical energy using fission as a source of heat.

Mr. Fournier left the Academic environment in 1972 because of the lack of opportunities in Canada for plasma physics researchers. In 2002 Mr. Fournier read an article in Scientific America entitled “Nuclear Energy’s Next Generation”. This article coupled with his contacts in the **Private Placement** world of financing for large-scale Energy and Oil Projects prompted him to re-ignite his passion for the Pebble Bed Reactor and continue his research on the Pebble Bed Modular Technology which had taken up some of his research time in the early 70’s.

Land and Sea Enterprises, Inc. has been investigating the **PBMR** option since that fateful day in 2002. The Senior Engineer has long realized that nuclear is the only possible non-hydrocarbon burning solution to the global energy production needs.

In th spring of 2002 **Land and Sea Enterprises, Inc.** and **Nuclear Technologies, Inc.** initiated a conceptual design and pricing exercise. The results of the exercise over the next two years confirmed the basic validity of the PBMR parameters. The **NTPBMR** project was born.

During the period 2002-2008 the Senior Engineer undertook extensive reviews of the project and created a conceptual design for the **NTPBMR.** In Houston, Texas, the pebble bed modular reactor (PBMR) has been under design and development for more than ten years.

Land and Sea Enterprises, Inc. has invested a substantial portion of its assets in the development of this pebble bed nuclear technology. We have been seriously involved with this technology since 2002, when the Senior Engineer of **Land and Sea Enterprises, Inc**., read an article in Scientific American written by the M.I.T. group outlining a reactor system which he had designed in the early 1970. The **Land and Sea Ent., Inc.** PBMR is modeled after the reactor developed in Germany in the early 1970.

ASCENTRUST LLC
The POWER of ENGINEERING

APPENDIX
INTRODUCTION TO
NUCLEAR POWER AND NTPBMR TECHNOLOGY

SECTION G: Introduction to NTPBMR

ASCENTRUST LLC
The POWER of ENGINEERING

AN INTRODUCTION TO
NUCLEAR TECHNOLOGY PEBBLE BED MODULAR REACTOR
(NTPBMR)
FOR
STAKEHOLDERS AND CONSULTANTS

SECTION ONE: PREAMBLE

Nuclear Power Industry activities can be broadly divided into fuel cycle activities, reactor activities and support activities. Fuel cycle activities include uranium mining and milling to produce ore concentrates (yellowcake), conversion of uranium ore concentrates into uranium hexafluoride, uranium enrichment, fuel fabrication, spent fuel reprocessing and nuclear waste management, and the design and construction of fuel cycle facilities. Reactor activities include reactor design, licensing and construction, reactor operation, maintenance and decommissioning.

AscenTrust, LLC. (**The Company**) and its strategic partners, own or control the intellectual property, the processes and the manufacturing facilities and control the engineering, procurement, construction and fabrication capabilities to design, license and build an American based infrastructure for the manufacturing of all the systems and sub-systems required to build a safe, clean **Nuclear Technology Pebble Bed Modular Reactor (NTPBMR)** electric generating power plants. The Company will mandate that 85% of the supply chain for the components of the project be manufactured in the **United States.**

The Company and its strategic partners, own or control the intellectual property, the processes and the manufacturing facilities and control the engineering, procurement, construction and fabrication capabilities to design, license and build an American based infrastructure for the manufacturing of all the systems and sub-systems required to build a carbon-dioxide charged, methane and oxygen fired closed cycle turbine generating plant.

Using only private funding, the company is working with the County Judges and Commissioners of Matagorda, Jefferson, Orange, Montgomery and Harris Counties, and will harness the support of the Governor of the State of Texas to design, license and build the main manufacturing plants required to fully implement the **NTPBMR Technology.** The primary fabrication facilities will all be situated in the State of Texas.

The environmentally benign aspects of nuclear power, compared to alternative energy sources are important to developing economies as well as Industrialized Nations. Our Nuclear Power Project can contribute significantly to the responsible use of natural resources found on the American Continent and create an energy production supply chain which is sustainable and has a very small carbon dioxide footprint. However, the Company and its nuclear industry partners are also aware of the serious safety and proliferation hazards associated with nuclear facilities and we are committed to developing the **NTPBMR** in a manner consistent with **NRC** and **IAEA** safety and non-proliferation standards. The Senior Engineer has already outline the design process for **Defence in Depth** to be used for the **NTPBMR.**

Both reactor and fuel cycle services rely upon a number of support activities, including consulting, legal services, parts manufacturing, fuel transportation and fuel supply fabrication, research and development (R&D) institutions (government, enterprise or university-based) and industry bodies. The Company will work with Dr. Gary Sorensen

and Mr. Howard Selman of **The Living Space Initiative** to flesh out the residential and commercial side of the support structures required for the successful implementation of this supply chain in the sixteen states belonging to the **Southern States Energy Board,** the epicenter of the project will be in Orange County, Texas.

One of the most attractive facets of the **NTPBMR** project is the number of high value jobs which we will be able to create across the Supply Chain of the Nuclear Fuel Cycle. We estimate that we will be able to create more than 100,000, direct, permanent, high value jobs in Engineering, Research and Development, Manufacturing, Construction, etc. The multiplication factor for these types of jobs is more than five so that we can expect to create over 500,000 permanent jobs all across the Southern States.

0.1. INTRODUCTION

The ramp-up in gasoline prices in the summer of 2008 gave national prominence, once again, to the issues of energy supply and demand. The crisis highlighted our dependence on fossil fuels for the production of this electrical energy. The energy ethos in the **U.S.** has been, for a large part of the history of its growth in the 20th Century : **Not in my back yard.**

Increased demand coupled with a strict regulatory environment has stopped the licensing and construction of new power plants. The "crisis" apparently came and went and was soon forgotten. What it did accomplish however was a more lively discussion, in the most liberal area of the **United States of America,** of the importance of supply, recognizing the ever increasing demand as we electrify. In this discussion of demand came the realization that approximately 20% of the nation's electricity was being generated by nuclear energy. This 20% also represents approximately 69% of the zero carbon footprint electrical energy production in the U.S.

The net consequence of a number of factors, such as a faulty deregulation schemes, were rolling blackouts due to lack of generation at any price. We, in Texas, will feel this consequence for the short term future. The Obama administration has given us clear indication that they intend to close down all coal-fired power plants. The construction of new plants fired by the use of hydrocarbons is frowned upon and the regulatory climate somewhat hostile, companies are leery in making generation investments.

On a different but somewhat parallel track, in terms of energy use versus planetary environmental health is the issue of the slight increases in low levels of carbon dioxide (**CO_2**) in the atmosphere over the last few years. The level of **CO_2**, in the atmosphere, ranges from 250 parts per million to 350 parts per million. This carbon dioxide becomes part of what the environmentally involved scientist call "greenhouse" gases. These greenhouse gases absorb light in the infrared and prevent the re-emission of photons in the lower bands of frequencies which allows the earth to cool itself. This absorbed energy gets trapped in the atmosphere and is causing an increase in the mean global temperature of the earth. Increased carbon dioxide emissions in the atmosphere have increased the amount of rhetoric, often vitriolic, in reference to the existence and implications of increasing greenhouse gases in our environment due to the burning of

these same fossil fuels. While the environmental ministers of nations from around the world seek to find ways to meet the 1992 Kyoto accords which call for reductions in CO_2 and other greenhouse gases to 10% below 1990 levels, the reality, 25 years later, is that CO_2 emissions have not decreased at all but increased by 10%.

As everyone involved with nuclear technologies know, one of the key advantages of nuclear energy is that it is essentially a greenhouse-gas-emission-free technology. Yet, at its most recent meeting in Copenhagen, Denmark, the **Conference of the Parties** COP 15, these same environmental ministers voted to specifically exclude nuclear energy from helping address the global warming problem.

Clearly, there is something wrong here since, in the United States, nuclear energy provided over 69% of the emission-free generation, far exceeding the 30% hydroelectric power. Solar and other renewable forms of energy provide the rest (~1%).

Concerns about global warming policies that might eventually lead to the inception of a CO_2 tax which will impair investments in coal-fired power plants, and coupled with attractive operating economics recently experienced in the production of electricity with the use of **Nuclear Power,** Public sentiment is slowly being led towards acceptance of **Nuclear Power** as a viable element of the energy production mix.

In the past 25 years, nuclear power plants have shown tremendous operational improvements and many have been up-rated to add generating capacity. Many of the existing nuclear facilities have applied and been approved for extensions to their operating licenses. Average capacity factors have increased from 66 percent in 1990 to about 90 percent in 2005, owing primarily to increased availability as refueling outages have been shortened from an average of 104 days to 38 days and to improved maintenance programs that have reduced forced outages.

Although existing nuclear plants have demonstrated high reliability and very low operating costs, the next generation of nuclear plants will almost certainly have higher capital costs than conventional fossil fuel units. However, interest in diversifying the fuel mix and the fact that nuclear power does not emit any **CO_2** have led to 10 proposals for new nuclear units, reflecting serious interest in reviving this technology as a base-load option.

Some of the project sponsors have already filed for **Early Site Permits**, and are expected to file for combined construction and operating licenses within the next few years, which could lead to construction beginning on some of the plants soon. The Energy Policy Act, EP Act 2005 also encourages new nuclear facilities with a combination of loan guarantees, production tax credits, and risk protections for initial project developers. The time horizon for new nuclear investments is such that these investments are not likely to contribute to upward rate pressures for the foreseeable future. However, utilities that are planning these units will incur some outlays, and future investments in the construction phase of their projects which are likely to be substantial in both size and risk.

For many years, nuclear energy, while arguably a non-CO_2 emitting energy source, has been judged to be unacceptable for reasons of safety, unstable regulatory climate, a lack of a waste disposal solution and, more recently, economics. In recent years, however, the nuclear industry has made a remarkable turnaround. While a number of older plants have been shutdown for largely economic reasons, the 104 operating nuclear plants' performance has increased to the point, that as an overall fleet, its capacity factor was 91% in 2014. This means that these plants were operating full power for over 91% of the year. This improvement in the last 20 years is essentially the same as building 23 new 1,000 Mwe plants in that time period, based on historical performance averages. In addition, all safety statistics, as measured by the **Nuclear Regulatory Commission**, have shown dramatic improvements as well. **The Three Mile Island** accident occurred over 30 years ago. The image of nuclear energy as an unsafe technology still persists. Yet the record is quite the opposite.

The utilities have not put in an application for a nuclear power plant since the mid 1970's. The reason for the lack of new orders was the high capital cost. When operating in a difficult regulatory environment, utility executives simply avoided new nuclear construction and went to the cheapest and fastest way to make on-line generation available, which was natural gas. Combined cycle gas plants were the generation source of choice for many years for those companies that needed to build plants.

Today, utility executives still do not have new nuclear plant construction in their future plans even though the regulatory regime has stabilized. Although the regulatory environment has stabilized the utility companies and still uncertain how the passive systems mandated by the **Nuclear Regulatory Commission** can be successfully implemented, within the budgetary constraints of competition with gas-fired electrical generation plants.

Nuclear plants are performing extremely well. Safety issues have been addressed with no new issues emerging and slow progress is being made to finally dispose of spent fuel at **Yucca Mountain**. What has happened is a consolidation of the utility and nuclear industry with some larger utilities purchasing existing nuclear plants from companies that do not want to be in the business.

To address the inevitable problem of replacing existing nuclear generation, utilities have chosen to re-license existing plants from the current 40 years to 60 years. Several nuclear plants have applied and received Nuclear Regulatory Commission approval to do so. These extensions will allow utilities to continue to use these plants as long as they are economic and continue to be safely operated. Unfortunately, we still don't see a rush to build new nuclear plant. One of the main reasons lies in the financial risk involved in the licensing and construction of a new nuclear plant. Combined with the uncertain costs associated with new nuclear construction and the low risk and cost of building a Combined-cycle, natural gas fired power plant, we do not see a rise in investment in nuclear in the next ten years.

This need for a new approach, in the construction of **Nuclear Power Plants** is the basis for the formation of **Nuclear Technologies, Inc.** to look into the production of a Prototype **PEBBLE BED MODULAR REACTOR** (PBMR).

The major challenge faced by the Nuclear Industry for the reintroduction of nuclear energy into the world energy mix, is the development of a nuclear power system that:

1. Does not include water as a coolant or a moderator.
2. Is competitive with other energy alternatives, such as natural gas. oil or coal.
3. A Nuclear Reactor system which can successfully go through a **LOCA (**Loss of Coolant accident**)**
4. Can address the issue of containment
5. Can address the issue of Terrorism
6. Has to address the issue of proliferation
7. Can address the issues of nuclear Waste

As the power of the **Global Warming** Lobby increases the pressure on politicians, including the President of the **U.S.,** increases for the **U.S.** to sign the **Kyoto Treaty.** If the U.S. signs on to the Treaty, we will see the adoption of a CO_2 emission tax as an associated penalty in the use of power generation facilities which produce carbon dioxide as a by-product of combustion of fossil fuels. The environmental imperative of nuclear energy is obvious. No greenhouse gases emitted, small amounts of fuel required and small quantities of waste to be disposed of.

Unfortunately, historically the capital costs of new nuclear plants is quite large relative to the fossil alternatives. Despite the fact that nuclear energy's operating costs in terms of operations and maintenance and, most importantly, fuel are much lower than fossil alternatives, the barrier of high initial investment is a significant one for utilities around the world. The associated regulatory risk makes the construction of a water cooled nuclear power plant a very distant possibility.

In order to deal with this challenge, the Senior Engineer of **The Company,** started the redevelopment of a technology that was originally invented, tested and prototyped in Germany in the 1970's and 80's. A pebble bed research and demonstration reactor operated at the Juelich Research Institute, in Germany, for over 22 years, demonstrating the soundness of the technology.

This **Pebble Bed Modular Reactor** technology is the central theme of this document because it is the technology which we at **The Company** have been working on for so long. Unfortunately, Germany has abandoned its nuclear program for all practical purposes but there is now a world wide resurgence of interest in the development of this technology. The Chinese, the South Africans, the group at M.I.T. and the Engineering group of **The Company** has been researching and testing this technology for many years.

The nuclear energy plant which we are developing is a modular, 110 Megawatt-electric (Mwe), high temperature, pebble bed reactor, using helium gas as a coolant and conversion fluid and gas turbine technology. The fundamental concept of the reactor is that it takes advantage of the high temperature and high pressure properties of the **Brayton Cycle**, using helium as a coolant. Use of the Brayton cycle in the production of electricity permit theoretical thermal efficiencies close to 50%.

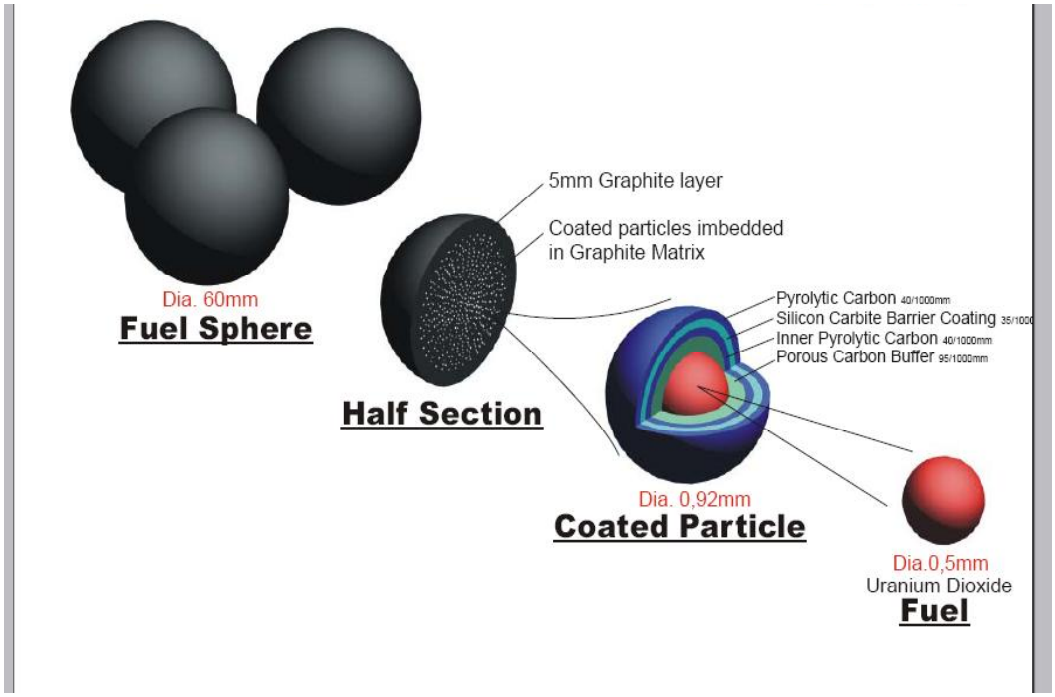
The PBMR utilizes an online refueling system that can yield capacity factors in the range of 95% because it does not have to be shut down to re-fuel. The pebble which form the fuel elements are constantly being re-circulated. Its modularity design concepts, in which all the systems and sub-systems of the plant can fit on specially designed railroad cars and flatbed truck and can be shipped from the factory, allows for a 3 to 5 year construction period, with expansion capabilities to meet merchant plant or large utility demand projections.

2. THE NTPBMR TECHNOLOGY

The **NTPBMR** technology consists of extensions of successfully designed, built and operated, helium cooled reactors built by the Germans in the 1970's and 1980's. The Principal characteristics of the **NTPBMR's** are;

2.1. THE FUEL ELEMENT

TRISO COATED FUEL ELEMENTS CREATED BY NUKEM FOR THE NTPBMR PROTOTYPE PROJECT



2.1.1. PROPERTIES OF TRISO COATED FUEL ELEMENTS

- The reactor core contains approximately 360,000 uranium fueled pebbles about the size of tennis balls. Each pebble contains about 9 grams of low enriched Uranium Oxide (**UO_2**) in 10,000 to 15,000 (depending on the design) tiny grains of sand-like micro-sphere coated particles each with its own a hard silicon carbide shell.
- The particle fuel consists of a spherical kernel of fissile or fertile fuel material encapsulated in multiple coating layers. The multiple coating layers form a miniature, highly corrosion resistant pressure vessel and an essentially impermeable barrier to release of gaseous and metallic fission products. This capability has been demonstrated at temperatures in excess of those predicted to be achieved under worst-case accident conditions in the **NTPBMR.**
- The micro-spheres are tri-coated with a porous layer of carbon, a layer of pyrolytic carbon and a layer of silicon carbide. The pyrolytic carbon layer absorbs the fission fragments and the Silicon Carbide coating retains these fission fragments and radioactive gasses within the micro-sphere. These micro-spheres are embedded in a graphite matrix material.
- The Uranium Oxide (**UO_2**) fuel micro-sphere has a melting temperature of approximately 2800^{o}C while the ceramic coating does not have a melting point and begins to degrade approximately at 2100^{o}C, and the degradation of the ceramic shell in the 50 or so hours required to empty the reactor would require temperatures in excess of 4000^{o}C. The temperature buildup in the core of the reactor in the event of a **Loss of Coolant Accident (LOCA)** is not expected to exceed 1600^{o}C

2.2 THE NUCLEAR ISLAND

2.2.2 PROPERTIES OF THE NUCLEAR ISLAND

A. On-line refueling capability: A unique feature of pebble bed reactors is the online refueling capability in which the pebbles are re-circulated with checks on integrity and consumption of uranium. This system allows new fuel to be inserted during operation and used or damaged fuel to be discharged and stored on site for the life of the plant. Overall burn-up is increased through this recycling. The online refueling capability allows for the extraction of all the nuclear fuel in the event of a **LOCA.** Extraction of all the fuel elements in the core in the case of a nuclear event will ensure that the fuel elements will remain intact through the nuclear event without the possibility the fuel pebbles will melt.

B. Graphite Moderator: The moderating environment of the **NTPBMR** is nuclear graphite. The **Reactor Pressure Vessel (RPV)** will house several hundred tons of Nuclear Graphite. The nuclear graphite has high thermal mass and will allow for passive cooling of the reactor core in the loss of coolant event.

C. Carbon Dioxide Emergency Core Fire Suppression System (ECFSS): The **ECFSS** is liquefied carbon dioxide. The carbon dioxide fire suppression system

will mitigate the risk of a graphite fire of the type which occurred at Windscale, in England, in the early days of the English gas-cooled Magnox program. The carbon dioxide will also act as a passive emergency core cooling system to extract heat from the core.

D. **Low Power Density:** The **NTPBMR** has very low power density in the core. Our preliminary design is for 3MWth per cubic meter. When one compares this figure with the 30 MWth power density in water cooled reactors, we can immediately see the increase in the level of safety in the **LOCA** event.

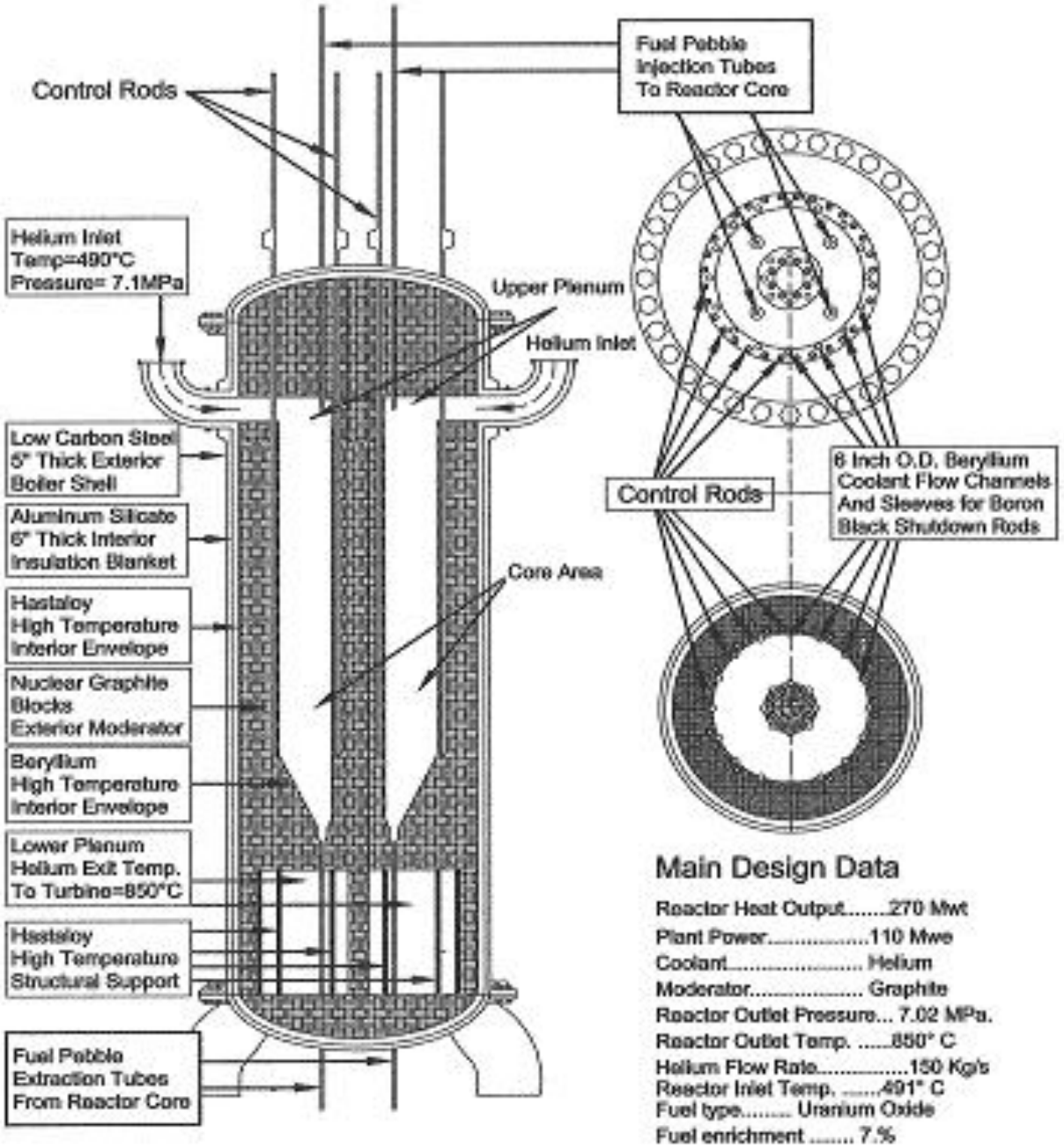
2.3 THE NUCLEAR HELIUM SUPPLY SYSTEM

Helium gas is used as the core coolant. Helium has a very small cross-section for neutron absorption, is inert and operating in a closed-loop, brayton cycle, single phase thermodynamic cycle which can power a turbine with high cycle efficiency.

- A Nuclear reactor using gas as the core coolant will eliminate completely the types of problem which occurred at Three Mile Island and Chernobyl, in their water-cooled nuclear reactor.
- Advances in gas turbine technologies will allow us to use helium as the coolant. Helium is an ideal cooling agent for a nuclear reactor since it is completely inert chemically, within the temperature ranges involved in a nuclear reactor vessel it remains in a single phase and it's neutron absorption cross-sections are quite low. and operating in a closed-loop, brayton cycle, single phase thermodynamic cycle which can power a turbine with high cycle efficiency.
- The inert nature of Helium will allow the filtration system of the **Nuclear Helium Gas Supply System (NHGSS)** to extract nearly 100% or radioactive fission products from the coolant. The **NHGSS** with filtration will reduce the radioactivity level in the turbine room by three orders of magnitude over existing water-cooled reactors.

The low radioactivity level in the turbine will ensure that an insignificant amount of radiation will be added to the cooling water which will return to our thermal heat sink or cooling pond.

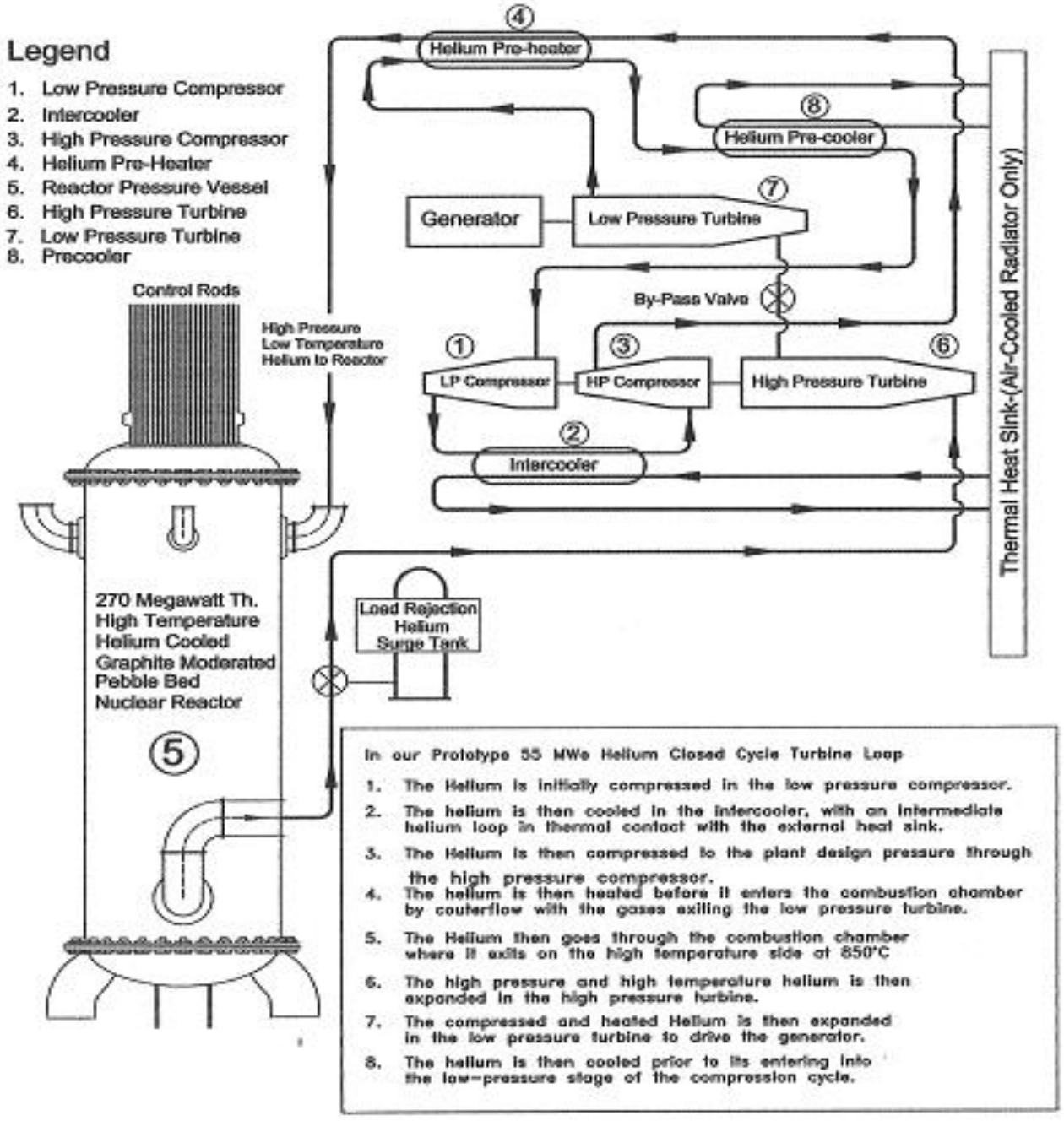
Nuclear Technology
Pebble Bed Modular Reactor
Preliminary Cross Section



Main Design Data

Reactor Heat Output.....270 Mwt
Plant Power.....110 Mwe
Coolant..... Helium
Moderator..... Graphite
Reactor Outlet Pressure... 7.02 MPa.
Reactor Outlet Temp. .....850° C
Helium Flow Rate.....150 Kg/s
Reactor Inlet Temp. .....491° C
Fuel type..... Uranium Oxide
Fuel enrichment 7.%

Nuclear Technology
Pebble Bed Modular Reactor
Flow Schematic For 55 MWe
Helium Closed Cycle Turbine Loop (Air-cooled)

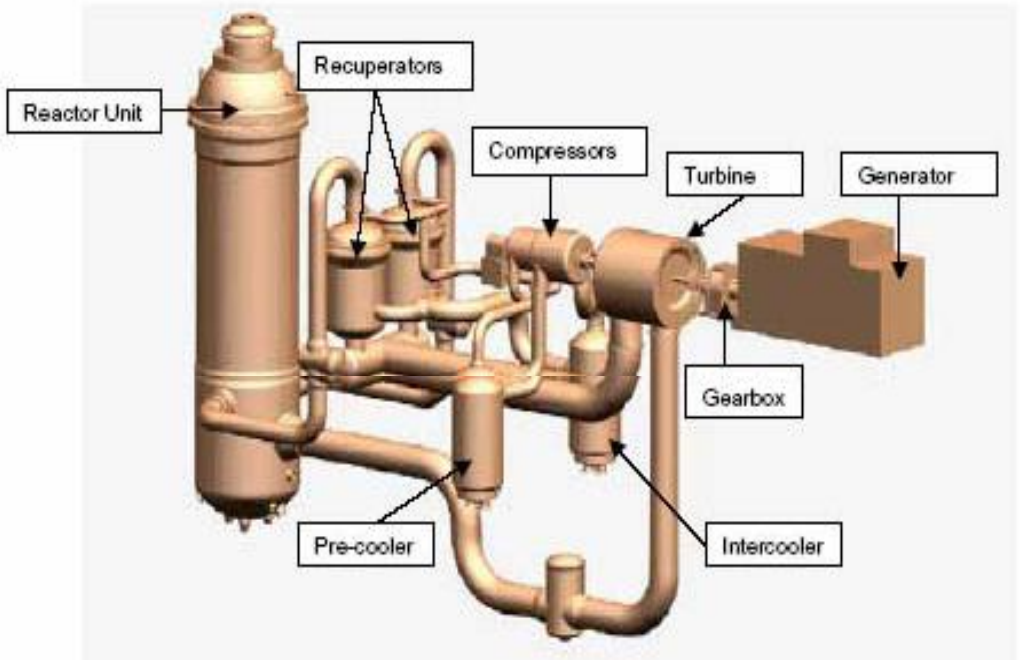


In our Prototype 55 MWe Helium Closed Cycle Turbine Loop

1. The Helium is initially compressed in the low pressure compressor.
2. The helium is then cooled in the intercooler, with an intermediate helium loop in thermal contact with the external heat sink.
3. The Helium is then compressed to the plant design pressure through the high pressure compressor.
4. The helium is then heated before it enters the combustion chamber by counterflow with the gases exiting the low pressure turbine.
5. The Helium then goes through the combustion chamber where it exits on the high temperature side at 850°C
6. The high pressure and high temperature helium is then expanded in the high pressure turbine.
7. The compressed and heated Helium is then expanded in the low pressure turbine to drive the generator.
8. The helium is then cooled prior to its entering into the low-pressure stage of the compression cycle.

4. THE THERMODYNAMIC CYCLE OF THE NTPBMR

ISOMETRIC VIEW OF THE NUCLEAR TECHNOLOGY PEBBLE BED MODULAR REACTOR

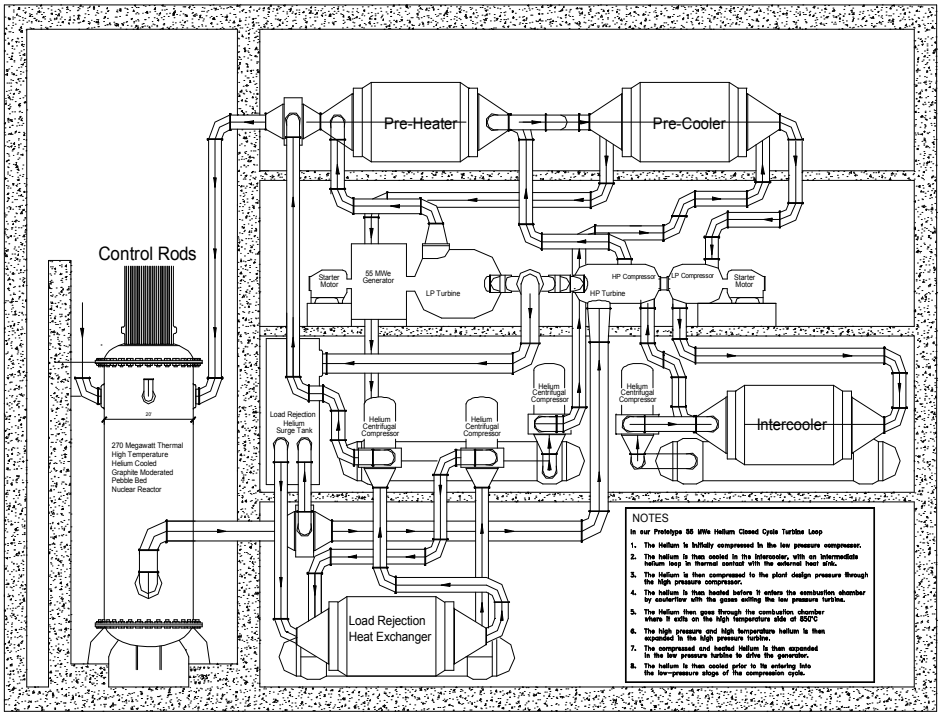


The Thermodynamic Cycle of the NTPBMR:

1. Fission in the Triso-coated micro-spheres creates kinetic energy through the recoil of the Uranium atoms which are split by the absorption of thermal neutrons.
2. The kinetic energy of recoil is transformed into thermal energy in the micro-spheres.
3. The thermal energy of the micro-sphere diffuses throughout the pebble and is transferred to the helium coolant by convective heat transfer.
4. The high pressure and high temperature helium is directed into the high pressure turbine. The high pressure turbine operates the compressors for the return of the helium to the reactor pressure vessel.
5. The helium is then directed to the low pressure turbine which operates the generator.
6. The helium is then cooled through a heat exchanger and the residual heat is exhausted to the atmosphere through an air powered radiator very much like an air conditioning unit on a house.
7. The cooled and compressed helium then re-enters the reactor pressure vessel

5. THE NTPBMR BALANCE OF PLANT

CROSS SECTION OF PLANT



COMPONENTS OF BALANCE OF PLANT

Each module produces 110MWe in two 55MWe turbine loops at shown in the cross-section above. The balance of plant consists of the following Systems and Sub-system which are important to the production of electricity and the safety of the technology in the event of a **LOCA.**

- **A.** The turbo-machinery:
- **B.** The on-line re-fueling system:
- **C.** Balance of Plant Control and Load Rejection equipment
- **D.** The heat exchangers
- **E.** The Carbon-Dioxide Fire Suppression System:
- **F.** Instrumentation and Control Systems
- **G.** The centrifugal compressors for secondary heat removal
- **H.** On site storage for fuel elements, helium and carbon dioxide

A. THE TURBO-MACHINERY:

All earlier **High Temperature Gas Reactors (HTGR)** installed steam cycles, because they were a mature technology at that time while helium gas turbine technology was not well understood. Use of the steam turbine cycles led to an indirect cycle with a steam generator coupled to the primary helium cycle which extracted heat from the core. The

use of the steam turbines introduces extra capital costs and increases the possibility of water ingress from the steam cycle through the heat exchangers and the water-cooled bearing assemblies.

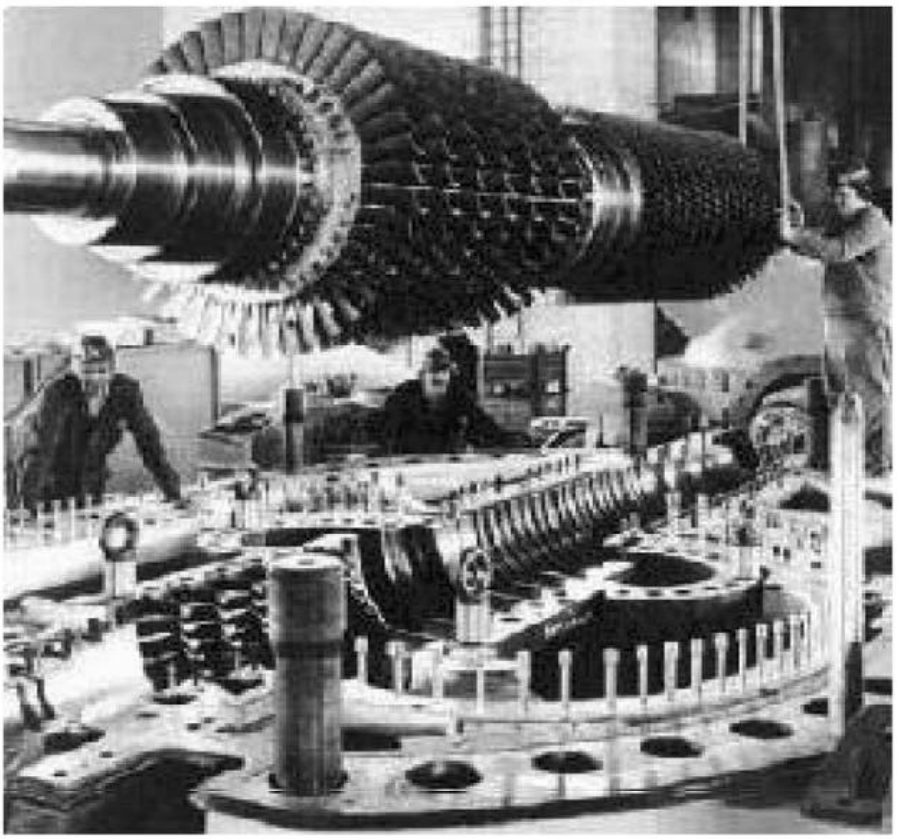
The **NTPBMR** technology proposes implementation of a helium gas turbine cycle rather than a steam turbine cycle. In our initial design we will even eliminate the water cooling on the exit side of the heat exchangers and will go directly to air cooling for the thermal heat sink.

This change leads to an increase in helium temperature, the direct cycle, and implementation of a modular concept with a compact, factory assembled helium cycle. The direct cycle enables elimination of the steam generator as well as the circulator. The size of the blades in a helium turbine is around 0.1 m whereas the blades are larger than 1 m in the steam turbine. As a result, the **NTPBMR** is economically competitive with large scaled water reactors even though the power level of the former is much lower than that of the latter. Therefore, the technology of the helium turbine cycle is essential in development of the **NTPBMR**.

The first and largest helium turbine to date was constructed in Germany in 1968. It was rated at 50 MWe at 750 C. Note that the largest helium turbine under design has an output of 400 MWe as GT-MHR. It was experimentally tested in a high-temperature, helium cooled nuclear reactor heat source generated by a fossil-fired heater with 53.5 MW for electricity generation (the HHT project) in 1968. The operating pressure for tests was up to around 1 MPa. The HHT project involved two experimental facilities. The first was an Oberhausen II helium turbine cogeneration plant operated from 1974 to 1988 by the German utility EVO (Energie Versorgung Oberhausen AG). The second facility was a high-temperature test plant (HHV) built in 1981. The main issues solved through these tests were material performance of the high temperature blades and disks and dynamic issues of rotor and magnetic bearings. The EVO was a milestone test facility that played an important role in the development of current **NTPBMR**.

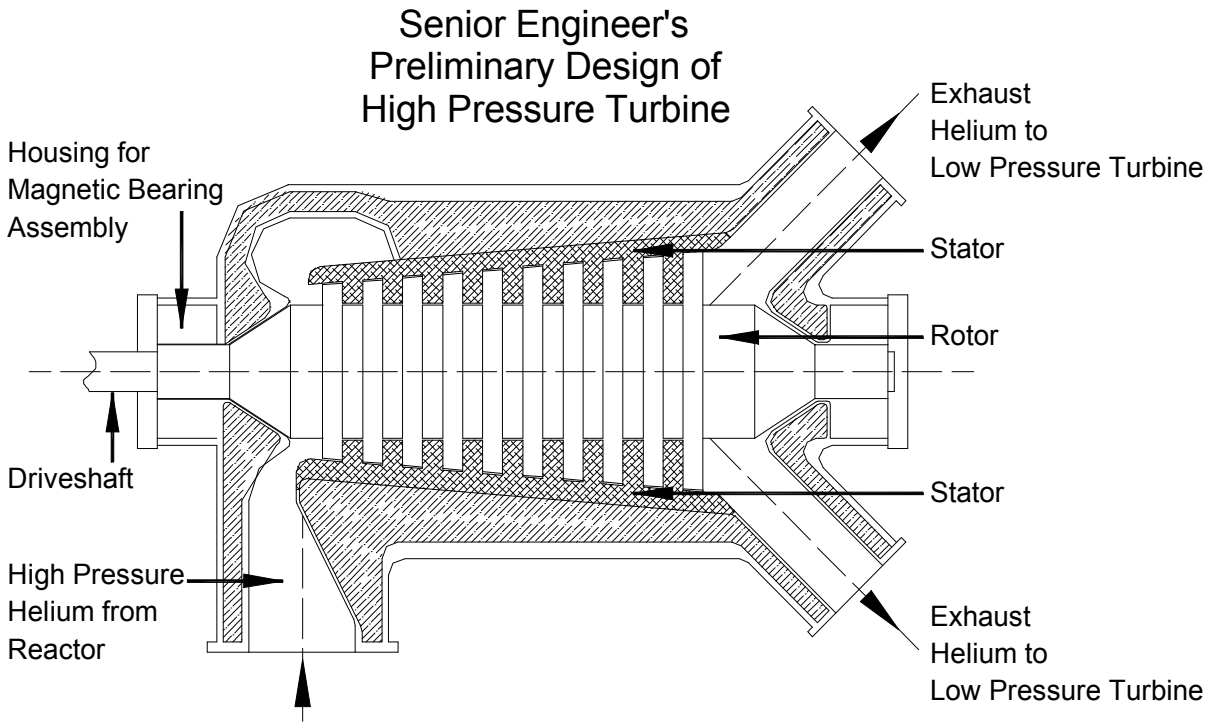
For the turbo-machinery, a two-shaft arrangement with an interconnected gear was selected. The high-pressure (HP) turbine, which has a rotational speed of 5,500 rpm, drives the low-pressure (LP) compressor and high-pressure (HP) compressor on the first shaft. The low-pressure (LP) turbine is directly connected to the generator with a synchronous rotational speed of 3,000 rpm. The mass flow rate of helium is 84.8 kg/s. A photograph of the HP turbine rotor is shown in the figure directly below, in figure 1. The HP turbine and the LP turbine have 7 stages and 11 stages, respectively. The HP compressor and the LP compressor have 15 stages and 10 stages, respectively, both with 100% reaction. The EVO facility was operated for approximately 24,000 hours. However, the maximum electricity power output of EVO was 30.5 MWe, which is much less than the design power.

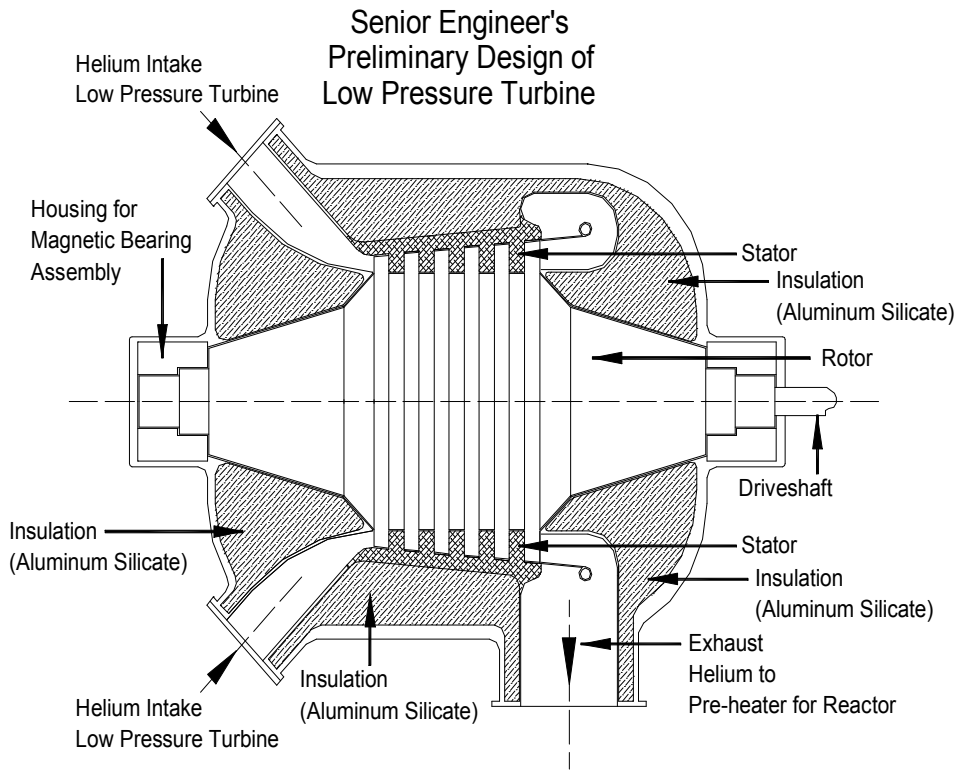
HIGH PRESSURE TURBINE ROTOR FOR OBERHAUSSEN II- 50MWE, HELIUM



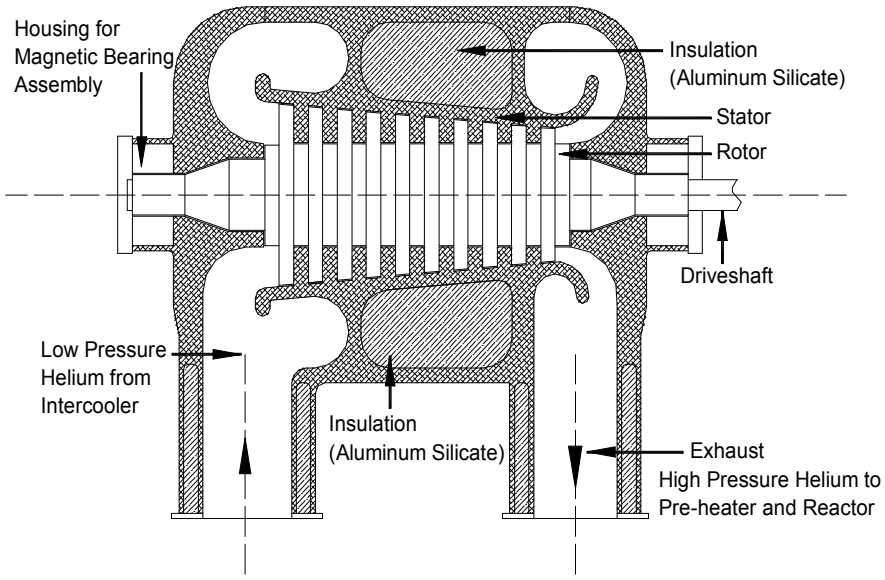
DESIGN CRITERIA FOR NTPBMR TURBOMACHINERY: For the turbo-machinery of the **NTPBMR Prototype,** a disconnected two-shaft arrangement has been selected.

- The high-pressure (HP) turbine which will have a design rotational speed of 7,200 rpm, drives the low-pressure (LP) compressor and high-pressure (HP) compressor on the first shaft.
- The low-pressure (LP) turbine is directly connected to the generator with a synchronous rotational speed of 3,600 rpm. The mass flow rate of helium is 184.8 kg/s. A preliminary design drawing of the HP turbine is shown in the figure directly below. The HP turbine and the LP turbine have 10 stages and 6 stages, respectively.
- The **NTPBMR** turbo-machinery is designed to operate up to 75MWe and will be optimized to operate with an output of 55MWe.

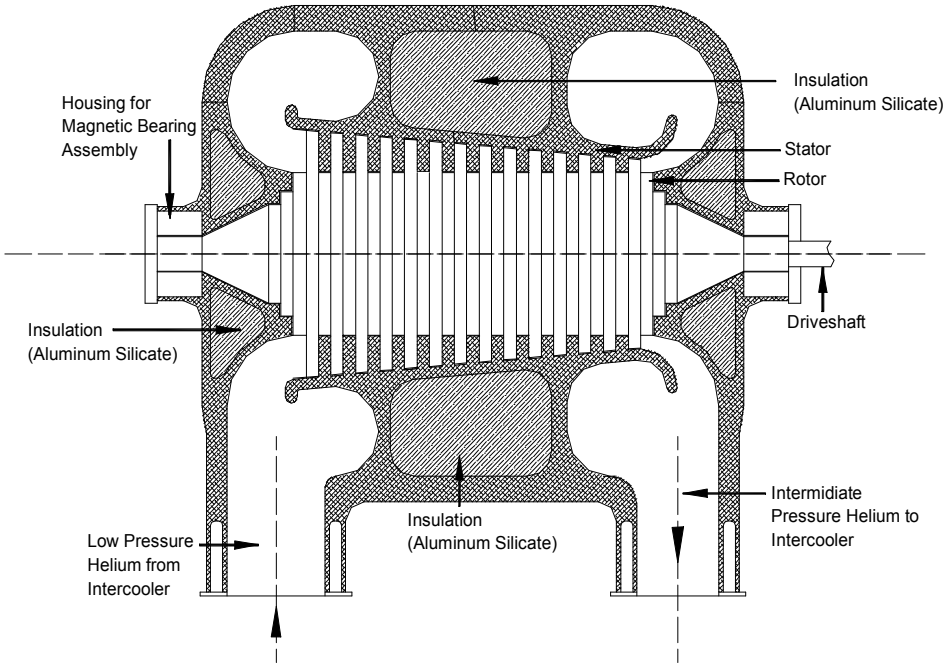

Senior Engineer's
Preliminary Design of
High Pressure Turbine
Housing for
Magnetic Bearing
Assembly
Exhaust
Helium to
Low Pressure Turbine
Stator
Rotor
Driveshaft
Stator
High Pressure
Helium from
Reactor
Exhaust
Helium to
Low Pressure Turbine


Senior Engineer's
Preliminary Design of
Low Pressure Turbine
Helium Intake
Low Pressure Turbine
Housing for
Magnetic Bearing
Assembly
Stator
Insulation
(Aluminum Silicate)
Rotor
Driveshaft
Insulation
(Aluminum Silicate)
Stator
Insulation
(Aluminum Silicate)
Exhaust
Helium to
Pre-heater for Reactor
Helium Intake
Low Pressure Turbine
Insulation
(Aluminum Silicate)

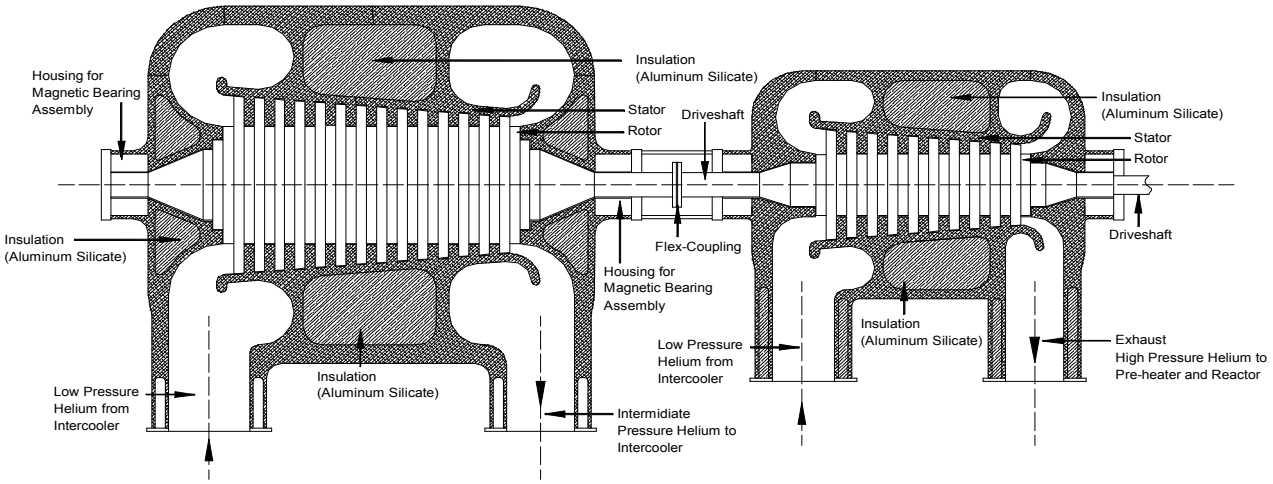
Senior Engineer's
Preliminary Design of
High Pressure Compressor



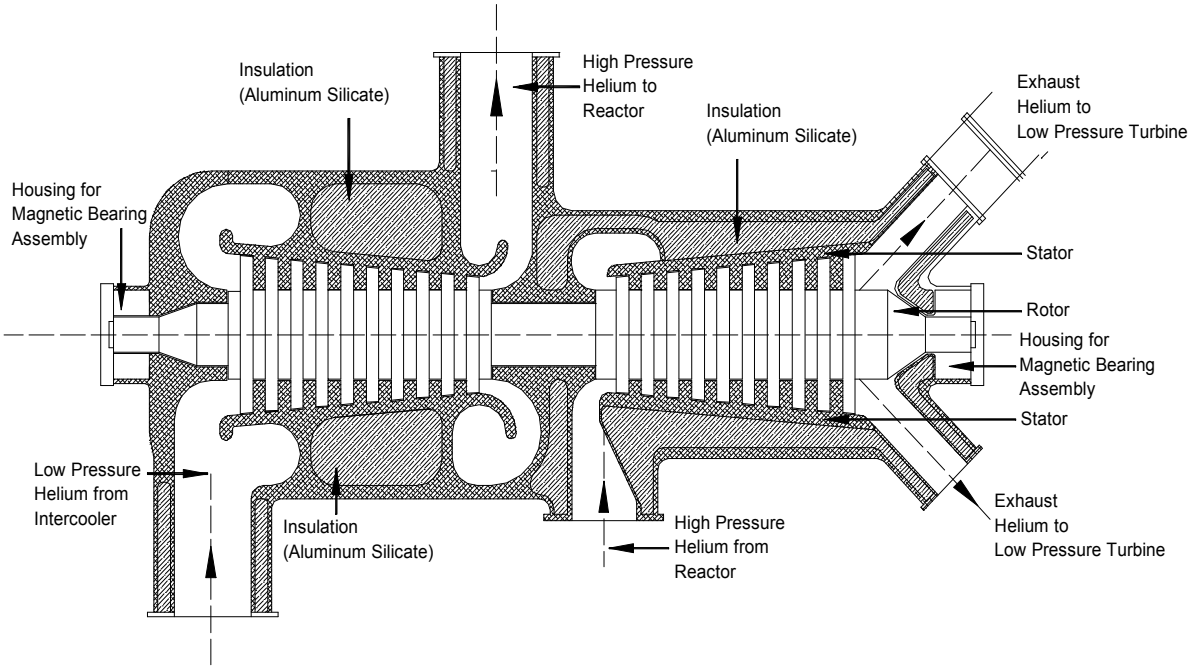
Senior Engineer's
Preliminary Design of
Low Pressure Compressor

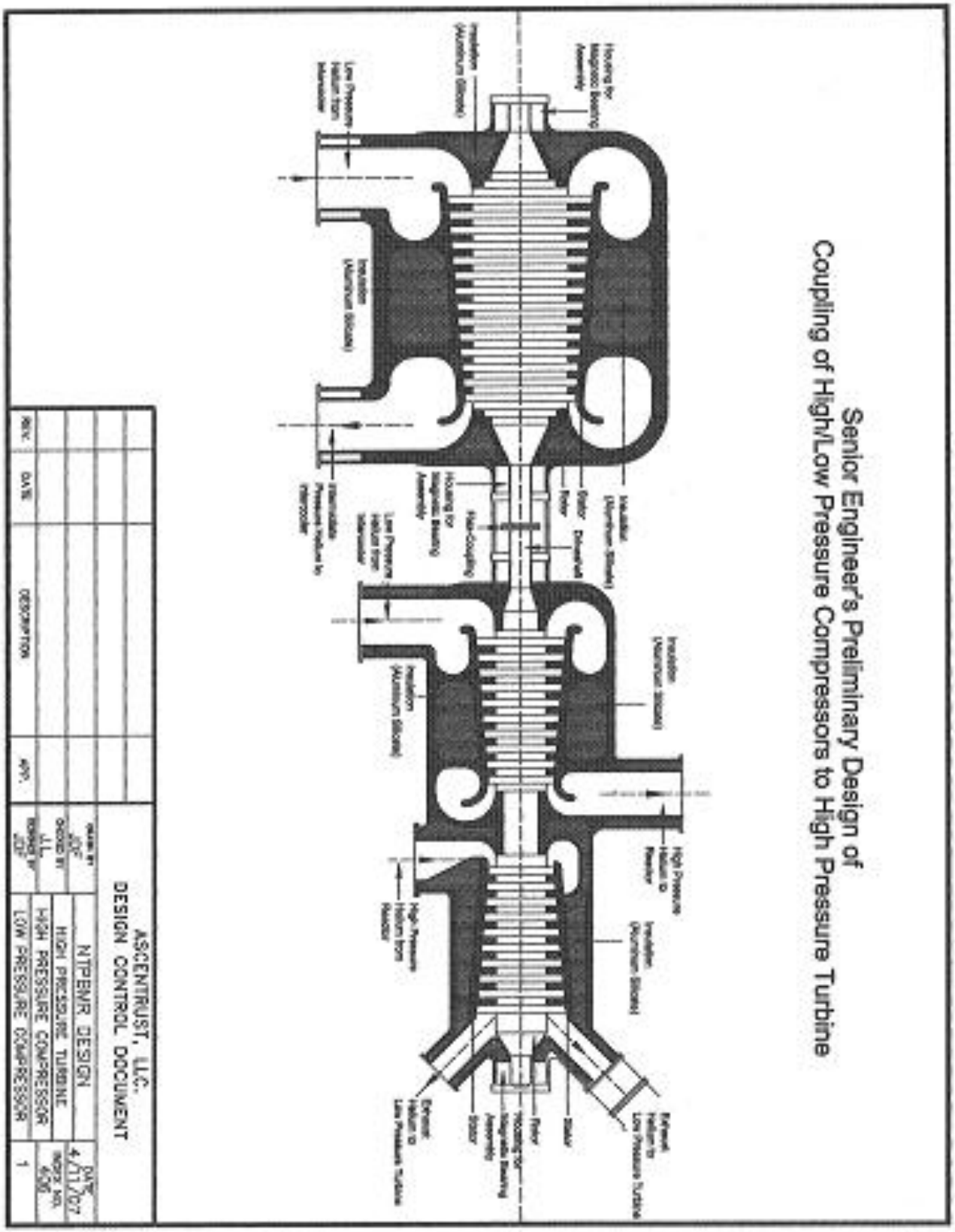


Senior Engineer's
Preliminary Design of
Coupling of High and Low Pressure Compressors



Senior Engineer's
Preliminary Design of
High Pressure Turbine/Compressor




Senior Engineer's Preliminary Design of
Coupling of High/Low Pressure Compressors to High Pressure Turbine
ASCENTRUST, LLC.
DESIGN CONTROL DOCUMENT
NTPBMR DESIGN
HIGH PRESSURE TURBINE
HIGH PRESSURE COMPRESSOR
LOW PRESSURE COMPRESSOR
DATE
4/11/07
REV.
DATE
DESCRIPTION
APP.
1

B. FUEL HANDLING AND STORAGE SYSTEM

The functions of the **FHSS** are:

- Initial loading of the core cavity with graphite spheres
- Loading the new fuel into the core
- Removing erroneously discharged fuel spheres from the graphite sphere system
- Preventing erroneously discharged graphite spheres initiating the loading of new fuel spheres, via radiation sensors fitted to the delivery line to the spent fuel storage tanks. A detected graphite sphere going the wrong way may not initiate the loading of a new fuel sphere.
- Removing fuel and graphite spheres from the discharge tube
- Separate out damaged spheres
- Separate fuel and graphite spheres
- Re-circulate partially used fuel spheres through the core.
- Measuring burn-up of partially used fuel spheres, and discharging spent fuel spheres into the spent fuel storage system
- De-fueling and refueling of the core, by transfer of the core inventory from the reactor into separate graphite and fuel storage tanks, during maintenance intervention requiring the venting of the main power system to the atmosphere
- Reloading the core from these tanks during refueling of the core.

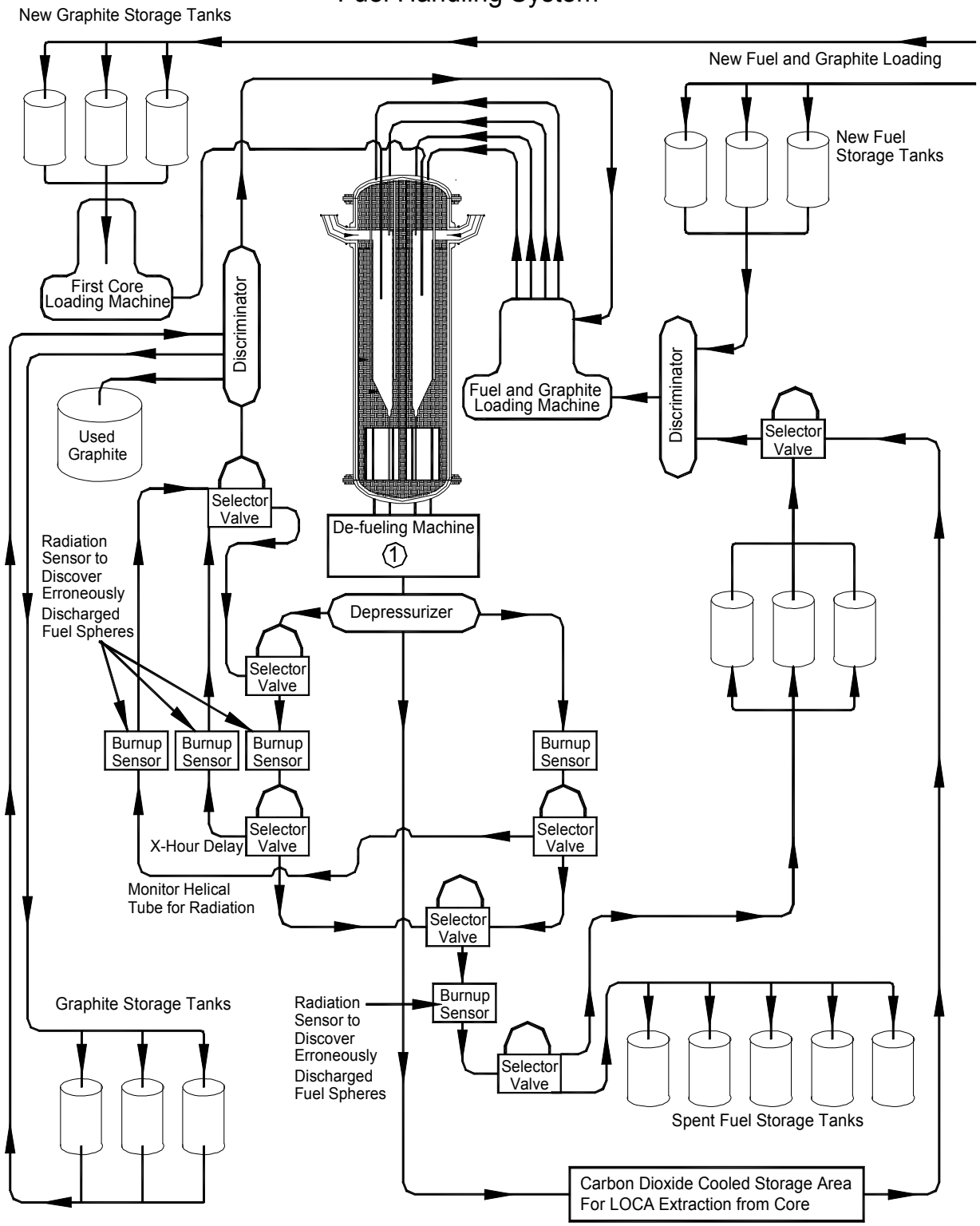
The **NTPBMR** core is to be operated according to the “multi-pass” fueling scheme: which means that fuel spheres are moved through the core more than once. In our particular case we anticipate that we will be able to circulate the fuel elements 10 times, before the fuel spheres reach the fuel burn-up levels which we are predicting to be achievable with this method.

One of the major benefits from the multi-pass fueling scheme is to provide for the uniform burn-up within the core, and thereby flattening the radial neutron flux profile and maximize the thermal power output of the modular unit.

The **FHSS** (see the figure on the next page), for the realization of the multi-pass fueling scheme, consists of the fresh fuel storage and feeding system, the fueling and de-fueling system, including the full discharge of the core in the event of a **LOCA (**Loss of Coolant Accident**).** The Storage Systems consists of the new fuel storage, graphite storage, spent fuel storage and the damaged fuel storage.

The main parts of the fuel handling system are located in the shielded, nuclear island portion of the reactor building. The spent fuel storage system will be designed to store the spent fuel of the power plant on site for the lifetime of the plant.

Nuclear Technology
Pebble Bed Modular Reactor
Fuel Handling System

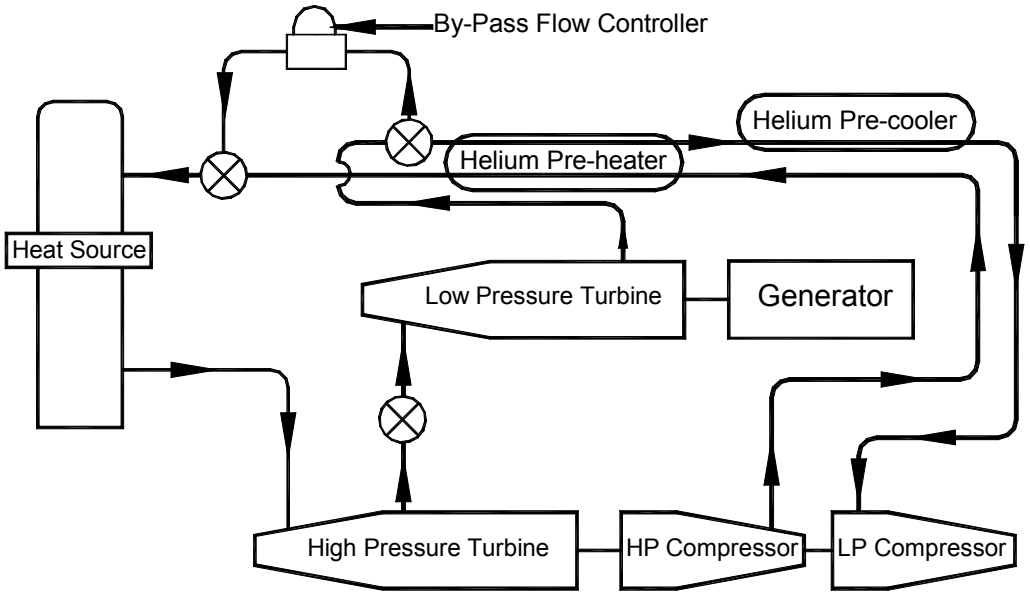


C. PLANT CONTROL AND LOAD REJECTION

a. By-pass Flow Control: As shown in the diagram below, a bypass valve bleeds high-pressure gas to short-circuit the heat source and the turbine. This throttling process is a source of irreversibility and thus reduces the cycle part load efficiency. One part of the high-pressure gas, bypassing the turbine, results in turbine output decrease. At the same time, the cycle pressure ratio is reduced, and thus the mass flow-rate through the compressor increases. If the rotational speed remains constant, the velocity triangles for the compressor and turbine are both not in the optimum condition, resulting in a decrease of the cycle efficiency.

The advantage of bypass valve control is that it can alter the turbine output rapidly to match the load variation. Thus, to achieve fast load change, bypass valve control will be included as one of the control functions in the closed gas turbine system, especially in a large system since the inventory control response is relatively slow. In the event of grid separation, the bypass valve control will also be used to prevent the shaft from over-speeding.

BY-PASS FLOW CONTROL OF A CLOSED BRAYTON CYCLE



b. Temperature modulation: Decreasing the turbine inlet temperature results in a decrease of the turbine output power and the turbine efficiency, and thus the cycle efficiency. The temperature modulation scheme utilizes this principle. For the **NTPBMR** gas turbine plant, adjusting the reactor power can alter the core outlet temperature, and thus the gas turbine inlet temperature.

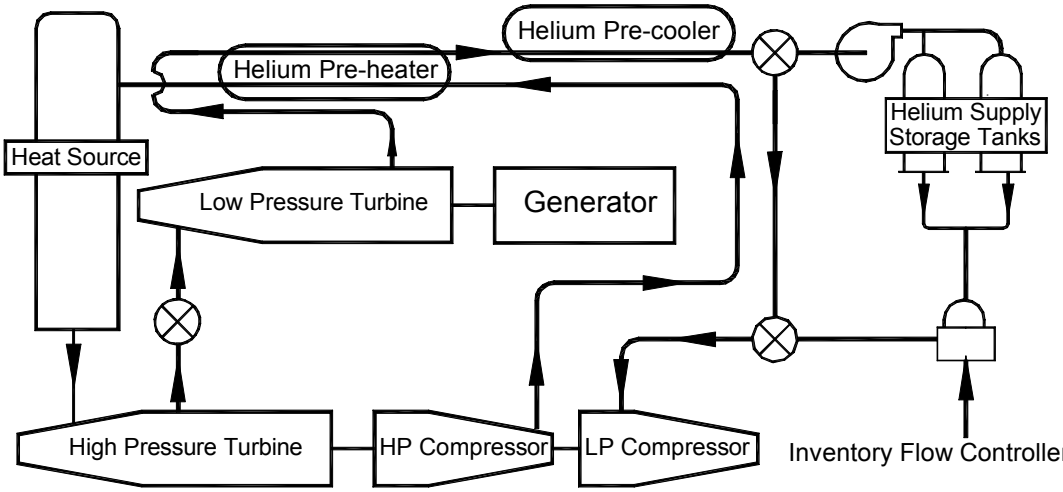
c. **Inventory Control:** As shown in the diagram below, the inventory of the working fluid in the closed power system is controlled by moving mass to or from a storage vessel. A compressor may be used to pump the working fluid from the system to the storage vessel as the load decreases although the ΔP across the compressor can also be used. The reduced mass inventory in the system results in a smaller mass flow rate, and thus a lower turbine power output.

When the load increases, the working fluid in the storage vessel is fed back to the system. To minimize the heat energy moving from the system to the storage vessel, the working fluid can be removed from a point with the lowest temperature of the cycle. With the reduced mass flow-rate, the temperatures and pressure ratio of the cycle remain constant, thus the thermodynamic cycle is unaltered.

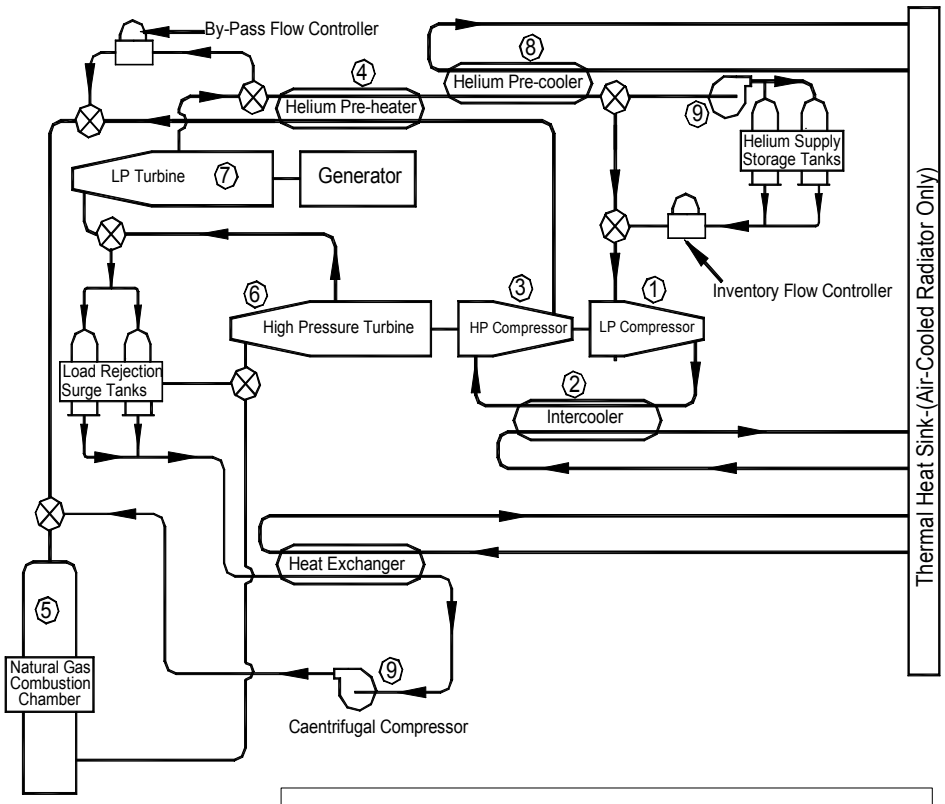
When the temperatures remain constant, the sonic speed of the working gas does not change as the mass flow-rate decreases. The blading and flow passage geometries fix the Mach number. This implies that the flow velocities along the cycle are constant and thus the mass flow-rate is proportional to the flow density. Also, the mass flow-rate is proportional to the pressure level.

As the pressure level decreases, the pressure losses will be slightly changed because the decrease in density also causes a decrease in the Reynolds number. The effect is that the cycle pressure ratio shifts from the design value and thus the cycle efficiency decreases slightly. Figure 2.8 Closed cycle with inventory control

INVENTORY FLOW CONTROL OF A CLOSED BRAYTON CYCLE



Nuclear Technology
Pebble Bed Modular Reactor
Turbo-machinery Control and Load Rejection



Legend

1. Low Pressure Compressor
2. Intercooler
3. High Pressure Compressor
4. Helium Pre-Heater
5. Heat Source
6. High Pressure Turbine
7. Low Pressure Turbine
8. Precooler
9. Centrifugal Compressor

Control of the methane-fired heater, helium closed loop turbine

There are two fundamentally different methods of controlling the closed loop brayton cycle

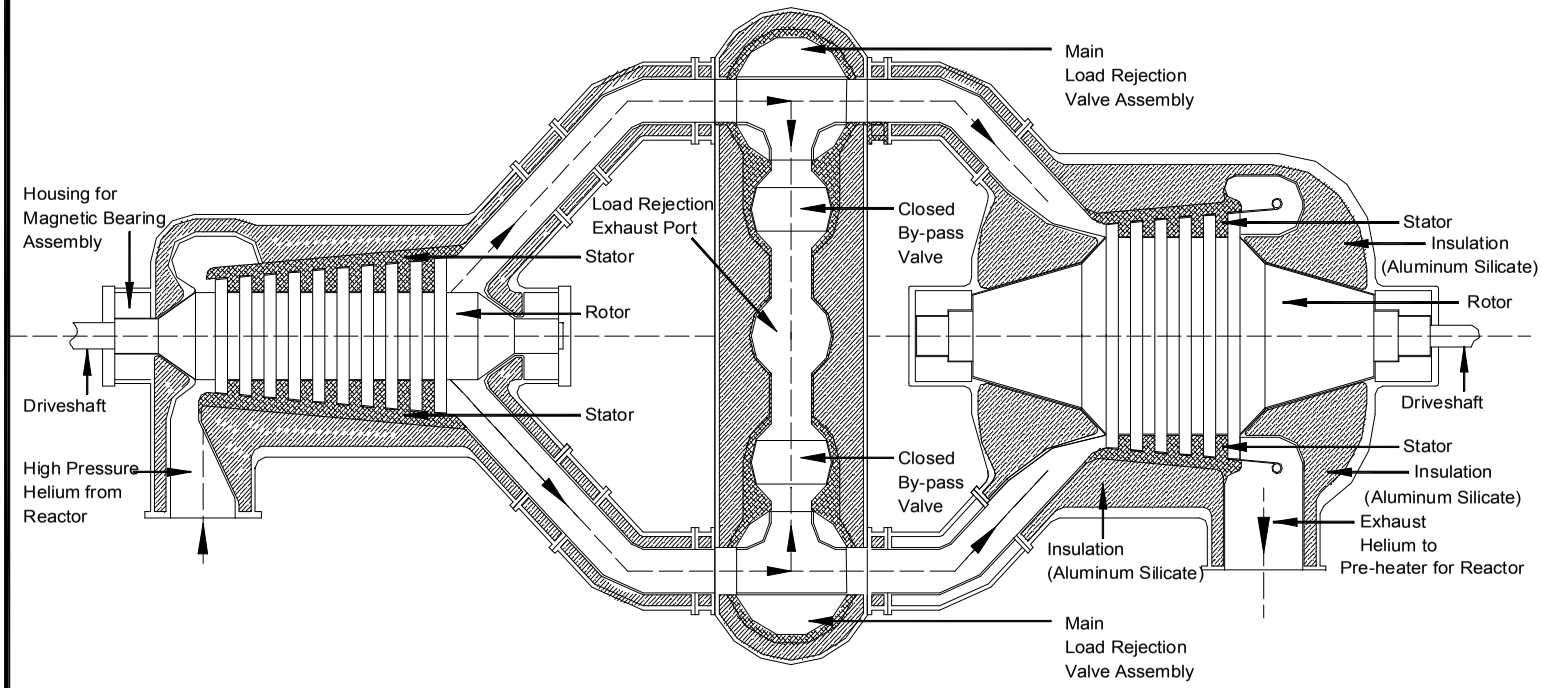
1. The first method is the bypass control at the heat source. A bypass valve bleeds high-pressure gas to short circuit the heat source and the turbine. The gas which bypasses the turbine causes a decrease in turbine output.

2. The second method which will be used to control the turbo-machinery is inventory control. As shown in the diagram above the inventory of the helium in the power system is controlled by moving mass to or from a storage vessel.

The load rejection portion of the control system is used in the event of a loss of connection to the grid, to prevent overspeed in the low pressure turbine.

d. Load Rejection between turbines: In the event in which the plant is tripped off-line, the low pressure turbine must immediately be by-passed in order to prevent over-speed due to the no-load situation created by the trip. The following diagram shows the preliminary design of the by-pass between the low and high speed turbines.

Senior Engineer's
Preliminary Design of
Load Rejection By-Pass System



REV.	DATE	DESCRIPTION	APP.

ASCENTRUST, LLC.
DESIGN CONTROL DOCUMENT

DRAWN BY JDF	NTPBMR DESIGN	DATE 4/11/07
CHECKED BY J.L.	HIGH PRESSURE TURBINE	INDEX NO. 406
DESIGNED BY JDF	LOW PRESSURE TURBINE	
	LOAD REJECTION ASSEMBLY	1

ASCENTRUST LLC ™
The POWER of ENGINEERING

6. OTHER CHARACTERISTICS OF THE NTPBMR TECHNOLOGY:

A. Loss of Coolant Proof: The low power density and high thermal mass of the technology and the online refueling capability will allow us to extract the pebble from the core in the event of a loss of coolant and allow the reactor to cycle through a loss of coolant event without raising the temperature in the core significantly.

B. Proliferation Proof; The fuel element is completely ceramic with the fuel inside of tiny micro-spheres. The extraction of sufficient quantities of plutonium from the fuel element to build a nuclear device will be impossible since it will require the acquisition of more than 200,000 fuel elements which have been in the core for more than three years. Since the on-line fuelling system is completely sealed, in a helium environment, the extraction of a single fuel element would have to break the pressurization of the core.

C. Ease of Waste Management: The **NTPBMR** fuel system leads itself easily to waste disposal: Either on-site or in an off-site permanent waste disposal facility. The fuel element completely contain the fission fragments and the whole fuel element is very robust. The spent fuel element can either be stored in dry storage above ground or can be sent to a burial facility.

D. Modular Design: The **NTPBMR** is modular in design and the comparatively small size and the lack of complexity in the design of the reactor adds to their economic feasibility. Each power module will produce approximately 110 megawatts (electric), with the use of two 55 MWe cooling loops.

The simplicity of design of our power plant is dramatic. These units will have only two dozen major plant subsystems which we believe can all be plant manufactured, licensed separately and moved to the proposed nuclear site. .

Each power module will produce approximately 110 megawatts electric, with the use of two 55 MWe cooling loops operating two closed loop brayton cycle gas turbines. The modules can easily be configured, in an energy park to produce up to 1.10 Gigawatts electrical power. The technology can also be scaled down to 55 megawatts by employing only one leg of the Helium cooling system.

E. Safety Characteristics: The **NTPBMR** has the highest level of safety available in a Nuclear Power Plant. Its safety is a result of the design, the materials used and the physical processes rather than engineered safety systems. The peak temperature that can be reached in the reactor core (1,600 degrees Centigrade under the most severe conditions) is far below any sustained temperature (2,000 degrees Centigrade) that will damage the fuel elements.

F. Economic Benefits: The **NTPBMR** modules will all be built in a factory. Only the reactor pressure vessel itself will have to be assembled in the Nuclear Island. This construction technique will allow the Company to capture the cost curve in

the construction of Nuclear Power facilities, where the stakeholders have an equity position in the manufacturing of the components of the modules of the power plants. The Company's goal is to be able to design and build a Nuclear power Plant for less than $2000.00 per KW of electrical production. With the added incentive given to the owner in that the fuel cost of operating a nuclear power are not a significant percentage of the operating costs.

7. SUMMARY OF THE NTPBMR TECHNOLOGY

The **NTPBMR** turbine plant is being developed as a generation IV nuclear energy system which offers advantages in the areas of economic competitiveness, safety and reliability. The **NTPBMR** promises a number of significant advantages over conventional commercial water-cooled technology. First, by fully using the high gas temperature, the **NTPBMR** will provide a thermal efficiency approaching 45%. Higher efficiency leads to improved economics.

The **NTPBMR** will be a demonstrably safe nuclear plant system. This implies that the system will be designed such that any postulate accidents will not result in fuel melt, fuel damage or damage to the core. Thus, no fuel damage and release of radioactivity to the environment will occur. This inherent safety is due to the fact that the core will be designed with a negative temperature coefficient of reactivity and the decay heat can be removed to the ground by a passive heat transfer mechanism. The passive heat transfer mechanism includes conduction and natural convection.

Since the coolant is inert helium in the **NTPBMR**, corrosion of the components is not a concern so that the cost for replacement of the degraded components caused by corrosion such as in water-cooled reactors is avoided. This simplifies operation and maintenance and thus improves the economics.

Overall, the objective of the **NTPBMR** is that its economics can compete with natural gas. With regard to the balance of plant design, the requirements can be summarized as follows:

A. High efficiency over a broad operating range;
B. Load following;
C. Low capital cost;
D. Constructability;
E. Modularity;
F. Transportability;
G. Code compliance.

These goals will require that the design provides high efficiency during full power operation and also high efficiency during partial power operation. From a control point of view, the plant must be capable of meeting the utility requirement for load following as an advanced nuclear system. Considering the components in the power conversion system, the constructability, complying with current codes and with no significant R&D effort need to be considered in making design decisions.

ASCENTRUST LLC
The POWER of ENGINEERING

APPENDIX
INTRODUCTION TO
NUCLEAR POWER AND NTPBMR TECHNOLOGY

SECTION H: Nuclear Graphite Development Plan

ASCENTRUST LLC

The POWER of ENGINEERING

INTRODUCTION TO
NUCLEAR GRAPHITE PROGRAM
FOR THE
NUCLEAR TECHNOLOGY PEBBLE BED MODULAR REACTOR
FOR
STAKEHOLDERS AND CONSULTANTS

INTRODUCTION TO NTPBMR GRAPHITE DEVELOPMENT PLAN

The **Nuclear Technology Pebble Bed Modular Reactor (NTPBMR)** is a helium-cooled High Temperature Gas Reactor (**NTPBMR**) with a large graphite core and predominantly graphite fuel element. Graphite physically contains the fuel and comprises the majority of the core volume. Graphite has been used effectively as a structural and moderator material in both research and commercial high-temperature gas-cooled reactors. This development has resulted in graphite being established as a viable structural material for **NTPBMR**s. While the general characteristics necessary for producing nuclear grade graphite are understood, historical "nuclear" grades no longer exist. A new supply chain must be created for the nuclear graphite which will be used in the **NTPBMR** nuclear power plants. This new graphite source must be fabricated, characterized, and irradiated to demonstrate that current grades of graphite exhibit acceptable non-irradiated and irradiated properties upon which the thermo-mechanical design of the structural graphite in the **NTPBMR** is based.

This **Graphite Development Plan** sets out the research and development (R&D) activities which will be undertaken by the Company to qualify nuclear grade graphite for use within the **NTPBMR** reactor. Background information from past graphite reactor experience, other relevant graphite grades, and the state of graphite technology developed for past gas reactors is presented to provide a perspective on what has been achieved previously in this area of research. The technology required to qualify the graphite for use in the **NTPBMR** is being developed based on the historical graphite fabrication and performance techniques used in the production of nuclear graphite for the English **Magnox** and **AGR,** gas-cooled reactor program. In the final years of the **AGR** program the nuclear graphite was being developed from Gilsonite found in the mountains of Colorado and Utah. the anticipated **NTPBMR** graphite design service conditions, and gaps in the fabrication and performance database.

The resultant data needs are outlined and justified from the perspective of reactor design, reactor performance, or the reactor safety case. The approach allows direct comparison between data needs and the resulting technology development activities. Because there are many variables (i.e., reactor designs, multiple graphite types, a range of operating temperatures and fluence, etc.) that can significantly affect the development of graphite technology for the **NTPBMR**, a "baseline" reactor design was chosen to simplify the identification of needed data. The **NTPBMR** design with an outlet temperature of 850°C was chosen as the baseline technology. In this case, the expected doses and operation service lifetimes are expected to be fairly moderate. Technology development needs to satisfy requirements beyond this baseline (i.e., much higher doses expected at graphite reflector surfaces facing the fuel pebbles in the pebble-bed core).

The irradiation program proposed for the **NTPBMR** design will consist of eight irradiations that span the proposed temperature-dose envelope for the moderation portion of the **NTPBMR**. These irradiations will contain specimens of sufficient size,

number, and type to support statistical assessments necessary to capture the inherent variability in graphite; to support American Society for Testing and Materials (ASTM) requirements for sample analysis; and to fully characterize the physical, thermal, and mechanical properties of the irradiated graphite.

The plan discusses in detail the specific material characterization techniques that will be used to characterize the graphite microstructure and establish the key material properties for both non-irradiated and irradiated specimens that will be used to support American Society of Mechanical Engineers (ASME) codification of graphite. Factors that can significantly affect the R&D program, such as graphite acquisition, test standard development, sample preparation (e.g., grain sizes, sample sizes, etc.), are discussed within each characterization section. In addition, the role of the modeling activities from the engineering-scale to the micro- or meso-scale to the nanoscale is discussed in the context of this qualification program, and the inter-relationships between the experimental and modeling activities are presented to establish a complete picture of the technology development required for the **NTPBMR** graphite qualification. Beyond the near-term the **NTPBMR** graphite qualification program presented here, a more complete evaluation of the processing route and raw material constituent's influence on graphite behavior is required for full commercialization of the **NTPBMR** graphite technology in the long term.

In addition, appropriate graphite recycling and disposal options must be considered to reduce the volume and costs of anticipated waste disposal. Recycle is considered as a long-term strategy and would only be pursued by vendors when large numbers of graphite moderated reactors are developed and a "nuclear graphite economy" is established. The magnitude of the R&D program necessary to establish a standard nuclear grade graphite, whether from a new coke source and/or from recycled material for use within any **NTPBMR** design, cannot be firmly estimated today given the current limited knowledge of the linkage between graphite fabrication, material properties, and in-reactor performance. It is anticipated that the work proposed to qualify graphite for the initial **NTPBMR** cores will provide the strong technical basis needed to establish a long-term graphite development and qualification program that meets these more ambitious commercialization goals.

NTPBMR NUCLEAR GRAPHITE PRODUCTION PLAN

1. INTRODUCTION:

Graphite has been used effectively as a structural and moderator material in both research and commercial high-temperature, gas-cooled nuclear reactors (i.e., Magnox, Advanced Gas Reactor [AGR], Albeitsgemeinschaft Versuchsreaktor [AVR], Reactor Bolshoi Moschnosti Kanalynyi [RBMK], Thorium Hochtemperatur Reaktor [THTR], Fort St. Vrain Reactor [FSVR], etc.). This development has resulted in graphite being established as a viable structural material for High Temperature Gas Reactors (HTGRs) and specifically for the **NTPBMR** technology.

While the general characteristics necessary for producing nuclear grade graphite are understood, historical "nuclear" grades no longer exist. New grades must be fabricated, characterized, and irradiated to demonstrate that current grades of graphite exhibit acceptable non-irradiated and irradiated properties so that the thermos-mechanical design of the structural graphite in the **NTPBMR** technology can be validated.

Beyond structural integrity, the reactor lifetime for specific graphite types cannot be established based on the current state of the art; establishing lifetime is complex because of the influence of fabrication and radiation damage on microstructural changes and associated changes in material properties. Lifetime predictions of graphite components with the service demands and reactor operating mode anticipated for the **NTPBMR** is a practical but much more complex problem than simply determining whether a graphite type is more stable or less stable in an irradiated environment.

Graphite properties, such as strain to failure, dimensional change rate, and irradiation dependence of thermal expansion coefficient, can constrain the reactor design by limiting lifetimes for critical components. For example, irradiation-induced dimensional changes to graphite can be severe enough to require limiting the temperature and flux gradients within graphite components or possibly requiring the need for added design features to physically hold components in position over time.

A complicating factor to establishing a qualified fabrication and performance dataset is the inherent variability in the graphite product. Variability within-billet, intra-billets, and lot-to-lot of the graphite must be accounted for in a statistical manner because of its influence on material properties. This variability must also be characterized to enable credible designs and to support the ongoing development of the probabilistic American Society of Mechanical Engineers (ASME) graphite design methodology.

The previous Fort St. Vrain design used deterministic performance models for H-451, which was unacceptably conservative given the understanding of graphite at that time. With our current knowledge, probabilistic performance models can be developed to characterize the new graphite grades for the **NTPBMR**.

Furthermore, to provide a consistent nuclear grade graphite material for eventual standardization and commercialization of **NTPBMR**s, an American Society for Testing and Materials (ASTM) standard specification for isotropic and near-isotropic nuclear graphites (D 7219-05) is being developed along with a standard specification for nuclear graphite suitable for components subjected to low neutron irradiation dose.

Additionally, ASME codes and guides for materials selection and qualification, design, fabrication, testing, installation, examination, inspection, and certification will be needed and thus are under development by the international graphite community. Development of these standards will be necessary to approve future grades of nuclear graphite for the **NTPBMR** technology.

The Senior Engineer has set in motion an R&D program for the nuclear graphite which includes acquisition of the Gilsonite mine in Colorado. This mine was the source of the

nuclear graphite being used in the **AGR** in England. From the proper source material the Company will access the expertise locked up in the brains of the engineers of the English Magnox and **AGR** Nuclear power projects.

Therefore, the overall objectives to qualify the **NTPBMR** graphite for initial operation are:

A. Establish statistical non-irradiated thermo-mechanical and thermo-physical properties. Characterize lot-to-lot and billet-to-billet variations (for probabilistic baseline data needs)

B. Establish irradiated thermo-mechanical and thermo-physical properties

C. Develop understanding of life-limiting phenomena at high dose and temperature (e.g., irradiation induced creep)

D. Develop appropriate constitutive relations

E. Establish reliable, predictive thermo-mechanical finite element models (FEM)

F. Establish relevant ASTM standards and ASME design rules.

Beyond the **NTPBMR** graphite production. The graphite research and development (R&D) program needs to evaluate processing route and raw material constituents influences on graphite as well as recycling and disposal issues. The current world market share for nuclear graphite is extremely small. While graphite manufacturers are willing to produce nuclear grade graphite, the petroleum industry, which produces the raw starting material – specialty coke, is much less interested. The material specifications for specialty coke are much more exacting than what is needed for electrode production, the majority market share for graphite. Since this material's market share is so small, the coke suppliers have very little financial interest in changing their production process to enable manufacture of these small batches of specialty coke necessary for nuclear graphite production.

The Company will develop its own nuclear graphite capabilities since there may not be enough specialty coke material needed for sustained production of nuclear graphite for the **NTPBMR** applications. In the longer term, a full evaluation of the processing route and raw material constituents influence on graphite behavior is required for full commercialization of the **NTPBMR** graphite technology. Being mindful that there are two separate but vital graphite technologies in the **NTPBMR** nuclear technology. These two technologies are the structural elements of the core and the fuel element. The magnitude of the program necessary to establish a standard nuclear grade graphite for use within any **NTPBMR** design is difficult to estimate given the current limited knowledge of the linkage between fabrication, material properties, and performance in reactor.

Finally, the lower power density of the **NTPBMR** and the large inner and outer graphite reflector volumes will generate large quantities of Low Level Waste that would have to be disposed of in the absence of recycling. Therefore it is being assumed by the Senior Engineer that a prudent way forward is to include the recycling of the nuclear graphite.

Appropriate graphite recycling and disposal options must be considered to reduce the volume and costs of anticipated waste disposal. Two options are currently envisioned:

- Reuse of blocks after heat treatment to anneal out radiation damage
- Form new blocks using reconstituted graphite material by crushing and jet milling irradiated blocks to fine powder. Such graphite fabrication methods have been employed before (e.g., BAN graphite). Recycle is considered as a long-term strategy and would only be pursued by vendors when large numbers of **NTPBMR**s are developed and a "nuclear graphite economy" is established. R&D would be needed to demonstrate that the recycled graphite demonstrated acceptable in-reactor performance.

2. BACKGROUND

The basic feasibility of graphite planned for the **NTPBMR** has previously been demonstrated in former high-temperature, gas-cooled reactor plants (e.g., DRAGON, Peach Bottom, AVR, THTR and FSVR). These reactor designs represent two design categories: the Pebble-bed Reactor (PBR) which is the family of the **NTPBMR** and the Prismatic Modular Reactor (PMR). Current commercial examples of potential HTGR candidates are the Gas Turbine-Modular Helium Reactor (GT-MHR) from General Atomics, the High Temperature Reactor concept (ANTARES) from AREVA, and the Pebble-bed Modular Reactor (PBMR) from the PBMR Pty, LTD consortium. Furthermore, the Japanese High-Temperature Engineering Test Reactor (HTTR) and Chinese High-Temperature Reactor (HTR) are demonstrating the feasibility of the reactor components and materials needed for the **NTPBMR** (HTTR reached a maximum coolant outlet temperature of 950°C in April 2004). This experience has in large part formed our understanding of graphite response within a HTGR nuclear environment.

2.1 RADIATION EFFECTS ON GRAPHITE

Radiation damage to a solid, crystalline microstructure occurs from either ballistic (atomic or subatomic kinetic collisions) or radiological (conversion of radiation-induced electronic excitations to kinetic energy) events. These events can result in significant atomic lattice disruptions, the magnitude of which is significantly dependent upon the bonding energy of the individual atoms.

Generally, ballistic events have higher damage efficiencies per event and thus provide a limiting case for materials exposed to such an environment (i.e., a high neutron flux in the **NTPBMR** core). Ballistic neutron damage of graphite and graphitic materials has been studied extensively for decades, and the mechanisms are well understood.

Neutron irradiation causes the displacement of carbon atoms from their equilibrium lattice positions into interstitial positions throughout the microstructure. Single vacancies and vacancy loops/clusters are left within the basal planes of the crystalline structure causing the basal planes to collapse/shrink (plane destruction) as further damage accumulates and vacancy clusters grow. Due to the anisotropic crystal structure of graphite, the interstitial atoms preferentially diffuse and accumulate in the lower energy

areas between the basal planes (van der Waals bonds between the covalently bonded basal plane atoms).

These small mobile groups of interstitial atoms aggregate into larger clusters, physically forcing the basal layer planes apart. The atoms within the clusters eventually rearrange themselves into new basal planes, resulting in the expansion of the graphite crystal in the c-axis direction. The corresponding contraction in the a-axis direction (parallel to the basal planes) occurs from vacancy collapse and plane destruction as discussed previously. The mechanical or material effects resulting from these basic radiation damage mechanisms are controlled by a number of factors, including the operating temperature, the degree of crystallinity within the microstructure, the variation of crystallite orientation, and the micro-damage within the formed graphitic microstructure during fabrication processes

All these parameters significantly affect the thermo-mechanical response of the graphite, but temperature plays the key role in determining the effects on graphitic structures. At lower irradiation temperatures (RT - 300°C), graphite structures with high crystallinity and low amount of fabrication defects show significant dimensional swelling in the c-axis direction (depending upon the dose and grades of graphite). Dimensional shrinkage in the a-axis direction occurs, but the rate is considerably smaller, indicating vacancy line/loop collapse or plane destruction. At these (and lower temperatures), significant levels of stored energy (in the form of damage) accumulate within the microstructure and can be released as heat in the graphite crystals.

This issue is effectively eliminated at higher operating temperatures where increased point defect mobility promotes significant recombination and the formation of more stable defect clusters. The physical and microstructural characteristics of the graphite significantly alter the graphite response at higher irradiation temperatures (> 400°C). Processing defects and crystallite misalignment imposed upon the microstructure during fabrication cool down (i.e., cracks parallel to the c-axis planes) physically accommodate the c-axis swelling, and the cracks close as the material swells perpendicular to the c-axis. This crystallite misalignment and damage within the microstructure provides a ready volume of space that can initially accommodate the crystallite swelling during irradiation. Since the c-axis swelling is mitigated, the macroscopic response is one of overall shrinkage due to a-axis shrinkage throughout the graphite volume.

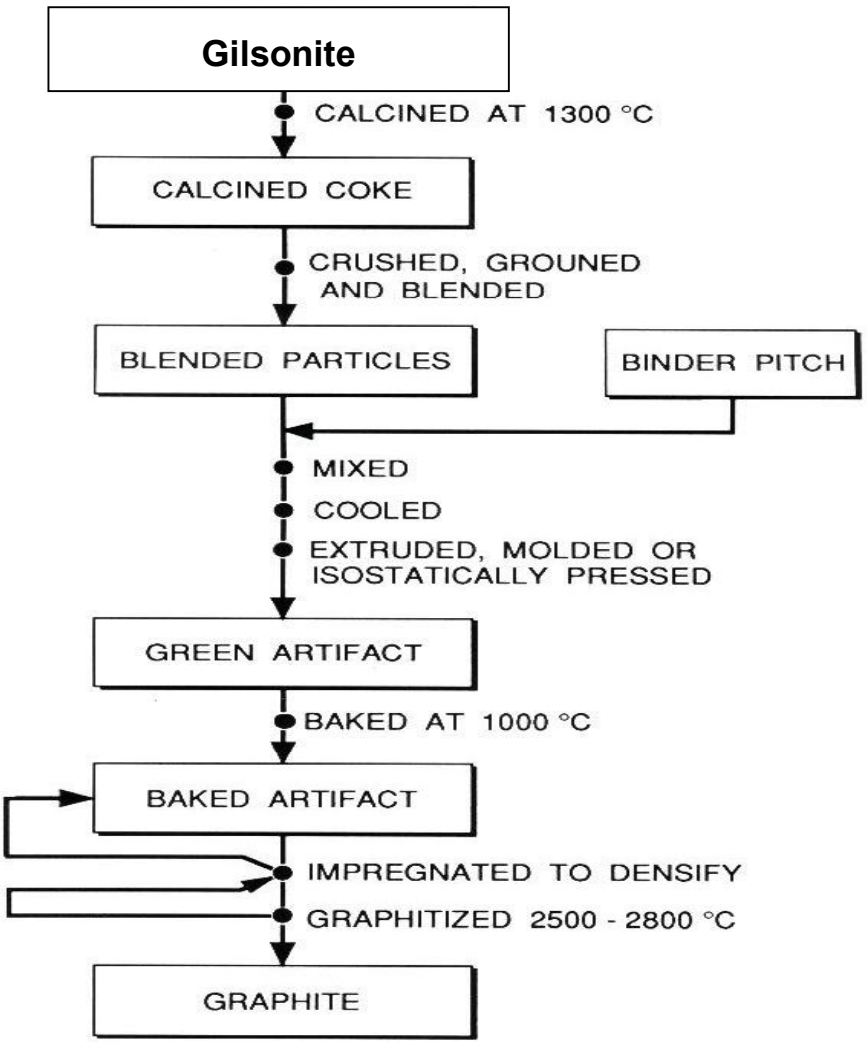
It is readily seen that the magnitude and rate of dimensional change and the point of “turnaround” are directly related to the degree of crystallinity within the microstructure, the variation of crystallite orientation, process conditions, and the resident micro-damage within the individual graphite types. In addition, the rate of dimensional change is also significantly affected by the irradiation temperature. Typically, the useful lifetime for a graphite type is defined as the time/dose it takes for the material to contract and then swell back to zero dimensional changes.

During operation, graphite components within the reactor core undergo neutron irradiation-induced dimensional change. Local differences in neutron dose and

temperature cause differential strains and resultant stresses to develop in the graphite. These stresses are relaxed by neutron irradiation induced creep strains. Thermal creep of graphite is not expected at the temperatures experienced in the reactor core (< 1100°C). The irradiation induced creep strains in graphite can be very large, exceeding several percent; premature failure of the graphite would occur if they were not accommodated by irradiation induced creep. This phenomenon has been shown to be particularly important for Magnox and RBMK plants in which creep is necessary to explain the absence of cracked core components.

2.2 NUCLEAR GRADE GRAPHITE

Nuclear grade graphite is a specially developed composite material manufactured from a filler coke and pitch binder. Nuclear graphites are usually manufactured from isotropic cokes (petroleum or coal-tar derived) and are formed in a manner to make them near-isotropic or isotropic materials. The Figure below shows the major processing steps in the manufacturing of nuclear graphite. After baking (i.e., carbonization), the artifact is typically impregnated with a petroleum pitch and re-baked to densify the part. Impregnation and re-bake may occur several times to attain the required density. Graphitization typically occurs at temperatures >2500_C with the entire process taking six to nine months.



Typical process steps in the manufacturing of nuclear graphite.

Nuclear grade graphite has been specially developed to meet reactor design requirements. Attributes required for modern nuclear grade graphite are:

- Acceptable dimensional change (isotropy): Near isotropic graphite
- High purity: Low elemental contamination, especially boron
- Fabricability: Ability to machine into large graphite components
- Characterized irradiated material performance: Must possess irradiation design database
- Each graphite type has a unique response to irradiation: Graphite of similar grade will not respond exactly the same to the same irradiation.

While these are minimum attributes necessary to achieve acceptable component lifetimes for use within an irradiation environment, they may not be sufficient to demonstrate adequate structural integrity for all design configurations. It is known that

individual “nuclear grade” graphites will have distinctly different responses to the irradiated environments based on the extent of anisotropy, grain size, microstructural defects, microstructure orientation, purity, and fabrication method. Thus, the response of each graphite type must be verified for use as a structural component within the **NTPBMR**.

As new types of nuclear graphite are being developed for the **NTPBMR**, a qualified properties database on these new candidate grades of graphite must be developed to support the design of graphite core components within the specific reactor service conditions of the **NTPBMR**. Non-irradiated and irradiated data are required for the physical, mechanical (including radiation induced creep), and oxidation properties of the new graphite. To meet these requirements, a radiation effects database must be developed for the NEW GRAPHITE which will be fabricated out of Gilsonite.

Component lifetime calculations using new graphite types will be determined from both the initial non-irradiated, “as-received” material properties and the property changes that will occur due to radiation damage or environmental degradation to the graphite during operation. The non-irradiated mechanical and material property values will be used as baseline data for initial reactor startup and operation. The “as-received” property values of the graphite components will be used to calculate the initial core thermal properties (e.g., conductivity, specific heat, etc.) and physical response (e.g., applied stresses, dimensional tolerances, etc.).

The evolution of these property changes is dependent upon a number of factors, including temperature, fluence/dose, graphite microstructure/orientation, chemical purity, and applied stresses during operations. Obviously, those components located physically closer to the fueled region of the core will experience higher temperatures and doses than components on the edge of the reactor, and a faster rate of change is expected. The extent of property changes include physical changes to the component (i.e., dimensional changes); changes in the thermo-mechanical properties, especially irradiation-induced and changes to thermo-physical properties, such as thermal conductivity, coefficient of thermal expansion, etc. All of these will affect the prediction of graphite lifetime.

2.3 REQUIREMENTS AND SERVICE CONDITIONS

Reactor design considerations, design service operating conditions, and reactor safety requirement are key considerations in determining the change in material properties of nuclear graphite. Physical parameters, such as component size, geometry and machining requirements may require specific billet sizes and grain size. Operating conditions will specify expected fluence/dose, temperatures, and initial imposed loads upon the graphite components. Finally, safety considerations may require additional material property measurements such as oxidation rates, properties after oxidation, wear/friction properties for dust formation and post-irradiation thermo-mechanical and thermo-physical properties.

3.1 PHYSICAL PARAMETERS OF CORE

3.1.1 MODERATOR BLOCKS AND PEBBLES

For the **NTPBMR** graphite program, a compromise between superior material properties and material cost is an important consideration in selecting a nuclear grade graphite. The Japanese IG-110 graphite with its very small grain size and isotropic microstructure shows excellent nuclear response (high stability) and is considered one of the best commercially available nuclear graphites on the market. However, it is prohibitively expensive and the fabrication technique is exacting. As a consequence, the Japanese only use IG-110 in limited applications within the harshest nuclear environments (i.e., inner components). These issues have lead the Japanese to evaluate different graphites other than IG-110 for future applications. Similar logic is being applied to the graphite selection for the **NTPBMR** program where in service conditions and applications for each component are evaluated and a suitable nuclear graphite is selected for optimal performance within those particular parameters

3.1.2 REFLECTOR

Obviously, those components located physically closer to the fueled region of the core will experience higher temperatures and doses than components on the edge of the reactor, and a faster rate of change is expected. The extent of property changes include physical changes to the component (i.e., dimensional changes); changes in the thermo-mechanical properties, especially irradiation-induced creep; and changes to thermo-physical properties, such as thermal conductivity. Reactor design considerations, design service operating conditions, and reactor safety requirement are key considerations in determining the change in material properties of nuclear graphite. Physical parameters, such as component size, geometry, and machining, may require specific billet sizes and grain size. Operating conditions will specify expected fluence/dose, temperatures, and initial imposed loads upon the graphite components.

3.1.3 FUEL PEBBLES

For the **NTPBMR** graphite program, a compromise between superior material properties and material cost is an important consideration in selecting a nuclear grade graphite. The Japanese IG-110 graphite with its very small grain size and isotropic microstructure shows excellent nuclear response (high stability) and is considered one of the best commercially available nuclear graphites on the market. However, it is prohibitively expensive and the fabrication technique is exacting. As a consequence, the Japanese only use IG-110 in limited applications within the harshest nuclear environments (i.e., inner components). These issues have lead the Japanese to evaluate different graphites other than IG-110 for future **NTPBMR** applications. Similar logic is being applied to the graphite selection for the **NTPBMR** program where in the service conditions and applications for each component are evaluate performance within those particular parameters.

Fuel blocks for the prismatic reactor design have a number of fuel and coolant channel holes dri axially down the length of the block. The pitch between these holes can be

quite small leaving the graphite webbing no thicker than 2 to 3 mm. If the graphite grain size is large, the webbing may only have one to two grains between channels. One to two grains of a material will not represent the true properties of a material providing uncertainty in irradiation stability and strength of the channel web Thus, it is necessary to select a small-grained graphite that can provide 10 or more grains (at least) between the channels. Pebble fuel has no such machining constraints and can easily accommodate

3.1.4 BLOCKS

A similar rationale is used when determining the parameters required for graphite blocks used in the inner and outer reflector regions. The continuous refueling design of the PBR allows it to operate without having to shut down for periodic re-fueling. However, the stationary reflector blocks do sustain radiation damage and must be replaced periodically. For economic reasons, a PBR is operated continuously for as long as possible, forcing the reflector blocks to withstand CONTINUOUS IRRADIATION.

Most pebble-bed designs would like the reactor to operate for at least 20 to 25 years before having to de-fuel the entire core to replace the reflector blocks. At expected **NTPBMR** fluence levels, this can equate to doses as high as 25 dpa, which is well past turnaround even for the most stable nuclear graphites. Such a high dose level will require careful analysis of the irradiation response of the graphite selected for reflector block use. For purely economic reasons, replacing the reflector blocks within a prismatic core as infrequently as possible would be desirable as well. The more stable the response of the graphite blocks is to irradiation, the lower the costs for replacement material and less down time for the reactor. These economic considerations are important factors for determining the appropriate graphite for the **NTPBMR** reactor.

3.1.5 PERIPHERAL GRAPHITE COMPONENTS

Graphite components outside of the central core region will receive considerably lower doses and operate at much lower temperatures. These include permanent reflector blocks, core support columns/structures, and other graphite components surrounding the core. As a result, the concern is not necessarily irradiation stability but environmental attack or abnormally large stress states on the graphite components. This can dramatically alter the fracture strength, compressive strength, changes to thermal conductivity, and/or emissivity resulting in structural integrity concerns and loss of efficient reactor heat flow.

Oxidation rates, both acute and long-term chronic degradation, are of specific concern during operation. Oxidation of graphitic components can lead to a loss of strength and may affect the thermo-mechanical and thermo-physical properties as well. Further, generation of dust and small particulates from wear and friction can provide a means for spreading contamination when the coolant is released from the primary system. Finally, seismic and large applied loads may exceed the strength of compromised graphite (such as the support columns) causing failure in critical core components. Material properties for all graphite components must be characterized to determine the response of the reactor core and support structures under normal and off-normal conditions.

3.2 NORMAL AND OFF-NORMAL OPERATING CONDITIONS

Normal operating conditions have been calculated from reactor models based on the **NTPBMR** design. Normal steady-state temperatures and fluxes shall be based on an 850°C coolant outlet. The range of expected dose is rather large and needs to be refined further as designs mature since higher dose rates will lead to a need for greater testing and longer irradiation programs. If the designs evolve toward the lower range of the estimated dose rates, some of the data requirements (and associated testing needs) driven by higher levels of irradiation damage may not be necessary. For example, if the inner reflector walls only receive a dose of 0.2 dpa, then it will take more than 20 years before the inner walls will obtain a dose approaching turnaround at 1000°C. Since turnaround is a function of temperature as well as dose, this also applies to lower operating temperatures within the reactor.

3.3 ANTICIPATED LICENSING DATA NEEDS

3.3.1 RESEARCH TOPICS IDENTIFIED FROM NRC PIRT

The Nuclear Regulatory Commission (NRC) Phenomena Identification and Ranking Table (PIRT) process was applied to the issue of nuclear grade graphite for the moderator and structural components of the **NTPBMR**. An international group of graphite experts used this process to identify and rank by importance any phenomena that may adversely affect the performance of a nuclear reactor during both normal and off-normal operation. Material property changes as well as material response during accident conditions are considered during this process.

A specified PIRT process has been developed to identify and rank phenomena affecting the performance of a nuclear reactor. The first part of the process is identification of phenomena that may significantly affect the performance of the reactor. Second, the phenomena are ranked as high, medium, or low importance and whether there is a high, medium, or low amount of data available to characterize the phenomena. Obviously, those phenomena that have a high impact on the performance with a low knowledge base will score significantly higher in the assessment. The initial ranked phenomena anticipated for graphite components within the **NTPBMR** core have been established in the NRC PIRT report, NUREG/CR-6944, Vol. 1-6.

Summarized below are those phenomena within the PIRT report that are deemed pertinent for the anticipated core design and operation requirements of the **NTPBMR** graphite R&D program. Both normal and off-normal operation (postulated accident conditions) were considered for either a prismatic reactor design or a pebble-bed reactor design within these main research areas. All of these areas of research are currently being integrated into the **NTPBMR** graphite R&D program to characterize reactor design, licensing, and operational performance of graphite.

Research areas containing the identified PIRT performance phenomena.

Structural integrity of graphite	Retention of long-term structural stability and mechanical strength under specified loads. Specified by ASME requirements.

Thermal response – normal operation	Changes in thermal properties at peak dose and temperatures.
Thermal response -off-normal operation	Verification that changes to thermal material properties is sufficiently small to guarantee the passively safe response of the reactor.
Changes to by-pass flow	Potential coolant flow issues due to shrinkage and swelling of graphite components.
Chemical and mechanical core stability	Oxidation and subsequent structural stability of oxidized graphite. For both acute (accident) and chronic (normal operation) conditions.

3.3.2 FULL OPERATION OR PARTIAL OPERATING LICENSE

The **NTPBMR** program will apply for a partial (or demonstration) license to the NRC for this first reactor. The assumption is that a fully qualified graphite will not be required for use within a demonstration reactor and the program will not need to perform some of the higher dose experiments to support a full operating license before startup can occur. In addition, the demonstration plant may not operate at full design power, thus producing less fluence and lower temperatures than expected for full power operation.

As a result, the regulator may be satisfied with some or only part of the data needed for full qualification effectively giving the program more time to gather the required data for full licensing of the graphite. Experiments necessitating longer times and higher irradiation dose can be delayed until the reactor will actually be operated at the higher temperatures and fluences expected at full design power.

3.3.3 FULL DATA SET OR EXTENSIVE CORE INSPECTION PROGRAM

Rather than fully characterizing the graphite before building the reactor, the **NTPBMR** program may elect to have an extensive core in-service inspection (ISI) program. As stated previously, one can be relatively certain that any of the current nuclear graphites (isotropic, pure “nuclear” grade graphite) will be stable for a short period of time within an **NTPBMR** core. However, an extensive core inspection program will be required to assure the NRC (and other regulatory groups) that the graphite is behaving as predicted since there will be insufficient verification data. In addition, since an ISI program can only monitor a fraction of the core, there will need to be additional verification data in the form of a characterization program (non- and irradiated material) conducted in parallel while the reactor is operating. The characterization program can be limited in scope since a large portion of the verification resides with the core inspection.

3.4 ASME STANDARD SPECIFICATION

To provide a consistent nuclear grade graphite material for eventual standardization and commercialization of **NTPBMR**s, an ASTM standard specification for isotropic and near-isotropic nuclear graphites (D 7219-05) is required. Additionally, ASME codes and guides for materials selection and qualification, design, fabrication, testing, installation, examination, inspection, and certification are needed. These standards will be necessary

to approve future grades of nuclear graphite for new **NTPBMR**s and are under development by the international graphite community. Establishing a nuclear graphite code case by the ASME will require specific and detailed data. The current direction from ASME is to specify a prescriptive method for gathering the requisite graphite characterization data. All gathered data, constitutive relationships, and the associated predictive models resulting from these relationships must be demonstrated to be adequate before the appropriate committees prior to approval of an ASME code case. Two approaches can be used to demonstrate the validity of the code case to the ASME committees – deterministic and probabilistic.

A probabilistic approach will be more accurate to describe failure within a ceramic material such as graphite (i.e., flaw size and distribution throughout the microstructure) than the deterministic approach. Furthermore, the probabilistic approach can (a) accommodate the inherent variability in as-fabricated and irradiation-induced changes to the graphite microstructure and properties and (b) reduce unnecessary conservatisms inherent in deterministic models that result in the need for excessive design margins.

4. MATERIAL PROPERTY NEEDS

Based on the parameters discussed previously, the technical areas for R&D are outlined below. Material property values within three primary areas (physical, thermal, and mechanical) will be required before any graphite can be used in the **NTPBMR**. Specific material properties within each area are identified and the reasoning for obtaining this data is defined for each property.

4.1 PHYSICAL

Generally, physical material properties are concerned with characterizing the microstructure and the effects of microstructure on the macroscopic response of the material (i.e., dimensional changes).

4.1.1 MICROSTRUCTURE CHARACTERIZATION

Determination of grain size, morphology/anisotropy, and pore size/distribution within the graphite microstructure is critical to determining the macroscopic physical, thermal, and mechanical properties. These parameters must be determined before an accurate analysis of the graphite performance can be made.

In addition, a key technological deficiency is the inability to determine microstructural features within a graphite material, specifically with non-destructive techniques. This is important for determining not only the evolution in test specimen microstructures as a function of irradiation, but also for determining defects within the large graphite billets. Inspecting billets without damage to ensure proper microstructural development is one of the largest problems facing any Quality Assurance (QA) program for purchasing of nuclear grade graphite. The implementation of non-destructive techniques will be necessary for accurate quality assurance.

4.1.2 MASS AND DIMENSIONAL MEASUREMENTS

Dimensional change is one of the key parameters defining a nuclear grade graphite. Determination of volumetric and density changes as a function of temperature and dose will be necessary to understand critical performance measures, such as turnaround, irradiation creep, and internal stresses imposed upon graphite components.

4.2 THERMAL

Thermal material properties are critical for protecting the fuel particles during off-normal events as well as for predicting thermally induced stress states within solid graphite components (i.e., reflector blocks). Degradation in thermal properties – conductivity, specific heat, and coefficient of thermal expansion (CTE) – will significantly impact the ability of the graphite to both absorb energy and transfer the heat load out of the core region during an off-normal event. Without adequate removal of the heat, fuel particle centerline temperatures will exceed the design limits resulting in unacceptable numbers of particle failures and radiation release levels. In addition, thermally induced stresses can be exacerbated between and within graphite blocks with significantly altered thermal properties. Elevated stress levels can exceed the structural strength of the graphite blocks, resulting in cracking, spallation, and structural stability.

4.2.1 THERMAL EXPANSION

The CTE for graphite components is critical for determining the dimensional changes that occur as a result of temperature increases. Localized external stresses can be imposed upon mechanically interlocked graphite core components as the individual pieces suffer differential expansion. Internal stresses can occur within larger graphite components if there is a temperature gradient causing differential expansion within the piece (i.e., one side has a higher temperature than the other). Finally, the thermal expansion is greatly dependent upon the graphite microstructure, such as orientation/anisotropy, pore size and distribution, and crystallinity.

Irradiation damage can alter graphite CTE values significantly, and changes must be quantified to determine the extent of change to the CTE. A reduction or increase in CTE can significantly affect the stresses (internally and externally) imposed upon the graphite components within a reactor core and will directly affect component lifetime. Determining the changes to the CTE as a function of irradiation dose and temperature will be a key parameter for reliable calculation of stress states within graphite components, volumetric changes, and irradiation creep rates.

4.2.2 THERMAL CONDUCTIVITY

The ability to conduct heat through the graphite core is critical to the passive removal of decay heat. Reduction of the thermal conductivity within graphite can significantly affect the passive heat removal rate and thus the peak temperature that the core and, subsequently, the fuel particles will experience during off-normal events. Determining changes to the conductivity as a function of irradiation dose and temperature is important for the safety analysis for the **NTPBMR**.

4.2.3 OXIDATION

The oxidation rate of graphite during an off-normal, air-ingress event is required to determine the effect of oxidation on the specific graphite properties as well as the entire core performance. There are two primary concerns: failure of individual graphite blocks (due to strength and thermal conductivity reduction as a result of pore formation and growth) and general core geometry configuration issues (the entire core fails due to acute oxidation and catastrophic graphite failure). Kinetic models resulting from experimental data will be required to predict weight loss in specific areas of the core. It is expected that the damage will be limited and that core geometry remains intact; however, some data will be required to confirm this assessment.

Additionally, based on regulatory requirements, thermal and mechanical testing of previously oxidized material will need to be performed to determine the chronic effects oxidation may have on graphite material properties. Mechanical and thermal properties will be investigated from both acute and chronic oxidized material. The affects due to chemical and physical (pores) differences for each graphite type will be required.

4.2.4 EMISSIVITY

Emissivity values for graphite must be high enough to allow heat energy to pass across the gap between the core and pressure vessel walls. Graphite emissivity is primarily a function of surface conditions for the graphite components (i.e., roughness, porosity, etc.). The as-received graphite is assumed to have high enough emissivity values to meet the heat conduction values. Confirmation that emissivity values do not degrade extensively due to oxidation and/or irradiation is needed.

4.2.5 SPECIFIC HEAT

There are concerns that energy stored within the graphite microstructure as a consequence of irradiation damage can be released if graphite is raised to a high temperature (the Wigner energy release phenomenon). If there is an off-normal event where the graphite is undergoing air oxidation, this additional stored energy, along with the heat generated from graphite oxidation, may exceed the specific heat value.

It is generally understood that irradiation damage energy (accumulation of Frenkel defects) is only available when it is “frozen” in the microstructure at low irradiation temperatures (< 300°C). When graphite is irradiated at temperatures higher than 300°C the increased point defect mobility affectively anneals out this accumulated damage within the graphite microstructure as fast as defect pairs can occur. Thus, at higher irradiation temperature it is assumed that Wigner internal energy release will be minimal, but some testing to determine the stored energy levels within graphite will be required.

4.2.6 ELECTRICAL RESISTIVITY/CONDUCTIVITY

Electrical conductivity values for graphite are not specifically required. Electrical conductivity is used as a rapid, simple means to determine grain orientation, structure, and crystallinity of the graphite. In conjunction with optical microscopy, it can be used to determine the microstructural texture of the graphite components without a great deal of sample preparation work.

4.3 MECHANICAL

The graphite single crystal is highly anisotropic due to strong covalent bonds between the carbon atoms in the basal in the plane and weak van der Waals bonds between the basal planes. This anisotropy is transferred to the filler coke particles and also to the crystalline regions in the binder phase. Thus, the mechanical and physical properties of graphite vary within a billet due to texture introduced during forming and thermal processing. Moreover, there is statistical variability in the properties between billets within the same lots, between lots, and between batches due to variations on raw materials, formulations, and processing conditions. Accurate characterization of the mechanical properties is fundamental to determining the induced and applied stresses to the graphite components. Determining the resulting stresses in (and on) the components from exposure to a reactor environment is necessary to calculate the ability of the graphite to withstand the imposed loads and continued service conditions during operation. Therefore, it is necessary to develop a statistical data base of the properties for a given graphite grade.

4.3.1 IRRADIATION CREEP

Strain relief of induced stresses (i.e., irradiation creep) within irradiated graphite microstructures allows the graphite to withstand irradiation damage. However, graphite will continue to suffer from irradiation creep even after initial internal stresses are relieved (i.e., primary irradiation creep), resulting in continued dimensional changes. The resulting macroscopic behavior is similar to the changes in CTE discussed above. Thermal creep does not occur at expected **NTPBMR** operating temperatures.

Because irradiation creep can alter the underlying microstructure, it can affect the material properties in nuclear graphite during long-term exposure. The graphite performance and changes to the material microstructure and properties during long-term exposure must be characterized and understood to validate the design and establish accurate lifetimes for our new graphite type.

Finally, since irradiation creep specimens are physically large, it is relatively easy to irradiate a large number of specimens simultaneously inside an irradiation creep experiment. Therefore, while investigating irradiation creep rates, all other irradiated material property values can also be determined utilizing both the creep samples and piggy-back irradiation samples within an irradiation capsule.

4.3.2 ELASTIC CONSTANTS AND STRESS-STRAIN CURVES

The mechanical properties of graphite are necessary to determine the structural integrity of graphitic components. These properties are vital to determining the viability of the structural strength and integrity of the reactor core. The as-received and irradiated values are needed for whole core models, which will be used for the graphite design code. Specific material properties required for the whole core modeling are listed in

4.3.3 TRIBOLOGY (wear/friction)

Tribology is primarily a concern with pebble-bed designs. The concern is that wear on the pebbles during movement can generate dust, which will act as a means for transporting fission products during loss of coolant. To determine the amount of dust to be generated, the tribological properties of the graphite must be determined.

5. TECHNOLOGY DEVELOPMENT PLAN

The scientific and engineering techniques described within this section encompass all the anticipated tests required to validate and qualify nuclear grade graphite for use within the **NTPBMR**. The plan presented here only represents the information needed for the granting of a license to construct a research prototype **NTPBMR.** The test matrixes shall be limited to reduce the scope of the testing in support of a limited licensing strategy (i.e., demonstration plant license) to meet **NTPBMR** deployment schedules. The high dose irradiation experiment are included here for completeness. Ultimately, data from all tests will be required for commercialization of the **NTPBMR** technology in order to use a new graphite type within an **NTPBMR**. Constitutive relationships and model development using the data acquired from this R&D program will be required for codification of the graphite. The appropriate role of model development and the extent of development are discussed as it pertains to perceived ASME and regulatory requirements.

5.1 EXPERIMENTAL DATA

Since many graphite components will be exposed to the full neutron flux generated in the **NTPBMR** core, any changes to pertinent material properties must be determined to understand the long-term behavior of the graphite in reactor. As a consequence, an extensive non-irradiated and irradiated material characterization program is planned. The non-irradiated characterization program focuses on developing a statistically valid material database for each of the graphite types selected for irradiation testing. This will establish baseline values for material properties that can be used to determine the quantitative changes during irradiation. A large irradiated specimen population will then be exposed to the expected **NTPBMR** reactor environment (temperature and dose ranges). As stated previously, this experiment is expected to yield pertinent information for all irradiated material property values necessary to qualify a new nuclear grade graphite. Specific descriptions of test sample preparation, non-irradiated material characterization, irradiation experiment descriptions, and material characterization comprising the experimental data needs are outlined below.

5.1.1 TEST SAMPLE PREPARATION

Before any material characterization testing (non-irradiated or irradiated) can be carried out, an optimal method of machining the graphite samples from the bulk material must be developed to ensure representative samples can be obtained. The **NTPBMR** program has to develop an extensive sample cutting and sectioning plan to guarantee not only statistically valid sample numbers but also spatial validity so that microstructural

changes within the bulk material (i.e., billet) affecting material property changes are well characterized. Particular attention will be given to the traceability of each specimen to its spatial location and orientation within a billet. These graphite billet cutting plans will be developed to promote a more complete or finer resolution material property "mapping" of material property changes within the billets. This will be achieved by maximizing the number of test specimens that could be obtained from each billet. However, to provide statistically significant results from the various test methods, a minimum of four samples are needed from each location/orientation within the same billet (per ASTM methodologies). Since this is physically impossible, it is assumed that the billets have some level of symmetry in material properties throughout the entire structure, which allows samples from different sections of the billet to effectively be "similar" with respect to material properties. Using samples from similar locations within each billet section will yield enough samples to provide for statistical validity within a single billet.

To facilitate machining, the billets are cut into successively smaller sections designated as "slabs", and each slab is sized to accommodate the proper grain orientation within the test specimens. All test specimen blanks machined for a given slab will have the same grain orientation (ag or wg) as the slab. The slabs will be further sectioned into sub-slabs to allow the rectangular test specimen blanks to be machined to the correct size. Finally, a tracking methodology is used that will account for every specimen machined from a graphite billet. A unique identification number will be assigned to each test specimen providing the exact location and orientation of the sample within the graphite billet. This identification system is based on the cutting methodology to provide an easy and concise method for identifying the different samples. This methodology (and the corresponding assumptions) will be used for producing all test specimens for material characterization. Since a large portion of the testing is with irradiated samples, a minimum specimen size must also be considered for volume restrictions within a materials test reactor. Each material test will depend on the specific graphite's grain size since ASTM test standards call for specimen sizes to have cross-sectional diameters of at least 5X the grain size across the stressed gauge section of the sample. This sizing requirement helps ensure representative and repeatable testing results. However, most graphite material tests use a minimum of 10X of the maximum grain size for the cross-sectional diameter across the gauge section. Thus, for a graphite with a 1-mm grain size, the minimum diameter in the test gauge area for a typical tensile specimen must be ~10-mm. Fabricating smaller test specimens is not allowed since they will not provide representative material properties across such a few number of grains in the material microstructure.

5.1.2 NON-IRRADIATED MATERIAL TESTING

Baseline, "as-received" material properties for each graphite type are needed to establish accurate thermal and mechanical response of the core. Since material properties are expected to vary throughout the rather large billets or blocks of graphite, mapping the magnitude and spatial positions of variability is important to determining an

individual component's material properties. To enable credible core designs and to support the ongoing development of a probabilistic graphite design methodology, the maximum variability within graphite components must be well characterized. For example, if the compressive strength is reduced significantly near the edges of the billets, a graphite support column fabricated from a position near the edge may not possess sufficient strength to support the weight of the core blocks above. Thus, determining where the strength begins to be reduced within the larger billet and by how much is important for determining where to fabricate an individual graphite component to meet specific design requirements within the core.

A complicating factor is the variability not only within the individual billets but also from billet-to-billet and finally lot-to-lot. These within-billet, intra-billets, and lot-to-lot variations of the graphite must be accounted for in a statistical manner to determine the maximum range of material property variations expected for components machined from an "average" billet. Such a statistical material property database can only be obtained from extensive non-irradiation characterization of samples taken within billets and compared to sample between different billets and different graphite lots.

Physical, thermal, and mechanical property testing of multiple graphite samples from a large billet sample matrix is necessary for determining the proper statistical ranges of values. The appropriate sample matrix size, sample geometry, and sample dimensions as described above will be important to establishing statistical validity. All material tests to be used to build this material property database are described later in this section. Once the non-irradiated, "as-received" material properties have been determined, the changes due to irradiation will be determined from post-irradiation examination (PIE) and characterization studies on representative graphite types.

5.1.3 IRRADIATION EXPERIMENTS

A series of irradiation experiments will be required to determine the graphite response under irradiation. After the graphite sample matrix is chosen, the irradiation conditions are determined based upon the expected reactor conditions. PIE and characterization entails performing the same physical, thermal, and mechanical tests as described for the non-irradiated materials, only this time with irradiated graphite samples. As discussed previously, thermal creep of graphite is not expected at the temperatures experienced in the reactor core (< 1000°C), so determining the non-irradiation creep rate is not required. However, irradiation induced creep in graphite is expected at these temperatures and will play an important role in the irradiated behavior of the graphite during reactor service. Thus, irradiation creep experiments will form a significant part of all irradiation studies for graphite types in the **NTPBMR**.

5.1.3.1 AGC EXPERIMENTS

The Advanced Graphite Capsule (AGC) experiments are designed to provide irradiation creep rates for moderate doses and higher temperatures of leading graphite types that will be used in the **NTPBMR** reactor design. The experiments are designed to provide not only static irradiation material property changes but also to determine irradiation

creep parameters for actively stressed (i.e., compressively loaded) specimens during exposure to a neutron flux.

As shown, the dose and temperature is bounding for the prismatic reactor design (dpa ~ 5 – 6 at 1100°C) for both fuel and front facing reflector blocks. This dose limit is intentionally below the expected point of turnaround for the current **NTPBMR** graphite types at normal **NTPBMR** operating temperatures. Only AGC-6 experiment (6 – 7 dpa at 1200°C) may approach expected turnaround limits for the selected **NTPBMR** graphite types.

To determine when (and if) turnaround will occur for the selected **NTPBMR** graphite during exposure in the AGC-6 capsule, an additional experiment –high-temperature vessel (HTV) has been postulated. The 18HTV 1 and 2 capsules are simple "drop-in" capsules with the exposure parameters illustrated in Figure 7. As shown, these experiments are operated at much higher temperatures (inducing faster turnaround) but at lower doses. As this is a simple dimensional change experiment to determine when turnaround may occur, the graphite is not loaded during irradiation. A detailed description of the HTV 1 and 2 experiment is presented in ORNL-GEN4/LTR-06-019 Since the prismatic **NTPBMR** design estimates that reflector blocks can be replaced well before turnaround should occur at normal operating temperature (<5 – 6 dpa) and fuel blocks are replaced after only two cycles (<4 – 5 dpa), the AGC experiment should fully bound the graphite experience within a prismatic design. The dpa levels achieved in the AGC experiment will not, however, fully bound the pebble-bed **NTPBMR** design for high-dose reflector blocks (see below). But, it will certainly provide preliminary data for the first 20 – 25% of the expected dpa levels for these graphite components. Graphite components located farther from the core region will have correspondingly less dose and operate at much lower temperatures than the fuel region. As a consequence, turnaround and irradiation creep levels for these peripheral graphite components will be at significantly longer times and lower rates and should be fully bounded by the AGC data.

All six capsules comprising the AGC experiment will be irradiated in the South Flux trap of the Advanced Test Reactor (ATR). This will require sequential irradiation campaigns for each capsule. Each capsule will contain approximately 400 specimens – 90 large irradiation creep specimen pairs and over 300 "piggy-back" specimens. The smaller and non-stressed "piggy back" specimens are located in the center channel in the experiment. Other "piggy backs" are used in the lower non-stressed creep specimen channels as offset specimens to account for the slight asymmetry in the ATR flux profile. The larger irradiation creep specimens are sub-divided into six columns of 15 specimens each. Each of the six columns contains stressed and non-stressed specimens. The symmetry of the flux buckling is used to irradiate each stressed and non-stressed specimens at the same fluence level. There are seven non-stressed specimens below core centerline and eight stressed specimens above core centerline. The creep measurement is made on the dimensional difference between the stressed and non-stressed specimens irradiated at the same fluence and temperature. Figure 8 shows the arrangement of the graphite specimens in the experiment.

All specimens are maintained at a constant temperature during exposure times of between six and 18 months depending on the required dose. PIE characterization is projected to take approximately 14 – 18 months for each capsule even though irradiated graphite samples can be contact handled after a short decay period (~ 6 months).

5.1.3.2 HIGH DOSE IRRADIATION EXPERIMENTS

The high-dose experiment is designed to provide irradiation exposure for very high doses and moderate temperatures. As noted above, the pebble-bed design expects the facing reflector blocks (inner and outer reflector) to operate at much longer times and thus withstand a maximum of irradiation damage before the core is shutdown, de-fueled, and the blocks replaced. Current expectations are for the reflector blocks to operate approximately 20 – 25 years before replacement. At the higher end of the dose range noted above for a pebble-bed **NTPBMR** design, this can correspond to as much as 25 dpa before change-out. While this appears to be a very large dose, the expected temperature ranges are lower than for a prismatic design resulting in longer turnaround times and slower irradiation creep rate. If a pebble-bed design is selected, the graphite material property changes for the higher expected dose levels will be required. A high dose creep experiment exposing selected graphite to much longer dose levels at moderate temperatures has been tentatively planned in support of this design selection.

As shown, these samples will be exposed to dose levels considerably higher than expected turnaround dose levels even at the moderate exposure temperatures. In addition, the irradiation creep samples will be tensile loaded during exposure to ensure optimal creep rates (Section 5.1.4). However, since these dose levels are expected after 25 years of service, the high dose experiments are not needed for initial material property ranges specifically required for reactor licensing and startup operations. Results stemming from this experiment can (and most likely will) be delayed for a few years until after reactor startup since data from the AGC experiments will provide sufficient data to support operations for at least 6 – 7 FPY operation of the reactor. The data from the high-dose experiment will be required if any graphite reflector block will be exposed to doses higher than 6 – 7 dpa.

5.1.4 MATERIAL CHARACTERIZATION

The following sections describe the material tests anticipated for all non-irradiated and irradiated examination and characterization studies. These material tests will be applied to both irradiated and “as received” graphite samples to ascertain the changes to the material properties resulting from a neutron radiation field. Where possible, ASTM standard test methods will be employed. If no test standard exists, some additional activity may be required to develop a test standard.

5.1.4.1 PHYSICAL TESTING

5.1.4.1.1 MICROSTRUCTURE CHARACTERIZATION

Optical microscopy measurements of grain size, morphology/anisotropy, and pore size/distribution will be used to determine the graphite microstructure. Adjacent optical samples will necessarily be taken as close to test specimens (with similar orientation) as

possible from within the graphite billet, and the microstructure will be inferred to the test sample microstructures. In conjunction with the optical microscopy, non-destructive X-ray tomography (CT) will be investigated for its ability to ascertain the microstructure within individual test samples. X-ray techniques will allow samples to be analyzed before being tested using additional techniques (both non-destructive and destructive). Image analysis techniques will be used to enhance information such as internal pore sizes and pore structure throughout the microstructure.

Changes to the microstructure due to thermal, irradiation, and stress history will be compared to the original microstructure. Microstructural evolution and modifications as a result of exposure to reactor environments can then be established. Non-destructive methods for large scale analysis (i.e., manufacturing, ISI methods, etc.) will need to be developed. CT methods may be possible but have limited resolutions for these large component sizes. Ultrasonic testing (UT), electrical resistivity/conductivity, impact echo, or other techniques will have to be developed to meet both ISI requirements as well as billet characterization for manufacturing QA.

5.1.4.1.2 MASS AND DIMENSIONAL MEASUREMENTS

Precision measurements of all irradiation test samples will allow macroscopic dimensional changes and pore formation estimates to be determined. Volumetric and density changes will be calculated and compared to pre-irradiation values for each test sample.

5.1.4.2 THERMAL TESTING

All thermal (and electrical) samples will be button samples having dimensions equal to or less than 12mm diameter x 6mm thickness. These small sample sizes allow for many specimens to be made available for both irradiated and non-irradiated testing. In addition, the small size also allows thermal samples to be machined from the ends of mechanical test specimens, if needed. This ensures special 22uniformity of measurements of relevant thermal, physical, and mechanical material properties within the graphite billet characterization. Additionally, “re-using” the same samples allows for larger sample batches within the irradiation test trains.

5.1.4.2.1 THERMAL EXPANSION AND CONDUCTIVITY

Thermal expansion and conductivity values will be obtained from graphite button samples within a laser flash diffusivity analyzer to temperatures of 1600°C (off-normal maximum temperature). Non-irradiated and irradiated button samples will be prepared for testing at all temperature ranges of interest.

5.1.4.2.2 OXIDATION

There are currently no approved ASTM test methods for measuring the oxidation rate of graphite. The **NTPBMR** program will assist in the development of this test standard. After development, the test will be used to ascertain the oxidation rate of selected

graphite types for a variety of operating conditions. Results will be used to develop kinetic models to predict weight loss in specific areas of the core.

Additionally, based on regulatory requirements, thermal and mechanical testing of previously oxidized material will need to be performed to determine the effects oxidation may have on graphite material properties. A core configuration issue as a result of air ingress is establishing that, whatever mitigation techniques are selected for this event in HTGR, an assessment that the damage is limited and that core geometry remains intact is important. However it should be mentioned that for the **NTPBMR** technology this event will not be an issue because of the passive injection of carbon-dioxide in the core in the event of a **LOCA.** In addition, mechanical and thermal properties will be investigated from both acute and chronic oxidized material.

5.1.4.2.3 EMISSIVITY

Limited confirmatory measurements of emissivity values for graphite will be measured using standard techniques (i.e., infrared based, etc.). NRC PIRT requirements will demand some comparative studies to determine any changes in emissivity resulting from oxidation and/or irradiation.

5.1.4.2.4 SPECIFIC HEAT

All thermal specimens will be subjected to analysis via Differential Scanning Calorimetry (DSC) to determine the specific heat for individual samples. Changes to the specific heat due to oxidation and/or irradiation will be compared to as-received values. In addition, previously irradiated samples from AGC capsules will be monitored to ascertain the potential reduction in specific heat due to the release of high temperature Wigner energy. These will be limited confirmatory studies to ascertain the potential for Wigner energy storage at the lower **NTPBMR** irradiation temperatures.

5.1.4.2.5 ELECTRICAL RESISTIVITY/CONDUCTIVITY

Electrical conductivity/resistivity values will be measured through sample button resistivity measurements. Microstructural characteristics will be compared to optical and CT results. These tests will be performed as possible based on the material geometry and size.

5.1.4.3 MECHANICAL TESTING

Mechanical testing is the most extensive and complex part of the graphite test program. Strength, irradiation creep, fracture toughness, and multi-axial testing procedures utilize complex sample geometries, complicated testing techniques, and take a long time to perform. Therefore, the techniques and plans outlined for these mechanical tests, such as the irradiation creep tests, require careful consideration.

5.1.4.3.1 IRRADIATION CREEP

An extensive irradiation creep program is needed to characterize graphite creep response as part of a larger irradiated materials characterization program. A large

sample population (both irradiation creep and piggy-back specimens) will need to be exposed to the expected **NTPBMR** PMR design in the AGC experiment. If property changes within graphite for higher doses are required, then a second, high-dose irradiation experiment will be implemented.

Generally, at doses below turnaround (0 – 6 dpa for **NTPBMR** graphite grades) in the normal operating temperature regime expected for **NTPBMR** (~1000 – 1200°C), both compressive and tensile irradiation creep rates are similar. As a consequence, conducting irradiation creep with a compressive load should yield the same response as in a tensile stress. This assumption is true until turnaround occurs. Since turnaround is a function of both temperature and dose (dpa), those graphite types exposed to higher temperatures will experience turnaround at correspondingly lower doses. After turnaround, graphite loaded in tension enters into a non-linear (tertiary) creep regime where the creep rate is significantly increased (c-axis growth and pore formation). Tensile stresses either promote or at the very least allow unhindered strain relief during irradiation, providing a "worst-case" creep rate for the graphite types exposed to higher doses (see Figure 11). Compressive loads, after turnaround, will tend to retard the creep rate and effectively delay the tertiary creep regime. Therefore, once turnaround has been achieved, graphite samples should be in a tensile stress state to determine the fastest rate of irradiation creep possible within the graphite.

5.1.4.3.2 AGC EXPERIMENT

The AGC experiment is designed to provide irradiation creep rates for moderate doses and higher temperatures of leading graphite types that will be used in the **NTPBMR** reactor design. The experiment is designed to induce irradiation creep within the secondary regime, thus allowing the graphite to be compressively loaded during irradiation, which simplifies the experiments considerably. Static compressive loads of 14.5 – 20 MPa (2-3 Ksi) are applied to the graphite during irradiation. The temperature and dose regimes covered by the AGC experiment are illustrated in Figure 7. As shown, the dose is intentionally below (or possibly at) the point of turnaround within the graphite at the normal **NTPBMR** operating temperatures. Only the AGC-6 experiment (6 – 7 dpa at 1200°C) will approach expected turnaround limits for current **NTPBMR** graphite types. Since the AGC experiments are a comparison measurement between stressed and unstressed irradiated specimens, if turnaround were to occur during AGC-6 exposure, the creep rate results would be affected by the compressive loading state of the graphite. Results from the HTV experiment will provide both turnaround and high temperature irradiation data for all selected graphite types. Turnaround data from these experiments will be used to adjust the exposure, loading, and temperature limits for AGC-6 to extrapolate as much accurate information from it as possible.

5.1.4.3.3 HIGH DOSE IRRADIATION CREEP

As noted previously, the pebble-bed design expects the facing reflector blocks (inner and outer reflector) to operate at much longer times and thus withstand a maximum of

irradiation damage before the core is shutdown, de-fueled, and the blocks replaced. Current expectations are for the reflector blocks to operate approximately 20 – 25 years before replacement. At the higher end of the dose range noted above for a, **NTPBMR** pebble-bed design, this can correspond to as much as 25 dpa before change out of the reflector blocks. While this appears to be a very large dose, the expected temperature ranges are lower than for a prismatic design, resulting in longer turnaround times and slower irradiation creep rate. For the pebble-bed design the creep rate and resultant strain from these higher doses must be determined for accurate lifetime predictions. This will require an extensive design development program to determine an optimal tensile loading configuration that can withstand long-term exposure (i.e., four years within ATR or 2.5 years within the High-Flux Isotope Reactor [HFIR]). In addition, sample size, geometry, and matrix size will need to be considered to determine the most advantageous MTR to use for this experiment.

One benefit is that only one graphite type will be required for these tests since the **NTPBMR** pebble-bed design is currently interested in only a single graphite type. Thus, the sample matrix can be significantly reduced allowing multiple MTRs to be considered. However, similar to the AGC experiment, the test temperatures, fluences, and tensile loads must be constant during the test.

5.1.4.3.4 ELASTIC CONSTANTS AND STRENGHT TESTING

Standard strength testing techniques using stress-strain (_-_) curve relationships will provide the bulk of the mechanical material properties. Extensive testing programs for both non-irradiated and irradiated graphite samples will be necessary to (1) prove consistency between billets and lots of graphite, (2) provide baseline material property data, and (3) quantitatively demonstrate the material property changes as a result of exposure to a **NTPBMR** environment.

ASTM test standards call for specimen sizes to have cross-sectional diameters of at least 5X the grain size across the stressed gauge section of the sample to provide representative and repeatable testing results. Traditional practices tend to use a minimum of 10X of the maximum grain size for the cross-sectional diameter of the gauge section. Thus, for a typical large-grained graphite (i.e., NBG-18) with a maximum grain size of 1.6 mm, the test gauge section will need to be at least 16mm across to provide accurate mechanical property values. Most other graphite types have considerably smaller grain sizes and may use smaller gauge sections.

This imposes a minimum sample size that in many cases may be too large for most MTRs. The ATR facility can and will be used to accommodate the larger-sized specimens, but multiple reactors are anticipated to meet all the irradiation needs. This may force the use of much smaller specimens that are not included in standard ASTM testing methods. A program to develop miniature test specimens for irradiation testing will need to be developed for graphite. This is similar to the on-going miniature sample development for irradiated metal samples. In either case, careful determination of the optimal sample size is required for each mechanical test before irradiation.

Specialized grips are required for the testing of graphite, as specified in the test standards. This is especially true for miniature test specimens that will require specialized fixtures to accommodate the small size. These specialized fixtures must be identified and developed prior to mechanical testing activities.

Finally, some elastic constants will be obtained using non-destructive UT methods. Standard testing procedures will be used to obtain dynamic elastic modulus values (tensile and compression) for specific samples.

TABLE OF MECHANICAL PROPERTIES

Property	Test Standard
Static and dynamic elastic modulus	ASTM C747-05, Standard Test Method of Elasticity and Fundamental Frequencies of Carbon and Graphite Materials by Sonic Resonance
Poisson's ratio	ASTM C747-05, Standard Test Method of Elasticity and Fundamental Frequencies of Carbon and Graphite Materials by Sonic Resonance
Strength values	
flex	ASTM C1161-02c, Standard Test Method for Flextural strength of Advanced Ceramics at Ambient Temperature
tensile	ASTM C749-02, Standard Test Method for Tensile Stress-Strain of Carbon and Graphite
compression	ASTM C695-05, Standard Test Method for Compressive Strength of Carbon and Graphite
Strain to failure	ASTM C565-93, Standard Test Method for Tension testing of Carbon and Graphite Mechanical Materials
Fracture toughness : (Klc, Glc, _f)	Under development
Multi-axial failure criteria	Under development

5.2 MULTI-SCALE MODEL DEVELOPMENT

Models are required to allow the designer to assess the condition of graphite components and core structure design margins at any point in the lifetime of the reactor. The models are needed to describe interactions between graphite components, specifically, the behavior of the stack of graphite blocks making up the core moderator and reflector. Specific models should be able to calculate external loads imposed upon the components, internal stresses resulting from radiation and temperature induced dimensional changes, movement of components (i.e., dimensional clearances for control rod insertion), and estimates of residual strength both with and without environmental attack (i.e., air-ingress during offnormal event).

Modeling the behavior of a graphite core is complex and will require some fundamental understanding of the graphite physical, thermal, and mechanical behavior as a function of irradiation temperature and neutron fluence. However, the primary objective of these models is to provide the ability to calculate in-service stresses and strains in graphite components and estimate the structural integrity of the core as a whole. Thus, understanding of fundamental mechanistic material behavior during operation will be limited to those aspects required to understand the response of the entire core both during normal operation and during off-normal events (e.g. predict seismic behavior of the core). A physics-based understanding of microstructural damage and its effects on materials structure and properties will provide an initial start to estimating the amount of changes to a graphite component can be estimated but the degree of change is unique to the specific nuclear graphite grade used and these fundamental principles must be supplemented with actual experimental material property data to provide a complete analysis of the core behavior.

For example, the existence of temperature and flux gradients within the core and individual components will generate differential changes in dimensions and, hence, stress. Such stresses will creep out (relax) at the expected temperature and fluence levels experienced during normal operation. In addition, stresses arising due to thermal gradients will also creep out during operation, but will reappear in the opposite sense when the core cools during reactor shutdown. To model the core and component stresses during operation (and cool down), the change in properties of the graphite as a function of temperature and neutron dose must be known. Since the point-to-point flux and temperature data will not be available for all combinations of dose and temperature for the required properties, behavioral models are required to estimate the stress states for all components throughout the core. Thus, the whole core models must use a combination of experimentally derived material properties underpinned by an understanding of the fundamental physics to account for all variations possible within the graphite components of the core. Consequently, a major goal is the development and validation of multi-scale models for the behavior of graphite, core components, and whole graphite cores for use in licensing and continued operational safety assessments.

5.2.1 WHOLE GRAPHITE CORE AND COMPONENT BEHAVIOR MODELS

Finite element models (FEM) are required to define the core condition at all times during core life. Such models will take core-physics and thermo-hydraulics inputs for point dose and temperature values and apply graphite material behavior models to calculate the changes in properties with neutron dose, temperature, and oxidative weight loss. Core and component-scale models will allow designers to predict core and core block (e.g., reflector or fuel element) dimensional distortion, component stresses, residual strength, and probability of failure during normal or off-normal conditions. Finite element based codes such as COMSOL™ and ABAQUAS™ offer platforms upon which the desired whole core/component behavioral models may be assembled. It is anticipated that reactor vendors will have their own custom codes to describe and predict the behavior of the core within their particular design for **NTPBMR**. However, independent validation of

these whole core-scale models may be requested by the **NTPBMR** project to ensure the safety envelope of the core during normal and off-normal operating conditions.
Finally, the development and utilization of such codes is an integral part of the design process and is recognized as such by the ASME graphite core components design code, which is currently under preparation by a sub group of ASME B&PV Sect III (nuclear). The sub-group is currently benchmarking core component stress models against a standard set of problems (data sets). Additional validating data for the developed models will come from large multiaxial load specimen testing and, ultimately, from full-scale core components tests.

5.2.2 MACRO-SCALE MATERIALS BEHAVIOR MODELS

Materials behavior models are needed to predict the effects of temperature, neutron dose, and oxidation weight-loss on key physical and mechanical properties. The material behavior results from these models are validated through an extensive program of non-irradiated and irradiation characterization experiments. The properties of interest include:

- CTE and thermal conductivity, specific heat
- Strength (tensile, compressive, flexural)
- Fracture behavior
- Elastic constants (Young's modulus, shear modulus, Poisson's ratio)
- Creep coefficient(s).

The material property models must also take into account the interaction of effects (e.g., neutron damage and weight loss) and the interdependency of effects (e.g., effects of stressed dimensional change [creep]) on the physical properties of graphite. Materials models must be physically based (i.e., based on the materials structural changes) and should incorporate structural damage models and existing physics based models (e.g., phonon conduction). Existing material property models (in some cases empirically derived models) must be evaluated and new or improved materials behavior models developed. Material property values needed for validation of these models will be obtained from the experimental characterization research, as described above (i.e., AGC experiment, non-irradiated property characterization, and possibly high-dose experiments). Particular emphasis will be placed on this aspect of the multi-scale modeling as it is most directly applicable to the **NTPBMR** R&D program. Individual vendor designs are expected to significantly influence the whole core-scale modeling efforts, and as such, the majority of the development effort for whole core-scale models is expected to reside with the vendors. However the material property models necessary for predicting graphite component and core behavior will be essential to developing and validating the whole core-scale models.

5.2.3 MICRO/NANO-SCALAE MODELS

Nano- and micro-scale modeling provides a fundamental understanding of material behavior. *Ab-Initio* models of the atomistic phenomena occurring on irradiation will allow prediction of the displacement damage that can occur and may shed light on the crystal deformation modes. Simulations (e.g., Density Function Theory) of defect structures for relevant combinations of dose and temperature can provide the basis of determining crystal strains. Understanding the physical interactions of the graphite crystallites and the inherent porosity within and around the crystallites is crucial to building microstructural models for the behavior of polycrystalline graphite. Similarly, the deformation processes that occur within the crystallites when graphite is subjected to stress, either externally applied or those that develop within the graphite due to dose and temperature gradients, must be understood and modeled. Surprisingly, after ~60 years of graphite use in reactors, the microstructural mechanism of irradiation creep and crystal deformation are still being questioned and are not fully elucidated. Recent fundamental studies by Heggie, et al (University of Sussex Group, UK), have suggested displacement damage structures previously considered improbable are indeed energetically favorable, indicating the need for further study. Crystallite damage observations using Transmission Electron Microscope (TEM), coupled with Scanning Electron Microscipe (SEM) and CT studies of irradiated graphite will provide mechanistic data for structural models.

As indicated above, development of nano- and micro-scale models will underpin the macro-scale materials property models, as well as provide valuable input for experimental validation requirements. However, fundamental studies and micro-scale modeling should be supportive of the material property models to enable a basic understanding of the mechanisms driving the material property changes. While important, less direct emphasis will be placed on complete development of these nano-models than on the material property models discussed previously. Some support will be required to fully understand the underlying principles that induce changes to the material properties, but the majority of the work will be left to long-range research funding sources.

ASCENTRUST LLC
The POWER of ENGINEERING

APPENDIX
INTRODUCTION TO NUCLEAR POWER
AND
INTRODUCTION TO NTPBMR TECHNOLOGY

SECTION I: Acronyms

LIST OF ABBREVIATIONS

ACNW	Advisory Committee on Nuclear Waste
ACR	Advanced CANDU Reactor
ACRS	Advisory Committee for Reactor Safeguards
AE	Architect/Engineer
AEC	Atomic Energy Commission
AECL	Atomic Energy of Canada, Ltd.
AGR	Advanced Gas Reactor
AHTR	Advanced High Temperature Reactor
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
ANPR	Advance Notification Of Proposed Rulemaking
ANS	American Nuclear Society
AO	Abnormal Occurrence
AOO	Anticipated Operation Occurrences
AOT	Allowable/allowed outage time
ASME	American Society of Mechanical Engineers
ASP	Accident Sequence Precursor
ATHEANA	A Technique for Human Event Analysis
ATWS	Anticipated Transient Without SCRAM
AVR	Arbeitsgemeinschaft Versuchsreaktor
BDBT	Beyond Design Basis Threat
BEIR VII	Biological Effects of Ionizing Radiation
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CANDU	Canadian Deuterium-Natural Uranium Reactor
C/C	Carbon/carbon-fiber composite
CCDF	Complementary Cumulative Distribution Function
CCF	Common-cause failure
CCFP	Conditional Containment Failure Probability
CCGT	Combined-Cycle Gas Turbine
CCWS	Closed Cooling Water System
	Component Cooling Water System
CD	Civil Defense
C/D	Cool-down
CDA	Containment Depressurization Actuation
	Core disruptive 'accident'
CDAs	Controls, Displays, and Alarms
CDBA	Containment Design-basis Accident

CDC	Center for Disease Control
	Computer Design Code
	Control Data Corp.
CDE	Condensate Demineralization Effluent
CDF	Core Damage Frequency
	Cumulative Damage Function
CDM	Central Data Management
.	Certified Design Material
CDN	Corporate Data Network
CDP	Cask Decontamination Pit
CDPA	Civil Defense Preparedness Agency
CDP	Core damage probability
CDR	Conceptual design requirement ,..
CDRG	Catastrophic Disaster Response Group
CD-ROM	Compact disk/read-only' memory
CDS	Cask Decontamination Station
	Component Disassembly Station
	Computer Data Screening
	Conceptual design study
	Condensate Demineralization Subsystem
	Current Disposal Site
CDV	Capacitance discharge vaporization
CE	Combustion Engineering Inc.
	Commonwealth Edison Co.
	Conductivity Element
	Consumer Electronics'
	Corps of Engineers, U.S. Army
C/E	Calculation/experiment
CEA	Cambridge Electron Accelerator
	Atomic Energy Commission (France)
CFC	Carbon-fiber Composite
CFR	U.S. Code of Federal Regulatio*ns*
CFR 10-50	U.S. Code of Federal Regulations for Construction Permit and Operator's License
CFR 10-52	U.S. Code of Federal Regulations for Combined Construction Permit and Operator's License
CHP	Cogeneration of Heat and Power
CLB	Current Licensing Basis
CLIIP	Consolidated Line Item Improvement Process

CNSI	Chem-Nuclear Systems, Inc.
CO	Carbon Monoxide
CO_2	Carbon Dioxide
CP	Construction Permit
CPEF	Conditional Probability of Early Fatality
CPLF	Conditional Probability of Latent Fatality
CRCPD	Conference of Radiation Control Program Directors
CRGR	Committee to Review Generic Requirements
CRMP	configuration risk management program
CSNI	Committee on the Safety of Nuclear Installations
DBA	Design Basic Accident
DBT	Design Basic Threat
DCF	Dose Conversion Factor
DG	diesel generator
	draft guide
DOE	Department of Energy
DPO	differing professional opinion
DSI	direction-setting issue
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
ECI	Emergency Cooling Injection
EDO	Executive Director of Operations
EF	Early Fatality
EIS	Environmental Impact Statement
EP	Emergency Preparedness
EPA	Environmental Protection Agency
EPIX	Equipment Performance and Information Exchange
EPRI	Electric Power Research Institute
EXF[1]	1 rem Exeedance Frequency
ESBWR	Economic and Simplified Boiling Water Reactor
ET	Executive Team
EVO	Energieversorgung Oberhausen
FAVOR	a probabilistic fracture mechanics code
FBR	Fast breeder Reactor
FC	Frequency Consequence
FCSS	Division of Fuel Cycle Safety and Safeguards (NMSS/FCSS)
FIMA	Fission per Initial Metal Ion
FSAR	Final Safety Analysis Report
FTE	Full-time Employees

F-V	Fussell-Vesely
FY	Fiscal Year
GAO	Government Accountability Office
GDC	General Design Criterion/criteria
GFR	Gas-Cooled Fast Reactor
GEM	graphical evaluation module
GIF	Generation IV International Forum
GL	generic letter
GPRA	Government Performance and Results Act
GQA	graded quality assurance
GSI	Generic Safety Issue
GTCC	Greater than Class C
GT-MHR	Gas Turbine-High Temperature Reactor
GTHTR	Gas Turbine High Temperature Reactor
HAZAP	Hazard and Operability Analysis
HC&SCC	Hydrogen Codes and Standards Coordinating Committee
HENDEL	Helium Engineering Demonstration Loop
HERA	Human Event Repository and Analysis
HEU	Highly enriched Uranium
HFIR	High Flux Isotope Reactor
HFR	High Flux Reactor
HHV	Hocktemperatur Helium Versuchanlage (High Temperature Helium Test Loop)
HIP	High Isostatic Temperature
HLR	High Level Requirement
HLW	high-level waste
HRA	human reliability analysis
HSE	Health and Safety Executive
HTE	High-temperature Electrolysis
HTGR	High Temperature Gas Reactor
HTTR	High Temperature Engineering Test Reactor
IAEA	International Atomic Energy Agency
I&C	Instrumentation and Control
ICRP	International Commission on Radiation Protection
IDCCS	Integrated Data Collection and Coding System
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and electronics Engineers
IER	Individual Early Risk
IHX	Intermediate Heat Exchanger

ILR	Individual Late Risk
IM	Importance Measure or Measures
IMC	Inspection Manual chapter
IMNS	Division of Industrial and Medical Nuclear Safety (NMSS/IMNS)
INEEL	Idaho National Engineering and Environmental laboratory
INPO	Institute of Nuclear Power Operations
INSRI	International Nuclear Energy Research Initiative
IPEEE	individual plant examination for external events
IPE	individual plant examination
ISFSI	Independent Spent Fuel Storage Installation
ISA	Integrated Safety Analysis
ISGTR	Induced Steam Generator Tube Rupture
ISI	In-service Inspection
ISPRA	Institute for Environment and Sustainability
IST	In-service testing
ITRG	Independent Technology Review Group
JAERI	Japan Atomic Energy Research Institute
JMTR	Japan Materials Test Reactor
KVK	Komponentenversuchskreislauf (German Test Facility)
LBE	Licensing Basis Events
LCO	Limiting Conditions for Operation
LER	Licensee Event Report
LERF	Large Early Release Frequency
LEU	Low Enriched uranium
LF	Latent Fatality
LFR	Lead-Cooled Fast Reactor
LLRF	Large Late Release Frequency
LMR	Liquid metal-Cooled Reactor
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LP/SD	low-power/shutdown
LPZ	Low Population Zone
LRS	low-risk-significant
LT	Leadership team
LTC	Long Term Cooling
LTR	License Termination Rule
LWR	Light Water Reactor
MACCS	MELCOR accident consequence code system
MOR	monthly operating report

MOX	Mixed Oxide
MSLB	Main Steam Line Break
MSPI	Mitigating Systems Performance Index
MSR	Molten Salt Reactor
MW_e	Mega-Watt Electric
MW_{th}	Mega-Watt Thermal
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NERI	Nuclear Energy Research Institute
NFI	Nuclear Fuel Industries
NFPA	National Fire Protection Association
NGNP	Next Generation Nuclear Plant
NMSS	NRC Office of Nuclear Material Safety and Safeguards
NOAK	n^{th} -of-a-kind
NOED	Notice of Enforcement Discretion
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NRS	non-risk-significant
NRR	NRC Office of Nuclear Reactor Regulation
NTPBMR	Nuclear Technology Pebble Bed Modular Reactor
NSSS	Nuclear Steam Supply System
OAS	Organization of Agreement States
OCFO	NRC Office of the Chief Financial Office
ODS	Oxide Dispersion Strengthened
OEDO	NRC Office of the Executive Director for Operations
OKBM	Experimental Designing Bureau of Machine Building (Russia)
O&M	Operation and Maintenance
OL	Operating License
OGC	Office of General Council
OSTP	NRC Office of State and Tribal Programs
PA	Performance Assessment
PAG	Protective Action Guidelines
PBMR	Pebble Bed Modular Reactor
PBPM	Planning, Budgeting, and Performance Management
PCHEs	Printed Circuit Heat Exchangers
PCT	Peak Cladding Temperature
PCU	Power Conversion Unit
PIE	Postulated Initiating Event

PNP	Prototypanlage Nukleare Prozesswaerme (Prototype plant Nuclear Process Heat)
PRA	Probabilistic Risk Assessment
PRASC	PRA steering committee
PRM	Petition for Rulemaking
PSAR	Preliminary Safety Analysis Report
PTS	Pressurized Thermal Shock
Pu	Plutonium
PV	Pressure Vessel
PWR	Pressurized-Water Reactor
QA	Quality Assurance
QC	Quality Control
QHO	Quantitative Health Objectives
RAD	Radiation Absorption Dose
RADS	Reliability and Availability Data System
RASP	Risk Assessment Standardization Project
RAW	Risk Achievement Worth
RBI	risk-based performance indicator
RCCS	Reactor Cavity Cooling System
RCS	Reactor Coolant System
R&D	Research and Development
REM	Roentgen Equivalent Man
RES	NRC Office of Nuclear Regulatory Research
RG	Regulatory Guide
RI	Risk Informed
RIE	Risk Informed Environment
RILP	Risk Informed Licensing Panel
RIPB	Risk Informed, Performance Based
RIRIP	Risk Informed Regulation Implementation Plan
RIS	Regulatory Issue Summary
ROP	Reactor Oversight Process
RPM	Revolutions per Minute
RPV	Reactor Pressure Vessel
RTG	Risk Task Group (NMSS)
SAMDA	Severe Accident Mitigation Design Alternative
SAR	Safety Analysis Report
SAPHIRE	Systems Analysis Program for Hands-on Integrated Reliability Evaluation
SBO	station blackout

SCRAM	super critical reactor axe man
SCSS	sequence coding and search system
SCWR	Super Critical Water Reactor
SDP	significance determination process
SFPO	Spent Fuel Project Office (NMSS)
SFR	Sodium-Cooled Fast Reactor
SI	Sulfur-iodine
SG	Steam Generator
SGTAP	Steam Generator Task Action Plan
SGTR	Steam Generator Tube Rupture
SMR	Steam Methane Reforming
SNM	Special Nuclear Material
SPAR	Standardized Plant Analysis Risk
SR	Stress Relief
SRM	Staff Requirements Memorandum
SRP	Standard Review Plan
SSC	Systems, Structures and Components
STP	South Texas Project
STS	Standard Technical Specifications
TBD	to be determined
TI	temporary instruction
TID	Technical Information Document
TMI	Three Mile Island
TP	Total Population
TRISO	Ceramic-coated-particle fuel
TS	Technical Specification
TSTF	Technical Specification Task Force
TTC	NRC Technical Training Center
TXS	(Siemens) Teleperm XS
UAI	(system) unavailability index
UCO	Uranium Oxycarbide
UL	Underwriters Laboratories
URI	(system) unreliability index
USI	Unresolved Safety Issue
VHTR	Very High Temperature Reactor
V&V	Verification and Validation
WOG	Westinghouse Owners Group

ASCENTRUST LLC
The POWER of ENGINEERING

APPENDIX
INTRODUCTION TO NUCLEAR POWER
AND
INTRODUCTION TO NTPBMR TECHNOLOGY

SECTION J: Glossary of Nuclear Terms

.

ASCENTRUST LLC
The POWER of ENGINEERING

NUCLEAR GLOSSARY

Absorber: Any material that stops ionizing radiation. Lead, concrete, and steel attenuate gamma rays. A thin sheet of paper or metal will stop or absorb alpha particles and most beta particles.

Accelerator: Device used to increase the energy of particles, which then collide with other particles. Major types are linear accelerators and circular accelerators. The name refers to the path taken by the accelerated particle.

Access Hatch: An airtight door system that preserves the pressure integrity of a reactor containment structure while allowing access to personnel and equipment.

Act: means the **Atomic Energy Act** of 1954 (68 Stat. 919) including any amendments thereto.

Activation: The process of making a radioisotope by bombarding a stable element with neutrons or protons.

Active Fuel Length: The end-to-end dimension of fuel material within a fuel element.

Activity: The rate of disintegration (transformation) or decay of radioactive material per unit time. The units of activity are the curie (Ci) and the becquerel (Bq).

Agreement State: A state that has signed an agreement with the Nuclear Regulatory Commission under which the state regulates the use of byproduct, source, and small quantities of special nuclear material in that state.

Air Sampling: The collection of samples to measure the radioactivity or to detect the presence of radioactive material, particulate matter, or chemical pollutants in the air.

Airborne Radioactivity Area: A room, enclosure, or area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations that (1) exceed the derived air concentration limits or (2) would result in an individual present in the area without respiratory protection exceeding, during those hours, 0.6 percent of the annual limit on intake or 12 derived air concentration-hours

ALARA: Acronym for "as low as (is) reasonably achievable." Means making every reasonable effort to maintain exposures to ionizing radiation as far below the dose limits as practical, consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest.

Alpha Particle: A positively charged particle ejected spontaneously from the nuclei of some radioactive elements. It is identical to a helium nucleus that has a mass number of 4 and an electrostatic charge of +2. It has low penetrating power and a short range (a few centimeters in air). The most energetic alpha particle will generally

fail to penetrate the dead layers of cells covering the skin and can be easily stopped by a sheet of paper. Alpha particles are hazardous when an alpha-emitting isotope is inside the body.

Alternate ac Source: means an alternating current (ac) power source that is available to and located at or nearby a nuclear power plant and meets the following requirements:

1. Is connectable to but not normally connected to the offsite or onsite emergency ac power systems;
2. Has minimum potential for common mode failure with offsite power or the onsite emergency ac power sources;
3. Is available in a timely manner after the onset of station blackout; and
4. Has sufficient capacity and reliability for operation of all systems required for coping with station blackout and for the time required to bring and maintain the plant in safe shutdown (non-design basis accident).

Anion: A negatively charged ion.

Annihilation: Annihilation of particles is the disappearance of the mass energy of a particle and its corresponding antiparticle, and its appearance as another sort of energy (possibly including a spray of particles of total quantum number zero for each of the additive quantum numbers).

Annual Limit on Intake (ALI): The derived limit for the amount of radioactive material taken into the body of an adult worker by inhalation or ingestion in a year. ALI is the smaller value of intake of a given radionuclide in a year by the reference man that would result in a committed effective dose equivalent of 5 rems (0.05 sievert) or a committed dose equivalent of 50 rems (0.5 sievert) to any individual organ or tissue.

Anticipated Transient Without Scram (ATWS): ATWS is one of the "worst case" accidents, consideration of which frequently motivates the NRC to take regulatory action. The accident could happen if the system that provides a highly reliable means of shutting down the reactor (scram system) fails to work during a reactor event (anticipated transient). The types of events considered are those used for designing the plant.

Antineutrino: Antiparticle to the neutrino. See antiparticles.

Antiparticle: Particle having the same mass, spin, isospin as a particle, but having all additive quantum numbers opposite to those of its respective particle. Antiparticles have the opposite charge of its corresponding particle. Anti-baryons are antiparticles to baryons, anti-leptons are antiparticles to leptons, anti-quarks are antiparticles to quarks. The antiparticle for a particular particle, for example a neutrino, is denoted an antineutrino.

Assumptions (for IPEs, IPEEs, and PRAs): In the context of PRAs, assumptions are those parts of the mathematical models that the analyst expects will hold true for the range of solutions used for making decisions. Without assumptions, even the most powerful computers may not be able to provide useful solutions for the models.

Asymptotic Freedom: Quark-quark interactions weaken as the energy gets higher, or, equivalently, as the quarks approach one another.

Atom: The smallest particle of an element that cannot be divided or broken up by chemical means. It consists of a central core of protons and neutrons, called the nucleus. Electrons revolve in orbits in the region surrounding the nucleus

Atomic Energy: Energy released in nuclear reactions. Of particular interest is the energy released when a neutron initiates the breaking up or fissioning of an atom's nucleus into smaller pieces (fission) or when two nuclei are joined together under millions of degrees of heat (fusion). It is more correctly called nuclear energy.

Atomic Energy Commission: Federal agency created in 1946 to manage the development, use, and control of nuclear energy for military and civilian applications. Abolished by the Energy Reorganization Act of 1974 and succeeded by the Energy Research and Development Administration (now part of the U.S. Department of Energy) and the U.S. Nuclear Regulatory Commission.

Atomic Mass (sometimes mistakenly called atomic weight): The mass of a neutral atom. Its value in atomic mass units (u) is approximately equal to the sum of the number of protons and neutrons in the nucleus of the atom.

Atomic Mass Number: *A*, the total number of nucleons (protons and neutrons) found in a nucleus.

Atomic Mass Unit (amu or u): Unit of mass defined by the convention that the atom **Carbon 12, C_{12}** has a mass of exactly 12 u; the mass of 1 u is **1.67×10^{-27} kg.**

Atomic Number: *Z*, the total number of protons found in a nucleus.

Atomic Weapon: means any device utilizing atomic energy, exclusive of the means for transporting or propelling the device (where such means is a separable and divisible part of the device), the principal purpose of which is for use as, or for development of, a weapon, a weapon prototype, or a weapon test device.

Atomic Weight: for an element is defined as the average atomic weight of the isotopes of the element. The atomic weight for an element can be calculated by summing the products of the isotopic abundance of the isotope with the atomic mass of the isotope.

Atom Percent: is the percentage of the atoms of an element that are of a particular isotope. Atom percent is abbreviated as **a/o.**

Attenuation: The process by which the number of particles or photons entering a body of matter is reduced by absorption and scattered radiation.

Auxiliary Building: Building at a nuclear power plant, frequently located adjacent to the reactor containment structure,that houses most of the reactor auxiliary and safety systems, such as radioactive waste systems, chemical and volume control systems, and emergency cooling water systems.

Auxiliary Feed-water: Backup water supply used during nuclear plant startup and shutdown to supply water to the steam generators during accident conditions for removing decay heat from the reactor.

Average Planar Linear Heat Generation Rate (APLGHR): The average value of the linear heat generation rate of all the control rods at any given horizontal plane along a fuel bundle.

Background Radiation: The radiation found in the natural environment originating primarily from the naturally radioactive elements of Earth and from cosmic rays. The term may also mean radiation extraneous to an experiment.

Barrel (petroleum): A unit of volume equal to 42 U.S. gallons

Baryon: A massive composite hadron (made of three quarks) such as the proton or the neutron.

Baryon Number: Quantum number characteristic of baryons. Each baryon has a value of +1, while each anti-baryon has a value of -1.

Basic Component: means, for the purposes of **10-CFR 50.55(e)** :

1. When applied to nuclear power reactors, any plant structure, system, component, or part thereof necessary to assure
   - **A.** The integrity of the reactor coolant pressure boundary,
   - **B.** The capability to shut down the reactor and maintain it in a safe shutdown condition
   - **C.** The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in **10CFR 50.34(a)(1), § 50.67(b)(2),** or § **100.11.**
2. When applied to other types of facilities or portions of such facilities for which construction permits are issued under **10CFR § 50.23,** a component, structure, system or part thereof that is directly procured by the construction permit holder for the facility subject to the regulations of this part and in which a defect or failure to comply with any applicable regulation in this chapter, order, or license issued by the Commission could create a substantial safety hazard.
3. In all cases, **basic component** includes safety related design, analysis, inspection, testing, fabrication, replacement parts, or consulting services that are associated with the component hardware, whether these services are performed by the component supplier or other supplier.

Bayesian Estimation: A mathematical formulation, using Bayes' theorem, by which the likelihood of an event can be estimated taking explicit consideration of certain contextual features (such as amount of data, nature of decision, etc.).

Bayesian Prior: A way to express the context of a Bayesian estimation in which initial data are updated as new data become available.

Becquerel (Bq): Unit of activity in the International System–one disintegration per second; 1 Bq = 27 pCi. The unit of radioactive decay equal to 1 disintegration per second. 37 billion (3.7×10^{10}) becquerels = 1 curie (Ci).

Beta Particle (beta radiation, beta ray): An electron of either positive charge (e^+ or b+) or negative charge (e, e^- or b-) emitted by an atomic nucleus or neutron in the process of a transformation. Beta particles are more penetrating than alpha particles but less than gamma rays or x-rays. Electron capture is a form of beta decay.

A charged particle emitted from a nucleus during radioactive decay, with a mass equal to 1/1837 that of a proton. A negatively charged beta particle is identical to an electron. A positively charged beta particle is called a positron. Large amounts of beta radiation may cause skin burns, and beta emitters are harmful if they enter the body. Beta particles may be stopped by thin sheets of metal or plastic

Beyond Design-Basis Accidents: This term is used as a technical way to discuss accident sequences that are possible but were not fully considered in the design process because they were judged to be too unlikely. As the regulatory process strives to be as thorough as possible, "beyond design-basis" accident sequences are analyzed to fully understand the capability of a design.

Big Bang: Beginning of the universe; a transition from conditions of unimaginable density and temperature to conditions of lower density and temperature.

Binding Energy: The minimum energy required to separate a nucleus into its component neutrons and protons.

Bioassay: The determination of kinds, quantities, or concentrations and, in some cases, locations of radioactive material in the human body, whether by direct measurement (in vivo counting) or by analysis and evaluation of materials excreted or removed (in vitro) from the human body.

Bioenergy: Energy derived from biomass as electricity or heat, or combinations of heat and power; in the form of liquid or gaseous fuels, it is often referred to as biofuels

Biological Half-life: The time required for a biological system, such as that of a human, to eliminate, by natural processes, half of the amount of a substance (such as a radioactive material) that has entered it.

Biological Shield: A mass of absorbing material placed around a reactor or radioactive source to reduce the radiation to a level safe for humans.

Biomass: Any organic matter available on a renewable or a recurrent basis, including agricultural crops and residues, wood and wood residues, urban and animal residues, and aquatic plants.

Blackbody: An object that is a perfect emitter and absorber of radiation.

Blackbody Radiation: Radiation emitted by a blackbody (the intensity depends on temperature).

Black Hole: An object so dense that light cannot escape from it.

Boiling Water Reactor (BWR): A reactor in which water, used as both coolant and moderator, is allowed to boil in the core. The resulting steam can be used directly to drive a turbine and electrical generator, thereby producing electricity

Bone Seeker: A radioisotope that tends to accumulate in the bones when it is introduced into the body. An example is strontium-90, which behaves chemically like calcium

Boson: A particle having spin that is an integer multiple of **(h-bar)**.

Breeder: A reactor that produces more nuclear fuel than it consumes. A fertile material, such as uranium-238, when bombarded by neutrons, is transformed into a fissile material, such as plutonium-239, which can be used as fuel.

British Thermal Unit (Btu): One British thermal unit, or BTU, is the quantity of heat required to raise the temperature of one pound of water one degree Fahrenheit.

Byproduct Material: Byproduct material is:

1. Any radioactive material (except special nuclear material) yielded in, or made radioactive by, exposure to the radiation incident to the process of producing or using special nuclear material (as in a reactor);
2. The tailings or wastes produced by the extraction or concentration of uranium or thorium from ore

Calibration: The adjustment, as necessary, of a measuring device such that it responds within the required range and accuracy to known values of input.

Capability: The maximum load that a generating station can carry under specified conditions for a given period of time without exceeding approved limits of temperature and stress.

Capacity Factor (gross): The ratio of the gross electricity generated, for the time considered, to the energy that could have been generated at continuous full-power operation during the same period.

Capacity Factor (net): The ratio of the net electricity generated, for the time considered, to the energy that could have been generated at continuous full-power operation during the same period.

Carbon Dioxide (CO_2): A colorless, odorless, non-poisonous gas that is a normal part of the ambient air. Carbon dioxide is a product of fossil fuel combustion.

Cask: A heavily shielded container used to store and/or ship radioactive materials. Lead and steel are common materials used in the manufacture of casks.

Cation: A positively charged ion.

Certified Fuel Handler: means, for a nuclear power reactor facility, a non-licensed operator who has qualified in accordance with a fuel handler training program approved by the Commission.

Chain Reaction: A reaction that initiates its own repetition. In a fission chain reaction, a fissionable nucleus absorbs a neutron and fissions spontaneously, releasing additional neutrons. These, in turn, can be absorbed by other fissionable nuclei, releasing still more neutrons. A fission chain reaction is self-sustaining when the number of neutrons released in a given time equals or exceeds the number of neutrons lost by absorption in non-fissionable material or by escape from the system.

Charged Particle: An ion. An elementary particle carrying a positive or negative electric charge.

Chemical Recombination: Following an ionization event, the positively and negatively charged ion pairs may or may not realign themselves to form the same chemical substance they formed before ionization. Thus, chemical recombination could change the chemical composition of the material bombarded by ionizing radiation.

Cherenkov Radiation: Light emitted by particles that move through a medium in which the speed of light is slower than the speed of the particles.

Cladding: The thin-walled metal tube that forms the outer jacket of a nuclear fuel rod. It prevents corrosion of the fuel by the coolant and the release of fission products into the coolant. Aluminum, stainless steel, and zirconium alloys are common cladding materials.

Cleanup System: A system used for continuously filtering and de-mineralizing a reactor coolant system to reduce contamination levels and to minimize corrosion.

Climate Change: The change in weather patterns and surface temperatures that is at the center of the controversy concerning man's use of carbon based fuels for the generation of energy. Some groups of scientist are attempting to link the change in average temperature on the earth to the miniscule increases in greenhouse gas concentrations in the earth's atmosphere.

Coastdown: An action that permits the reactor power level to decrease gradually as the fuel in the core is depleted.

Cogeneration: The production of electrical energy and another form of useful energy (such as heat or steam) through the sequential use of energy.

Cold Shutdown: The term used to define a reactor coolant system at atmospheric pressure and at a temperature below 200 degrees Fahrenheit following a reactor cooldown.

Collective Dose: The sum of the individual doses received in a given period by a specified population from exposure to a specified source of radiation.

Combined Cycle: An electric generating technology in which electricity is produced from otherwise lost waste heat exiting from one or more gas (combustion) turbines. The exiting heat is routed to a conventional boiler or to a heat recovery steam generator for utilization by a steam turbine in the production of electricity. Such designs increase the efficiency of the electric generating unit.

Commission: means the **Nuclear Regulatory Commission** or its duly authorized representatives.

Committed dose equivalent: means the dose equivalent to organs or tissues of reference that will be received from an intake of radioactive material by an individual during the 50-year period following the intake.

Committed effective dose equivalent: is the sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs or tissues. The committed dose equivalent for a given organ multiplied by a weighting factor.

Common defense and security: means the common defense and security of the United States.

Cost of Service Regulation: means the traditional system of rate regulation, or similar regulation, including "price cap" or "incentive" regulation, in which a rate regulatory authority generally allows an electric utility to charge its customers the reasonable and prudent costs of providing electricity services, including capital, operations, maintenance, fuel, decommissioning, and other costs required to provide such services.

Compact: A group of two or more states formed to dispose of low-level radioactive waste on a regional basis. Forty-two states have formed nine compacts.

Compound: A chemical combination of two or more elements combined in a fixed and definite proportion by weight.

Condensate: Water that has been produced by the cooling of steam in a condenser.

Condenser: A large heat exchanger designed to cool exhaust steam from a turbine below the boiling point so that it can be returned to the heat source as water. In a pressurized water reactor, the water is returned to the steam generator. In a boiling water reactor, it returns to the reactor core. The heat removed from the steam by the condenser is transferred to a circulating water system and is exhausted to the environment, either through a cooling tower or directly into a body of water.

Conservation Law: A relation asserting that a specific quantity is conserved. For example, conservation of energy, conservation of momentum, conservation of electron number. Conservation laws are connected to symmetries through Noether's theorem.

Construction Recapture: The maximum number of years that could be added to the license expiration date to recover the period from the construction permit to the date when the operating license was granted. A licensee is required to submit an application for such a change.

Construction or Constructing: means, for the purposes of **10-CFR-50.55(e**), the analysis, design, manufacture, fabrication, quality assurance, placement, erection, installation, modification, inspection, or testing of a facility or activity which is subject to the regulations in this part and consulting services related to the facility or activity that are safety related.

Containment Structure: A gaslight shell or other enclosure around a nuclear reactor to confine fission products that otherwise might be released to the atmosphere in the event of an accident.

Contamination: Undesired radioactive material that is deposited on the surface of or inside structures, areas, objects, or people.

Control Rod: A rod, plate, or tube containing a material such as hafnium, boron, etc., used to control the power of a nuclear reactor. By absorbing neutrons, a control rod prevents the neutrons from causing further fissions.

Controls: when used with respect to nuclear reactors means apparatus and mechanisms, the manipulation of which directly affects the reactivity or power level of the reactor.

When used with respect to any other facility means apparatus and mechanisms, the manipulation of which could affect the chemical, physical, metallurgical, or nuclear process of the facility in such a manner as to affect the protection of health and safety against radiation.

Control Room: The area in a nuclear power plant from which most of the plant power production and emergency safety equipment can be operated by remote control.

Controlled Area: At a nuclear facility, an area outside a restricted area but within the site boundary, access to which the licensee can limit for any reason.

Coolant: A substance circulated through a nuclear reactor to remove or transfer heat. The most commonly used coolant in the United States is water. Other coolants include heavy water, air, carbon dioxide, helium, liquid sodium, and a sodium-potassium alloy.

Cooldown: The gradual decrease in reactor fuel rod temperature caused by the removal of heat from the reactor coolant system after the reactor has been shutdown.

Cooling Tower: A heat exchanger designed to aid in the cooling of water that was used to cool exhaust steam exiting the turbines of a power plant. Cooling towers transfer exhaust heat into the air instead of into a body of water.

Core: The central portion of a nuclear reactor containing the fuel elements, moderator, neutron poisons, and support structures.

Core Damage Frequency: An expression of the likelihood that, given the way a reactor is designed and operated, an accident could cause the fuel in the reactor to be damaged.

Core Meltdown Accident: An event or sequence of events that result in the melting of part of the fuel in the reactor core.

Cosmic Radiation: Penetrating ionizing radiation, both particulate and electromagnetic, originating in outer space. Secondary cosmic rays, formed by interactions in the Earth's atmosphere, account for about 45 to 50 millirem of the 360 millirem background radiation that an average individual receives in a year.

Cost of Service Regulation: means the traditional system of rate regulation, or similar regulation, including "price cap" or "incentive" regulation, in which a rate regulatory authority generally allows an electric utility to charge its customers the reasonable and prudent costs of providing electricity services, including capital, operations, maintenance, fuel, decommissioning, and other costs required to provide such services.

Counter: A general designation applied to radiation detection instruments or survey meters that detect and measure radiation. The signal that announces an ionization event is called a count.

Criteria Pollutant: A pollutant determined to be hazardous to human health and regulated under the Environmental Protection Agency's (EPA) National Ambient Air Quality Standards. The 1970 amendments to the Clean Air Act require EPA to describe the health and welfare impacts of a pollutant as the "criteria" for inclusion in the regulatory regime.

Criticality: A term used in reactor physics to describe the state when the number of neutrons released by fission is exactly balanced by the neutrons being absorbed (by the fuel and poisons) and escaping the reactor core. A reactor is said to be "critical" when it achieves a self-sustaining nuclear chain reaction, as when the reactor is operating.

Critical Mass: The smallest mass of fissionable material that will support a self-sustaining chain reaction.

Critical Organ: That part of the body that is most susceptible to radiation damage under the specific conditions under consideration.

Cross-Section: The cross-section of a Nuclear Reaction denoted by the Greek letter σ is a measure of the probability of the occurrence of a particular reaction under prescribed conditions.

Crud: A colloquial term for corrosion and wear products (rust particles, etc.) that become radioactive (i.e., activated) when exposed to radiation. Because the activated deposits were first discovered at Chalk River, a Canadian nuclear plant, "crud" has been used as shorthand for Chalk River Unidentified Deposits.

Crude Oil: A mixture of hydrocarbons that exists in the liquid phase in natural underground reservoirs and remains liquid at atmospheric pressure after passing through surface separating facilities. Crude oil production is measured at the wellhead and includes lease condensate.

Cumulative Dose: The total dose resulting from repeated exposures of ionizing radiation to an occupationally exposed worker to the same portion of the body, or to the whole body, over time.

Curie (Ci): The basic unit used to describe the intensity of radioactivity in a sample of material. The curie is equal to 37 billion (3.7×10^{10}) disintegrations per second, which is approximately the activity of 1 gram of radium. A curie is also a quantity of any radionuclide that decays at a rate of 37 billion disintegrations per second. It is named for Marie and Pierre Curie, who discovered radium in 1898.

The original unit used to describe the intensity of radioactivity in a sample of material. One curie equals thirty-seven billion disintegrations per second, or approximately the radioactivity of one gram of radium. This unit is no longer recognized as part of the International System of units. It has been replaced by the becquerel.

Cyclotron: Circular accelerator in which the particle is bent in traveling through a magnetic field, and an oscillating potential difference causes the particles to gain energy.

Cyclotron Frequency: Frequency at which the electric field is switched in order to accelerate the particles in the cyclotron. The frequency is related to the mass and charge of the particle to be accelerated.

Daughter: A nuclide formed by the radioactive decay of a different (parent) nuclide.

Daughter Products: Isotopes that are formed by the radioactive decay of some other isotope. In the case of radium-226, for example, there are 10 successive daughter products, ending in the stable isotope lead-206.

Decay Heat: The heat produced by the decay of radioactive fission products after a reactor has been shut down.

Decay, Radioactive: The decrease in the amount of any radioactive material with the passage of time due to the spontaneous emission from the atomic nuclei of either alpha or beta particles, often accompanied by gamma radiation.

The change of one radioactive nuclide into a different nuclide by the spontaneous emission of radiation such as alpha, beta, or gamma rays, or by electron capture. The end product is a less energetic, more stable nucleus. Each decay process has a definite half-life.

Decay Rate: The ratio of activity to the number of radioactive atoms of a particular species.

Decay Time: The time required for a quantity to fall to ***1/e*** times the original value.

Declared Pregnant Woman: A woman who is an occupational radiation worker and has voluntarily informed her employer, in writing, of her pregnancy and the estimated date of conception

Decommission: means to remove a facility or site safely from service and reduce residual radioactivity to a level that permits:

1. Release of the property for unrestricted use and termination of the license;
2. Release of the property under restricted conditions and termination of the license.

Decommissioning: The process of closing down a facility followed by reducing residual radioactivity to a level that permits the release of the property for unrestricted

DECON: A method of decommissioning in which the equipment, structures, and portions of a facility and site containing radioactive contaminants are removed and safety buried in a low-level radioactive waste landfill or decontaminated to a level that permits the property to be released for unrestricted use shortly after cessation of operations.

Decontamination: The reduction or removal of contaminating radioactive material from a structure, area, object, or person. Decontamination may be accomplished by (1) treating the surface to remove or decrease the contamination, (2) letting the material stand so that the radioactivity is decreased as a result of natural radioactive decay, or (3) covering the contamination to shield or attenuate the radiation emitted

Deep-Dose Equivalent (DDE): The external whole-body exposure dose equivalent at a tissue depth of 1 cm (1000 mg/cm^2).

Defect: means, for the purposes of **10-CFR 50.55(e)**:

1. A deviation in a basic component delivered to a purchaser for use in a facility or activity subject to a construction permit under this part, if on the basis of an evaluation, the deviation could create a substantial safety hazard;
2. The installation, use, or operation of a basic component containing, a defect as defined in paragraph (1) of this definition;
3. A deviation in a portion of a facility subject to the construction permit of this part provided the deviation could, on the basis of an evaluation, create a substantial safety hazard.

Defense-in-depth: A design and operational philosophy with regard to nuclear facilities that calls for multiple layers of protection to prevent and mitigate accidents. It includes the use of controls, multiple physical barriers to prevent release of radiation, redundant and diverse key safety functions, and emergency response measures.

Density: The ratio of an object's mass to its volume.

Department and Department of Energy: means the Department of Energy established by the Department of Energy Organization Act (Pub. L. 95-91, 91 Stat. 565, 42 U.S.C. 7101 et seq.), to the extent that the department, or its duly authorized representatives, exercises functions formerly vested in the Atomic Energy Commission, its Chairman, members, officers and components and transferred to the U.S. Energy Research and Development Administration and to the Administrator thereof pursuant to sections 104 (b), (c) and (d) of the Energy Reorganization Act of 1974 (Pub. L. 93-438, 88 Stat. 1233 at 1237, 42 U.S.C. 5814) and retransferred to the Secretary of Energy pursuant to section 301(a) of the Department of Energy Organization Act (Pub. L. 95-91, 91 Stat. 565 at 577-578, 42 U.S.C. 7151).

Departure From Nuclear Boiling Ratio (DNBR): The ratio of the heat flux to cause departure from nucleate boiling to the actual local heat flux or a fuel rod.

Departure From Nucleate Boiling (DNB): The point at which the heat transfer from a fuel rod rapidly decreases due to the insulating effect of a steam blanket that forms on the rod surface when the temperature continues to increase.

Depleted Uranium: Uranium having a percentage of uranium-235 smaller than the 0.7 percent found in natural uranium. It is obtained from spent (used) fuel elements or as byproduct tails, or residues, from uranium isotope separation.

Derived Air Concentration (DAC): The concentration of radioactive material in air and the time of exposure to that radionuclide in hours. An NRC licensee may take 2000 hours to represent one ALI, equivalent to a committed effective dose equivalent of 5 rems (0.05 sievert).

Derived Air Concentration-Hour (DAC-hour): The product of the concentration of radioactive material in air (expressed as a fraction or multiple of the derived air concentration for each radionuclide) and the time of exposure to that radionuclide, in hours. A licensee may take 2,000 DAC-hours to represent one ALI, equivalent to a committed effective dose equivalent of 5 rem (0.05 Sv).

Design Bases: means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be:

1. **r**estraints derived from generally accepted "state of the art" practices for achieving functional goals,

2. requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals.

Design-Basis Accident: A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to assure public health and safety.

Design-Basis Phenomena: Earthquakes, tornadoes, hurricanes, floods, etc., that a nuclear facility must be designed and built to withstand without loss of systems, structures, and components necessary to assure public health and safety.

Design-basis Threat: A profile of the type, composition, and capabilities of an adversary. The NRC and its licensees use the design-basis threat (DBT) as a basis for designing safeguards systems to protect against acts of radiological sabotage and to prevent the theft of special nuclear material. The DBT is described in detail in Title 10, Section 73.1(a), of the Code of Federal Regulations This term is applied to clearly identify for a licensee the expected capability of its facility to withstand a threat.

Detector: A material or device that is sensitive to radiation and can produce a response signal suitable for measurement or analysis. A radiation detection instrument.

Deterministic (probabilistic): Consistent with the principles of "determinism," which hold that specific causes completely and certainly determine effects of all sorts. As applied in nuclear technology, it generally deals with evaluating the safety of a nuclear power plant in terms of the consequences of a predetermined bounding subset of accident sequences. The term "probabilistic" is associated with an evaluation that explicitly accounts for the likelihood and consequences of possible accident sequences in an integrated fashion.

Deterministic Effect: The health effects of radiation, the severity of which varies with the dose and for which a threshold is believed to exist. Radiation-induced cataract formation is an example of a deterministic effect (also called a non-stochastic effect)

Deuterium: An isotope of hydrogen with one proton and one neutron in the nucleus.

Deuteron: The nucleus of deuterium. It contains one proton and one neutron. See also heavy water.

Deviation: means, for the purposes of **10 CFR-50.55(e)**, a departure from the technical or quality assurance requirements defined in procurement documents, safety analysis report, construction permit, or other documents provided for basic components installed in a facility subject to the regulations of this part.

Differential Pressure (dp or dP): The difference in pressure between two points of a system, such as between the inlet and outlet of a pump.

Director: means, for the purposes of **10 CFR-50.55(e)**, an individual, appointed or elected according to law, who is authorized to manage and direct the affairs of a corporation, partnership or other entity.

Discount Rate: The interest rate used to assess the value of future cost and revenue streams; an essential factor in assessing true returns from an investment in energy efficiency, as well as opportunity costs associated with not making that investment. In this report, we always use real discount rates that do not include inflation. To obtain the equivalent nominal discount rate including inflation, simply add the percentage annual inflation rate to the real discount rate

Discovery: means, for the purposes of**10 CFR-50.55(e),** the completion of the documentation first identifying the existence of a deviation or failure to comply potentially associated with a substantial safety hazard within the evaluation procedures discussed in **10 CFR-50.55(e)**.

Distillate Fuel Oil: The lighter fuel oils distilled off during the refining process. Included are products known as ASTM grades numbers 1 and 2 heating oils, diesel fuels, and number 4 fuel oil. The major uses of distillate fuel oils include heating, fuel for on- and off-highway diesel engines, and railroad diesel fuel.

Doppler Coefficient: Another name used for the fuel temperature coefficient of reactivity.

Dose: The absorbed dose, given in rads (or in SI units, grays), that represents the energy absorbed from the radiation in a gram of any material. Furthermore, the biological dose or dose equivalent, given in rem or sieverts, is a measure of the biological damage to living tissue from radiation exposure.

Dose Equivalent: The product of absorbed dose in tissue multiplied by a quality factor and then sometimes multiplied by other necessary modifying factors at the location of interest. It is expressed numerically in rems or sieverts

Dose Rate: The ionizing radiation dose delivered per unit time. For example, rem or sieverts per hour.

Dosimeter: A small portable instrument (such as a film badge or thermoluminescent or pocket dosimeter) for measuring and recording the total accumulated personal dose of ionizing radiation.

Dosimetry: The theory and application of the principles and techniques involved in the measurement and recording of ionizing radiation doses.

Drywell: The containment structure enclosing a boiling water reactor vessel and its recirculation system. The drywell provides both a pressure suppression system and a fission product barrier under accident conditions.

Earthquake, Operating Basis: An earthquake that could be expected to affect the reactor plant site, but for which the plant power production equipment is designed to remain functional without undue risk to public health and safety.

Effective Dose Equivalent: The sum of the products of the dose equivalent to the organ or tissue and the weighting factors applicable to each of the body organs or tissues that are irradiated.

Effective Half-life: The time required for the amount of a radioactive element deposited in a living organism to be diminished 50 percent as a result of the combined action of radioactive decay and biological elimination.

Efficiency, Plant: The percentage of the total energy content of a power plant's fuel that is converted into electricity. The remaining energy is lost to the environment as heat.

Elastic Scattering: In this interaction of radiation with matter. The impinging particle approaches the target and

Electrical Generator: An electromagnetic device that converts mechanical (rotational) energy into electrical energy.

Electric Dipole Moment: The product of charge and distance of separation for an electric dipole.

Electric Utility Restructuring: With some notable exceptions, the electric power industry historically has been composed primarily of investor-owned utilities. These utilities have been predominantly vertically integrated monopolies (combining electricity generation, transmission, and distribution), whose prices have been regulated by State and Federal government agencies. Restructuring the industry entails the introduction of competition into at least the generation phase of electricity production, with a corresponding decrease in regulatory control. Restructuring may also modify or eliminate other traditional aspects of investor-owned utilities, including their exclusive franchise to serve a given geographical area, assured rates of return, and vertical integration of the production process.

Electromagnetic Radiation: A traveling wave motion resulting from changing electric or magnetic fields. Familiar electromagnetic radiation range from x-rays (and gamma rays) of short wavelength, through the ultraviolet, visible, and infrared regions, to radar and radio waves of relatively long wavelength.

Electron: An elementary particle with a negative charge and a mass 1/1837 that of the proton. Electrons surround the positively charged nucleus and determine the chemical properties of the atom.

Electron-volt (eV): Energy unit used as the basis of measurement for atomic processes. One electron-volt is equal to the amount of energy gained by an electron dropping through a potential difference of one volt, which is 1.6 ´ 10-19 joules.

Electron Capture: A radioactive decay process in which an orbital electron is captured by and merges with the nucleus. The mass number is unchanged, but the atomic number is decreased by one.

Electroweak Interaction: A theory which unifies the electromagnetic and weak interactions.

Element: One of the 103 known chemical substances that cannot be broken down further without changing its chemical properties. Some examples include hydrogen, nitrogen, gold, lead, and uranium.

Emergency Classifications: Response by an offsite organization is required to protect local citizens near the site. A request for assistance from offsite emergency response organizations may be required.

Emergency Core Cooling Systems (ECCS): Reactor system components (pumps, valves, heat exchangers, tanks, and piping) that are specifically designed to remove residual heat from the reactor fuel rods should the normal core cooling system (reactor coolant system) fail.

Emergency Feedwater: Another name that may be used for auxiliary feed-water.

Energy: The capacity for doing work as measured by the capability of doing work (potential energy) or the conversion of this capability to motion (kinetic energy). Energy has several forms, some of which are easily convertible and can be changed to another form useful for work.

Energy Saving Performance Contract: An agreement with a third party in which the overall performance of installed energy conservation measures is guaranteed by that party.

Energy Services Company: A company which designs, procures, finances, installs, maintains, and guarantees the performance of energy conservation measures in an owner's facility or facilities.

ENTOMB: A method of decommissioning in which radioactive contaminants are encased in a structurally long-lived material, such as concrete. The entombment structure is appropriately maintained and continued surveillance is carried out until the radioactivity decays to a level permitting decommissioning and ultimate unrestricted release of the property.

Enthalpy: In thermodynamics the Quantity called enthalpy, denoted by **H** or **h** (for the specific enthalpy)

$$H = U + pV .$$

Where **U** is the internal energy,
p is the internal pressure
V is the volume.

Enthalpy is a property of a gas or liquid and it's units in the British System are **Btu/lbm.**

Entropy: In thermodynamics the Quantity called entropy, denoted by **S** or **s** (for the specific entropy) is a measure of the amount of energy in a physical system not available to do work. As a physical system becomes more disordered, and its energy becomes more evenly distributed, that energy becomes less able to do work. The amount of entropy is often thought of as the amount of disorder in a system.

Environmental Qualification: A process for ensuring that equipment will be capable of withstanding the ambient conditions that could exist when the specific function to be performed by the equipment is actually called upon to be performed under accident conditions.

Ethanol: A denatured alcohol (**C_2H_5OH**) intended for motor gasoline blending.

Evaluation: means, for the purposes of **10-CFR-50.55(e**), the process of determining whether a particular deviation could create a substantial safety hazard or determining whether a failure to comply is associated with a substantial safety hazard.

Event Notification (EN) System: An internal NRC automated event tracking system used by the NRC Operations Center to track information on incoming notifications of the occurrence of significant material events that have or may affect public health and safety. Significant material events are reported to the NRC Operations Center by NRC licensees, staff of the Agreement States, other Federal agencies, and the public.

Excited State: The state of an atom or nucleus when it possesses more than its normal energy. Typically, the excess energy is released as a gamma ray.

Exclusion Area: The area surrounding the reactor where the reactor licensee has the authority to determine all activities, including exclusion or removal of personnel and property.

Excursion: A sudden, very rapid rise in the power level of a reactor caused by super-criticality. Excursions are usually quickly suppressed by the negative temperature coefficient, the fuel temperature coefficient or the void coefficient (depending upon reactor design), or by rapid insertion of control rods.

Exposure: Being exposed to ionizing radiation or to radioactive material.

External Radiation: Exposure to ionizing radiation when the radiation source is located outside the body.

Externalities: Benefits or costs, generated as a byproduct of an economic activity, that do not accrue to the parties involved in the activity.

Extremities: The hands, forearms, elbows, feet, knees, leg below the knees, and ankles. (Permissible radiation exposures in these regions are generally greater than in the whole body because they contain fewer blood forming organs and have smaller volumes for energy absorption.)

Fast Fission: Fission of a heavy atom (such as uranium-238) when it absorbs a high energy (fast) neutron. Most fissionable materials need thermal (slow) neutrons in order to fission.

Fast Neutron: A neutron with kinetic energy greater than its surroundings when released during fission.

Federal Government: funding for conversion means funds appropriated to the Department of Energy or to any other Federal Agency to pay directly to or to reimburse non-power reactor licensees for costs attendant to conversion.

Federal Licensee; means any NRC licensee, the obligations of which are guaranteed by and supported by the full faith and credit of the United States Government.

Feedwater: Water supplied to the reactor pressure vessel (in a BWR) or the steam generator (in a PWR) that removes heat from the reactor fuel rods by boiling and becoming steam.

Fermion: A particle having a spin that is an odd integer multiple of **(h-bar)**/2.

Fertile Material: A material, which is not itself fissile (fissionable by thermal neutrons), that can be converted into a fissile material by irradiation in a reactor. There are two basic fertile materials: uranium-238 and thorium-232. When these fertile materials capture neutrons, they are converted into fissile plutonium-239 and uranium-233, respectively.

Film Badge: Photographic film used for measurement of ionizing radiation exposure for personnel monitoring purposes. The film badge may contain two or three films of differing sensitivities, and it may also contain a filter that shields part of the film from certain types of radiation.

Fiscal Year: The 12-month period, from October 1 through September 30, used by the Federal Government in budget formulation and execution. The fiscal year is designated by the calendar year in which it ends.

Fissile Material: Although sometimes used as a synonym for fissionable material, this term has acquired a more restricted meaning. Namely, any material fissionable by thermal (slow) neutrons. The three primary fissile materials are uranium-233, uranium-235, and plutonium-239.

Fission (fissioning): The splitting of a nucleus into at least two other nuclei and the release of a relatively large amount of energy. Two or three neutrons are usually released during this type of transformation.

Fission Gases: Those fission products that exist in the gaseous state. In nuclear power reactors, this includes primarily the noble gases, such as krypton and xenon.

Fission Products: The nuclei (fission fragments) formed by the fission of heavy elements, plus the nuclide formed by the fission fragments' radioactive decay.

Fissile Nucleus: A nucleus that may fission after collision with a thermal (slow) neutron or that fissions spontaneously (by itself).

Fission: The splitting of a heavy nucleus into two roughly equal parts (which are nuclei of lower-mass elements), accompanied by the release of a relatively large amount of energy in the form of kinetic energy of the two parts and in the form of emission of neutrons and gamma rays.

Fission products: Nuclei formed by the fission of higher mass elements. They are of medium atomic mass and almost all are radioactive. Examples: ^{90}Sr, ^{137}Ce.

Fusion: A process whereby low mass nuclei combine to form a more massive nucleus plus one or move massive particles.

Fluorescent Lamps: Fluorescent lamps produce light by passing electricity through a gas, causing it to glow. The gas produces ultraviolet light; a phosphor coating on the inside of the lamp absorbs the ultraviolet light and produces visible light. Fluorescent lamps produce much less heat than incandescent lamps and are more energy efficient. Linear fluorescent lamps are used in long narrow fixtures designed for such lamps. Compact fluorescent light bulbs have been designed to replace incandescent light bulbs in table lamps, floodlights, and other fixtures.

Fissionable Material: Commonly used as a synonym for fissile material, the meaning of this term has been extended to include material that can be fissioned by fast neutrons, such as uranium-238.

Flux: A term applied to the amount of some type of particle (neutrons, alpha radiation, etc.) or energy (photons, heat, etc.) crossing a unit area per unit time. The unit of flux is the number of particles, energy, etc., per square centimeter per second.

Formula Quantity: Strategic special nuclear material in any combination in a quantity of 5000 grams or more computed by the formula, grams = (grams contained U-235) + 2.5 (grams U-233 + grams plutonium). This class of material is sometimes referred to as a Category I quantity of material

Fossil Fuel: Any naturally occurring organic fuel formed in the Earth's crust, such as petroleum, coal, and natural gas.

Fuel acceptable: to the Commission means that the fuel replacing the existing **Highly Enriched Uranium (HEU)** fuel in a specific non-power reactor

1. Meets the operating requirements of the existing license or, through appropriate **NRC** safety review and approval, can be used in a manner which protects public health and safety and promotes the common defense and security;
2. Meets the Commission's policy of limiting, to the maximum extent possible, the use of **HEU** fuel in that reactor.

Fuel Assembly: A cluster of fuel rods (or plates). Also called a fuel element. Many fuel assemblies make up a reactor core.

Fuel Cells: One or more cells capable of generating an electrical current by converting the chemical energy of a fuel directly into electrical energy. Fuel cells differ from conventional electrical cells in that the active materials such as fuel and oxygen are not contained within the cell but are supplied from outside.

Fuel Cycle: The series of steps involved in supplying fuel for nuclear power reactors. It can include mining, milling, isotopic enrichment, fabrication of fuel elements, use in a reactor, chemical reprocessing to recover the fissionable material remaining in the spent fuel, re-enrichment of the fuel material, re-fabrication into new fuel elements, and waste disposal.

Fuel Reprocessing: The processing of reactor fuel to separate the unused fissionable material from waste material.

Fuel Rod: A long, slender tube that holds fissionable material (fuel) for nuclear reactor use. Fuel rods are assembled into bundles called fuel elements or fuel assemblies, which are loaded individually into the reactor core.

Fuel Temperature Coefficient of Reactivity: The change in reactivity per degree change in the fuel temperature. The physical property of fuel pellet material (uranium-238) that causes the uranium to absorb more neutrons away from the fission process as fuel pellet temperature increases. This acts to stabilize power reactor operations. This coefficient is also known as the Doppler coefficient.

Full-time Equivalent: A measurement equal to one staff person working a full-time work schedule for one year.

Fusion Reaction: A reaction in which at least one heavier, more stable nucleus is produced from two lighter, less stable nuclei. Reactions of this type are responsible for enormous release of energy, as in the energy of stars, for example.

Gamma Radiation: High-energy, short wavelength, electromagnetic radiation emitted from the nucleus. Gamma radiation frequently accompanies alpha and beta emissions and always accompanies fission. Gamma rays are very penetrating and are best stopped or shielded by dense materials, such as lead or depleted uranium. Gamma rays are similar to x-rays.

Gamma Ray: A highly penetrating type of nuclear radiation, similar to x-radiation, except that it comes from within the nucleus of an atom, and, in general, has a shorter wavelength.

Gap: The space inside a reactor fuel rod that exists between the fuel pellet and the fuel rod cladding.

Gas Centrifuge: A uranium enrichment process that uses a large number of rotating cylinders in a series. These series of centrifuge machines, called trains, are interconnected to form cascades. In this process, **Uranium Hexafluoride, (UF_6)** gas is placed in a drum or cylinder and rotated at high speed. This rotation creates a strong gravitational field so that the heavier gas molecules (containing U-238) move

toward the outside of the cylinder and the lighter gas molecules (containing U-235) collect closer to the center. The stream that is slightly enriched in U-235 is withdrawn and fed into the next higher stage, while the slightly depleted stream is recycled back into the next lower stage. Significantly more U-235 enrichment can be obtained from a single unit gas centrifuge than from a single unit gaseous diffusion barrier

Gas-Cooled Reactor: A nuclear reactor in which a gas is the coolant.

Gaseous Diffusion Plant: A facility where uranium hexafluoride gas is filtered. Uranium-235 is separated from uranium-238, increasing the percentage of uranium-235 from 1 to about 3 percent. The process requires enormous amounts of electric power.

Gases: A substance possessing perfect molecular mobility and the property of indefinite expansion, as opposed to a solid or liquid; any such fluid or mixture of fluids other than air. Normally, these formless substances completely fill the space, and take the shape of, their container.

Gas-Turbine Electric Power Plant: A plant in which the prime mover is a gas turbine. A gas turbine typically consists of an axial-flow air compressor and one or more combustion chambers which liquid or gaseous fuel is burned. The hot gases expand to drive the generator and then are used to run the compressor.

Gauge Boson: Particle mediating an interaction. By exchange of the gauge particle, the interaction between two particles is accomplished.

Geiger Counter: A **Geiger-Müller** detector and measuring instrument. A radiation detection and measuring instrument. It consists of a gas-filled tube containing electrodes, between which there is an electrical voltage, but no current, flowing. When ionizing radiation passes through the tube, a short, intense pulse of current passes from the negative electrode to the positive electrode and is measured or counted. The number of pulses per second measures the intensity of the radiation field. It was named for Hans Geiger and W. Mueller, who invented it in the 1920s. It is sometimes called simply a Geiger counter or a G-M counter and is the most commonly used portable radiation instrument.

Generation (gross): The total amount of electric energy produced by a generating station as measured at the generator terminals

Generation (net): The gross amount of electric energy produced less the electric energy consumed at a generating station for station use.

Global Warming: Global warming is the increase in global temperatures that the earth has been experiencing this century. Gases that are thought by many to contribute to global warming through the greenhouse effect include carbon dioxide, methane, nitrous oxides, chlorofluorocarbons (CFCs), and halocarbons (the replacements for CFCs). Carbon dioxide emissions are primarily caused by the use of fossil fuels for energy.

Gluon: A gauge particle mediating the color strong interaction.

Gigawatt: One billion watts.

Gigawatthour: One billion watt-hours.

Government Agency: means any executive department, commission, independent establishment, corporation, wholly or partly owned by the United States of America which is an instrumentality of the United States, or any board, bureau, division, service, office, officer, authority, administration, or other establishment in the executive branch of the Government.

Graphite: A form of carbon, similar to that used in pencils, used as a moderator in some nuclear reactors.

Gray (Gy): The international system (SI) unit of absorbed dose. One gray is equal to an absorbed dose of 1 Joule/kilogram (one gray equals 100 rads)

Greenhouse Gas: Any gas that absorbs infrared radiation in the atmosphere.

Hadron: A strongly interacting particle.

Half-life: The time in which one half of the atoms of a particular radioactive substance disintegrate into another nuclear form. Measured half-lives vary from millionths of a second to billions of years. Also called physical or radiological half-life.

Half-life, Biological: The time required for the body to eliminate one half of the material taken in by natural biological means.

Half-life, Effective: The time required for a radionuclide contained in a biological system, such as a human or an animal, to reduce its activity by one-half as a combined result of radioactive decay and biological elimination.

Half-thickness: Any given absorber that will reduce the intensity of an original beam of ionizing radiation to one-half of its initial value.

Head, Reactor Vessel: The removable top section of a reactor pressure vessel. It is bolted in place during power operation and removed during refueling to permit access of fuel handling equipment to the core.

Health Physics: The science concerned with the recognition, evaluation, and control of health and environmental hazards that may arise from the use and application of ionizing radiation.

Heap Leach: A method of extracting uranium from ore using a leaching solution. Small ore pieces are placed in a heap on an impervious material (plastic, clay, asphalt) with perforated pipes under the heap. Acidic solution is then sprayed over the ore, dissolving the uranium. The solution in the pipes is collected and transferred to an ion-exchange system for concentration of the uranium.

Heat Exchanger: Any device that transfers heat from one fluid (liquid or gas) to another fluid or to the environment.

Heat Pump: A device that extracts available heat from one area (the heat source) and transfers it to another (the heat sink) to either heat or cool an interior space. Geothermal heat pumps can operate more efficiently than the standard air-source heat pumps, because during winter the ground does not get as cold as the outside air (and during the summer, it does not heat up as much).

Heat Sink: Anything that absorbs heat. It is usually part of the environment, such as the air, a river, or a lake.

Heatup: The rise in temperature of the reactor fuel rods resulting from an increase in the rate of fission in the core.

Heavy Water (D_2O): Water containing significantly more than the natural proportions (one in 6,500) of heavy hydrogen (**deuterium, D**) atoms to ordinary hydrogen atoms. Heavy water is used as a moderator in some reactors because it slows down neutrons effectively and also has a low probability of absorption of neutrons.

Heavy Water Moderated Reactor: A reactor that uses heavy water as its moderator. Heavy water is an excellent moderator and thus permits the use of un-enriched uranium as a fuel.

High-enriched Uranium: Uranium enriched to 20 percent or greater in the isotope uranium-235.

High-level Waste: Radioactive materials at the end of a useful life cycle that should be properly disposed of, including--

1. The highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste directly in reprocessing and any solid material derived from such liquid waste that contains fission products in concentrations;
2. Irradiated reactor fuel; and
3. Other highly radioactive material that the Commission, consistent with existing law, determines by rule require permanent isolation.

High-level waste (HLW) is primarily in the form of spent fuel discharged from commercial nuclear power reactors. It also includes HLW from activities and a small quantity of reprocessed commercial HLW

High Radiation Area: Any area with dose rates greater than 100 millirems (1 millisievert) in one hour 30 centimeters from the source or from any surface through which the ionizing radiation penetrates. Areas at licensee facilities must be posted as "high radiation areas" and access into these areas is maintained under strict control.

Highly Enriched Uranium: (**HEU**) fuel means fuel in which the weight percent of U-235 in the uranium is 20% or greater. Target material, special instrumentation, or experimental devices using HEU are not included.

Historical Site Assessment: means the identification of potential, likely, or known sources of radioactive material and radioactive contamination based on existing or derived information for the purpose of classifying a facility or site, or parts thereof, as impacted or non-impacted.

Homolog (or homologs): Elements in the same periodic table group that tend to exhibit similar, but not identical, chemical properties.

Hormesis: Controversial theory which argues that there is a benefit to health, or decrease in biological damage from radiation as dose in increased (valid only for very small doses).

Hot: A colloquial term meaning highly radioactive.

Hot Spot: The region in a radiation/contamination area where the level of radiation/contamination is significantly greater than in neighboring regions in the area.

Hubble Constant: Ratio of outward speed of galaxies to their distances from Earth.

Impacted Areas: mean the areas with some reasonable potential for residual radioactivity in excess of natural background or fallout levels.

Incentive Regulation: means the system of rate regulation in which a rate regulatory authority establishes rates that an electric generator may charge its customers that are based on specified performance factors, in addition to cost-of-service factors.

Independent Power Producer: A wholesale electricity producer (other than a qualifying facility under the Public Utility Regulatory Policies Act of 1978), that is unaffiliated with franchised utilities. Unlike traditional utilities, IPPs do not possess transmission facilities that are essential to their customers and do not sell power in any retail service territory where they have a franchise.

Induced Radioactivity: Radioactivity that is created by bombarding a substance with neutrons in a reactor or with charged particles produced by particle accelerators.

Infrared Radiation: Electromagnetic radiation of longer wavelength than visible light.

In Situ Leach: A process using a leaching solution to extract uranium from underground ore bodies in place (in other words, in situ). The leaching agent, which contains an oxidant such as oxygen with sodium carbonate, is injected through wells into the ore body in a confined aquifer to dissolve the uranium. This solution is then pumped via other wells to the surface for processing.

In Vitro: From the Latin for "in glass," isolated from the living organism and artificially maintained, as in a test tube.

In Vivo: From the Latin for "in one that is living," occurring within the living.

Individual Plant Examination (IPE): As requested by the NRC in Generic Letter 88-20, “Individual Plant Examination for Severe Accident Vulnerabilities” (November 23,

1988), a risk analysis that considers the unique aspects of a particular nuclear power plant, identifying the specific vulnerabilities to severe accident of that plant.

Individual Plant Examination for External Events (IPEEE): While the "individual plant examination" takes into account events that could challenge the design from things that could go awry internally (in the sense that equipment might fail because components do not work as expected), the "individual plant examination for external events" considers challenges such as earthquakes, internal fires, and high winds.

Induced Radioactivity: Radioactivity that is created when stable substances are bombarded by ionizing radiation. For example, the stable isotope cobalt-59 becomes the radioactive isotope cobalt-60 under neutron bombardment.

Integrated Plant Evaluation: An evaluation that considers the plant as a whole rather than system by system.

Iodine Spiking Factor: The magnitude of a rapid, short-term increase in the appearance rate of radioiodine in the reactor coolant system. This increase is generally caused by a reactor transient that results in a rapid drop in reactor coolant system pressure relative to the fuel rod internal pressure.

Ion: **1**. An atom that has too many or too few electrons, causing it to have an electrical charge, and therefore, be chemically active.

2. An electron that is not associated (in orbit) with a nucleus.

Ion-exchange: A common method for concentrating uranium from a solution. The uranium solution is passed through a resin bed where the uranium-carbonate complex ions are transferred to the resin by exchange with a negative ion like chloride. After build-up of the uranium complex on the resin, the uranium is eluted with a salt solution and the uranium is precipitated in another process.

Ionization: The process of adding one or more electrons to, or removing one or more electrons from, atoms or molecules, thereby creating ions. High temperatures, electrical discharges, or nuclear radiations can cause ionization.

Ionization Chamber: An instrument that detects and measures ionizing radiation by measuring the electrical current that flows when radiation ionizes gas in a chamber, making the gas a conductor of electricity.

Ionizing Radiation: Any radiation capable of displacing electrons from atoms or molecules, thereby producing ions. Some examples are alpha, beta, gamma, x-rays, neutrons, and ultraviolet light. High doses of ionizing radiation may produce severe skin or tissue damage.

.**Irradiate:** To expose to some form of radiation.

Isomer: Nuclides with the same number of neutrons and protons in different states of excitation.

Isomeric Transition: A relatively long-lived radioactive decay in which a nucleus goes from a higher to a lower energy state. The mass number and the atomic number are unchanged.

isotope: Isotopes of a given element have the same atomic number (same number of protons in their nuclei) but different mass numbers (different number of neutrons in their nuclei). ^{238}U and ^{235}U are isotopes of uranium.

Any two or more forms of an element having identical or very closely related chemical properties and the same atomic number but different atomic weights or mass numbers.

Isotope Separation: The process of separating isotopes from one another, or changing their relative abundances, as by gaseous diffusion or electromagnetic separation. Isotope separation is a step in the isotopic enrichment process.

Isotopic Enrichment: A process by which the relative abundance of the isotopes of a given element are altered, thus producing a form of the element that has been enriched in one particular isotope and depleted in its other isotopic forms.

joule (J): Unit of energy, equivalent to the work done in lifting a one-newton weight a distance of one meter.

K-capture: The capture by an atom's nucleus of an electron from the innermost electron orbital (K-shell) surrounding the nucleus.

kelvin (K): Unit of temperature equal in size to the Celsius degree, but with the zero set by the absolute zero of temperature, -273.15°C. Ice freezes at 273 K, room temperature is about 293 K, and water boils at 373 K, at sea level. human body temperature is 310 K.

keV: One thousand electron-volts.

Kerosene: A petroleum distillate that is used in space heaters, cook stoves, and water heaters; it is suitable for use as an illuminant when burned in wick lamps

Kilo: A Greek prefix meaning "thousand" in the nomenclature of the metric system. This prefix multiplies a unit by 1000.

Kilovolt: The unit of electrical potential equal to 1000 volts.

Kilowatt (kW): One thousand watts of electricity (see Watt).

Kilowatthour (kWh): One thousand watthours.

Kinetic energy: The energy that a body possesses by virtue of its mass and velocity. Also called the energy of motion.

Lens Dose Equivalent: The external exposure dose equivalent to the lens of the eye at a tissue depth of 0.3 centimeters (300 mg/cm^2).

Lepton: A particle (such as the electron or neutrino) not subject to strong interactions.

Lepton Number: Additive quantum number defining leptons; the three lepton numbers are electron number, muon number, and tau number. These numbers remain the same in all reactions.

Lifetime: The mean life of a particle or radioactive nucleus. This is equivalent to the decay time.

Linac: Another name for a linear accelerator.

Linear Accelerator: Particle accelerator laid out in a straight line.

Lethal Dose (LD): The dose of radiation expected to cause death to 50 percent of an exposed population within 30 days (LD 50/30). Typically, the LD 50/30 is in the range from 400 to 450 rem (4 to 5 sieverts) received over a very short period.

Licensed Material: Source material, special nuclear material, or byproduct material received, possessed, used, transferred or disposed of under a general or specific license issued by the NRC.

Licensing Basis: The collection of documents or technical criteria that provides the basis upon which the NRC issues a license to possess radioactive materials, conduct operations involving emission of radiation, use special nuclear materials, or dispose of radioactive waste.

Light Truck: Two-axle, four-tire trucks with a gross vehicle weight less than 10,000 pounds.

Light Water: Ordinary water as distinguished from heavy water..

Light Water Reactor: A term used to describe reactors using ordinary water as coolant, including boiling water reactors (**BWR**s) and pressurized water reactors (**PWR**s), the most common types used in the United States.

Limiting Condition for Operation: The section of Technical Specifications that identifies the lowest functional capability or performance level of equipment required for safe operation of the facility.

Limiting Safety System Settings: Settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting will ensure that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

Linear Heat Generation Rate: The heat generation rate per unit length of fuel rod, commonly expressed in kilowatts per foot (kw/ft) of fuel rod.

Liquefied Natural Gas: Natural gas (primarily methane) that has been liquefied by reducing its temperature to -260F at atmospheric pressure.

Liquefied Petroleum Gas: Ethane, ethylene, propane, propylene, normal butane, butylene, and isobutane produced at refineries or natural gas processing plants.

Loop: In a pressurized water reactor, the coolant flow path through piping from the reactor pressure vessel to the steam generator, to the reactor coolant pump, and back to the reactor pressure vessel. Large PWRs may have as many as four separate loops.

Loss of Coolant Accident (LOCA): Those postulated accidents that result in a loss of reactor coolant at a rate in excess of the capability of the reactor makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.

Low enriched uranium: (**LEU**) fuel means fuel in which the weight percent of U-235 in the uranium is less than 20%.

Low Population Zone (LPZ): An area of low population density often required around a nuclear installation before it's built. The number and density of residents is of concern in emergency planning so that certain protective measures (such as notification and instructions to residents) can be accomplished in a timely manner

Low-level Waste: A general term for a wide range of wastes having low levels of radioactivity. Industries; hospitals and medical, educational, or research institutions; private or government laboratories; and nuclear fuel cycle facilities (e.g., nuclear power reactors and fuel fabrication plants) that use radioactive materials generate low-level wastes as part of their normal operations. These wastes are generated in many physical and chemical forms and levels of contamination

Low-level radioactive wastes containing source, special nuclear, or byproduct material are acceptable for disposal in a land disposal facility. For the purposes of this definition, low-level waste has the same meaning as in the Low-Level Radioactive Waste Policy Act, that is, radioactive waste not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or byproduct material as defined in section 11e.(2) of the Atomic Energy Act (uranium or thorium tailings and waste).

Major Decommissioning Activity: means, for a nuclear power reactor facility, any activity that results in permanent removal of major radioactive components, permanently modifies the structure of the containment, or results in dismantling components for shipment containing greater than class C waste in accordance with **10-CFR § 61.55**.

Major Radioactive Components: means, for a nuclear power reactor facility, the reactor vessel and internals, steam generators, pressurizers, large bore reactor coolant system piping, and other large components that are radioactive to a comparable degree.

Mass Energy: Energy a particle has by virtue of its mass (given by $\mathbf{E = MC^2}$).

Mass Number: The total number of protons and neutrons in the nucleus**: A=Z+N.** This is also the total nucleon number of the nucleus.

Mass-energy Equation: The equation developed by Albert Einstein, which is usually given as $\mathbf{E = mc^2}$**,** showing that, when the energy of a body changes by an amount E (no matter what form the energy takes), the mass (m) of the body will change by an amount equal to E/c^2. The factor c squared, the speed of light in a vacuum (3×10^8), may be regarded as the conversion factor relating units of mass and energy. The equation predicted the possibility of releasing enormous amounts of energy by the conversion of mass to energy. It is also called the Einstein equation.

Maximum Dependable Capacity (gross): In a nuclear power reactor, dependable main-unit gross generating capacity, winter or summer, whichever is smaller. The dependable capacity varies because the unit efficiency varies during the year due to temperature variations in cooling water. It is the gross electrical output as measured at the output terminals of the turbine generator during the most restrictive seasonal conditions (usually summer).

Maximum dependable capacity (net): In a nuclear power reactor, gross maximum dependable generating capacity less the normal station service loads.

Mega: A prefix that multiplies a basic unit by 1,000,000 (10 to the sixth power).

Megacurie: One million curies.

Megawatt (MW): One million watts.

Megawatt Hour (MWh): One million watt-hours.

Methanol: A light volatile alcohol ($\mathbf{CH_3OH}$) used for motor gasoline blending.

Metric Ton: Approximately 2200 pounds in the English system of measurements. (Note: In the international system of measurements, 1 metric ton = 1000 kg.)

Micro: A prefix that divides a unit into one million parts (0.000001).

Microcurie: One millionth of a curie. That amount of radioactive material that disintegrates (decays) at the rate of 37 thousand atoms per second.

Milli: A prefix that divides a basic unit by 1000.

Millirem: One thousandth of a rem (0.001 rem).

Milliroentgen (mR): One thousandth of a roentgen (R). $1mR = 10^{-3}$ R = 0.001 R.

Mixed Oxide (MOX) Fuel: A mixture of uranium oxide and plutonium oxide used to fuel a reactor. Mixed oxide fuel is often called "MOX."

Moderator: A material, such as ordinary water, heavy water, or graphite that is used in a reactor to slow down high-velocity neutrons thus increasing the likelihood of fission.

Moderator Temperature Coefficient of Reactivity: As the moderator (water) increases in temperature, it becomes less dense and slows down fewer neutrons, which results in a negative change of reactivity. This negative temperature coefficient acts to stabilize atomic power reactor operations.

Molecule: A group of atoms held together by chemical forces. A molecule is the smallest unit of a compound that can exist by itself and retain its chemical properties.

Monitoring of Radiation: Periodic or continuous determination of the amount of ionizing radiation or radioactive contamination present in a region, as a safety measure, for the purpose of health or environmental protection. Monitoring is done for air, surface and ground water, soil and sediment, equipment surfaces, and personnel (for example, bioassay or alpha scans).

Meson: A particle (such as the pion) made of quark-antiquark pairs.

MeV: One million electron-volts.

Microwaves: Electromagnetic radiation with wavelength intermediate between radio wave and infrared radiation.

Multiwire Proportional Counter: Particle detector using changes in the current in wires due to the passage of ionizing particles nearby.

Muon: A charged lepton about 200 times more massive than an electron.

Muon Number: Additive quantum number characterizing muons and muon neutrinos

Nano: A prefix that divides a basic unit by one billion (10^{-9}).

Nanocurie: One billionth 10^{-9} of a curie.

Natural Circulation: The circulation of the coolant in the reactor coolant system without the use of the reactor coolant pumps. The circulation is due to the natural convection resulting from the different densities of relative cold and heated portions of the system.

Natural Gas: A mixture of hydrocarbons (principally methane) and small quantities of various non-hydrocarbons existing in the gaseous phase or in solution with crude oil in underground reservoirs.

Natural Uranium: Uranium as found in nature. It contains 0.7 percent uranium-235, 99.3 percent uranium-238, and a trace of uranium-234 by weight. In terms of the amount of radioactivity, it contains approximately 2.2 percent uranium-235, 48.6 percent uranium-238, and 49.2 percent uranium-234.

Net Summer Capability: The steady hourly output that generating equipment is expected to supply to system load exclusive of auxiliary power, as demonstrated by tests at the time of summer peak demand.

Neutrino: An electrically neutral particle with negligible mass. It is produced in processes such as beta decay and reactions that involve the weak force.

Neutron: One of the basic particles that make up a nucleus. A neutron and a proton have about the same mass, but the neutron has no electrical charge.

Neutron Capture: The reaction that occurs when a nucleus captures a neutron. The probability that a given material will capture a neutron is proportional to its neutron capture cross section and depends on the energy of the neutrons and the nature of the material.

Neutron Chain Reaction: A process in which some of the neutrons released in one fission event cause other fissions to occur. There are three types of chain reactions:

1. Non-sustaining--An average of less than one fission is produced by the neutrons released by each previous fission (reactor sub-criticality);
2. Sustaining--An average of exactly one fission is produced by the neutrons released by each previous fission (reactor criticality);
3. Multiplying--An average of more than one fission is produced by the neutrons released by previous fission (reactor super-criticality).

Neutron Flux: A measure of the intensity of neutron radiation in neutrons/cm^2-sec. It is the number of neutrons passing through 1 square centimeter of a given target in 1 second. Expressed as nv, where n = the number of neutrons per cubic centimeter and v = their velocity in centimeters per second.

Neutron Generation: The release, thermalization, and absorption of fission neutrons by a fissile material and the fission of that material producing a second generation of neutrons. In a typical nuclear power reactor system, there are about 40,000 generations of neutrons every second.

Neutron Leakage: Neutrons that escape from the vicinity of the fissionable material in a reactor core. Neutrons that leak out of the fuel region are no longer available to cause fission and must be absorbed by shielding placed around the reactor pressure vessel for that purpose.

Neutron Number: The total number of neutrons in the nucleus, N.

Neutron Source: Any material that emits neutrons, such as a mixture of radium and beryllium, that can be inserted into a reactor to ensure a neutron flux large enough to be distinguished from background to register on neutron detection equipment.

Neutron, Thermal: A neutron that has (by collision with other particles) reached an energy state equal to that of its surroundings, typically on the order of 0.025 eV (electron volts).

Nuclear Binding Energy: The energy that free nucleons give up in order to be bound inside a nucleus.

Nuclear Reactor: A device in which a fission chain reaction can be initiated, maintained, and controlled. Its essential components are fissionable fuel, moderator, shielding, control rods, and coolant.

Nucleon: A constituent of the nucleus; that is, a proton or a neutron.

Nucleus: The core of the atom, where most of its mass and all of its positive charge is concentrated. Except for ^{1}H, the nucleus consists of a combination of protons and neutrons.

Nuclide: Any species of atom that exists for a measurable length of time. A nuclide can be distinguished by its atomic mass, atomic number, and energy state.

Nitrogen Oxides (NOx): A product of combustion of fossil fuels whose production increases with the temperature of the process. It can become an air pollutant if concentrations are excessive.

Noble Gas: A gaseous chemical element that does not readily enter into chemical combination with other elements. An inert gas. Examples are helium, argon, krypton, xenon, and radon.

Non-Bypassable Charges: mean those charges imposed over an established time period by a Government authority that affected persons or entities are required to pay to cover costs associated with the decommissioning of a nuclear power plant. Such charges include, but are not limited to, wire charges, stranded cost charges, transition charges, exit fees, other similar charges, or the securitized proceeds of a revenue stream.

Non-impacted Areas: mean the areas with no reasonable potential for residual radioactivity in excess of natural background or fallout levels.

Non-power Reactor: means a research or test reactor licensed under **10-CFR §§ 50.21(c)** or **50.22** of this part for research and development.

Non-stochastic Effect: The health effects of radiation, the severity of which vary with the dose and for which a threshold is believed to exist. Radiation-induced cataract formation is an example of a non-stochastic effect (also called a deterministic effect)

Non-vital Plant Systems: Systems at a nuclear facility that may or may not be necessary for the operation of the facility (i.e., power production) but that would have little or no effect on public health and safety should they fail. These systems are not safety related.

Non-power Reactor: Reactors used for research, training, and test purposes, and for the production of radioisotopes for medical and industrial uses.

Notification: means the telephonic communication to the NRC Operations Center or written transmittal of information to the NRC Document Control Desk.

Not Applicable (NA): Specifies that a particular field is not applicable to the event.

Not Reported (NR): Specifies that information applicable to the particular field was not included in the event report.

Nozzle: As used in power water reactors and boiling water reactors, the interface (inlet and outlet) between reactor plant components (pressure vessel, coolant pumps, steam generators, etc.) and their associated piping systems.

NRC Operations Center: Rockville, Maryland, serves as the focal coordination point for communicating with NRC licensees, State agencies, and other Federal agencies about operating events in both the nuclear reactor and nuclear material industry. The Operations Center is staffed 24 hours a day by an NRC Headquarters Operations Officer (HOO), who is trained to receive, evaluate, and respond to events reported to the Operations Center.

Nuclear Electric Power: Electricity generated by an electric power plant whose turbines are driven by steam generated in a reactor by heat from the fissioning of nuclear fuel.

Nuclear Energy: The energy liberated by a nuclear reaction (fission or fusion) or by radioactive decay.

Nuclear Force: A powerful short-ranged attractive force that holds together the particles inside an atomic nucleus.

Nuclear Power Plant: An electrical generating facility using a nuclear reactor as its heat source to provide steam to a turbine generator.

Nuclear Reactor: means an apparatus, other than an atomic weapon, designed or used to sustain nuclear fission in a self-supporting chain reaction.

Nuclear Steam Supply System (NSSS): The reactor and the reactor coolant pumps (and steam generators for a pressurized water reactor) and associated piping in a nuclear power plant used to generate the steam needed to drive the turbine generator unit.

Nuclear Waste: A particular type of radioactive waste that is produced as part of the nuclear fuel cycle (i.e., those activities needed to produce nuclear fission, or splitting of the atom). These include extraction of uranium from ore, concentration of uranium, processing into nuclear fuel, and disposal of byproducts. Radioactive waste is a broader term that includes all waste that contains radioactivity. Residues from water treatment, contaminated equipment from oil drilling, and tailings from the processing of metals such as vanadium and copper also contain radioactivity but are not "nuclear waste" because they are produced outside of the nuclear fuel cycle. NRC generally regulates only those wastes produced in the nuclear fuel cycle (uranium mill tailings, depleted uranium, spent fuel rods, etc.).

Nucleon: Common name for a constituent particle of the atomic nucleus. At present, applied to protons and neutrons

Nucleus: The small, central, positively charged region of an atom. Except for the nucleus of ordinary hydrogen, which has only a proton, all atomic nuclei contain both protons and neutrons. The number of protons determines the total positive charge or atomic number. This number is the same for all the atomic nuclei of a given chemical element. The total number of neutrons and protons is called the mass number.

Nuclide: A general term referring to all known isotopes, both stable (279) and unstable (about 2,700), of the chemical elements.

Occupational Dose: The dose received by an individual in the course of employment in which the individual's assigned duties involve exposure to radiation or to radioactive material from licensed and unlicensed sources of radiation, whether in the possession of the licensee or other person. Occupational dose does not include dose received from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive materials and released in accordance with NRC regulations, from voluntary participation in medical research programs, or as a member of the general public.

Operable: A system, subsystem, train, component, or device is operable or has operability when it is capable of performing its specified functions and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication, or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its functions are also capable of performing their related support functions.

Operational mode: In a nuclear power reactor, an operational mode corresponds to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature.

Orphan Source: See unwanted radioactive material

Oxygenates: Any substance which, when added to motor gasoline, increases the amount of oxygen in that motor gasoline blend.

Ozone: Three-atom oxygen compound (03) found in two layers of the Earth's atmosphere. One layer of beneficial ozone occurs at 7 to 18 miles above the surface and shields the Earth from ultraviolet light.

Several holes in this protective layer have been documented by scientists. Ozone also concentrates at the surface as a result of reactions between byproducts of fossil fuel combustion and sunlight, having harmful health effects.

Parent: A radionuclide that decays to another nuclide.

A radionuclide that upon radioactive decay or disintegration yields a specific nuclide (the daughter).

Particulates: Visible air pollutants consisting of particles appearing in smoke or mist.

Parts Per Million (ppm): Parts (molecules) of a substance contained in a million parts of another substance (e.g., water).

Pellet, Fuel: As used in pressurized water reactors and boiling water reactors, a pellet is a small cylinder approximately 3/8-inch in diameter and 5/8-inch in length, consisting of uranium fuel in a ceramic form--uranium dioxide, UO_2. Typical fuel pellet enrichments in nuclear power reactors range from 2.0 percent to 3.5 percent uranium-235.

Performance-based Regulation: Required results or outcome of performance rather than a prescriptive process, technique, or procedure.

Performance-based Regulatory Action: Licensee attainment of defined objectives and results without detailed direction from the NRC on how these results are to be obtained. (See the Communication Plan for Performance-Based Regulation by using accession number ML021120533 in ADAMS.)

Periodic Table: An arrangement of chemical elements in order of increasing atomic number. Elements of similar properties are placed one under the other, yielding groups or families of elements. Within each group, there is a variation of chemical and physical properties, but in general, there is a similarity of chemical behavior within each group.

Permanent Cessation of Operations: means, for a nuclear power reactor facility, a certification by a licensee to the NRC that it has permanently ceased or will permanently cease reactor operation(s), or a final legally effective order to permanently cease operation(s) has come into effect.

Permanent Fuel Removal: means, for a nuclear power reactor facility, a certification by the licensee to the NRC that it has permanently removed all fuel assemblies from the reactor vessel.

Person: means (1) any individual, corporation, partnership, firm, association, trust, estate, public or private institution, group, government agency other than the Commission or the Department, except that the Department shall be considered a person to the extent that its facilities are subject to the licensing and related regulatory authority of the Commission pursuant to section 202 of the Energy Reorganization Act of 1974, any State or any political subdivision of, or any political entity within a State, any foreign government or nation or any political subdivision of any such government or nation, or other entity; and (2) any legal successor, representative, agent, or agency of the foregoing.

Personnel Monitoring: The use of portable survey meters to determine the amount of radioactive contamination on individuals, or the use of dosimetry to determine an individual's occupational radiation dose.

Petroleum: A generic term applied to oil and oil products in all forms.

Photon: A quantum (or packet) of energy emitted in the form of electromagnetic radiation. Photons have momentum and energy, but no rest mass or electrical charge. Gamma rays and x-rays are examples of photons.

Photomultiplier: Commonly used device for detecting photons by converting them to an electrical signal.

Photovoltaic Cell: An electronic device consisting of layers of semiconductor materials fabricated to convert incident light directly into electricity (direct current).

Photovoltaic Module: An integrated assembly of interconnected photovoltaic cells designed to deliver a selected level of working voltage and suited for incorporation in photovoltaic power systems.

Pico: A prefix that divides a basic unit by one trillion (10^{-12}).

Picocurie: One trillionth (10^{-12}) of a curie.

Pig: A colloquial term describing a container (usually lead or depleted uranium) used to ship or store radioactive materials. The thick walls of this shielding device protect the person handling the container from radiation. Large containers used for spent fuel storage are commonly called casks.

Pile: A colloquial term describing the first nuclear reactors. They are called piles because the earliest reactors were "piles" of graphite and uranium blocks.

Pion: The least massive known spin-0 meson. The three charge states of the pion (negative, neutral and positive) are involved in the long-range force between the nucleons.

Planned Special Exposure: An infrequent exposure to radiation, separate from and in addition to the annual dose limits

Plausible Accidents: Postulated events that meet a probability test rather than the more challenging test represented by a design-basis event.

Plutonium (Pu): A heavy, radioactive, manmade metallic element with atomic number 94. Its most important isotope is fissile plutonium-239, which is produced by neutron irradiation of uranium-238. It exists in only trace amounts in nature.

Pocket Dosimeter: A small ionization detection instrument that indicates ionizing radiation exposure directly. An auxiliary charging device is usually necessary.

Poison, Neutron: In reactor physics, a material other than fissionable material in the vicinity of the reactor core that will absorb neutrons. The addition of poisons, such as control rods or boron, into the reactor is said to be an addition of negative reactivity.

Pool Reactor: A reactor in which the fuel elements are suspended in a pool of water that serves as the reflector, moderator, and coolant. Popularly called a "swimming pool reactor," it is used for research and training, not for electrical generation.

Positron: Particle equal in mass but opposite in charge to the electron. A positive electron.

Possession-only License: A form of license that allows possession but not operation.

Power Coefficient of Reactivity: The change in reactivity per percent change in power. The power coefficient is the summation of the moderator temperature coefficient of reactivity, the fuel temperature coefficient of reactivity, and the void coefficient of reactivity.

Power Defect: The total amount of reactivity added due to a given change in power. It can also be expressed as the integrated power coefficient over the range of the power change.

Power Reactor: A reactor designed to produce heat for electric generation (as distinguished from reactors used for research), for producing radiation or fissionable materials or for reactor component testing.

Preliminary Notification (PN): A brief summary report issued by the NRC staff to notify the Commission of the occurrence of a significant event that appears to have health and safety significance or major public or media interest. PNs are based on information provided by State radiation control program staff.

Pressure Vessel: A strong-walled container housing the core of most types of power reactors. It usually also contains the moderator, neutron reflector, thermal shield, and control rods.

Pressurized Water Reactor (PWR): A power reactor in which heat is transferred from the core to an exchanger by high temperature water kept under high pressure in the primary system. Steam is generated in a secondary circuit. Many reactors producing electric power are pressurized water reactors.

Pressurizer: A tank or vessel that acts as a head tank (or surge volume) to control the pressure in a pressurized water reactor.

Price-cap Regulation: means the system of rate regulation in which a rate regulatory authority establishes rates that an electric generator may charge its customers that are based on a specified maximum price of electricity.

Primary System: A term that may be used for referring to the reactor coolant system.

Probabilistic Risk Analysis: A systematic method for addressing the risk triplet as it relates to the performance of a complex system to understand likely outcomes, sensitivities, areas of importance, system interactions, and areas of uncertainty. The risk triplet is the set of three questions that the NRC uses to define “risk”:

1. What can go wrong?
2. How likely is it?
3. What are the consequences?

Procurement Document: means, for the purposes of § 50.55(e) of this chapter, a contract that defines the requirements which facilities or basic components must meet in order to be considered acceptable by the purchaser.

Produce: when used in relation to special nuclear material, means

1. To manufacture, make, produce, or refine special nuclear material
2. To separate special nuclear material from other substances in which such material may be contained; or
3. To make or to produce new special nuclear material.

Production Expense: Production expenses are a component of generation expenses that includes costs associated with operation, maintenance, and fuel.

Production facility: means:

1. Any nuclear reactor designed or used primarily for the formation of plutonium or uranium-233;
2. Any facility designed or used for the separation of the isotopes of plutonium, except laboratory scale facilities designed or used for experimental or analytical purposes only;
3. Any facility designed or used for the processing of irradiated materials containing special nuclear material, except
   - **A.** laboratory scale facilities designed or used for experimental or analytical purposes,
   - **B.** Facilities in which the only special nuclear materials contained in the irradiated material to be processed are uranium enriched in the isotope U-235 and plutonium produced by the irradiation, if the material processed contains not more than 106 grams of plutonium per gram of U-235 and has fission product activity not in excess of 0.25 millicuries of fission products per gram of U-235,
   - **C.** Facilities in which processing is conducted pursuant to a license issued under parts 30 and 70 of **10-CFR 50.55(e)**, or equivalent regulations of an Agreement State, for the receipt, possession, use, and transfer of irradiated special nuclear material, which authorizes the processing of the irradiated material on a batch basis for the separation of selected fission products and limits the process batch to not more than 100 grams of uranium enriched in the isotope 235 and not more than 15 grams of any other special nuclear material.

Proportional Counter: A radiation instrument in which an electronic detection system receives pulses that are proportional to the number of ions formed in a gas-filled tube by ionizing radiation.

Proprietary Information: Privately owned knowledge or data, such as that protected by a registered patent, copyright, or trademark.

Proton: One of the basic particles that makes up an atom. The proton is found in the nucleus and has a positive electrical charge equal to the negative charge of an electron and a mass similar to that of a neutron**:** a hydrogen nucleus.

Proton Number: The total number of protons in the nucleus, Z.

Primary Energy: The energy that is embodied in resources as they exist in nature (e.g., coal, crude oil, natural gas, or sunlight). For the most part, primary energy is transformed into electricity or fuels such as gasoline or charcoal. These, in turn, are referred to as secondary or site energy.

Propane: A normally gaseous straight-chain hydrocarbon (C3H8). It is a colorless paraffinic gas that is extracted from natural gas or refinery gas streams.

Public Dose: The dose received by a member of the public from exposure to radiation or to radioactive material released by a licensee, or to any other source of radiation under the control of a licensee. Public dose does not include occupational dose or doses received from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive materials and released in accordance with 10 CFR 35.75, or from voluntary participation in medical research programs.

Quadrillion Btu (Quad): Equivalent to 10 to the 15th power Btu (1 quad = 1.055 x 10e18 joules).

Quality Factor: The factor by which the absorbed dose (rad or gray) is to be multiplied to obtain a quantity that expresses, on a common scale for all ionizing radiation, the biological damage (rem or sievert) to an exposed individual. It is used because some types of radiation, such as alpha particles, are more biologically damaging internally than other types.

Quantum Theory: The concept that energy is radiated intermittently in units of definite magnitude, called quanta, and absorbed in a like manner.

QCD: Quantum chromodynamics, the gauge theory describing the color strong interaction.

QED: Quantum electrodynamics, the gauge theory describing electromagnetism.

Quark: A strongly-interacting fermion that is a building block of hadronic matter. Quarks come in six flavors: up, down, charm, strange, top, bottom.

Rad (Radiation Absorbed Dose): The special unit for radiation absorbed dose, which is the amount of energy from any type of ionizing radiation (e.g., alpha, beta, gamma, neutrons, etc.) deposited in any medium (e.g., water, tissue, air). A dose of one rad means the absorption of 100 ergs (a small but measurable amount of energy) per gram of absorbing tissue (100 rad = 1 gray).

Radiation Area: Any area with radiation levels greater than 5 millirems (0.05 millisievert) in one hour at 30 centimeters from the source or from any surface through which the radiation penetrates.

Radiation Detection Instrument: A device that detects and displays the characteristics of ionizing radiation.

Radiation (Ionizing Radiation): Alpha particles, beta particles, gamma rays, x-rays, neutrons, high-speed electrons, high-speed protons, and other particles capable of producing ions. Radiation, as used in 10 CFR Part 20, does not include non-ionizing radiation, such as radio- or microwaves, or visible, infrared, or ultraviolet light (see also 10 CFR 20.1003).

Radiation, Nuclear: Particles (alpha, beta, neutrons) or photons (gamma) emitted from the nucleus of unstable radioactive atoms as a result of radioactive decay.

Radiation Shielding: Reduction of radiation by interposing a shield of absorbing material between any radioactive source and a person, work area, or radiation-sensitive device.

Radiation Sickness (Syndrome): The complex of symptoms characterizing the disease known as radiation injury, resulting from excessive exposure (greater than 200 rads or 2 gray) of the whole body (or large part) to ionizing radiation. The earliest of these symptoms are nausea, fatigue, vomiting, and diarrhea, which may be followed by loss of hair (epilation), hemorrhage, inflammation of the mouth and throat, and general loss of energy. In severe cases, where the radiation exposure has been approximately 1000 rad (10 gray) or more, death may occur within two to four weeks. Those who survive six weeks after the receipt of a single large dose of radiation to the whole body may generally be expected to recover.

Radiation Source: Usually a sealed source of radiation used in tele-therapy and industrial radiography, as a power source for batteries (as in use in space craft), or in various types of industrial gauges. Machines, such as accelerators and radioisotope generators, and natural radio-nuclides may be considered sources.

Radiation Standards: Exposure standards, permissible concentrations, rules for safe handling, regulations for transportation, regulations for industrial control of radiation, and control of radioactive material by legislative means.

Radiation Warning Symbol: An officially prescribed symbol (a magenta or black trefoil) on a yellow background that must be displayed where certain quantities of radioactive materials are present or where certain doses of radiation could be received.

Radiative capture: In radiative capture the incident neutron enters the target nucleus forming a compound nucleus. The compound nucleus then decays to its ground state by gamma emission.

Radioactive Contamination: Deposition of radioactive material in any place where it may harm persons or equipment.

Radioactive Dating: A technique for estimating the age of an object by measuring the amounts of various radioisotopes in it.

Radioactive Decay: Large unstable atoms can become more stable by emitting radiation. This process is called radioactive decay. This radiation can be emitted in the form of a positively charged alpha particle, a negatively charged beta particle, or gamma rays or x-rays.

Radioactive Series: A succession of nuclides, each of which transforms by radioactive disintegration into the next until a stable nuclide results. The first member is called the parent, the intermediate members are called daughters, and the final stable member is called the end product.

Radioactive Waste: Materials that are radioactive and for which there is no further use.

Radioactivity: The spontaneous emission of radiation, generally alpha or beta particles, often accompanied by gamma rays, from the nucleus of an unstable isotope. Also, the rate at which radioactive material emits radiation. Measured in units of becquerels or disintegrations per second.

Radiography: The making of a shadow image on photographic film by the action of ionizing radiation.

Radioisotope: An unstable isotope of an element that decays or disintegrates spontaneously, emitting radiation. Approximately 5,000 natural and artificial radioisotopes have been identified.

Radiological Sabotage: Any deliberate act directed against a plant or transport in which an activity licensed pursuant to 10 CFR Part 73 of NRC's regulations is conducted or against a component of such a plant or transport that could directly or indirectly endanger the public health and safety by exposure to radiation.

Radiological Survey: The evaluation of the radiation hazards accompanying the production, use, or existence of radioactive materials under a specific set of conditions. Such evaluation customarily includes a physical survey of the disposition of materials and equipment, measurements or estimates of the levels of radiation that may be involved, and a sufficient knowledge of processes affecting these materials to predict hazards resulting from expected or possible changes in materials or equipment.

Radiology: That branch of medicine dealing with the diagnostic and therapeutic applications of radiant energy, including x-rays and radioisotopes

Radionuclide: A radioactive nuclide. An unstable isotope of an element that decays or disintegrates spontaneously, emitting radiation.

Radiosensitivity: The relative susceptibility of cells, tissues, organs, organisms, or other substances to the injurious action of radiation.

Radium (Ra): A radioactive metallic element with atomic number 88. As found in nature, the most common isotope has a mass number of 226. It occurs in minute quantities associated with uranium in pitchblende, camotite, and other minerals.

Radon (Rn): A radioactive element that is one of the heaviest gases known. Its atomic number is 86. It is a daughter of radium.

Reaction: Any process involving a chemical or nuclear change.

Reactivity: A term expressing the departure of a reactor system from criticality. A positive reactivity addition indicates a move toward supercriticality (power increase). A negative reactivity addition indicates a move toward subcriticality (power decrease).

Reactor Coolant Pressure Boundary: means all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are:

1. Part of the reactor coolant system, or
2. Connected to the reactor coolant system, up to and including any and all of the following:
   - **A.** The outermost containment isolation valve in system piping which penetrates primary reactor containment,
   - **B.** The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment,
   - **C.** The reactor coolant system safety and relief valves.

For nuclear power reactors of the direct cycle boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valve in the main steam and feedwater piping.

Reactor Coolant System: The system used to remove energy from the reactor core and transfer that energy either directly or indirectly to the steam turbine.

Reactor, Nuclear: A device in which nuclear fission may be sustained and controlled in a self-supporting nuclear reaction. The varieties are many, but all incorporate certain features, including fissionable material or fuel, a moderating material (unless the reactor is operated on fast neutrons), a reflector to conserve escaping neutrons, provisions of removal of heat, measuring and controlling instruments, and protective devices. The reactor is the heart of a nuclear power plant.

Reasonable: Rational, sensible, or resulting from sound judgment.

Reference Man: A person with the anatomical and physiological characteristics of an average individual that is used in calculations assessing internal dose.

Reflector: A layer of material immediately surrounding a reactor core that scatters back (or reflects) into the core many neutrons that would otherwise escape. The returned neutrons can then cause more fissions and improve the neutron economy of the reactor. Also called moderator.

Rem (Roentgen Equivalent Man): The acronym for roentgen equivalent man is a standard unit that measures the effects of ionizing radiation on humans. The dose equivalent in rems is equal to the absorbed dose in rads multiplied by the quality factor of the type of radiation (see 10 CFR 20.1004).

Research and Development: means (1) theoretical analysis, exploration, or experimentation; or (2) the extension of investigative findings and theories of a scientific or technical nature into practical application.

Responsible Officer: means, for the purposes of § 50.55(e) of this chapter, the president, vice-president, or other individual in the organization of a corporation, partnership, or other entity who is vested with executive authority over activities subject to this part.

Restricted Data: means all data concerning (1) design, manufacture, or utilization of atomic weapons; (2) the production of special nuclear material; or (3) the use of special nuclear material in the production of energy, but shall not include data declassified or removed from the Restricted Data category.

Restricted Area: Any area to which access is controlled for the protection of individuals from exposure to radiation and radioactive materials.

Risk: The combined answers to

1. What can go wrong?
2. How likely is it?
3. What are the consequences?

Risk-based Decision Making: An approach to regulatory decision making in which such decisions are made solely based on the results of a probabilistic risk analysis.

Risk-informed Decision Making: An approach to decision making in which insights from probabilistic risk analyses are considered with other engineering insights.

Risk-informed Regulation: Incorporating an assessment of safety significance or relative risk in NRC regulatory actions. Making sure that the regulatory burden imposed by individual regulations or processes is commensurate with the importance of that regulation or process to protecting public health and safety and the environment.

Risk-significant: When used to qualify an object, such as a system, structure, component, accident sequence, or cut set, this term identifies that object as exceeding a predetermined criterion related to its contribution to the risk from the facility being addressed.

Roentgen (R): A unit of exposure to ionizing radiation. It is the amount of gamma or x-rays required to produce ions resulting in a charge of 0.000258 coulombs/kilogram of air under standard conditions. Named after Wilhelm Roentgen, the German scientist who discovered x-rays in 1895.

Rubblization: A decommissioning technique involving demolition and burial of formerly operating nuclear facilities. All equipment from buildings is removed and the surfaces are decontaminated. Above-grade structures are demolished into rubble and buried in the structure's foundation below ground. The site surface is then covered, regraded and, landscaped for unrestricted use.

rem (röntgen equivalent, man): A measure of dose deposited in body tissue, averaged over the body. One rem is approximately the dose from any radiation corresponding to exposure to one röntgen of g radiation. The rem is no longer accepted for use with the International System. One rem is equivalent to 0.01 sievert.

Renewable Energy: Energy obtained from sources that are essentially inexhaustible (unlike, for example, the fossil fuels, of which there is a finite supply). Renewable sources of energy include conventional hydroelectric power, wood, waste, geothermal, wind, photovoltaic, and solar thermal energy.

Residual Strong Force: Force between composite objects (made of quarks) due to the remaining effect of the color force on colorless objects. These forces are much weaker than the strong color force.

Röntgen or **roentgen (R):** Unit of exposure measuring the ionizing ability of g radiation; one röntgen produces one electric charge (1.6×10^{-19} C) per 10^6 m^3 of dry air at 0° C and atmospheric pressure. This corresponds to an energy loss of 0.0877 joule per kilogram in air. The röntgen is no longer accepted for use with the International System.

Safe Shutdown Earthquake: Is the maximum earthquake potential for which certain structures, systems, and components, important to safety, are designed to sustain and remain functional.

Safeguards: As used in regulation of domestic nuclear facilities and materials, the use of material control and accounting programs verify that all special nuclear material is properly controlled and accounted for, and the physical protection (also referred to as physical security) equipment and security forces. As used by the International Atomic Energy Agency (IAEA), verifying that the "peaceful use" commitments made in binding non-proliferation agreements, both bilateral and multilateral, are honored.

Safety Injection: The rapid insertion of a chemically soluble neutron poison (such as boric acid) into the reactor coolant system to ensure reactor shutdown.

Safety Limit: A restriction or range placed upon important process variables that are necessary to reasonably protect the integrity of the physical barriers that guard against the uncontrolled release of radioactivity.

Safety Related: In the regulatory arena, this term applies to systems, structures, components, procedures, and controls of a facility or process that are relied upon to remain functional during and following design-basis events. Their functionality ensures that key regulatory criteria, such as levels of radioactivity released, are met. Examples of safety related functions include shutting down a nuclear reactor and maintaining it in a safe shutdown condition.

Safety-Related Structures: systems and components means those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

1. The integrity of the reactor coolant pressure boundary
2. The capability to shut down the reactor and maintain it in a safe shutdown condition
3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

Safe Shutdown: (non-design basis accident (non-DBA)) for station blackout means bringing the plant to those shutdown conditions specified in plant technical specifications as Hot Standby or Hot Shutdown, as appropriate.

Safety-significant: When used to qualify an object, such as a system, structure, component, accident sequence, or cut set, this term identifies that object as having an impact on safety, whether determined through risk analysis or other means, that exceeds a predetermined significance criterion.

SAFSTOR: A method of decommissioning in which the nuclear facility is placed and maintained in such condition that the nuclear facility can be safely stored and subsequently decontaminated to levels that permit release for unrestricted use.

Scattered Radiation: Radiation that, during its passage through a substance, has been changed in direction. It may also have been modified by a decrease in energy. It is one form of secondary radiation.

Scintillation Detector: The combination of phosphor, photomultiplier tube, and associated electronic circuits for counting light emissions produced in the phosphor by ionizing radiation.

Scram: The sudden shutting down of a nuclear reactor, usually by rapid insertion of control rods, either automatically or manually by the reactor operator. May also be called a reactor trip. It is actually an acronym for "safety control rod axe man," the worker assigned to insert the emergency rod on the first reactor (the Chicago Pile) in the U.S.

Sealed Source: Any radioactive material or byproduct encased in a capsule designed to prevent leakage or escape of the material.

Secondary Radiation: Radiation originating as the result of absorption of other radiation in matter. It may be either electromagnetic or particulate in nature.

Secondary System: The steam generator tubes, steam turbine, condenser, and associated pipes, pumps, and heaters used to convert the heat energy of the reactor coolant system into mechanical energy for electrical generation. Most commonly used in reference to pressurized water reactors.

Seismic category I: Structures, systems, and components that are designed and built to withstand the maximum potential earthquake stresses for the particular region where a nuclear plant is sited.

Severe Accident: A type of accident that may challenge safety systems at a level much higher than expected.

Shallow-Dose Equivalent (SDE): The external exposure dose equivalent to the skin or an extremity at a tissue depth of 0.007 centimeters (7 mg/cm^2) averaged over an area of 1 square centimeter.

Shielding: Any material or obstruction that absorbs radiation and thus tends to protect personnel or materials from the effects of ionizing radiation.

Shutdown: A decrease in the rate of fission (and heat production) in a reactor (usually by the insertion of control rods into the core).

Shutdown Margin: The instantaneous amount of reactivity by which the reactor is sub-critical or would be sub-critical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth that is assumed to be fully withdrawn.

Source Material: means source material as defined in subsection 11z. of the Act and in the regulations contained in part 40 of this chapter.

Source Term: refers to the magnitude and mix of the radio-nuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release.

Special Nuclear Material: means (1) plutonium, uranium-233, uranium enriched in the isotope-233 or in the isotope-235, and any other material which the Commission, pursuant to the provisions of section 51 of the act, determines to be special nuclear material, but does not include source material; or (2) any material artificially enriched by any of the foregoing, but does not include source material.

Standard Industrial Classification (SIC): A set of codes developed by the Office of Management and Budget which categorizes industries according to groups with similar economic activities.

Scaler: An electronic instrument for counting radiation induced pulses from radiation detectors such as a Geiger-Müller tube.

scintillation Counter: An instrument that detects and measures gamma radiation by counting the light flashes (scintillations) induced by the radiation.

Scintillator: Material that emits light when particles traverse it.

Secular Equilibrium: A state of parent-daughter equilibrium that is achieved when the half-life of the parent is much longer than the half-life of the daughter. In this case, if the two are not separated, the daughter will eventually decay at the same rate at which it is being produced. At this point, both parent and daughter will decay at the same rate until the parent is essentially exhausted.

Shielding: A protective barrier, usually a dense material, that reduces the passage of radiation from radioactive materials to the surroundings by absorbing it.

Sievert (Sv): A measure of dose (technically, dose equivalent) deposited in body tissue, averaged over the body. Such a dose would be caused by an exposure imparted by ionizing x-ray or gamma radiation undergoing an energy loss of 1 joule per kilogram of body tissue (l gray). One sievert is equivalent to 100 rem.

Somatic Effects of Radiation: Effects of radiation limited to the exposed individual, as distinguished from genetic effects, that may also affect subsequent unexposed generations.

Source Material: Uranium or thorium, or any combination thereof, in any physical or chemical form or ores that contain by weight 1/20 of one percent (0.05 percent) or more of:

1. uranium,
2. thorium,
3. any combination thereof.

Source material does not include special nuclear material.

Special Nuclear Material: Plutonium, uranium-233, or uranium enriched in the isotopes uranium-233 or uranium-235.

Spent (depleted) Fuel: Nuclear reactor fuel that has been used to the extent that it can no longer effectively sustain a chain reaction.

Spent Fuel Pool: An underwater storage and cooling facility for spent (used) fuel elements that have been removed from a reactor.

Spent Nuclear Fuel: Fuel that has been removed from a nuclear reactor because it can no longer sustain power production for economic or other reasons.

Stable Isotope: An isotope that does not undergo radioactive decay.

Standard Review Plan: A document that provides guidance to the staff for reviewing an application to obtain an NRC license to construct or operate a nuclear facility or to possess or use nuclear materials.

Standard Technical Specifications: NRC staff guidance on model technical specifications for an operating license. (See also Technical Specifications.)

Startup: An increase in the rate of fission (and heat production) in a reactor (usually by the removal of control rods from the core).

Station blackout: means the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system). Station blackout does not include the loss of available ac power to buses fed by station batteries through inverters or by alternate ac sources as defined in this section, nor does it assume a concurrent single failure or design basis accident. At single unit sites, any emergency ac power source(s) in excess of the number required to meet minimum redundancy requirements (i.e., single failure) for safe shutdown (non-DBA) is assumed to be available and may be designated as an alternate power source(s) provided the applicable requirements are met. At multi-unit sites, where the combination of emergency ac power sources exceeds the minimum redundancy requirements for safe shutdown (non-DBA) of all units, the remaining emergency ac power sources may be used as alternate ac power sources provided they meet the applicable requirements. If these criteria are not met, station blackout must be assumed on all the units.

Stay Time: The period during which personnel may remain in a restricted area in a reactor before accumulating some permissible occupational dose.

Steam Generator: The heat exchanger used in some reactor designs to transfer heat from the primary (reactor coolant) system to the secondary (steam) system. This design permits heat exchange with little or no contamination of the secondary system equipment.

Stochastic Effects: Effects that occur by chance, generally occurring without a threshold level of dose, whose probability is proportional to the dose and whose severity is independent of the dose. In the context of radiation protection, the main stochastic effects are cancer and genetic effects.

Source: A radioactive material that produces radiation for experimental or industrial use.

Stable: Non-radioactive.

Standard Model: Gauge theory encompassing the electroweak and strong interactions.

Strong Interaction: The interaction due to exchange of color. Also called strong force.

Sub-critical Mass: An amount of fissionable material insufficient in quantity or of improper geometrical configuration to sustain a fission chain reaction.

Sub-criticality: The condition of a nuclear reactor system when the rate of production of fission neutrons is lower than the rate of production in the previous generation owing to increased neutron leakage and poisons.

Substantial Safety Hazard: means, for the purposes of § 50.55(e) of this chapter, a loss of safety function to the extent that there is a major reduction in the degree of protection provided to public health and safety for any facility or activity authorized by the construction permit issued under this part.

Supercritical Reactor: A reactor in which the power level is increasing with time.

Supercriticality: The condition for increasing the level of operation of a reactor. The rate of fission neutron production exceeds all neutron losses, and the overall neutron population increases.

Superheating: The heating of a vapor, particularly steam, to a temperature much higher than the boiling point at the existing pressure. This is done in some power plants to improve efficiency and to reduce water damage to the turbine.

Survey Meter: Any portable radiation detection instrument especially adapted for inspecting an area or individual to establish the existence and amount of radioactive material present.

Symmetry: Invariance of equations of motion under changes in condition.

Tailings: Naturally radioactive residue from the processing of uranium ore into yellowcake in a mill. Although the milling process recovers about 93 percent of the uranium, the residues, or tailings, contain several naturally-occurring radioactive elements, including uranium, thorium, radium, polonium, and radon.

Testing Facility: A nuclear reactor of a type described in **10-CFR-50.21(c)** of this part and for which an application has been filed for a license authorizing operation at:

1. A thermal power level in excess of 10 megawatts
2. A thermal power level in excess of 1 megawatt, if the reactor is to contain:
    - **A.** A circulating loop through the core in which the applicant proposes to conduct fuel experiments
    - **B.** A liquid fuel loading
    - **C.** An experimental facility in the core in excess of 16 square inches in cross-section.

Thermal Energy: Random kinetic energy possessed by objects in a material at finite temperature.

Tracer: A small amount of radioactive isotope introduced into a system in order to follow the behavior of some component of that system.

Transmutation: The transformation of one element into another by a nuclear reaction.

Technical Specifications: Part of an NRC license authorizing the operation of a nuclear production or utilization facility. A Technical Specification establishes requirements for items such as safety limits, limiting safety system settings, limiting

control settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls. (See also Standard Technical Specifications.)

Terrestrial Radiation: The portion of the natural background radiation that is emitted by naturally occurring radioactive materials, such as uranium, thorium, and radon in the earth.

Thermal Breeder Reactor: A breeder reactor in which the fission chain reaction is sustained by thermal neutrons.

Thermal power: The total core heat transfer rate to the reactor coolant.

Thermal Reactor: A reactor in which the fission chain reaction is sustained primarily by thermal neutrons. Most current reactors are thermal reactors.

Thermal shield: A layer, or layers, of high-density material located within a reactor pressure vessel or between the vessel and the biological shield to reduce radiation heating in the vessel and the biological shield.

Thermalization: The process undergone by high-energy (fast) neutrons as they lose energy by collision.

Thermoluminescent dosimeter: A small device used to measure radiation by measuring the amount of visible light emitted from a crystal in the detector when exposed to ionizing radiation.

Thermonuclear: An adjective referring to the process in which very high temperatures are used to bring about the fusion of light nuclei, such as those of the hydrogen isotopes deuterium and tritium, with the accompanying liberation of energy.

Total Effective Dose Equivalent (TEDE): The sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

Transient: A change in the reactor coolant system temperature and/or pressure due to a change in power output of the reactor. Transients can be caused by:

1. adding or removing neutron poisons,
2. increasing or decreasing electrical load on the turbine generator,
3. accidental conditions.

Transuranic Element: An artificially made, radioactive element that has an atomic number higher than uranium in the periodic table of elements such as neptunium, plutonium, americium, and others.

Transuranic waste: Material contaminated with transuranic elements that is produced primarily from reprocessing spent fuel and from use of plutonium.

Trip, Reactor: A term that is used by pressurized water reactors for a reactor scram (see Scram).

Tritium: A radioactive isotope of hydrogen (one proton, two neutrons). Because it is chemically identical to natural hydrogen, tritium can easily be taken into the body by any ingestion path. It decays by beta emission. It has a radioactive half-life of about 12.5 years.

Thorium: Thorium is an element with an atomic number of 90. This element occurs in nature almost entirely as a single nuclear isotope, with mass number of 232. Thorium is called a fertile material because when it absorbs a neutron it becomes $\mathbf{U^{233}}$ which is fissile.

Turbine: A machine for generating rotary mechanical power from the energy of a stream of fluid (such as water, steam, or hot gas). Turbines convert the kinetic energy of fluids to mechanical energy through the principles of impulse and reaction, or a mixture of the two.

A rotary engine made with a series of curved vanes on a rotating shaft, usually turned by water or steam. Turbines are considered the most economical means to turn large electrical generators.

Turbine generator (TG): A steam (or water) turbine directly coupled to an electrical generator. The two devices are often referred to as one unit.

Ultraviolet: Electromagnetic radiation of a wavelength between the shortest visible violet and low energy x-rays.

Ultraviolet Radiation: Electromagnetic radiation having wavelengths between the visible part of the spectrum and x-rays.

Uncertainty Range: Defines an interval within which a numerical result is expected to lie within a specified level of confidence. The interval often used is the 5-95 percentile of the distribution reporting the uncertainty.

Unique Purpose: means a project, program, or commercial activity which cannot reasonably be accomplished without the use of **Highly Enriched Uranium (HEU**) fuel, and may include:

1. A specific experiment, program, or commercial activity (typically long-term) that significantly serves the U.S. national interest and cannot be accomplished without the use of **HEU** fuel
2. Reactor physics or reactor development based explicitly on the use of **HEU** fuel
3. Research projects based on neutron flux levels or spectra attainable only with **HEU** fuel
4. A reactor core of special design that could not perform its intended function without using **HEU** fuel.

United States: when used in a geographical sense, includes Puerto Rico and all territories and possessions of the United States.

Unnecessary Regulatory Burden: Regulatory criteria that go beyond the levels that would be reasonably expected to be imposed on licensees given that regulations apply to conditions that incorporate normal operation and design-basis conditions.

Unrestricted area: The area outside the owner-controlled portion of a nuclear facility (usually the site boundary). An area in which a person could not be exposed to radiation levels in excess of 2 millirems in any one hour from external sources (see 10 CFR 20.1003).

Unstable isotope: A radioactive isotope (see also stable isotope).

Unwanted Radioactive Material (Orphan Sources): refers to sealed sources of radioactive material contained in a small volume (but not radioactively contaminated soils and bulk metals) in any one or more of the following conditions (taken from the NRC Orphan Source Initiative):

1. In an uncontrolled condition that requires removal to protect public health and safety from a radiological threat;
2. Controlled or uncontrolled, but for which a responsible party cannot be readily identified;
3. Controlled, but the material's continued security cannot be assured. If held by a licensee, the licensee has few or no options for, or is incapable of providing for, the safe disposition of the material;
4. In the possession of a person, not licensed to possess the material, who did not seek to possess the material; or
5. In the possession of a state radiological protection program for the sole purpose of mitigating a radiological threat because of one of the above conditions, and for which the state does not have a means to provide for the material's appropriate disposition.

Uranium: A radioactive element with the atomic number 92 and, as found in natural ores, an atomic weight of approximately 238. The two principal natural isotopes are uranium-235, $\mathbf{U^{235}}$**.** (0.7 percent of natural uranium), which is fissile, and uranium-238, $\mathbf{U^{238}}$ (99.3 percent of natural uranium), which is fissionable by fast neutrons and is fertile. Natural uranium also includes a minute amount of uranium-234, $\mathbf{U^{234}}$.

Uranium Fuel Fabrication Facility: A facility that:

1. Manufactures reactor fuel containing uranium for any of the following:
   - **A.** preparation of fuel materials;
   - **B.** formation of fuel materials into desired shapes;
   - **C.** application of protective cladding;
   - **D.** recovery of scrap material;
   - **E.** storage associated with such operations;
2. Conducts research and development activities.

Uranium Hexafluoride Production Facility: A facility that receives natural uranium in the form of ore concentrate, processes the concentrate, and converts it into uranium hexafluoride (UF_6).

Utilization Facility means any nuclear reactor other than one designed or used primarily for the formation of plutonium or U^{233}.

Van de Graaff Accelerator: Device using a high voltage terminal to accelerate charged particles.

Vapor: The gaseous form of substances that are normally in liquid or solid form.

Very High Radiation Area: An area accessible to individuals, in which radiation levels exceed 500 rad (5 gray) in one hour at 1 meter from the source or from any surface that the radiation penetrates (see 10 CFR 20.1003).

Viability Assessment: A Department of Energy decision making process to judge the prospects for geologic disposal of high-level radioactive wastes at Yucca Mountain based on;

1. Specific design work on the critical elements of the repository and waste package,
2. A total system performance assessment that will describe the probable behavior of the repository,
3. A plan and cost estimate for the work required to complete a license application,
4. An estimate of the costs to construct and operate the repository(see 10 CFR Part 60). The viability assessment was required by the **Energy and Water Development Appropriations Act,** 1997 (Public Law 104-206). After the viability assessment was completed, site-specific environmental standards at 40 CFR Part 197 and implementing regulations (see 10 CFR Part 63).

Void: In a nuclear power reactor, an area of lower density in a moderating system (such as steam bubbles in water) that allows more neutron leakage than does the more dense material around it.

Void Coefficient of Reactivity: A rate of change in the reactivity of a water reactor system resulting from a formation of steam bubbles as the power level and temperature increase.

Waste, Radioactive: Radioactive materials at the end of a useful life cycle or in a product that is no longer useful and should be properly disposed of.

Watt: An electrical unit of power. 1 watt = 1 Joule/second. It is equal to the power in a circuit in which a current of one ampere flows across a potential difference of one volt.

Watt-hour: An electrical energy unit of measure equal to 1 watt of power supplied to, or taken from, an electrical circuit steadily for 1 hour.

Weighting factor (WT): Multipliers of the equivalent dose to an organ or tissue used for radiation protection purposes to account for different sensitivities of different organs and tissues to the induction of stochastic effects of radiation

Well-logging: All operations involving the lowering and raising of measuring devices or tools that contain licensed material or are used to detect licensed materials in wells for the purpose of obtaining information about the well or adjacent formations that may be used in oil, gas, mineral, groundwater, or geological exploration (see 10 CFR 39.2).

Wheeling Service: The movement of electricity from one system to another over transmission facilities of intervening systems. Wheeling service contracts can be established between two or more systems.

Watt (Electric): The electrical unit of power. The rate of energy transfer equivalent to one ampere of electric current flowing under a pressure of one volt at unity power factor.

Watthour (Wh): The electrical energy unit of measure equal to 1 watt of power supplied to, or taken from, an electric circuit steadily for one hour.

Weak Interaction: The interaction responsible for weak decays of particles, mediated by the exchange of W± and Z0 gauge bosons.

Whole-body Counter: A device used to identify and measure the radioactive material in the body of human beings and animals. It uses heavy shielding to keep out naturally existing background radiation and ultrasensitive radiation detectors and electronic counting equipment.

Whole-body Exposure: Whole body exposure includes at least the external exposure, head, trunk, arms above the elbow, or legs above the knee. Where a radioisotope is uniformly distributed throughout the body tissues, rather than being concentrated in certain parts, the irradiation can be considered as whole-body exposure (see also 10 CFR 20.1003).

Wipe Sample: A sample made for the purpose of determining the presence of removable radioactive contamination on a surface. It is done by wiping, with slight pressure, a piece of soft filter paper over a representative type of surface area. It is also known as a "swipe" or "smear" sample.

Wind Energy: The kinetic energy of wind converted into mechanical energy by wind turbines (i.e., blades rotating from a hub) that drive generators to produce electricity.

X-radiation: Electromagnetic radiation usually produced in transitions of the inner electrons of atoms. The wavelength is between ultraviolet and gamma rays.

X-ray: Electromagnetic radiation with wavelengths between ultraviolet and gamma rays.

Radiation from cosmic sources; naturally occurring radioactive materials, including radon (except as a decay product of source or special nuclear material) and global

fallout as it exists in the environment from the testing of nuclear explosive devices. It does not include radiation from source, byproduct, or special nuclear materials regulated by the Nuclear Regulatory Commission. The typically quoted average individual exposure from background radiation is 360 millirems per year.

Penetrating electromagnetic radiation (photon) having a wavelength that is much shorter than that of visible light. These rays are usually produced by excitation of the electron field around certain nuclei. In nuclear reactions, it is customary to refer to photons originating in the nucleus as x-rays.

Yellowcake: Yellowcake is the product of the uranium extraction (milling) process; early production methods resulted in a bright yellow compound, hence the name *yellowcake*. The material is a mixture of uranium oxides that can vary in proportion and in color from yellow to orange to dark green (blackish) depending at which temperature the material was dried (level of hydration and impurities). Higher drying temperatures produce a darker, less soluble material. Yellowcake is commonly referred to as U3O8 and is assayed as pounds U3O8 equivalent. This fine powder is packaged in drums and sent to a conversion plant that produces uranium hexafluoride (UF6) as the next step in the manufacture of nuclear fuel.