



“十二五”普通高等教育本科国家级规划教材

HEDIAN ZHUANYE YINGYU

核电专业英语

(第二版)

阎昌琪 编



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阎昌琪 编
阮於珍 主审



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内 容 提 要

本书为“十二五”普通高等教育本科国家级规划教材。

本书的内容涵盖了与核电有关的基础理论知识和专业知识,介绍了反应堆结构、核动力系统和设备,也包括了近年来核电的新发展,例如第三代和第四代反应堆的介绍。书中涉及的核电专业词汇广泛、内容丰富、知识性强。

为了便于读者掌握专业词汇,每课课文后对重要的关键词作了英、中文两种解释,同时还列出了重要的词汇解释、课文中的难点注释。为了加深读者对课文内容的理解,课后还附有习题和答案。

本书可作为普通高等教育本科核工程与核技术专业的英语阅读教材,也可作为核电工程技术人员的培训和自学用书,同时可作为能源动力类等相关专业人员的阅读材料。

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前 言

核工程与核技术专业是我国拥有核专业的大学设置的主要本科专业，核电技术是核工程专业的主要专业课内容，核电所涉及的学科领域广泛，包含很多当今世界的高新技术。美、英等发达国家在该领域处于领先地位，很多核电技术新的研究成果和新技术的介绍都以英文发表，学生毕业参加工作后都要阅读大量的英文资料，为此我们编写了这本核电专业英语教材，供核工程专业本科生提高专业英语阅读能力所用。

本书内容涵盖了核工程类专业所涉及的主要学科，既有与核反应堆有关的基础知识，也有核电站动力系统及主要设备的内容，还包括了核安全的基本常识介绍。书中内容选自英文原版教材和设计说明书，保证了语言的纯正性，同时兼顾内容新、知识面宽；力求课文内容概念性和知识性强、难度适中，不涉及复杂的专业理论。本书的内容安排由浅入深、由基础到专业，每一课都相对独立，可以根据学生的兴趣和专业方向选择使用。

本书第一版自出版以来，受到使用单位比较好的评价，目前几个有核工程类专业的学校都在使用本书作为本科生教材，如西安交通大学、哈尔滨工程大学、华北电力大学等学校都在使用，使用效果良好。同样，由于本书与核电技术紧密结合，目前一些核电人员培训单位已把本书列为岗前培训和订单培养的授课教材，因此本书在核电人才培养方面发挥了比较好的作用。

在本书的教学使用过程中，一些老师和同学都提出了宝贵的意见，对此作者在本次再版时进行了认真修改，在此向在本书编写和校对过程给予帮助的老师 and 同学一并表示感谢。

编 者

2016年6月

第一版前言

核电工程技术涉及的学科领域广泛, 包含很多高新技术。目前, 美、英等发达国家在该领域处于领先地位, 很多新的研究成果和新技术的介绍都以英文发表, 每个从事核电工程研究、设计和运行的专业人员都需要阅读大量的英文资料。为了满足越来越多的核电工程专业人员学习专业英语的需要, 编写了本书。

本书内容涵盖了核电工程所涉及的主要学科, 在介绍核反应堆物理及核反应堆热工等基础知识的同时, 也介绍了各种类型的核反应堆及主要设备的原理和使用特性等。主要内容包括: 核反应堆物理的基础知识、核材料、核反应堆热工水力学、反应堆结构、核动力系统和主要设备、核反应堆运行及核安全、核动力的发展及新一代反应堆等。本书内容广泛, 选用不同风格的文章, 做到内容新、知识面宽; 选用的课文概念性和知识性强、难度适中, 不涉及复杂的专业理论。为了便于读者掌握专业词汇, 每课课后对重要的关键词作了英、中文两种解释, 同时还列出了重要的词汇解释、课文中的难点注释。为了加深读者对课文内容的理解, 课后还附有习题和答案。

本书的内容安排由浅入深、由基础到专业, 可适合不同层次的学生使用。全书围绕着核电工程这一主题, 在专业部分中以新一代压水堆核电站动力装置为主线, 介绍了目前运行的核电技术以及新一代的核电技术。书中的每一课都有相当的独立性, 可以根据学生的兴趣和专业方向选择使用, 同时也考虑了课文内容满足总体需要, 保证全书内容是一个完整的整体。

本书的阅读内容取自英文的原版教科书和工程设计说明书。书中的引文, 编者尽力与版权所有取得联系, 但仍有部分联系不上, 在此深表歉意。

全书共 20 个单元, 由哈尔滨工程大学阎昌琪教授编写, 由中国原子能科学研究院阮於珍研究员主审, 阮老师提出了许多宝贵的意见和建议, 在此深表谢意。

由于编者水平所限, 书中难免存在缺点和不足, 敬请读者提出宝贵意见。

编 者

2010 年 5 月

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Unit 1 The Basic Concepts for Nuclear Physics

1.1 Atoms and Nuclei

The atoms of all elements, which at one time were thought to be the fundamental particles of nature, consist of numbers of three more fundamental particles—protons, neutrons and electrons. The arrangement of these particles within the atom, and in particular the number of protons and electrons, determine the chemical identity of the element. The atom consists of nucleus in which all the positively charged protons and uncharged neutrons are closely grouped together, and a number of negatively charged electrons moving in orbital paths around the nucleus. In an electrically neutral or unionized atom, the number of protons is equal to the number of electrons, and this number, Z , is the atomic number of a particular element and identifies it. (This number corresponds to the position of the element in the Periodic Table.) The number of neutrons in the nucleus is denoted by N , and the sum of the number of neutrons and protons in the nucleus is called, for reasons that will shortly be apparent, the mass number, A .

$$N + Z = A$$

The term nucleon is applied to all particles, both protons and neutrons, in the nucleus.

Fig.1.1 is a useful, though not strictly accurate representation of an atom of carbon with six protons and six neutrons in the nucleus, and six orbital electrons. To be more accurate, the radius of the innermost electron orbit should be about ten thousand times the radius of the nucleus.

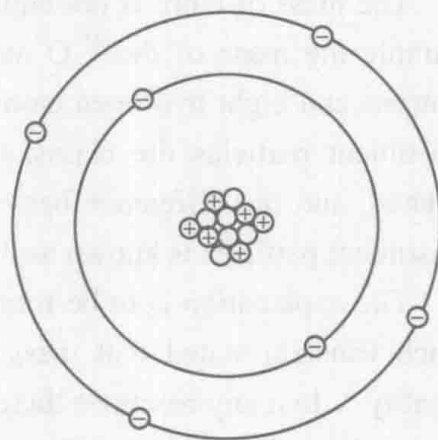


Fig.1.1 atomic structure of carbon 12

1.2 Isotopes

Atoms having the same atomic number, Z , but different numbers of neutrons N are called isotopes of the element identified by Z , and all elements have a number of isotopes, in some cases twenty or more. The naturally occurring elements each have one or more stable isotopes which exist naturally, and other isotopes which are unstable or radioactive and can be produced by artificial means. Different isotopes of an element behave identically as far as their chemistry is concerned, which is not surprising as chemical bonds exist between electrons. Isotopes differ from one another physically in that the masses and other characteristics of their nuclei are different, which is to be expected as it is in the nuclei that the difference between two isotopes lies¹.

The complete identification of an isotope is made by giving its chemical symbol, the atomic number Z as a subscript and the mass number A as a superscript. For example, the symbol ${}^{16}_8\text{O}$ identifies the isotope of oxygen that has eight protons and eight neutrons in its nucleus. The isotope ${}^{17}_8\text{O}$ has eight protons and nine neutrons in its nucleus. Naturally occurring oxygen consists of a mixture of three isotopes, ${}^{16}_8\text{O}$, ${}^{17}_8\text{O}$ and ${}^{18}_8\text{O}$. There are also three radioactive isotopes of oxygen which do not occur naturally. The subscript Z is in fact unnecessary as the name oxygen identifies that element with eight protons in its nucleus and the symbols may be written as ${}^{16}\text{O}$, ${}^{17}\text{O}$ and ${}^{18}\text{O}$.

Hydrogen is an important element in nuclear engineering. Naturally occurring hydrogen consists of two isotopes, 99.985 percent of the isotope ${}^1\text{H}$ and 0.015 percent of the isotope ${}^2\text{H}$ called heavy hydrogen or deuterium. There is a third isotope ${}^3\text{H}$ called tritium which is radioactive. This is the only case in which the different isotopes of an element have different names. Usually they are identified by their mass numbers.

1.3 Mass Defect

The mass of atom is not equal to the sum of the masses of its constituent particles. For example the mass of the ${}^{16}\text{O}$ atom is obviously less than the sum of the masses of eight neutrons and eight hydrogen atoms. Somewhere in the process of building the atom with its constituent particles the classical principle of conservation of mass appears to have been violated, and the difference between the mass of an atom and the sum of the masses of its constituent particles is known as the mass defect.

The explanation is to be found in the principle of the equivalence of mass and energy in which Einstein stated that mass and energy are different forms of the same fundamental quantity². In many reactions there is an interchange of mass and energy so that, particularly on an atomic scale, the laws of conservation of mass and conservation of energy are not valid when applied separately to a reaction, and must be replaced by the law of conservation of mass plus energy. In any reaction in which mass changes, a decrease of mass is accompanied by the release of energy, and an increase of mass corresponds to the absorption of energy.

The equivalence between mass and energy is expressed by the famous equation:

$$E = mc^2$$

where c —the speed of light, $2.998 \times 10^8 \text{ m/s}$.

1.4 Binding Energy

The force of electrostatic repulsion between like charges, which varies inversely as the square of their separation, would be expected to be so large that nuclei could not be formed. The fact that they do exist is evidence that there is an even larger force of attraction. This nuclear force acts only when the nucleons are very close to each other and binds them into

a compact stable structure. Associated with the net force is a potential energy of binding. To disrupt a nucleus and separate it into its component nucleons, energy must be supplied from the outside. Recalling Einstein's relation between mass and energy, this is the same as saying that mass must be supplied to the nucleus. A given nucleus is lighter than the sum of its separate nucleons, the difference being the binding mass-energy. Let the mass of an atom including nucleus and external electrons be M , and let m_n and m_H be the masses of the neutron and the proton plus matching electron. Then the binding energy is $B = \text{total mass of separate particles} - \text{mass of the atom}$ or

$$B = Nm_n + Zm_H - M$$

(Neglected in this relation is a small energy of atomic or chemical binding.) Let us calculate B for tritium, the heaviest hydrogen atom. Fig.1.2 shows the dissociation that would take place if a sufficient energy were provided. Now $Z=1$, $N=2$, $m_n=1.008\,665$, $m_H=1.007\,825$, and $M=3.016\,049$, Then

$$B = 2(1.008\,665) + 1(1.007\,825) - 3.016\,049 = 0.009\,106(\text{u})$$

Converting by use of the relation $1\text{u}=931\text{MeV}$, the binding energy is $B=8.48\text{MeV}$.

Calculations such as these are required for several purposes to compare the stability of one nucleus with that of another, to find the energy release in a nuclear reaction, and to predict the possibility of the fission of a nucleus.

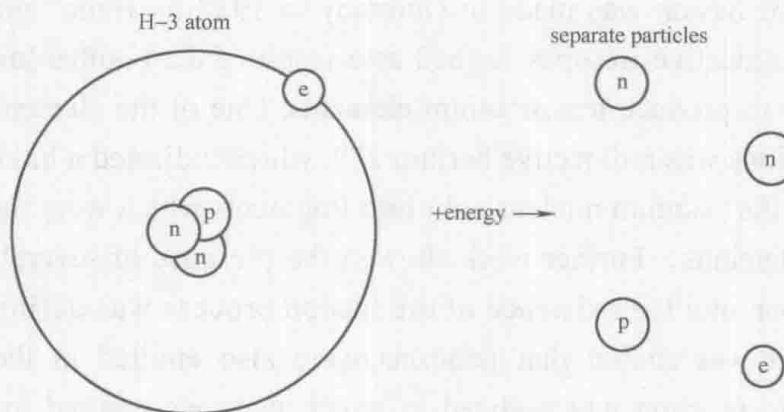


Fig.1.2 dissociation of tritium

We can speak of the binding energy associated with one particle such as a neutron. Suppose that M_1 is the mass of an atom and M_2 is its mass after absorbing a neutron. The binding energy of the neutron of mass m_n is then

$$B_n = M_1 + m_n - M_2$$

1.5 Energy Levels

Normally atomic nuclei exist in an equilibrium or stable condition known as their ground state of energy. However, as a result of nuclear reactions (which might be caused by the bombardment of atoms by protons, neutrons or other light particles), nuclei can be produced in

an excited or unstable condition in which one or a number of nucleons are raised to an excited state. The excited states or levels in a nucleus are similar to the excited state of atoms. In the case of the latter, excitation results in an electron jumping from its normal orbit to another orbit further from the nucleus, and an atom may have a number of discrete excited states corresponding to an electron having made one or more such jumps. In the nucleus the situation is more complicated because excitation can result in several nucleons being raised to excited levels simultaneously, and some nuclei can have a very large number of closely spaced excited levels. In general light nuclei have more widely spaced excited levels, and in all nuclei the spacing of the levels decreases as the excitation energy increases.

Most excited nuclei exist in this state for only a very short time, a typical average lifetime being about 10^{-14} seconds, and they decay, or become de-excited, by the emission of high energy electromagnetic radiation known as gamma radiation, or particles such as neutrons, or both. In most reactions of interest to nuclear engineers involving the formation and decay of excited nuclei, the lifetime of the excited nucleus is so short that the process of formation and decay can be regarded as instantaneous.

1.6 Fission

The discovery of fission was made in Germany in 1938 by Hahn³ and Strassmann who were studying the radioactive isotopes formed as a result of the bombardment of uranium by neutrons in an effort to produce transuranium elements. One of the elements identified in the products of the reactions was radioactive barium 139, which indicated a hitherto unknown type of reaction in which the uranium nucleus split into fragments which were themselves nuclei of intermediate mass elements. Further work showed the presence of several other elements of medium mass number, and the existence of the fission process was definitely established. Shortly afterwards it was shown that neutrons were also emitted in the process and the possibility of a chain reaction was realized in which neutrons emitted in one fission event might be able to cause further fission, thus establishing a continuous reaction.

The isotope of uranium that is principally responsible for fission is ^{235}U , which is present in naturally occurring uranium to the extent of 0.715 percent. In this isotope fission can be caused by neutrons of any energy, low energy neutrons being the most effective. Fission in ^{238}U , which comprises 99.285 percent of natural uranium, can only be caused by neutrons of energy greater than 1 MeV.

There are three other isotopes of importance which can undergo fission. ^{232}Th , the only naturally occurring isotope of that element, is fissionable with neutrons with energy greater than about 1.4 MeV, and two isotopes, ^{233}U and ^{239}Pu , which do not occur naturally but can be produced artificially by nuclear reactions, undergo fission with neutrons of all energy, low energy neutrons being again the most effective. It is customary to refer to the five isotopes mentioned above (and any other isotopes which undergo fission with neutrons of energy less

than about 10MeV) as fissionable, and to reserve the term fissile for the three isotopes ^{233}U , ^{235}U and ^{239}Pu which undergo fission with low energy neutrons.

The characteristics of fission will be described by considering ^{235}U , however the fission of the other four isotopes is essentially the same in all respects. The first stage of the reaction is the absorption of a neutron in ^{235}U to form ^{236}U at an excited state. In some cases the ^{236}U goes to its ground state of energy by the emission of gamma radiation, an example of an(n, γ) reaction, however in the majority of cases the ^{236}U nucleus splits as described above. The products of fission are two fission fragments whose mass numbers vary between 70 and 160, a number of neutrons varying between none and five, beta particles, gamma radiation, neutrinos and energy. These products are shown in Fig.1.3.

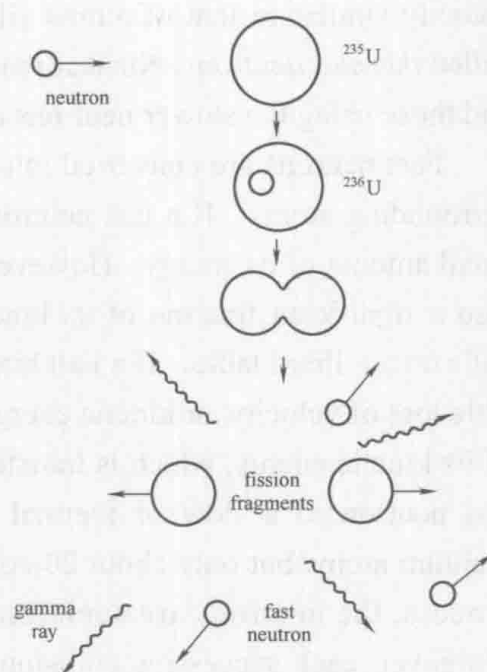


Fig.1.3 The fission process

The release of energy in the fission process may be illustrated by considering the fission of a ^{235}U atom, which splits up into barium and krypton atoms and releases three more neutrons:



If we could weigh components in this reaction we would find that those on the right-hand side of the equation weighed 0.091% less than those on the left-hand side. Thus, during the reaction, approximately 0.1% of the original mass is converted into energy. This energy appears as kinetic energy of the fission products and neutrons, which then collide with surrounding atoms and increase their thermal vibration that is, release heat. For each kilogram of ^{235}U totally fissioned by the above reaction, 80 million (8×10^{13}) joules are released. This is equivalent to the energy available from 3000 tons of coal.

^{235}U is described as a fissile isotope; unfortunately, naturally occurring uranium consists mainly (99.3%) of a non-fissile isotope, ^{238}U . Thus, only a small part of natural uranium can be burned in the fission process to produce energy. The proportion of ^{235}U in natural uranium is 0.71% by weight, and thus 1kg of natural uranium is equivalent in energy potential to about 20 tons of coal. However the energy potential of uranium can be increased about 100-fold by conversion of the non-fissile ^{238}U into another fissile material, namely, plutonium-239 (^{239}Pu). We shall return to this below.

The three neutrons emitted in the above fission reaction have an initial velocity of typically 20 000km/s (about 6% of the velocity of light). Although these fast neutrons can interact with other atoms of ^{235}U , their chance of doing so can be increased by about 1000-fold if their velocity can be reduced, say, to 2km/s.⁴ These slower-moving neutrons would have a

velocity similar to that of atoms vibrating due to thermal motions, and hence they are often called *thermal neutrons*. Nuclear reactors using the fast neutrons are often termed *fast reactors*, and those using the slower neutrons are termed *thermal reactors*.

Fast neutrons are converted into thermal neutrons as a result of a series of collisions with surrounding atoms. If a fast neutron hits a large atom, it tends to bounce off and lose only a small amount of its energy. However, if it hits a small atom such as hydrogen or carbon, it will lose a significant fraction of its kinetic energy. (An analogy may be made to the motion of balls on a billiard table. If a ball hits the massive cushion of the table, it bounces off with very little loss of velocity, or kinetic energy. If it hits a stationary ball, it may lose a large proportion of its kinetic energy, which is transferred to the other ball in the collision.) Thus, to convert a fast neutron to a slow or thermal neutron requires about 2000 successive collisions with uranium atoms but only about 20 collisions with the lightest atom, hydrogen. In the collision process, the neutrons are sometimes absorbed without leading to a subsequent fission. Moreover, each successive collision may lead either to a fission reaction or to the neutron's combining with the atom with which it is colliding to make another isotope. Thus, there is an advantage in surrounding the uranium with lighter material that can lead to the conversion of fast neutrons to thermal neutrons, which can then pass back into the uranium. This process is known as moderation, and the light material used is termed a moderator. Moderators used in thermal reactors have included hydrogen (in the form of its oxide, water), the hydrogen isotope deuterium (also in the form of its oxide, heavy water) and carbon (usually in the form of graphite). The best moderator is heavy water, which absorbs neutrons only weakly. However, heavy water is an expensive material and it is often preferable to use ordinary (light) water even though it absorbs neutrons much more strongly.

Because ^{238}U absorbs neutrons, it is not possible to produce a self-sustaining chain reaction by simply assembling a large enough mass of natural uranium, which is 99.3% of ^{238}U . However, if pieces of natural uranium are distributed within heavy water or graphite, the neutrons produced in the fission reaction are converted to thermal neutrons (which, as mentioned above, are 1000 times more effective than fast neutrons in continuing the chain reaction), and a self-sustaining chain reaction is possible. This idea was first demonstrated by Enrico Fermi at Stagg Field, Chicago, on December 2, 1942; Fermi employed pieces of uranium distributed in a "pile" of graphite. Light water cannot be used to sustain a chain reaction with natural uranium because of the high absorption of neutrons by hydrogen. However nuclear reactors may be constructed with light water as a kind of moderator provided the concentration of ^{235}U is increased from 0.71% to about 3%.

As we shall see later, various generic types of nuclear reactors have arisen from the various possible combinations of fuel and moderator. These can be classified as follows.

(1) Heavy water-moderated, heavy water-cooled reactors. These are the basis of the Canadian line of development and are called CANDU reactors.

(2) Graphite-moderated, gas-cooled natural uranium reactors. These are the basis of the

British Magnox reactors.

(3) Light water-moderated, light water-cooled reactors with fuel enriched in ^{235}U . These are the basis of the U.S. boiling-water reactor (BWR) and pressurized-water reactor (PWR) development.

The further development of the British system, the advanced gas-cooled reactor (AGR), uses graphite as a moderator and a somewhat enriched fuel to compensate for the fact that the fuel is contained in stainless steel, which absorbs a significant fraction of the neutrons.

We shall describe and discuss all the above reactors, and in particular their cooling problems, in the following chapters. However, before doing so, it may be interesting to glance back to prehistoric times. The light water-cooled and moderated reactors have been around far longer than one might imagine. In fact, the invention of Enrico Fermi was preempted by nature approximately 2 billion years earlier. In 1972, evidence was found of the dormant remains of a natural fission reactor located at Oklo, in the West African Republic of Gabon. This natural reactor operated for a period of hundreds of thousands of years. Its existence was discovered by an intriguing piece of detective work by French nuclear scientists.

In May 1972, H. Bouzigues obtained a curious result during a routine analysis of standard samples of uranium ore from Gabon. He found that they contained about 0.4% less ^{235}U than expected. This was not due to an error in his analysis or to a natural variation. On this planet, at any particular time, the ratio of ^{235}U to ^{238}U is fixed; some other explanation had to be found for the discrepancy. A careful investigation carried out by the French Atomic Energy Agency traced the abnormal ore to one particular location in Oklo. It was concluded that the deficiency in ^{235}U could be explained only by the occurrence of a natural fission reaction at the site. At the time this natural reactor was operating, the ore was buried deep underground and natural groundwater served as a moderator and to some extent as a coolant. Such a reactor would not be possible with the present-day concentration of ^{235}U in naturally occurring uranium, as we explained above. However, it should be remembered that the half-life of ^{235}U is about 700 million years and that of ^{238}U is about 4.5 billion years. Thus, in prehistoric times, the concentration of ^{235}U was much higher than it is today. When the earth was formed some 4.6 billion years ago, the concentration of ^{235}U in natural uranium was about 25%, and it had decreased to about 3% at the time when the Oklo reactor was operating.

It is thought that the natural reactor at Oklo operated under considerable pressure and temperature and that the rate of reaction was controlled by variations of the water (moderator) density. Cooling was provided mainly by conduction, with some limited circulation by permeation. The power level is estimated to have been somewhat less than 100 kW and the total energy released over the period of operation to have been about 4.7×10^{17} joules (15 000 MW • year), representing the fission of about 6 metric tons of ^{235}U . This amount of energy is about that released in a modern pressurized-water reactor in 4 years.

It is possible that a combination of local circumstances may have led to other naturally occurring reactors. Though the search continues, none has been located so far. Such naturally

occurring reactors have been impossible for the past 2 billion years because the ^{235}U concentration has been below the required 3%. A detailed review of the Oklo reactor phenomenon is given by Cowan (1976). It is interesting to note that 2 tons of ^{239}Pu would have been produced at the Oklo natural reactor though the amount that remains is infinitesimal because of the comparatively short half-life (25 000 years) of ^{239}Pu . Thus, it cannot be claimed that plutonium is an entirely human creation.

Self-test

1. ____ is composed of protons and neutrons, and constitutes practically all the mass of the atom.
A. A nucleon B. A nucleus C. The nuclei D. An atom
2. The protons and neutrons are called _____.
A. nuclei B. isotopes C. nucleons D. atoms
3. In the nuclear reactions mass and energy is _____.
A. interchanged B. built C. made D. replaced
4. The atoms consist of numbers of _____.
A. three fundamental particles B. protons
C. γ -rays D. nuclei
5. Atoms having the same atomic number Z , but different numbers of neutrons N are called ____ of the element.
A. nuclei B. isotopes C. units D. masses
6. During the nuclear reaction ____ of the particles is the same before and after the reaction.
A. the energy B. the mass C. the momentum D. the protons
7. ____ are emitted during fission and the possibility of a chain reaction is realized.
A. Elements B. Fragments C. γ -rays D. Neutrons
8. The excited nuclei become de-excited by emission of _____.
A. γ -rays or particles B. X-rays
C. energy D. deuterons
9. For heavier elements it is necessary to have an excess of ____ over ____ for stability.
A. protons/neutrons B. electrons/ γ -rays
C. neutrons/nucleons D. neutrons/protons
10. The fission fragments ____ most of the mass-energy released.
A. are B. bear C. carry out D. take
11. The binding force acts only when the nucleons are very _____.
A. large B. close to each other C. similar D. excited
12. Fission in ^{238}U _____.
A. is easy

B. can be caused by neutrons of 0.1MeV

C. can be caused by neutrons of 1.5MeV

D. is never realized

关键字解释

Avogadro's number: The number of atoms in 12g of the pure isotope ^{12}C ; i.e. the reciprocal of the atomic mass unit in grams. It is also by definition the number of molecules (or atoms, ions, electrons) in a molar of any substance and has the value of $6.022\ 52 \times 10^{23}/\text{mol}$.

阿伏伽德罗常数: 12 克纯同位素 ^{12}C 中的原子的数目; 也就是以克为单位的原子质量的倒数。按定义, 它也是 1 摩尔任何物质中分子 (或原子、离子、电子) 的数目, 其数值为 $6.022\ 52 \times 10^{23}/\text{摩尔}$ 。

Binding energy: All nuclei have rest masses less than the total rest mass of their constituent protons and neutrons. The mass difference m is the mass defect. This arises because all nucleons bound to the nuclei must have negative energy (potential well). So if free nucleons are combined to form a nucleus, the total energy of the system must decrease by an amount B , the binding energy of the nucleus.

结合能: 所有原子核的静质量都比它们的组分质子和中子的总静质量小, 这个质量差值 m 称为质量亏损。这是由于束缚在原子核中的所有核子必定具有负能 (势井), 于是, 如果自由核子要结合成核, 系统的总能量就必须减少 B , 这就是原子核的结合能。

CANDU: Canadian Deuterium and Uranium reactor.

加拿大重水铀反应堆。

Chemical bond: The linkage between two atoms; as a rule, it consists of a pair of shared electrons.

化学键: 两原子之间的键合; 通常是由一对共享电子组成。

Coulomb force: Electrostatic attraction or repulsion between two charged particles.

库仑力: 两个带电粒子之间的静电吸引或排斥的力。

Electromagnetic radiation: The emission and propagation of electromagnetic energy from a source including long waves, heat radiations, light, X-rays and γ -rays.

电磁辐射: 源自包括长波、热辐射、光波、X 射线和 γ 射线的电磁能的发射和传播。

Electron: A negatively charged particle which is present in every atom; beta rays consist of free electrons; the mass of an electron at rest is $9.107 \times 10^{-28}\text{g}$ and its charge is 4.774×10^{-10} electrostatic unit.

电子: 见于每个原子内的一种负电荷粒子; β 射线由自由电子组成; 静止的电子质量为 9.107×10^{-28} 克, 其电荷为 4.774×10^{-10} 静电单位。

Excited nuclei: The nuclei raised to an excited state with an excess of energy over its ground state. Nuclear reactions frequently leave the product nuclei in an excited state.

激发态核: 带有超过基态能量的核, 核反应往往造成核处于激发态。

Fission: The spontaneous or induced disintegration of a heavy atomic nucleus into two or

more lighter fragments. The energy released in the process is referred to as nuclear energy.

裂变: 自发的或某种原因引起的重原子核分裂成两个或多个较轻碎片的分裂过程。这个过程释放的能量称为核能。

Isotopes: Atoms of the same element which have the same atomic numbers and chemical properties, but different atomic masses; for example, uranium -235 and uranium -238.

同位素: 同一元素中原子序数和化学性质相同, 但原子质量不同的原子, 如铀-235 和铀-238。

Mass defect: The difference between the mass of an atom and the sum of the masses of its constituent particles.

质量亏损: 原子的质量与组成这个原子的各粒子质量之和的差值。

Neutron: A neutral particle found in the nucleus of an atom, which has about the same mass as a proton.

中子: 存在于原子核中的中性粒子, 其质量与质子的质量近乎相等。

Nucleon: A constituent of the nucleus of the atom, proton or neutron.

核子: 原子核的一个组成部分, 质子或中子。

Nucleus: Composed of protons (positively charged) and neutrons (no charge), and constitutes practically all the mass of the atom. Its charge equals the atomic number; its diameter is from 10^{-15} to 10^{-14} m.

原子核: 由质子(带正电)和中子(不带电)组成, 它几乎构成了原子的所有质量。它的电荷等于原子序数; 直径从 10^{-15} 到 10^{-14} 米。

Proton: A positively charged particle in the nucleus of an atom; its charge is equal but opposite to that of an electron; its mass is 1.672×10^{-24} g.

质子: 原子核内带正电荷的粒子, 其电荷与电子所带电荷数量相等, 极性相反, 其质量为 1.672×10^{-24} 克。

词 汇

abbreviate [ə'brɪ:vieɪt] v. 缩写

analogous [ə'næləgəs] = Similar a. 类似的

barium ['beəriəm] n. 钡

bismuth ['bɪzməθ] n. 铋

bombardment [bɒm'ba:dmənt] n. 轰击

boron ['bɔ:rən] n. 硼

bounce [baʊns] v. & n. 弹, 反冲

commissariat [kəmi'sɛəriət] n. 给养

configuration [kən'figju'reɪʃən] n. 结构, 配置, 形状, 方位

constituent [kən'stɪtjuənt] a. 组成的

coulomb ['ku:lɒm] n. 库(仑)

curious ['kjʊəriəs] a. 好奇的, 奇妙的

- decay [di'kei] *v. & n.* 衰变
- deficiency [di'fiʃənsi] *n.* 缺乏
- deuteron [dju'ti:riən] *n.* 氘核, 重氢核
- deuterium [dju'tiəriəm] *n.* 氘, 重氢
- discrepancy [dis'krepənsi] *n.* 不同, 差异, 亏损
- discrete [dis'kri:t] *a.* 不连续的, 分立的
- disrupt [dis'rʌpt] *v.* 使分离
- dormant ['dɔ:mənt] *a.* 固定的
- dumbbell ['dʌmbel] *n.* 哑铃
- electrostatic [i,lektəu'stætik] *a.* 静电的
- emission [i'miʃən] *n.* 发射
- equivalence [i'kwivələns] *n.* 等效, 等价
- fissile ['fisail] *a.* 易裂变的
- fragment ['frægmənt] *n.* 碎片
- generic [dʒi'nerik] *a.* 同属的, 通用的
- glance [glɑ:ns] *vi. & n.* 一瞥
- helium ['hi:ljəm] *n.* 氦
- hitherto [,hiðə'tu:] *ad.* 至今
- infinitesimal [infini'tesiməl] *a.* 无限小的
- instantaneous [instən'teinjəs] *a.* 瞬时的
- intriguing [intrigɪŋ] *vt. & n.* 引起……兴趣; 阴谋
- lithium ['liθiəm] *n.* 锂
- macroscopic [mækəu'skɒpik] *a.* 宏观的
- mega ['megə] *n.* 兆, 百万
- moderation [,mɒdə'reiʃən] *n.* 慢化, 减速
- mol [məul] *n.* 摩尔
- momentum [məu'mentəm] *n.* 动量
- particle ['pɑ:tɪkl] *n.* 粒子, 微粒
- permeation [,pə:mi'eɪʃən] *n.* 渗透
- plutonium [plu:'təuniəm] *n.* 钚
- preempt [pri'empt] *v.* 先取(占), 优先
- prehistoric ['pri:his'tɒrik] *a.* 很久以前的, 史前的
- radiation [,reidi'eɪʃən] *n.* 放射性, 发射, 射线
- repulsion [ri'pʌlʃən] *n.* 排斥
- SI=System International 国际单位
- sodium['səudjəm] *n.* 钠
- subscript ['sʌbskript] *n.* 下脚标
- superscript ['sju:pəskript] *n.* 上脚标
- thorium ['θɔ:riəm] *n.* 钍

transuranium [trænsjue'reinjəm] *n.* 铀后元素

unionized [ʌn'aɪənaɪzd] *a.* 未(非)电离的

valid ['vælid] *a.* 有效的, 正确的

violate ['vaɪəleɪt] *vt.* 违反

as far as 就……而论

more or less 有点

tend to 趋于

课 文 注 释

1. Isotopes differ from...isotopes lies.

句中副词 *physically* 是修饰不及物动词 *differ* 的, 说明同位素之间的物理特性不同; *which* 引导的是非限定性定语从句, 修饰它前面的句子, 而不是修饰某个单词; *as* 在这里是因为的意思; *it is in the nuclei that...* 是强调句, 意思是“两个同位素的差别是在于原子核内”。整句译为: 同位素之间的物理特性不同, 由于两个同位素的差别是在原子核内, 可以预计到它们的原子核的质量和特性都有差别。

2. The explanation is to be found in the principle of the equivalence of mass and energy in which Einstein stated that...

Einstein, 爱因斯坦(1879—1955), 物理学家。生于德国犹太人家庭, 后迁居美国。创立了狭义相对论和广义相对论。句中 *in which* 引导定语从句, 说明 *principle*; *that* 引导的是 *stated* 的宾语从句。整句译为: 在质量与能量的等效原理中可找到解释, 在这一原理中, 爱因斯坦证明了质量和能量是同一基本量的两个不同形式。

3. Hahn

哈恩(1879—1968), 德国核化学家, 最先发现了中子轰击铀核会产生裂变。1944 年获诺贝尔化学奖。

4. Although these fast...so can be reduced, say, to be 2 km/s.

句中 *so* 是代词, 表示前面所提到的与其他原子核反应; *fold* 是倍数; *say* 在这里是例如的意思。全句译为: 虽然这些快中子能够与其他的 ^{235}U 核裂变, 但是如果它们的速度被减小, 例如减至 2km/s, 它们反应的几率会增加大约 1000 倍。

Unit 2 Radiation

2.1 Radioactivity

Radioactivity is a commonplace natural phenomenon, the discovery of which predates the discovery of nuclear fission by more than 40 years. Radioactivity is harmful to living tissue and yet humans and other animals have not evolved sensor organs with which to detect the danger¹. It is the invisible nature of radiation that contributes much to public anxiety concerning nuclear power. Scientific instruments have, however, been developed that are sensitive to all types of radiation and these are widely used to ensure that workers and the public are not put at risk by nuclear processes.

The property of radiation that is potentially harmful is its ability to ionize atoms within our bodies; that is, to knock electrons out of atoms in our cells altering their chemical properties. In very large doses such radiation can cause tissue death. At lower levels the concern is that if the ionization occurs during the process of cell division then a mutation to the genetic programming of the cell might occur. In most cases this would also lead to cell death. but there is a risk that harmful cell mutation can survive within the body and possibly cause various types of cancer. Cancer is a particularly horrible disease and this is another reason for public concern about nuclear power.

Radioactivity is a spontaneous, natural and random process that can occur in three basic forms known as alpha, beta and gamma decay. Alpha radiation is the emission of a clump of protons and neutrons from the nucleus equivalent to the nucleus of a helium atom (two protons and two neutrons). As a consequence of its relatively large mass, an alpha particle can carry a significant amount of energy, sufficient to do serious damage to any living tissue it encounters. However, also as a consequence of its bulk, such radiation is easily blocked, such as by just a few centimeters of air or a piece of paper. The skin, which is relatively insensitive to radiation, also acts as a barrier to alpha radiation, so external sources of alpha radiation are not especially dangerous. The real risk from alpha radiation would occur if alpha emitting material were to be ingested into the lungs or the digestive system.

Beta radiation consists of very fast moving electrons. Its potential for biological harm is lower than with alpha radiation, but its penetrating power is somewhat greater. Beta radiation can be screened by a sheet of perspex or a metal foil. Without such screening, a substantial external source of beta radiation would be serious cause for concern.

Gamma radiation is a very high-energy form of light. Unlike the other two forms of radiation it is not electrically charged and it has no mass. It is basically a very high-energy X-ray. While fundamentally not as biologically damaging as either alpha or beta radiation, gamma radiation is a particular concern as it is extremely penetrating. Several centimetres of

lead or several metres of concrete are required to fully shield gamma radiation.

The idea that we are vulnerable to differing forms of radioactivity (requiring, in some cases, substantial shielding), that we cannot detect it without artificial devices and that it has the potential to damage our genes is somewhat unnerving². The reality, however, is that mankind has always lived with radiation and radioactivity. Radiation is all around us. It comes from outer space. Radioactivity is in the rocks beneath our homes, and it is in our drinking water. It always has been and always will be. X-rays are similar to gamma rays and we routinely volunteer to expose ourselves to intense bursts of this kind of radiation for the benefits it gives us in medicine and dentistry. These doses are enormously larger than the radiation exposure one should ever get from living near a nuclear power plant.

The key to radioactivity is probability. Even a small sample of radioactive material contains a vast number of atoms. More than we can even imagine. For a radioactive material each atom has a probability every second of decaying releasing a radioactive particle. For natural materials these probabilities are tiny, but because of the large numbers of atoms involved the effect is noticeable. For radioactivity all atoms of a given isotope are equivalent and each has the same small chance of decaying. This means that the amount of radioactivity in a sample is proportional to the number of atoms of that given isotope in the sample. Once an alpha or beta radioactive decay occurs the atom will change from one isotope to another. Let us assume that the isotope produced is not itself radioactive. This means that each radioactive decay reduces the number of atoms of the radioactive isotope remaining to decay. In mathematical terms such a process is described as follows.

If N is the number of atoms of the radioactive isotope in the sample and the Greek letter λ defines the random probability of a radioactive decay then

$$\frac{dN}{dt} = -\lambda N$$

This implies that if the sample has initially N_0 atoms of the radioactive isotope, then

$$N = N_0 e^{-\lambda t}$$

This in turn makes clear that the time for exactly one half of the initial number of radioactive atoms to decay is a constant, whatever the starting number. This time is known as the half-life $T_{1/2}$:

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

Just as the probability of radioactive decay is fixed for a given radioactive isotope, then so is the half-life.

There are numerous units in circulation associated with radioactivity and its effects. In this book we follow the units specified in the primary sources quoted. In order to cross compare, it is worth noting that one becquerel (1 Bq) is equivalent to one disintegration per second, and that 1 curie (1 Ci) equals 3.7×10^{10} Bq.

2.2 Excitation and Ionization by Electrons

These processes occur in the familiar fluorescent light bulb, or in a vacuum tube used in electrical devices, in an X-ray machine, or in matter exposed to beta particles. If an electron that enters a material has a very low energy, it will merely migrate without affecting the molecules significantly. If its energy is larger, it may impart energy to atomic electrons as described by the Bohr theory, causing excitation of electrons to higher energy states or producing ionization, with subsequent emission of light. When electrons of inner orbits in heavy elements are displaced, the resultant high energy radiation is classed as X-rays. These rays, which are so useful for internal examination of the human body, are produced by accelerating electrons in a vacuum chamber to energies in the kilovolt range and allowing them to strike a heavy element target. In addition to the X-rays due to transitions in the electronic orbits, a similar radiation called bremsstrahlung (German: braking radiation) is produced. It arises from the deflection and resulting acceleration of electrons as they encounter nuclei or atomic electrons.

Beta particles as electrons from nuclear reactions have energies in the range of 0.01~1 MeV, and thus are capable of producing large amounts of ionization as they penetrate a substance. As a rough rule of thumb, about 32 eV of energy is required to produce one ion pair. The beta particles lose energy with each event, and eventually are stopped. For electrons of 1 MeV energy, the range, as the typical distance of penetration, is no more than a few millimeters in liquids and solids or a few meters in air.

2.3 Heavy Charged Particle Slowing by Atoms

Charged particles such as protons, alpha particles, or ions such as the fragments of fission are classed as heavy particles, being much more massive than the electron. For the same particle energy they have far less speed than an electron, but they are less readily deflected in their motion than electrons because of their inertia. The mechanism by which heavy ions slow down in matter is primarily electrostatic interaction with atomic electrons. As the positively charged projectile approaches and passes, with the attraction to electrons varying with distance of separation as $1/r^2$ the electron is displaced and gains energy, while the heavy particle loses energy³. Fig. 2.1 shows conditions before and after the collision schematically. It is found that the kinetic energy lost in one collision is proportional to the square of Z , the number of external electrons in the target atom, and inversely proportional to the kinetic energy of the projectile. A great deal of ionization is produced by the heavy ion as it moves through matter. Although the projectile of energy in the million-electron-volt range loses only a small fraction of its energy in one collision, the amount of energy imparted to the electron can be large compared with its binding to the

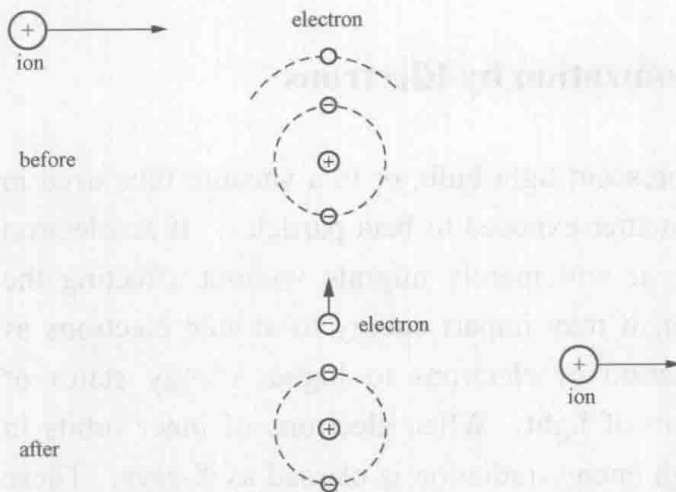


Fig.2.1 interaction of heavy ion with electron

atom or molecule, and the electron is completely removed. As the result of these interactions, the energy of the heavy ion is reduced and it eventually is stopped in a range that is very much shorter than that for electrons. A 2MeV alpha particle has a range of about 1cm in air; it can be stopped by a sheet of paper or the outer layer of skin of the body. Because of these short ranges, there is little difficulty in providing protective shielding against alpha particles.

2.4 Heavy Charged Particle Scattering by Nuclei

When a high-speed charged ion such as an alpha particle encounters a very heavy charged nucleus, the mutual repulsion of the two particles causes the projectile to move on a hyperbolic path, as in Fig.2.2. Such a collision can take place in an ionized gas or in a solid if the incoming particle avoids interaction with the external electrons of the atom. The projectile is scattered through an angle that depends on the detailed nature of the collision, i.e., the initial energy and direction of motion of the incoming ion relative to the target nucleus, and the magnitude of electric charge of the interacting particles. Unless the energy of the bombarding particle is very high and it comes within the short range of the nuclear force, there is a small chance that it can enter the nucleus and cause a nuclear reaction.

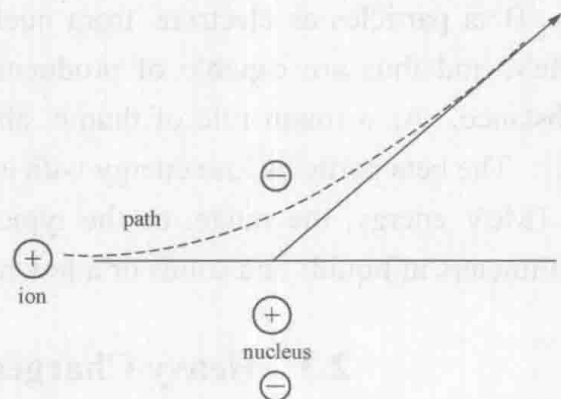


Fig.2.2 interaction of heavy ion with nucleus

2.5 Gamma Ray Interactions with Matter

We now turn to a group of three related processes involving gamma ray photons produced by nuclear reactions. These have energies as high as a few MeV. The interactions include simple scattering of the photon, ionization by it, and a special nuclear reaction known as pair production.

(1) Photon-electron Scattering

One of the easiest processes to visualize is the interaction of a photon of energy $E=h\nu$ and an electron of rest mass m_0 . Although the electrons in a target atom can be regarded as moving

and bound to their nucleus, the energies involved are very small (eV) in comparison with those of typical gamma rays (keV or MeV). Thus the electrons may be viewed as free stationary particles. The collision may be treated by the physical principles of energy and momentum conservation. As sketched in Fig.2.3, the photon is deflected in its direction and loses energy, becoming a photon of new energy $E'=h\nu'$. The electron gains energy and moves away with high speed v and total mass-energy mc^2 , leaving the atom ionized. In this Compton effect, named after its discoverer, one finds that the greatest photon energy loss occurs when it is scattered backward (180°) from the original direction. Then, if E is much larger than the rest energy of the electron $E_0 = m_0c^2 = 0.51\text{MeV}$, it is found that the final photon energy E' is equal to $E_0/2$. On the other hand, if E is much smaller than E_0 , the fractional energy loss of the photon is $2E/E_0$. The derivation of the photon energy loss in general is complicated by the fact that the special theory of relativity must be applied.

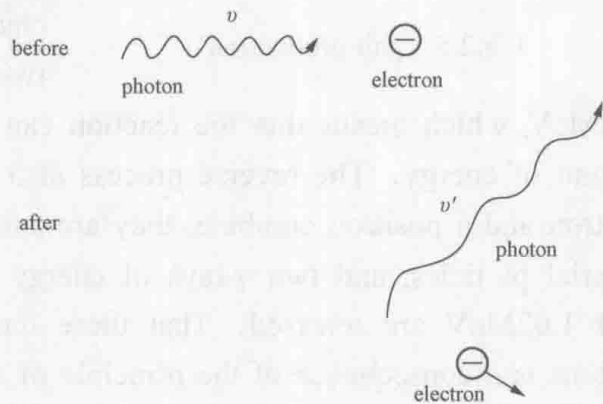


Fig.2.3 photon-electron scattering (Compton effect)

The probability of Compton scattering is expressed by a cross section, which is smaller for larger gamma energies for the element lead, a common material for shielding against X-rays or gamma rays. We can deduce that the chance of collision increases with each successive loss of energy by the photon, and eventually the photon disappears.

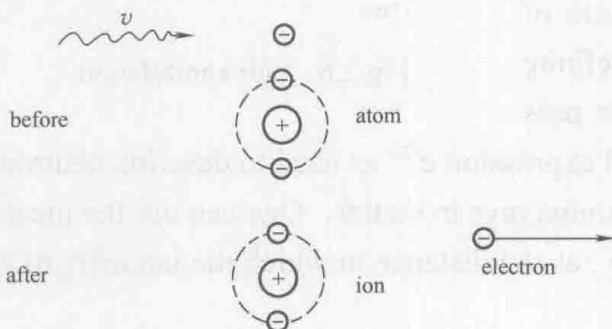


Fig.2.4 photoelectric effect

(2) Photoelectric Effect

This process is in competition with scattering. An incident photon of high enough energy dislodges an electron from the atom, leaving a positively charged ion⁴. In so doing, the photon is absorbed and thus lost, see Fig.2.4.

The above two processes are usually treated separately even though both result in ionization. In the Compton effect, a photon of lower energy survives; but in the photoelectric effect, the photon is eliminated. In each case, the electron released may have enough energy to excite or ionize other atoms by the mechanism described earlier. Also, the ejection of the electron is followed by light emission or X-rays production, depending on whether an outer shell or inner shell is involved. The secondary particles and rays may be more important in the long run than the primary particle.

(3) Electron-Positron Pair Production

The third process to be considered is one in which the photon is converted into

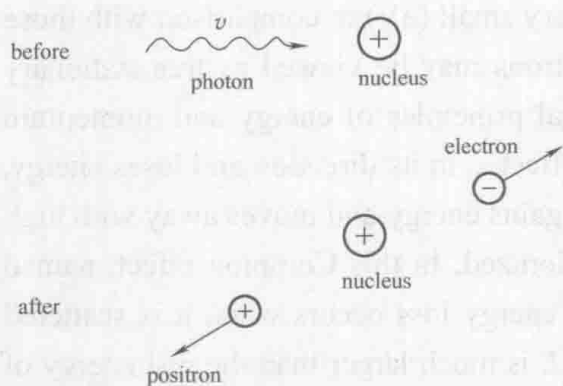


Fig.2.5 pair production

matter. This is entirely in accord with Einstein's theory of the equivalence of mass and energy. In the presence of a nucleus, as sketched in Fig.2.5, a gamma ray photon disappears and two particles appear—an electron and a positron. Since these are of equal charge but of opposite sign, there is no net charge after the reaction, just as before, the gamma ray having zero charge. The law of conservation of charge is thus met. The total new mass produced is twice the mass-energy of the electron, $2(0.51)=1.02\text{MeV}$, which means that the reaction can occur only if the gamma ray has at least this amount of energy. The reverse process also takes place. As sketched in Fig.2.6, when an electron and a positron combine, they are annihilated as material particles, and two γ -rays of energy totaling at least 1.02MeV are released. That these must be two photons is a consequence of the principle of momentum conservation⁵.

We note that the total gamma ray cross section curve for a substance, as the sum of the components for Compton effect, photoelectric effect, and pair production, exhibits a minimum around 3MeV energy. This implies that γ rays in this vicinity are more penetrating than those of higher or lower energy. In contrast with the case of beta particles and alpha particles, which have a definite range, a certain fraction of incident gamma rays can pass through any thickness of material. The exponential expression $e^{-\Sigma z}$ as used to describe neutron behavior can be carried over to the attenuation of gamma rays in matter. One can use the mean free path $\lambda=1/\Sigma$ or better, the half-thickness $0.693/\Sigma$, as the distance in which the intensity of a gamma ray beam is reduced by a factor of two.

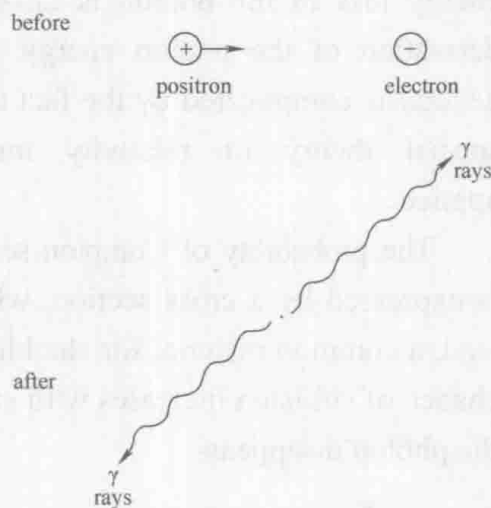


Fig.2.6 pair annihilation

2.6 The Properties of Alpha, Beta and Gamma Radiation

Alpha particles are nuclei of helium -4 with a mass of approximately $4u$ and a double positive charge. They are, of course, much heavier than electrons so that they are not significantly deflected as a result of an ionizing interaction with an electron, and they travel through matter in straight lines. They cause intense ionization due to their double positive charge, and so lose energy rapidly. Consequently alpha particles have a short and well-defined range in matter which depends on their energy and the density of the material through which they pass. The values for the range of 5MeV alpha particles in air and aluminum are about

3.5cm and 0.002 5cm respectively, from which it is evident that alpha particles can be easily stopped and present no shielding problem in nuclear physics or engineering, although they do present a health hazard if an alpha-emitting isotope is ingested in the body.

Unlike alpha particles which are emitted with discrete energies, beta particles are emitted from a radioactive isotope with a spectrum of energies. They interact with matter by causing ionization, but the effect is less intense than that due to alpha particles because of the smaller charge and much smaller mass of the beta particles. The range of beta particles emitted by a radioactive isotope is not well defined for two reasons, firstly the fact that they are produced with a spectrum of energies from zero up to their maximum energy, and secondly the fact that they do not travel in straight lines, but in zigzag paths resulting from a change of direction at each ionization interaction. As a guide, however, the ranges of beta particles of 5MeV energy in air and aluminum are about 2000cm and 1cm respectively, so that although they are much more penetrating than alpha particles they do not present a serious shielding problem; there is however a similar ingestion hazard.

Gamma radiation has a much greater penetrating power in matter than beta particles and consequently presents a serious shielding problem in any situation where it is produced in large quantities. A nuclear reactor is an example of this and further consideration of the characteristics, range and attenuation of gamma radiation in matter will be left to the chapter devoted to radiation hazards and reactor shielding.

Self-test

1. When electrons of___in heavy elements are displaced, the resultant high energy radiation is classed as X-rays.

- A. high energy
- B. inner orbits
- C. outer orbits
- D. low energy

2. Beta particles are capable of producing___as they penetrate a substance.

- A. large amounts of ionization
- B. energy
- C. photons
- D. particles

3. Alpha particles are___in their motion than electrons.

- A. fast
- B. slowed
- C. heavy
- D. less readily deflected

4. A lot of ionization is produced by the heavy ion___it moves through matter.

- A. as
- B. for
- C. with
- D. that

5. Alpha particle can___the outer layer of skin of the body.

- A. penetrate
- B. pass
- C. be accelerated
- D. be stopped by

6. In the photoelectric effect, the photon is___.

- A. produced
- B. found
- C. eliminated
- D. increased

7. Electron and positron are___equal charge but of opposite sign.

A. divided to B. produced C. of D. with

8. The γ -ray has _____ charge(s).

A. many B. one C. positive D. zero

9. Beta Particles present _____ shielding problem.

A. no B. no serious C. a D. serious

10. Low energy neutrons usually interact with nucleus _____ than high energy neutrons.

A. more readily B. fast C. slower D. more difficult

11. Alpha particles are much _____ than electrons.

A. larger B. heavier C. lighter D. smaller

12. When radioactive isotopes decay the _____ are formed.

A. protons B. daughter products

C. radioactive isotopes D. neutron and electron

关 键 字 解 释

Annihilation: Spontaneous conversion of a particle and its antiparticle into radiation e.g. positron and electron yielding two γ -ray photons each of energy 0.511MeV.

湮灭: 粒子及其反粒子变为辐射的自发转换, 例如正电子和电子产生两个能量各为 0.511MeV 的 γ 射线光子。

Bohr theory: The theory on explanation of the structure and behavior of atoms, representing them as consisting of a positively charged nucleus surrounded by planetary negative electrons revolving in definite orbits; when an electron jumps to a smaller orbit, monochromatic radiation is emitted.

玻尔理论: 阐明原子结构和行为的理论, 它假设原子中的负电子像行星一样按确定的轨道绕带正电的核运动, 当一个电子跃迁到较小的轨道时, 就放出单色辐射。

Bremsstrahlung: Electromagnetic radiation emitted when a charged particle changes its velocity. Thus when electrons collide with a target and suffer large decelerations, the X-radiation emitted constitutes the continuous X-ray spectrum.

轫致辐射: 带电粒子速度改变时发射的电磁辐射, 因而当电子撞击一个靶并经受很大减速度时, 所发射的 X 辐射构成连续 X 射线谱。

Compton scattering: Elastic scattering of photons, i.e. scattering in which both momentum and energy are conserved. A change in the direction of travel of a photon due to the interaction between the photon and the medium.

康普顿散射: 光子的弹性散射, 即动量和能量都守恒的散射。由于光子和介质之间的相互作用, 使光子运动方向发生改变。

Cosmic rays: Strongly penetrating rays, discovered by Milliken in 1925, coming from beyond the earth's atmosphere and containing particles that move at extremely high speed.

宇宙射线: 穿透力很强的射线, 1925 年由密利根发现, 来自地球大气层之外, 含有极高速度运动的粒子。

Decay: The process of spontaneous transformation of a radionuclide.

衰变: 放射性核素自然蜕变的过程。

Ion pair: A positive ion and a negative ion or electron, having charges of the same magnitude, and formed from a neutral atom or molecule by the action of radiation or by any other agency that supplies energy.

离子对: 由于辐射或任何其他能提供能量的媒介的作用, 从中性原子或分子上产生的一个正离子和一个等电荷的负离子或电子。

Ionization: An electrically neutral atom or molecule acquiring a charge.

电离(作用): 一中性原子或分子获得电荷的过程。

Kinetic energy: The ability of a moving body to perform work.

动能: 运动物体做功的能力。

Particle accelerator: A device in which atomic particles are speeded up by the application of electric and magnetic fields; for example, cyclotron and cosmotron.

粒子加速器: 一种利用电场和磁场的作用将原子中的粒子加速的装置, 如回旋加速器和质子同步加速器。

Photon: A light quantum; a unit of light or other radiant energy; the energy emitted when an electron is transferred from an inner orbit of the atom to the next.

光子, 光量子: 光或其他辐射能的单位; 电子由一个内层轨道转入另一个轨道时放出的能量。

Positron: Positron electron, of the same mass as and opposite charge to the electron, produced in the decay of radio-isotopes and pair production by X-rays of energy much greater than 1MeV.

正电子: 正电子和电子有相同的质量和相反的电荷, 产生于放射性同位素的衰变以及能量远大于 1MeV 的 X 射线导致的电子偶生成中。

Radioactivity: The spontaneous uncontrollable disintegration of the nucleus of an atom with the emission of α , β and/or γ rays.

放射性: 原子核自发的难以控制的衰变, 放射出 α 、 β 和/或 γ 射线。

Vacuum tube: Thermionic tube; an evacuated tube containing a heated electron-emitting cathode, an anode that receives the electrons, and additional electrodes that control the flow of electrons from the cathode to the anode.

真空管: 热离子管, 热电子管; 一种被抽成真空的管, 含有一个加热发射电子的阴极, 一个接受电子的阳极和附加用于控制从阴极到阳极电子流量的电极。

词 汇

aluminum [ə'ljʊ:minəm] *n.* 铝

analogy [ə'nælədʒi] *n.* 类似

annihilate [ə'naɪəleɪt] *v.* 湮灭

anxiety [æŋ'zaɪəti] *n.* 担心, 忧虑

- attenuation [ə'tenju'eɪʃən] *n.* 衰减
- beryllium [bə'rɪljəm] *n.* 铍
- biological [baɪə'lɒdʒɪkəl] *a.* 生物(学)的
- bulk [bʌlk] *n.* 大块, 大多数
- clump [klʌmp] *n.* 群
- collision [kə'liʒən] *n.* 碰撞
- competition [kəmpi'tɪʃən] *n.* 竞争
- culminate ['kʌlmineɪt] *v.* 达到极点, 结束
- deduce [dɪ'dju:s] *v.* 推论, 导出
- deflect [dɪ'flekt] *v.* 折射
- deflection [dɪ'flekʃən] *n.* 偏转, 折射
- derivation [deri'veɪʃən] *n.* 导出, 起源
- diffraction [dɪ'frækʃən] *n.* 衍射
- dislodge [dis'lɒdʒ] *v.* 移动, 撞出
- distinguish [dɪ'stɪŋɡwɪʃ] *v.* 区别, 分类
- division [dɪ'vɪʒən] *n.* 分, 划
- ejection [i'dʒekʃən] *n.* 放射, 排出
- eliminate [ɪ'lɪmineɪt] *v.* 消除
- embrace [ɪm'breɪs] *v. & n.* 包含; 接受
- encounter [ɪn'kaʊntə] *v.* 碰到, 遇到
- exponential [eks'pəʊ'nɛnʃəl] *a.* 指数的
- foil [fɔɪl] *n.* 叶, 箔
- fractional ['frækʃənəl] *a.* 分部的
- helium ['hi:lɪəm] *n.* 氦
- hyperbolic [ˌhaɪpə:'bɒlɪk] *a.* 双曲线的
- hypothesis [haɪ'pɒθɪsɪs] *n.* 假说
- incident ['ɪnsɪdənt] *a.* 入射的
- inertia [ɪ'nɜ:ʃiə] *n.* 惯性, 惰性
- ingest [ɪn'dʒest] *v.* 吸入
- interaction [ɪntər'ækʃən] *n.* 相互作用
- irradiate [ɪ'reɪdieɪt] *v.* 照射, 辐照
- kilovolt ['kɪləvɒlt] *n.* 千伏
- massive ['mæsɪv] *a.* 大的, 大量的
- mechanism ['mekənɪzəm] *n.* 机理, 作用原理
- migrate [maɪ'greɪt] *v.* 迁移
- mineral ['mɪnərəl] *n.* 矿物(质), 无机物
- momentum [məʊ'mentəm] *n.* 动量
- mutation [mju:'teɪʃən] *n.* 变化, 转变
- mutual ['mju:tʃuəl] *a.* 相互的, 共同的

- penetrate ['penitreit] *v.* 穿透, 透入
 perspex ['pə:speks] *n.* 有机玻璃
 predate ['pri:'deit] *v.* 在……之前
 projectile [prɒ'dʒɪktail] *n.* 抛射体, 入射粒子
 propagate ['pɒpəgeit] *v.* 传播
 quantum ['kwɒntəm] *n.* (复数 quanta) 量子
 rigorously *ad.* 严格地, 精确地
 scatter ['skætə] *v. & n.* 散射, 扩散
 schematically [ski'mætikəli] *ad.* 用示意图, 大略地
 screen [skri:n] *n. & v.* 屏蔽, 隔离
 shielding ['ʃi:ldiŋ] *n.* 屏蔽; *a.* 防护的
 sketch [sketʃ] *v.* 画示意图
 spectrum ['spektrəm] *n.* 谱
 stationary ['steiʃənəri] *a.* 静止的
 target ['tɑ:git] *n.* 靶, 目标
 vicinity [vi'siniti] *n.* 附近
 visualize ['vizjuəlaiz] *v.* 目测, 观察
 as a rule of thumb 根据经验

课 文 注 释

1. Radioactivity is harmful to...the danger.

句中 yet 是副词, 表示到目前为止; sensor organs 是感觉器官。整句译为: 放射性对生命组织是有害的, 到目前为止, 人类和其他动物还没有进化出能够监测到它危害的感觉器官。

2. The idea that we are vulnerable to...is somewhat unnerving.

此句中的主语是 The idea, 后面的 that..., that...and that...是并列的定语从句, 说明 The idea, 这里 is somewhat unnerving 是系表结构, 它的主语是 The idea。整句译为: 我们容易受到不同形式的放射性(有些情况下需要屏蔽)的危害, 如果没有人工制成的仪器, 我们监测不到它的存在以及它对我们人类基因的潜在危害, 这些想法使人颇为不安。

3. As the positively charged...loses energy.

句中 varying 引导的现在分词短语作 attraction 的定语。整句译为: 当带正电荷的入射粒子接近或穿过重粒子时, 与电子之间产生吸引力, 吸引力的大小与它们距离之间的关系为 $1/r^2$, 电子产生位移并获得能量, 重粒子损失了能量。

4. An incident photon of high enough...charged ion.

句中 enough 是副词, 修饰形容词 high。整句译为: 一个能量足够高的入射光子从原子中撞出一个电子, 产生了一个带有正电荷的离子。

5. That there must be two...conservation.

句中 That 引导的是主语从句。整句译为: 必须有两个光子, 这是动量守恒产生的结果。

Unit 3 Nuclear Materials

3.1 Fuels

3.1.1 Uranium

Uranium in one form or another is by far the most common fuel material for nuclear reactors¹. (By comparison, the use of thorium and plutonium has so far been on a very small scale.) It can be used either as pure uranium, a metal, or as a compound such as uranium dioxide UO_2 or uranium carbide UC.

Uranium is a rather soft and ductile metal which oxidizes readily in air and water at high temperature. Its melting point is 1133°C . It exists in one of three allotropic forms, depending on its temperature. These three different forms are called the alpha, beta and gamma phases, and changes from one phase to another due to temperature changes accompanied by density changes. Alpha phase uranium has a density of $19\text{g}/\text{cm}^3$ and a thermal conductivity which varies from $25\text{W}/(\text{m} \cdot \text{k})$ at 25°C to $42\text{W}/(\text{m} \cdot \text{k})$ at 665°C . The transition from the alpha to the beta phase takes place at 665°C and is accompanied by dimensional changes in the crystalline structure of the uranium, expansion along one axis and contraction along the others. To avoid distortion due to these anisotropic dimensional changes 665°C is considered to be the maximum operating temperature for uranium.

Metallic uranium is also very susceptible to radiation damage which produces dimensional changes and swelling above about 450°C . Consequently high burnup of metallic uranium fuel are not possible. In the British gas cooled Magnox reactors, which are the principal users of this type of fuel, the burnup is limited to about $3500\text{MW}(\text{d}/\text{tU})$.

To summarize, the low operating temperature, susceptibility to radiation damage and low permissible burnup of uranium are serious disadvantages to its choice as a reactor fuel, and account for its very limited use².

Uranium dioxide UO_2 is a black powder which can be fabricated by cold pressing and sintering at high temperature to produce small cylindrical pellets, and in this form it is by far the most common material for the fuel of commercial reactors. In this ceramic form UO_2 has good stability at high temperature and good resistance to radiation damage which enables it to be used to high burnup. The melting point is 2865°C and the theoretical density is $10.96\text{g}/\text{cm}^3$, although in practice the density of UO_2 pellets produced as described above is about $10\text{g}/\text{cm}^3$. The thermal conductivity is low, being about $2.5\text{W}/(\text{m} \cdot \text{k})$ in the temperature range from 1000 to 2000°C , however this low thermal conductivity is compensated by the very high melting point which permits high maximum fuel temperatures.

Uranium dioxide does not react with water at high temperature, a very valuable characteristic as otherwise cladding failures in water cooled reactors would lead to serious

reactions. It can retain a large fraction of the gaseous fission products at temperatures below 1000°C , but as the fuel temperature at the centre of a pellet is likely to be greatly in excess of this value, provision must be made for fission product gas release³. This is usually done by having an empty space at the top of each fuel tube into which the gases can diffuse.

During operation in reactor UO_2 pellets suffer structural changes, principally as a result of the high operating temperatures and high temperature gradients, but also as a result of prolonged irradiation. The effects may include swelling, formation of cracks and voids in the pellet and changes in the grain structure of the UO_2 . This type of fuel is normally subjected to much higher burnup than pure uranium, and 5 percent or more of the original uranium atoms in the fuel may undergo fission and be changed, each one to two intermediate mass fission product atoms⁴.

Uranium carbide, UC, is another ceramic fuel of possible interest, but it has not been developed or used to anything like the same extent as UO_2 . It may have some advantages over UO_2 , principally its higher thermal conductivity and higher density which leads to more uranium atoms per unit volume of fuel, which is an advantage in a reactor. UC reacts with water, which makes it unsuitable for use in water cooled reactors, but it does not react with sodium below 500°C , so it might be used in fast reactors. Its melting point, 2380°C , is rather lower than that of UO_2 , but this is compensated by its higher thermal conductivity. The development of uranium carbide so far has been principally as the fuel for high temperature gas-cooled reactors.

3.1.2 Plutonium

Pure plutonium metal is not suitable as a reactor fuel due to the large number of crystalline phases which exist up to its melting point of 640°C . The thermal conductivity is also very low, about 4.2W/mK at room temperature. Plutonium metal is highly reactive in moist air, but it can be stored in dry air at low temperature. It is a very dangerous material, being radioactive, toxic and an essential component of nuclear weapons, and is potentially a serious health hazard, particularly if it exists as dust in the atmosphere and is taken into the lungs by inhalation.

As a reactor fuel plutonium is used as the PuO_2 . Its melting point is 2400°C . Plutonium dioxide is mixed with uranium dioxide to form mixed oxide fuel (MOX) which for fast reactors typically contains 20 to 25 percent of PuO_2 . The properties of this mixed oxide fuel are similar to those of UO_2 alone.

3.1.3 Thorium

Thorium has not been used as a reactor fuel to any great extent yet except in a few high temperature gas-cooled reactors. ^{232}Th is the fertile isotope from which uranium 233 is produced, and it is theoretically possible to obtain high breeding ratios in thermal as well as fast reactors using this combination. Pure metallic thorium has a melting point of about 1700°C . It is superior to uranium due to its better stability, but it is not used as a fuel in its pure form. Instead it is used either as thorium dioxide ThO_2 or thorium carbide ThC_2 . To date these

compounds have only been used to a very small extent in a few high temperature gas-cooled reactors.

Thorium dioxide is similar in many respects to uranium dioxide. It is produced by the same methods of powder metallurgy, and it is chemically inert and has a good resistance to radiation damage. Thorium carbide has been used in the form of coated particle fuel in HTGRs⁵. Very small spherical particles less than 1mm diameter of mixed ThC₂ and UC₂ (highly enriched in ²³⁵U) are coated with thin layers of pyrolytic carbon and silicon carbide to retain fission products. These particles are dispersed in graphite to form a homogeneous mixture of fuel and moderator which has a very high operating temperature and good resistance to radiation damage.

3.2 Moderators

The requirements of the moderator for a thermal reactor, namely low mass number, very low neutron capture cross-section and high scattering cross-section, limit the choice to only a few materials. Hydrogen and its isotope deuterium, carbon and beryllium are the only elements that are suitable. Hydrogen and deuterium, being gases, are not sufficiently dense and must be used in the form of compounds, water and heavy water being the obvious choices. The use of hydrocarbon compounds has been tried, but has not been successful and such materials are not used as moderators. It is interesting to recall, however, that Fermi used paraffin wax in his early experiments in the 1930s to slow down neutrons and study their interactions with the elements, so he was one of the first scientists to be aware of the effects of neutron moderation.

Beryllium has a very low neutron capture cross-section (0.000 9 barns), high melting point (about 1300°C) and good strength, and at one time it seemed possible that it would find an application either as the moderator or the fuel cladding in thermal reactors. However, it and its compounds are toxic, and beryllium itself has low ductility and poor corrosion resistance. Beryllium oxide BeO also has undesirable properties. As a result of this neither beryllium nor its oxide has found any use in power reactor, and it is unlikely that they will be used in the future.

The choice of moderators for thermal reactors is thus limited to three materials-water, heavy water and carbon in the form of graphite.

3.2.1 Water

Water is an obvious choice for the moderator of a thermal reactor, and it can also serve as the coolant. It has excellent neutron slowing down properties which enable water moderated reactors to have much more compact cores than are possible in other types of thermal reactors. The capture cross-section of water is rather high (0.66 barns per molecule) so that water moderated and cooled reactors require enriched uranium for criticality. It is, of course, abundant, cheap and easily obtainable with high purity.

The main problem associated with the use of water as the moderator and coolant in a

power reactor concerns its rather unfavorable thermodynamic characteristics. The saturation pressure and temperature relationship is such that high pressures are required to prevent boiling at high temperatures, e.g. a pressure of 150 bar is required to allow water to reach a temperature of 340°C without boiling⁶. Pressures of 15 to 16 MPa are typical of pressurized water reactors, in which the temperature is limited to about 325°C.

It is important to maintain water purity in a water cooled and moderated reactor, firstly to minimize corrosion and secondly to prevent the water from becoming radioactive due to (n, γ) reactions with the impurities as the water flows through the reactor core. Radiation levels in the water can influence the radiation dose levels to which power station operating and maintenance staff are exposed, and the maintenance of high water purity assists in reducing operator exposures.

3.2.2 Heavy Water

Heavy water is similar to ordinary water in most of its physical and thermodynamic properties. The principal difference is that deuterium has a much lower capture cross-section than hydrogen, and the capture cross-section of heavy water is only 0.001 barns. Deuterium is, however, not such a good moderator as hydrogen. Consequently, heavy water moderated and cooled reactors use natural uranium as their fuel, but their core sizes are rather larger than water moderated reactor cores. An important difference is that heavy water is very expensive to produce by separation from ordinary water, and losses by leakage must be minimized.

3.2.3 Graphite

The first nuclear reactor in the world, CP-1 (Chicago Pile 1) was moderated with graphite, and although this material has not been used subsequently in American commercial power reactors, it has been used extensively in British reactors. Its nuclear characteristics, i. e. slowing down power and capture cross-section are not as good as those of heavy water, but it is readily obtainable in a pure form at reasonable cost and is easily machined. Its structural and thermal properties are good although it reacts with oxygen and air at high temperatures. Graphite has a high thermal conductivity [130~190W/(m·K)] and it sublimates at 3650°C without melting, so there is for all practical purposes no limit on its maximum operating temperature. Graphite cores are assembled from large numbers (many thousands) of blocks of rectangular shape in which holes are provided for fuel elements and control rods.

The effect of prolonged neutron irradiation on graphite is very important, as such irradiation causes dimensional changes and the buildup of stored energy within the crystalline structure. The dimensional changes are due to the anisotropic crystalline structure of graphite, and they depend on its temperature during irradiation. Below 300°C there is a contraction along one axis and expansion along another. At higher temperature there is contraction in all directions. The design of a graphite core and keys which hold one block in position relative to its neighbors must make these dimensional changes possible without damage or distortion of the core.

The stored energy, sometimes called Wigner energy, is due to the energy required to

displace atoms in the graphite's crystalline structure. Neutron irradiation causes the displacement of atoms, and if they do not return to their original positions, this energy remains stored in the graphite. This is the case at temperatures below 100°C . At higher temperatures partial self-annealing of the graphite takes place and some of the stored energy is released. It is necessary to release the stored energy in a controlled way otherwise, if it is released too rapidly, the graphite temperature can rise disastrously. This occurred at one of Britain's two original gas-cooled plutonium producing reactors at Windscale Cambria⁷ in 1957. On that occasion an uncontrolled Wigner energy release caused the graphite temperature to reach such a high level that the graphite burned and the core of the reactor was destroyed.

Present British gas-cooled reactors operate with graphite temperatures of about 400°C at which there is continuous self-annealing of the Wigner strain energy, and no possibility exists of a repeat of the Windscale accident.

3.3 Coolants

The principal requirements of the coolant for a nuclear reactor are follows.

- (1) Good thermodynamic properties, namely high thermal conductivity, density and specific heat, and low viscosity.
- (2) Chemically non-reactive with other components of the reactor.
- (3) Very low neutron capture cross-section.
- (4) It should not become radioactive as a result of (n, γ) reactions which may occur when the coolant is passing through the core of the reactor.

Among the gaseous coolants, some can be eliminated from consideration for one reason or another. Oxygen and hydrogen are both reactive, the latter explosively so. Nitrogen has a significant capture cross-section (1.8 barns). Air, being a mixture of oxygen and nitrogen can also be ruled out. Oxygen 16 undergoes an (n, γ) reaction with high energy neutrons (e.g. fission neutrons) to form nitrogen 16 which is radioactive, but its half-life is only 7 seconds, so the radioactive hazard is short-lived. The two most suitable gaseous coolants are carbon dioxide and helium.

Carbon dioxide is a fairly unreactive gas, but it does react at high temperatures with certain types of steel and with graphite. Both these reactions have proved troublesome in British gas-cooled reactors in which carbon dioxide has been chosen as the coolant. Its advantages are its inertness, availability and cheapness, and the very low capture cross-sections of both carbon and oxygen. Carbon 13, which is a very small constituent of naturally occurring carbon, captures neutrons to a very small extent to form radioactive carbon 14 which poses a minor hazard if carbon dioxide leaks or is vented from a reactor to the atmosphere.

Helium is inert, has good thermodynamic properties and does not pose a radioactive hazard, so it might be regarded as the ideal gaseous reactor coolant. Unfortunately, it is not readily available in large quantities and is expensive. Its use as a reactor coolant is at present

confined to the few high temperature gas-cooled reactors operating in the USA and Germany.

The properties of water and heavy water as reactor coolants have been described in the preceding section, but their moderating ability makes them unsuitable for use as coolants in fast reactors.

Liquid metals are potentially excellent reactor coolants because of their good thermodynamic properties, in particular their high thermal conductivity which leads to very good heat transfer coefficients. Sodium, lithium, mercury and sodium-potassium alloys are all possibilities, but of these only sodium has been used to any great extent, exclusively in fast breeder reactors. Sodium-potassium alloys may become more commonly used. Mercury is very expensive and is toxic, as well as having too high a capture cross-section for use in thermal reactors. Lithium is similar to sodium in many respects, but has a higher melting point and is more expensive.

Sodium is the standard coolant for the still fairly small number of fast breeder reactors operating in the world. Its melting point is 98°C and its boiling point at atmospheric pressure is 890°C . So it is not necessary to use sodium at higher than atmospheric pressure, which is a distinct advantage. It is highly reactive with air and water, so high integrity pipework and heat exchangers are necessary to avoid leakage. Sodium has a fairly low capture cross-section (0.5 barns) but it does undergo the $^{23}\text{Na}(n, \gamma) ^{24}\text{Na}$ reaction, and intermediate heat exchangers are required to contain the radioactive ^{24}Na within the biological shield of the reactor. Sodium is not corrosive to most structural materials provided its oxygen content is maintained low⁸. The formation of sodium oxide in the coolant circuit can lead to plugging unless it is removed in cold traps.

Sodium-potassium eutectic alloy with a melting point of -11°C can be used for the decay heat removal systems of fast reactors as the coolant temperature in these systems may fall below the melting point of sodium.

3.4 Cladding Materials

The fuel cladding materials require a very low neutron capture cross-section, good thermal conductivity and good strength at high temperatures to resist thermal stresses, deformation of the fuel and buildup of fission product gas pressure inside the cladding. In addition the cladding should be easy to fabricate and not liable to corrosion or other chemical reactions with either the fuel or the coolant. The following metals and their alloys satisfy some if not all the above requirements, and have been considered for use as fuel cladding: aluminium, beryllium, magnesium, zirconium and stainless steel.

Beryllium has not proved to be suitable material for use in reactors, and it has not been developed as fuel cladding.

Aluminium has been used as the cladding material for low power research reactors in fairly low (0.23 barns) capture cross-section but its mechanical properties such as strength and

hardness are quite low and aluminium cannot be used in situations of high stress or temperatures greater than 300°C . Consequently it cannot be used as a cladding or structural material in power reactors, and its use as noted above is solely in research reactor, in which it is often used in the form of an inter-metallic compound with uranium.

Magnesium has a very low capture cross-section (0.063 barns) but like aluminium it is a soft metal of low strength with a maximum operating temperature of about 450°C . It is also very reactive, oxidizing readily in air. The characteristics of magnesium particularly as regards oxidation, can be greatly improved by the addition of small quantities (less than 1 percent) of aluminium and beryllium, and the alloy thus formed, called Magnox, has been used extensively as the cladding material in British gas-cooled reactors. However, the maximum operating temperature of Magnox is about 450°C , and this places a limit on the thermodynamic performance of these reactors.

Zirconium has a low capture cross-section (0.185 barns), a high melting point (1850°C), good mechanical properties and a high resistance to corrosion by water. These properties make it an excellent cladding material for reactors. The mechanical properties and corrosion resistance can be further improved by alloying zirconium with small quantities of tin (1.5 percent), iron (0.15~0.2 percent), chromium (0.1 percent) and nickel (up to 0.05 percent). The resulting alloys, Zircaloy -2 and Zircaloy -4, are used extensively as the cladding and in-core structural materials for water and heavy water moderated and cooled reactors. They are the preeminent fuel cladding materials at the present time.

At very high temperatures, above about 1000°C , zirconium and the zircaloys can react with steam to form hydrogen, which is potentially very hazardous. The conditions which might give rise to this reaction can occur after a reactor loss-of-coolant accident when the fuel and cladding overheat as a result of inadequate decay heat removal. If the cladding temperature exceeds 1000°C and the cladding comes into contact with steam, then the hydrogen producing reaction occurs, possibly with explosive results.

Stainless steel is well known for its excellent mechanical properties and corrosion resistance. Unfortunately its capture cross-section, which depends on the type of steel and the precise quantities of chromium and nickel in it, is too high for stainless steel to regard as ideal for fuel cladding or other in-core uses⁹. It is not now used as fuel cladding in pressurized water reactors, but it is used in the advanced gas-cooled reactor and in fast breeder reactors (FBRs). In the latter the rather high capture cross-section is of less importance than in thermal reactors. Stainless steel is used extensively for out-of-core uses in PWRs and FBRs where its excellent corrosion resistance is invaluable.

3.5 Control Materials

Materials for controlling reactors need to have high capture cross-section. Several such materials are available. In addition other materials of lower cross-section can be used in the

reactor core for “flux-shaping” and fine control¹⁰. Stainless steel rods can be used for this purpose.

Of the control materials, boron ($\sigma_c=760$ barns) is the most common. It cannot be used by itself, but may be alloyed with steel or used in the form of boron carbide encased in steel. Boron can also be used to compensate for long term changes of reactivity (such as fuel burnup) in the form of a burnable poison, boric acid, dissolved in the coolant of a pressurized water reactor.

Indium and cadmium both have high capture cross-sections (195 and 2450 barns respectively) but both have melting points which are too low to permit their use in power reactors. In the form of an alloy containing 80 percent silver, 15 percent indium and 5 percent cadmium, these elements are used as control rods in pressurized water reactors. The alloy has an adequately high melting point, but must be encased in stainless steel to protect it from corrosion.

Hafnium ($\sigma_c=113$ barns) is a good control material, having adequate mechanical strength and good corrosion resistance. However, it is too expensive to be used on a large scale for commercial reactors.

Gadolinium ($\sigma_c=46\,000$ barns) is used as a burnable poison in some reactors such as the advanced gas-cooled reactor.

Self-test

1. Boron is the most common materials ____ reactors.
A. controlling B. to control C. for D. for controlling
2. Uranium dioxide can ____ high burnup.
A. not be used to B. be used to C. use to D. be used with
3. After long time operation in reactor, UO_2 pellets ____ structural changes.
A. don't suffer B. have C. suffer D. present
4. The melting point of uranium dioxide is ____ plutonium dioxide.
A. higher than that of B. the same as
C. lower than that of D. similar to
5. During common reactor operation zirconium ____ steam.
A. reacted with B. cannot react with
C. call react with D. cannot be reacted
6. The mechanical properties of zirconium are ____ stainless steel.
A. better than that of B. poorer than that of
C. same as that of D. same to
7. Hydrogen and deuterium ____ moderators.
A. cannot be used as B. are suitable
C. are good D. are used as

8. Beryllium as fuel cladding_____in power reactor.

A. is used

B. use

C. has been used

D. hasn't been used

9. The density of heavy water is_____water.

A. different from

B. the same as

C. heavier than

D. heavier than that of

10. Graphite cores are assembled_____large numbers of blocks of rectangular shape.

A. of

B. with

C. from

D. for

11. If Wigner energy is released too rapidly, the graphite temperature can rise_____.

A. dangerously

B. more

C. violent

D. highly

12. Liquid metals are potentially excellent reactor_____.

A. moderator

B. structure material

C. coolant

D. material

关键字解释

Burnable poison: Neutron absorber introduced into reactor system to reduce initial reactivity but becoming progressively less effective as burn-up proceeds. This helps to counteract the fall in reactivity as the fuel is burned up.

可燃毒物: 一种中子吸收剂, 放入反应堆系统中, 用以降低初始反应性, 而后在燃烧过程中它的有效性逐渐降低。它可用来补偿由于燃料消耗而造成的反应性下降。

Burnup: ①In nuclear fuel, amount of fissile material burned up as a percentage of total fissile material originally present. ②Of fuel element performance, the amount of heat released from a given amount of fuel, expressed as megawatt days per tonne.

燃烧: ①在核燃料中, 烧掉的可裂变材料与初始存在的总可裂变材料的百分比。②给定量燃料中所释放出的热量, 以(MW·d/t)表示, 作为燃料性能的描述。

Cladding: Thin protective layer, usually metallic, of reactor fuel units to contain fission products and to prevent contacting between fuel and coolant.

包壳: 通常是金属材料制成, 包容裂变产物和防止燃料和冷却剂接触, 是反应堆燃料的保护层。

Coolant: The gas, liquid or liquid metal circulates through a reactor core to carry the heat generated in it by fission and radioactive decay to steam generators.

冷却剂: 循环流过反应堆堆芯, 将裂变和放射性衰变产生的热传输到蒸汽发生器去的气体、液体或液态金属。

Fertile isotope: A nonfissionable isotope in a nuclear reactor which can be converted by the capture of a neutron into fissile isotope.

增殖同位素: 在反应堆中通过俘获一个中子能转换为易裂变同位素的非裂变同位素。

Graphite: Naturally occurring allotropic form of carbon; used as moderator in nuclear reactors.

石墨: 天然存在的同素异形碳, 在反应堆中作慢化剂。

Heavy water: Deuterium oxide D_2O ; its chemical properties are the same as those of water but its physical properties are different.

重水: 氧化氘 D_2O , 重水的化学性质与普通水相同, 但其物理性质有所区别。

Moderator: A substance, such as graphite, paraffin, water or heavy water, used in a nuclear reactor to slow down neutrons.

慢化剂: 在核反应堆中用来减慢中子速度的物质, 如石墨、石蜡、水或重水等。

Plutonium: A radioactive element formed by the loss of an electron from the nucleus of neptunium; chemical symbol Pu; atomic number 94.

钚: 一种放射性元素, 由镎的原子核失去一个电子而形成, 化学符号 Pu, 原子序数 94。

Reactivity: The departure of the multiplication constant of a reactor from unity, measured in different ways.

反应性: 反应堆倍增系数与 1 的差额, 可用不同的方法来度量。

Sintering: Compacting metal powder into a continuous mass by compressing and then heating below the fusion temperature; the particles are held together by cohesive force.

烧结: 通过压缩将金属粉末压实成一个连续的物质, 然后以低于熔化温度进行加热, 颗粒因黏聚力而结合在一起。

Swelling: Change of volume of fuel which may occur during irradiation.

肿胀: 燃料受辐照时可能发生的体积变化。

Thorium: A Corrosion-resistant, ductile dark-gray radioactive metal; chemical symbol Th; atomic number 90.

钍: 一种耐腐蚀, 有延性的深灰色放射性金属, 化学符号 Th; 原子序数 90。

Wigner energy: Energy stored within a crystalline substance. Changes in physical properties of graphite resulting from the displacement of lattice atoms by high-energy neutrons and other energetic particles in a graphite reactor.

维格纳能量: 储存在晶体物质中的能量。在石墨堆中石墨由于受到高能中子和其他常能量粒子的作用, 造成点阵原子位移而引起的物理性质改变。

Zirconium: A metallic element, symbol Zr, atomic number 40, atomic mass 91.22, melting point 2130K. Its low neutron absorption and its retention of mechanical properties at high temperature make it an useful construction material of nuclear reactors.

锆: 金属元素, 符号 Zr, 原子序数 40, 原子质量 91.22, 熔点 2130K。由于其中子吸收截面很低, 且高温下能保证其机械性能, 因此它是反应堆中重要的结构材料。

词 汇

allotropic [æləu'trɒpɪk] *a.* 同素异形的

anisotropic [ə,naɪsəu'trɒpɪk] *a.* 各向异性的

beryllium [bə'rɪljəm] *n.* 铍

boric acid 硼酸

boron ['bɔ:rən] *n.* 硼

- cadmium ['kædmiəm] *n.* 镉
carbide ['kɑ:baid] *n.* 碳化物
ceramic [si'ræmik] *a.* 陶瓷的
chromium ['krəʊmjəm] *n.* 铬
component [kəm'pəʊnənt] *n.* 元素, 组成部分
conductivity [kɒndʌk'tiviti] *n.* 传导率
corrosion [kə'rəʊʒən] *n.* 腐蚀
crack [kræk] *v. & n.* 破裂
criticality [kriti'kælitɪ] *n.* 临界 (性, 状态)
crystalline ['kristəlain] *a.* 晶状的
diffuse [di'fju:z] *n.* 扩散
dimensional [di'menʃənəl] *a.* 尺寸的
dioxide [daɪ'ɒksaɪd] *n.* 氧化物
distortion [dis'tɔ:ʃən] *n.* 变形, 扭曲, 畸变
dose [dəʊs] *n.* 剂量
ductile ['dʌktaɪl] *a.* 可延展的, 易变形
enrich [in'ritʃ] *v.* 富集, 浓缩
eutectic [ju:'tektik] *a.* 共晶的
exposure [iks'pəʊʒə] *n.* 辐照
fabricate ['fæbrikeit] *v.* 制造, 生产
fraction ['frækʃən] *n.* 分数, 部分
gadolinium [,gædə'liniəm] *n.* 钆
hafnium ['hæfniəm] *n.* 铪
homogeneous [hɒmə'dʒi:njəs] *a.* 均匀的
hydrocarbon ['haɪdrəu'kɑ:bən] *n.* 碳氢化合物
indium ['indiəm] *n.* 铟
inert [i'nə:t] *a.* 惰性的
inhalation [,inhə'leiʃən] *n.* 吸入
integrity [in'tegriti] *n.* 完整, 一体化
liable ['laɪəbl] *a.* 易于……的
magnesium [mæg'ni:ziəm] *n.* 镁
Magnox ['mægnɒks] *n.* 镁诺克斯合金
metallurgy [me'tælədʒi] *n.* 冶金
moist [mɔɪst] *a.* 湿的
nickel ['nikl] *n.* 镍
oxide ['ɒksaɪd] *n. & a.* 氧化物 (的)
oxidize ['ɒksi,daɪz] *v.* 使氧化, 生锈
paraffin ['pærəfin] *n.* 石蜡
pellet ['pelɪt] *n.* 燃料芯块, 片

- phase [feɪz] *n.* 相
 potassium [pə'tæsjəm] *n.* 钾
 powder ['paʊdə] *n.* 粉末
 pre-eminent [pri:'eminənt] *a.* 优秀的
 pyrolytic *a.* 热解的
 retain [ri'tein] *v.* 保持, 保留
 saturation [ˌsætʃə'reɪʃən] *n.* 饱和
 scale [skeɪl] *n.* 规模, 尺度
 self-annealing *n.* 自退火
 silicon ['sɪlɪkən] *n.* 硅
 sodium ['səʊdʒəm] *n.* 钠
 sublime [sə'blaɪm] *v.* 升华
 superior [sju'piəriə] *a.* 高级的
 susceptible [sə'septəbl] *a.* 敏感的
 thermodynamic [θə:məʊdaɪ'næmɪk] *a.* 热力学的
 tin [tɪn] *n.* 锡
 toxic ['tɒksɪk] *a.* 有毒性的
 transition [træn'sɪʃən] *n.* 转变
 trap [træp] *n.* 井, 收集器
 viscosity [vis'kɒsɪti] *n.* 黏性
 void [vɔɪd] *n.* 空隙
 wax [wæks] *n.* 蜡
 zircaloy ['zə:kəlɔɪ] *n.* 锆合金

课文注释

1. Uranium in one form...nuclear reactors.

句中 by far the most common 表示最常用的, by far 是加重说明了 the most 的程度。整句译为: 不同形式的铀是核反应堆中最常用的一种燃料。

2. To summarize, the low...limited use.

句中有两个并列的谓语动词, 一个是系动词 are, 另一个是动词 account。整句译为: 总的说来, 允许的使用温度低、对放射性敏感以及燃耗低, 使选择金属铀作为反应堆的燃料有很大的缺点, 因此它的使用很有限。

3. It can retain...gas release.

句中 It 指 uranium dioxide, as 是因为的意思, 它引导的原因状语从句修饰后面的主句。整句译为: 当温度低于 1000℃ 时, 二氧化铀能保留住大部分裂变产生的气体, 但是燃料芯块中心的温度一般都高于这一温度, 因此必须考虑裂变气体的释放问题。

4. This type of fuel...product atoms.

句中 each one 指 uranium atom; be changed 与后面的 to 相连接, 说明改变成了……。

整句译为: 这种类型的燃料与纯铀相比有很高的燃耗, 燃料中铀原子的百分之五或更多的份额发生了裂变, 每一个铀原子裂变成两个中等质量的裂变产物核。

5. Thorium carbide...fuel in HTGRs.

句中 HTGRs 是 High Temperature Gas-cooled Reactors 的缩写。整句译为: 碳化钍以涂层颗粒的形式作为高温气冷堆的燃料。

6. The saturation pressure...without boiling.

句中 such that 中的 such 作代词, such that 表示(是)这样(以至于), that 引导的从句解释说明 such。整句译为: 饱和温度和压力的关系是这样: 要在高温下不产生沸腾就需要一个高的压力, 例如让水达到 340℃ 而不沸腾需要 150 巴的压力。

7. This occurred at...Windscale Cambria.

Cambria 坎布里亚, 郡名, 位于英格兰西端, 西、北、南三面濒临爱尔兰海, 首府卡莱尔。Windscale 温德斯格尔是坎布里亚的一个地名。

8. Sodium is not...maintained low.

句中 provided 是连词, 意思是“只要”, 在句中引导条件状语从句。整句译为: 只要氧的含量保持在比较低的水平, 钠对大部分结构材料都没有腐蚀作用。

9. Unfortunately its...in-core uses.

句中 which 引导的是定语从句, 修饰 capture cross-section。整句译为: 遗憾的是, 不锈钢的中子吸收截面太高, 使它不能作为燃料包壳和其他堆芯的理想材料, 它的中子吸收截面与钢的类型及铬和镍的精确含量有关。

10. In addition...for “flux-shaping” and fine control.

句中 flux-shaping 表示“中子通量展平”。整句译为: 另外, 较低吸收截面的其他材料可用在反应堆堆芯内, 为了展平堆芯中子通量和进行精细微调。

Unit 4 The Theory of Nuclear Reactors

It is now appropriate to consider the necessary requirements for a chain reaction, and also some other topics that are of importance in the development of nuclear energy for peaceful purposes.

The achievement of a system in which a controlled, self-sustaining fission chain reaction takes place is the first requirement, as this is the way in which the energy of fission can be released, also in a controlled way, and put to good use¹. The system in which the chain reaction takes place is called a nuclear reactor, and there are many possible types of reactor, depending on the materials of construction and the energy of the neutrons which cause fission. In this lesson we will consider qualitatively the conditions necessary for a chain reaction and will identify a number of different possible reactor types. Some types of reactor require enriched uranium to achieve a chain reaction, and the processes for enrichment are briefly described.

The complete utilization of the world's resources of uranium to provide energy is an important aspect of nuclear power, and the discussion of this topic leads to a description of the types of reactor and the fuel cycles that will enable not only uranium, but also thorium to be used as a long-term source of energy.

Finally, the role which nuclear power can play in supplying a world in which energy demands are increasing and resources are being rapidly depleted will be discussed.

4.1 The Chain Reaction

The condition that is necessary for a stable, self-sustaining chain reaction is that exactly one of the neutrons produced in one fission event proceeds to cause a second fission from which one neutron goes on to cause a third fission, and so on². In such a reaction the neutron density and fission rate remain constant. This condition can be expressed by means of a multiplication factor, k , which is defined as the ratio of the number of neutrons in one generation to the number of neutrons in the preceding generation. When this factor is exactly one the condition for a stable chain reaction is satisfied and the reactor is said to be critical. If this factor is greater than one the reactor is supercritical and a divergent chain reaction exists in which the neutron density and fission rate increase, possibly at an explosive rate as in an atomic bomb. If the multiplication factor is less than one the reactor is subcritical and the chain reaction decreases and eventually dies out.

A nuclear reactor is an assembly of many components of which at this stage we need mention a few of the most important. The most important component of any reactor is the fuel in which fission takes place and energy, in the form of heat, is released. At the present time uranium is the

most widely used nuclear fuel, although the isotope ^{239}Pu is becoming increasingly important. In many reactors a light element is included for the specific purpose of slowing down neutrons from fission to thermal energy at which they are most effective for causing further fission. This material is called the moderator. Finally cladding materials are required to contain and support the fuel and prevent the release of radioactive fission products, and in the case of all reactors, except those operating at very low power, a coolant is required which is circulated through the core to transport the energy released in the fuel by fission to external heat exchangers.

Returning to the conditions for a critical reactor we can say that the rate at which neutrons are used up or lost in a reactor must be exactly equal to the rate at which they are produced by fission. Neutrons are used up or lost in two ways, either they leak out of the reactor or they are absorbed within the reactor. Absorption within the reactor includes fission and capture in the fuel and capture in the other materials of the reactor such as the moderator, coolant, cladding and control rods. The relative rates at which these processes take place depend on the size and composition of the reactor. It is the purpose of reactor theory to analyze the various processes in a reactor, determine their rates and calculate for a given composition and size of core the value of the multiplication factor. Alternatively the calculation may be directed towards finding the critical size or composition of a reactor for a given value of the multiplication factor.

As an introduction to this type of problem let us consider the possibility of a chain reaction in an infinitely large system of pure natural uranium which consists of 99.285 percent ^{238}U and 0.715 percent ^{235}U . (The ratio of ^{238}U atoms to ^{235}U atoms is approximately 139 to 1.) The purpose of considering an infinite system is to avoid for the present time the question of neutron leakage as neutrons cannot leak out of an infinite system.

4.2 The Neutron Cycle in a Thermal Reactor

Accurate calculations of the multiplication factor for any reactor must take full account of all the processes involving neutrons between the time when they are produced by fission and time when they eventually disappear either by absorption or by leakage from the reactor³. The procedure is simplified if we consider various parts of the neutrons lifetime separately, and we will apply this method to a thermal reactor of finite size fueled with natural or enriched uranium.

Consider n neutrons at an average energy of 2MeV produced by thermal fission in ^{235}U . Before these neutrons slow down below 1MeV there is a possibility that a few cause fission, referred to as fast fission, in ^{238}U . The fast fission factor, ϵ , is defined as the number of neutrons slowing down below 1MeV per neutron produced by thermal fission.

Now $n\epsilon$ neutrons are slowing down below 1MeV and continue to slow down as a result principally of elastic scattering collisions with the moderator. During the slowing down process some neutrons leak out of the reactor, and some neutrons are captured in the ^{238}U

resonances. The resonance escape probability, ρ is defined as:

The fraction of neutrons which escape capture in the ^{238}U resonances during slowing down in the reactor.

The fast non-leakage probability, P_{NLf} , is defined as the fraction of neutrons which do not leak out of the reactor during slowing down.

The number of neutrons which slow down and become thermal is $n\varepsilon\rho P_{\text{NLf}}$.

Once the slowing down process is complete and neutrons are thermalized they will continue to diffuse in the reactor until they are absorbed or leak out. Only a fraction of the absorption of neutrons is in the fuel, and some neutrons are captured in the moderator, coolant or structural materials of the core. The thermal non-leakage probability, P_{NLth} , is defined as the fraction of thermal neutrons which do not leak out of the reactor.

The thermal utilization factor, f , is defined as the fraction of thermal neutrons absorbed in the reactor, which are absorbed in the fuel.

It is evident that the number of neutrons which is absorbed in the fuel, is $n\varepsilon\rho f P_{\text{NLf}} P_{\text{NLth}}$. Not all these neutrons cause fission, in fact in natural uranium our earlier calculation indicates that about 55 percent of thermal neutrons absorbed in the fuel cause fission, and that the number of neutrons produced is about 1.33 per neutron absorbed in the fuel. This quantity, the average number of fission neutrons produced per neutron absorbed in the fuel, has the symbol η and is usually just called eta.

Finally, therefore, at the beginning of the second generation $n\varepsilon\rho f\eta P_{\text{NLf}} P_{\text{NLth}}$ fission neutrons are produced from n fission neutrons at the beginning of the first generation. From our definition of the multiplication factor, which is called more explicitly for a finite reactor the effective multiplication factor, k_{eff} we can write⁴:

$$k_{\text{eff}} = \varepsilon\rho f\eta P_{\text{NLf}} P_{\text{NLth}}$$

It is often convenient to consider an infinitely large reactor, as this enables us neglect neutron diffusion and leakage, and $P_{\text{NLf}} = P_{\text{NLth}} = 1$. The multiplication factor is then referred to as the infinite multiplication factor, k_{∞} , and is expressed by the important Four Factor Formula:

$$k_{\infty} = \varepsilon\rho f\eta$$

The relationship between the two multiplication factors is:

$$k_{\text{eff}} = k_{\infty} P_{\text{NLf}} P_{\text{NLth}}$$

It was pointed out that for certain types of reactor enriched uranium is necessary to achieve criticality. The most important example is the pressurized water reactor which requires slightly enriched uranium with 2 to 3 percent of ^{235}U . Some current British graphite moderated reactors use fuel of a similar composition. Uranium fueled fast reactors (of which very few have been built) require highly enriched uranium with 25 to 50 percent of ^{235}U .

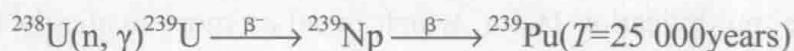
The process of enriching uranium involves a partial separation of the ^{235}U and ^{238}U so that the product has a higher concentration of ^{235}U than that of natural uranium. The waste, known as the tails, has a lower concentration. Two processes are available for uranium enrichment on

a commercial scale. In both of them; natural uranium is converted to the gaseous compound uranium hexafluoride UF_6 and the two isotopes of uranium produce two gases of slightly different density, the $^{238}\text{UF}_6$ being slightly more dense than the $^{235}\text{UF}_6$. Both processes make use of the slight difference in density to achieve separation of the isotopes. In the gaseous diffusion process the UF_6 gas diffuses through a series of semi-permeable membranes, and in the centrifuge process the UF_6 gas is spun at high rotational speed in a centrifuge.

4.3 Conversion and Breeding

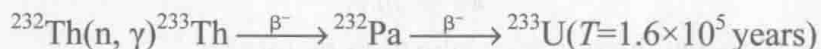
One important point to emerge from earlier section of this lesson is that in thermal reactors fueled with uranium, either natural or enriched, practically all the fission occurs in ^{235}U . In fast reactors, which contain no moderator and in which neutron energies are much higher than in thermal reactor, ^{238}U fission occurs to a small extent, but even in this type of reactor, it is ^{235}U fission which predominates and sustains the chain reaction. It is nearly correct to say, therefore, that only the ^{235}U in natural uranium contributes energy directly from its own fission.

Although ^{238}U cannot itself be used as the fuel in a nuclear reactor, it does have a vital role to play as an isotope from which new fissile fuel can be created⁵. In a uranium fueled reactor, a significant fraction of the neutrons produced by fission, possibly 30~40 percent, are captured in ^{238}U and produce ^{239}U by an (n, γ) reaction. ^{239}U is the start of a radioactive decay chain which produces ^{239}Np , also radioactive and ^{239}Pu , which has a very long half-life and almost be regarded as a stable isotope. The processes involved are:



^{239}Pu , as already pointed out, is fissile, its characteristics as far as fission is concerned are similar to those of ^{235}U , and it can be used as the fuel in both fast and thermal reactors.

Another isotope which has characteristics similar to ^{238}U is ^{232}Th , the only naturally occurring isotope of the element thorium. This isotope can only undergo fission with neutrons of energy greater than about 1.4MeV, so it cannot sustain a chain reaction and be used directly as a nuclear fuel. However, as a result of neutron capture in ^{232}Th the following processes take place:



^{233}U is fission with neutrons of all energies and like ^{235}U and ^{239}Pu it can be used as the fuel for nuclear reactors.

The importance of ^{238}U and ^{232}Th lies in their ability to act as fertile materials from which, as a result of neutron capture in nuclear reactors, the fissile isotopes ^{239}Pu and ^{233}U are produced. This process is known as breeding or conversion, and it provides the method whereby ^{238}U and ^{232}Th can be used as sources of energy through fission.

An important characteristic of a nuclear reactor, particularly one which is designed to produce new fissile material by one of the processes just described as well as power, is the ratio of the rates at which fertile material and fissile material are used up⁶. This ratio is the

breeding ratio B and is defined as the number of new fissile atoms produced in a reactor per atom of existing fissile fuel consumed by fission and neutron capture.

If the breeding ratio is exactly 1 then the quantity of fissile fuel remains constant, if it is greater than 1 the quantity of fissile fuel increases. And if it is less than 1 the quantity of fissile fuel decreases. In order to utilize all the world's resources of ^{238}U and ^{232}Th it is essential that some, though not necessarily all, of the world's reactors are designed so that their breeding ratios are greater than 1.

The following simplified argument illustrates this point. Consider a reactor (or a number of reactors) whose breeding ratio is B , fueled with natural uranium, the total mass of ^{235}U being M tons. If all the ^{235}U is burned up, the quantity of ^{238}U converted to ^{239}Pu is approximately BM tons. If this plutonium is used to provide the second charge of fuel for the reactors, and if the value of B is the same for a plutonium fueled reactor as for a uranium fueled reactor (which is not strictly true as we shall see), then the use of BM tons of ^{239}Pu results in another B^2M tons of ^{238}U being converted to ^{239}Pu , and so on. The total amount of uranium used up would be $M + BM + B^2M + \dots$. If $B < 1$, then this is equal to $M/(1-B)$. If this results were exact, which it is not, the value of B necessary to use up all the natural uranium originally in the reactors would be about 0.993, or for all practical purposes 1⁷. In fact it is necessary that the value of B should be slightly greater than 1 as the foregoing argument neglects the inevitable losses of plutonium and uranium during chemical processing, separation of fission products and manufacture of fuel elements.

If the value of B is much less than 1 then only a fraction of the available uranium is used up, for example if $B=0.75$, then theoretically 2.86 percent of the natural uranium can be used, still a very small fraction which in practice would be even smaller.

The dependence of the breeding ratio on other reactor parameters can be deduced from the following argument. When a neutron is absorbed in an atom of fissile fuel, that atom is consumed (in the sense used in the definition of the breeding ratio) and η_f neutrons are produced, where η_f is the average number of neutrons produced per neutron absorbed in the fissile fuel. For a steady chain reaction in a reactor one of these neutrons must be absorbed in another atom of fissile fuel to keep the reaction going, and according to the definition of the breeding ratio, B neutrons must be captured in the fertile material. Some neutrons will inevitably be captured in non-fuel materials, and some will leak out of the reactor. The sum of these two processes is represented by $(C+L)$ neutrons per neutron absorbed in the fissile fuel. These processes are shown in Fig.4.1. It is clear that in order to maintain a steady chain reaction the following items should be met.

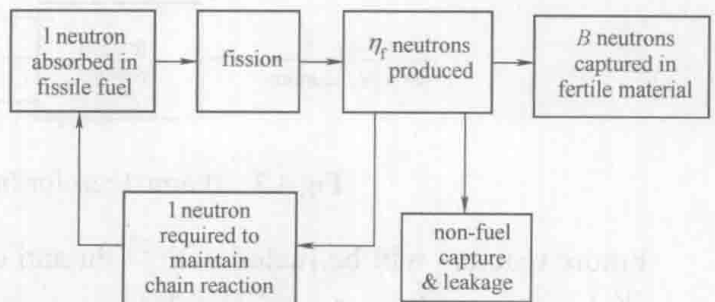


Fig.4.1 neutron cycle to illustrate breeding

Clearly if B is to be greater than 1, η_f must exceed 2 by an amount that allows for the term $(C+L)$, whose value is likely to be about 0.2. Values of η_f for the three fissile isotopes for fission caused by thermal and high energy neutrons are given in Table 4.1.

It will be seen from Table 4.1 that, allowing for reasonable non-fuel neutron capture and leakage, only a ^{233}U fueled thermal reactor can achieve a breeding ratio greater than 1. In fast reactors there is a considerable improvement, and reactors of this type are capable of giving breeding ratios greater than 1. The value of B in a fast reactor is actually greater than that given by equation due to the effect of fast fission in ^{238}U or ^{232}Th , which is very slight in a thermal reactor but may be considerable in a fast reactor in which as much as 20 percent of the fission may be in the ^{238}U (less in ^{232}Th).

Table 4.1 typical values of η_f for the fissile isotopes

Isotope	Energy of neutrons causing fission	
	Thermal (0.025eV)	High energy (0.5MeV)
^{235}U	2.07	2.35
^{239}Pu	2.15	2.90
^{233}U	2.29	2.40

4.4 Fuel Cycles

Fig.4.2 shows a thermal reactor fueled initially with natural or enriched uranium, it is typical of the vast majority of reactors built up to the present. The breeding ratio of such a reactor is less than 1 so the amount of ^{239}Pu produced is less than the amount of ^{235}U used. When the reactor is refueled, although the ^{239}Pu might be recycled with the depleted uranium (which contains much less than 0.715 percent of ^{235}U), some additional fissile fuel is needed to make up the deficit. Practically all the plutonium produced in this way has been stockpiled for nuclear weapons and future reactor, so that up to the present nearly all nuclear reactors have been fueled with natural or enriched uranium and stocks of plutonium and depleted uranium are steadily accumulating in several countries.

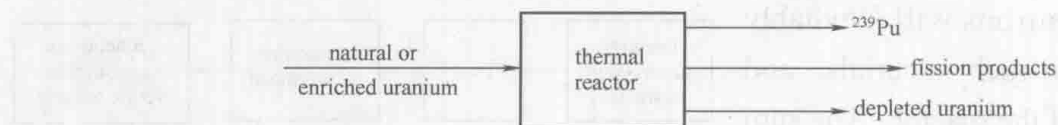


Fig.4.2 thermal reactor fueled with uranium

Future reactors will be fueled with ^{239}Pu and depleted uranium, which can be regarded as pure ^{238}U , and in such reactors ^{238}U will be converted to ^{239}Pu , but only in fast reactors will the breeding ratio be greater than 1 and in such reactors more ^{239}Pu will be produced than is destroyed. Fig.4.3 shows the fuel cycle of a fast reactor fueled with ^{239}Pu and ^{238}U in which the

breeding ratio is greater than 1.

When the first fuel charge is unloaded from the reactor it contains more ^{239}Pu and less ^{238}U than it did when new, and it is contaminated with fission products which must be removed during reprocessing. Some ^{239}Pu is available for other uses such as fueling thermal reactors, while the rest recycled. The ^{238}U is recycled but there is less of this isotope than in the original charge, so an additional supply ^{238}U is required for the second and subsequent fuel loadings. The important point about this fuel cycle is that the reactor is kept going by fresh supplies of non-fissile ^{238}U , and in this way ^{238}U can be completely converted to ^{239}Pu and used as a source of energy.

If the reactor of Fig.4.3 had been a thermal reactor in which the breeding ratio is less than 1, it would require a continuous supply of ^{239}Pu as well as ^{238}U to keep it going. It is obvious that in a planned long-term nuclear power program a combination of fast and thermal reactors will be able to utilize all the world's resources of uranium.

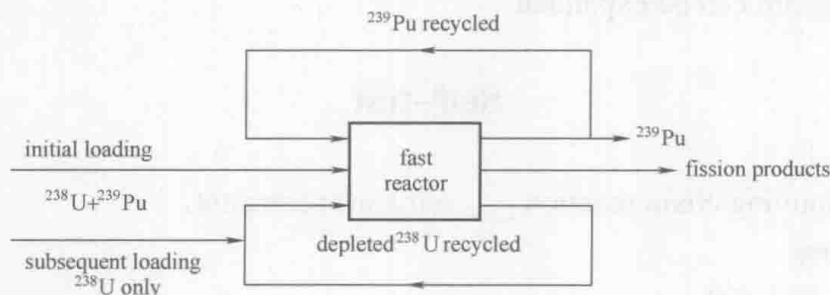


Fig.4.3 fuel cycle for a fast reactor fueled with ^{239}Pu and ^{238}U

Similar ideas apply to reactors making use of the $^{232}\text{Th} - ^{233}\text{U}$ breeding process. A reactor fueled initially with either ^{235}U or ^{239}Pu as the fissile material, and ^{232}Th as the fertile material will produce ^{233}U . When sufficient ^{233}U is produced it can be used as the fuel charge in either a fast or thermal reactor with ^{232}Th , and in such a reactor a breeding ratio greater than 1 is possible. In such a reactor the ^{233}U can be recycled, and the only fuel requirement is ^{232}Th , which can thus be completely used for the production of energy.

In reactors which are designed for breeding as well as power production the core may be subdivided into two regions. In this type of reactor the inner region, called the core, contains nearly all the fissile material (^{239}Pu), and it is in this part of the reactor that most of the energy is released by fission. The outer region is called the blanket and contains the fertile material (^{238}U). At the start of the reactor's life there is very little fission in the blanket, however neutrons produced by fission in the core may diffuse into the blanket and be captured in ^{238}U to produce ^{239}Pu . As the operation of this reactor proceeds, fissile material builds up in the blanket and provision must be made to remove the energy released by fission from the blanket as well as from the core.

Another point worthy of mention is that in any reactor designed for breeding, neutron capture in the moderator, structural materials, etc., and neutron leakage from the core should

be reduced to a minimum. This capture and leakage is the term $(C+L)$, and it is clear from this equation that any increase of the term $(C+L)$ reduces the possible value of the breeding ratio. As an example of this, it is generally true to say that the breeding ratio in a water-moderated reactor is less than in a graphite or heavy water-moderated reactor due to the rather high capture cross-section of water. In a reactor with a reasonably high value of the breeding ratio (say between 0.8 and 1.0) new fissile fuel is being produced almost as fast as it is being consumed, and the fuel in such a reactor can be used to a very high burn-up, thus prolonging the periods between refueling and reducing fuel costs.

In an expanding program of nuclear power using breeder reactors to produce new fuel for later reactors, an important parameter is the doubling time, T_d , which is defined as the time required for the quantity of fissile fuel in a breeder reactor to double itself. Clearly, it can be interpreted as the time required for a breeder reactor to produce enough new fuel to provide the first fuel charge for another identical reactor, and this time will control the rate at which a breeder reactor program can be expanded.

Self-test

- In self-sustaining chain reaction _____ remain(s) constant.
 - the neutrons
 - the fissile material
 - the neutron density and fission rate
 - the fission rate
- If the multiplication factor is less than one the reactor is _____.
 - subcritical
 - critical
 - explosive
 - supercritical
- The fraction of thermal neutrons absorbed in the reactor which are absorbed in the fuel is defined as _____.
 - the multiplication factor
 - thermal utilization factor
 - the fast fission factor
 - non-leakage factor
- In the PWR _____ is necessary to achieve critical.
 - uranium
 - natural uranium
 - high enriched uranium
 - slightly enriched uranium
- The fast reactors contain no _____.
 - uranium
 - moderator
 - coolant
 - thermal neutron
- ^{239}Pu is fissile, its characteristics as far as fission is concerned are _____ those of ^{235}U .
 - similar to
 - better than
 - less than
 - more than
- ^{238}U and ^{232}Th are good _____.
 - similar to
 - better than
 - less than
 - more than

- A. fissile materials B. fertile materials
C. moderators D. structural materials
8. If the breeding ratio is much less than 1, then ____ is used up.
A. most of the available uranium B. no uranium
C. a lot of uranium D. only a fraction of the available uranium
9. In fast reactor the breeding ratio is ____ than thermal reactor.
A. no more B. greater C. less D. rather
10. When the first fuel charge is unloaded from the reactor it contains ____ than it had loaded.
A. more ^{239}Pu B. more ^{235}U C. more ^{238}U D. less ^{239}Pu
11. In the case of power reactor it is necessary to circulate ____ through the reactor core to remove the energy released by fission.
A. water B. moderator C. a coolant D. fluid
12. In a uranium fueled reactor a significant fraction of neutrons are ____.
A. captured in ^{238}U B. escaped from reactor
C. captured in ^{239}Pu D. useless

关键字解释

Blanket: Region of fertile material surrounding reactor core in which the neutrons leaking from the core breed more fissile fuel, e.g. ^{233}U from thorium.

再生区: 在反应堆堆芯周围放置可转换材料的区域, 由堆芯泄漏出的中子在这里增殖出更多的易裂变材料, 例如由钍变成 ^{233}U 。

Breeding ratio: In a reactor, a comparison between the amount of the new fissionable fuel produced and the amount consumed during fission.

增殖比: 在核反应堆中, 新产生的易裂变燃料量与裂变过程中消耗的燃料量的比值。

Chain reaction: A reaction which, once “triggered”, continue to supply enough energy and particles to continue the reaction.

链式反应: 这种反应一旦触发, 会连续的提供足够的能量和粒子以继续进行反应。

Critical size: the minimum size of a nuclear reactor core required to sustain a chain reaction.

临界尺寸: 能够维持链式反应的最小核反应堆堆芯尺寸。

Criticality: State in nuclear reactor when multiplication factor for neutron flux reaches unity and external neutron supply is no longer required to maintain power level, i.e. chain reaction is self-sustaining.

临界: 核反应堆中子通量的增殖系数达到 1, 且不再需要外加中子源来维持功率水平的状态, 即此时链式反应可以自持。

Depleted uranium: Sample of uranium having less than its natural content of ^{235}U .

贫铀: 其 ^{235}U 含量低于天然铀中 ^{235}U 含量的铀。

Enriched uranium: Uranium in which the proportion of the fissile isotope ^{235}U has been increased above its natural abundance.

浓缩铀: 可裂变同位素 ^{235}U 所占的比例增加到它的天然丰度以上的铀。

Fertile material: The isotope in a nuclear reactor which can be converted by the capture of a neutron into fissile isotope, e.g. ^{238}U is converted by series of reactions into ^{239}Pu .

增殖材料: 同位素中可以通过在核反应堆中俘获一个中子转变为易裂变同位素的材料, 例如 ^{238}U 通过一系列反应转换成 ^{239}Pu 。

Fission neutron: Those released by nuclear fission, having a continuous spectrum of energy.

裂变中子: 核裂变释放的, 具有连续能谱的中子。

Fuel cycle: The stages involved in the supply and use of fuel in nuclear power plant. The main steps are mining, milling, extraction, purification, enrichment, fuel fabrication, irradiation in the reactor, cooling, reprocessing, recycling, and waste management and disposal.

燃料循环: 指核动力装置中燃料供应和使用的全部过程。主要涉及的步骤为采矿、矿石加工、提炼、精制、富集、燃料制造、堆内辐照、冷却、后处理、再循环和废物管理及处置。

Multiplication factor: The ratio of the average number of neutrons produced by fission in one neutron lifetime to the total number of neutrons absorbed and leaking out in the same interval.

倍增系数: 在中子的寿命中裂变产生的平均中子数与在同一时期内被吸收和泄漏出的中子数之比。

Thermal neutrons: Neutrons of a very low speed and energy; their energy is of the same order as that of the substance through which they are passing: they are responsible for various nuclear action.

热中子: 速度和能量都很低的中子; 其能级与它们所通过的物质的能级相同; 它们可以引起各种核反应。

Thermal reactor: One for which the fission chain reaction is propagated mainly by thermal neutrons and therefore contains a moderator.

热堆: 裂变链式反应主要是由于热中子引起的反应堆, 所以一定要有慢化剂。

词 汇

argument ['ɑ:gjʊmənt] *n.* 论证, 讨论

assembly [ə'sembli] *n.* 组合, 组合件; 组件, 系统

breeding ['bri:diŋ] *n.* 增殖

centrifuge ['sentrifju:dʒ] *n.* 离心

collision [kə'liʒən] *n.* 碰撞

composition [kəm'pə'ziʃən] *n.* 组成, 结构

contaminate [kən'tæmineit] *v.* 污染

critical ['kritikəl] *a.* 临界的

- deficit ['defisit] *n.* 亏损
- deplete [di'pli:t] *n.* 用尽
- diffuse [di'fju:z] *n.* 扩散
- elastic [i'læstik] *a.* 弹性的
- evident ['evidənt] *a.* 明显的
- explicitly [iks'plisitli] *ad.* 明显地, 清楚地
- fertile ['fə:tail] *a.* 可变成裂变物质的, 增殖的
- gaseous ['geizjəs] *a.* 气体的
- graphite ['græfait] *n.* 石墨
- hexafluoride *n.* 六氟化物
- infinite ['infini:t] *a.* 无限的
- membrane ['membrein] *n.* 薄膜
- multiplication [mʌltipli'keiʃən] *n.* 倍增
- neptunium [nep'tju:njəm] *n.* 镎
- predominant [pri'dominənt] *a.* 主要的
- probability [ˌprɒbə'biliti] *n.* 概率
- refuel [ˌri:'fjuəl] *v.* 换燃料
- reprocessing [ˌri:'prəusesiŋ] *n.* 再加工, 后处理
- resonance ['rezənəns] *n.* 共振
- scattering ['skætəriŋ] *n.* 散射
- self-sustaining *a.* 自持的
- semi-permeable *a.* 半渗透的
- spun [spʌn] *v.* spin 的过去分词, 旋转
- stockpile ['stɒkpail] *n. & v.* 储存, 堆放, 储存
- subcritical [sʌb'kritikəl] *a.* 次临界
- supercritical [ˌsju:pə'kritikəl] *a.* 超临界
- sustain [səs'tein] *v.* 维持
- tail [teɪl] *n.* 尾渣
- thermalize ['θə:məlaiz] *v.* 使热能化
- waste ['weist] *n.* 废物
- emerge from 出现
- die out 消失, 灭绝

课文注释

1. The achievement of..., as this is the...to good use.

这里 *as* 是因为的意思, 它引出一个状语从句, 进一步说明主句。整句译为: 一个系统达到可控的、自持的裂变链式反应是首先的要求, 因为这样裂变能才能被可控地释放出来, 并进行很好的利用。

2. The condition...and so on.

句中前一个 **that** 引导一个定语从句, 后一个 **that** 引导一个表语从句, 其中 **proceeds** 的主语是 **one**。整句译为: 稳定、自持的链式反应的必要条件是: 一次裂变产生的中子中的一个要引起第二次裂变, 第二次裂变产生的中子再引起第三次裂变并持续下去。

3. Accurate calculation...between the time...and time...from the reactor.

这里 **between** 引出的介词短语做 **neutrons** 的定语, 是指在裂变产生和最后消失两个时间之间的中子。整句译为: 任一个反应堆倍增系数的精确计算, 必须完全考虑所有从裂变产生中子开始、一直到这些中子最后被吸收或从反应堆泄漏而消失的整个过程中所涉及的中子。

4. From our definition...we can write:

句中 **for a finite reactor** 是介词短语, 在从句中作条件状语, **the effective multiplication factor** 是从句的宾语。整句译为: 根据我们对倍增系数的定义, 对一个有限的反应堆, 倍增系数应更确切的称为有效倍增系数 k_{eff} , 我们可以写出:

5. Although ^{238}U cannot...can be created.

这里 **does have** 是强调语气, **a vital role to play** 是指发挥着非常重要的作用。整句译为: 虽然 ^{238}U 本身不能用来作为核反应堆的燃料, 但它确实是一个发挥着重要作用的同位素, 由它可产生新的可裂变燃料。

6. An important...used up.

句中 **which** 引导定语从句修饰前面的 **one**; **particularly...as power** 这部分是 **nuclear reactor** 的补语, 进一步说明是哪一种反应堆。整句译为: 增殖材料和裂变材料消耗速率的比值, 是核反应堆的一个重要特性, 特别是那种用前面讲过的方法产生新裂变材料以及动力的反应堆的一个重要特性。

7. If this results...purpose 1.

句中 **which it is not** 是插入语, 说明这个结果实际上是不精确的; **for all practical purposes** 是词组, 表示实际上。整句译为: 如果这一结果是精确的话, (但实际上不精确) 反应堆中所有天然铀被用尽所需要的 B 值应该大约是 0.993, 或实际上是 1。

Unit 5 Nuclear Reactor Concepts

5.1 Power Reactors

The large-scale reactors used for the production of thermal energy that is converted to electrical energy are much more complex than the fast assembly. To illustrate, we can identify the components and their functions in a modern pressurized water reactor. Fig.5.1 gives some indication of the sizes of the various parts.

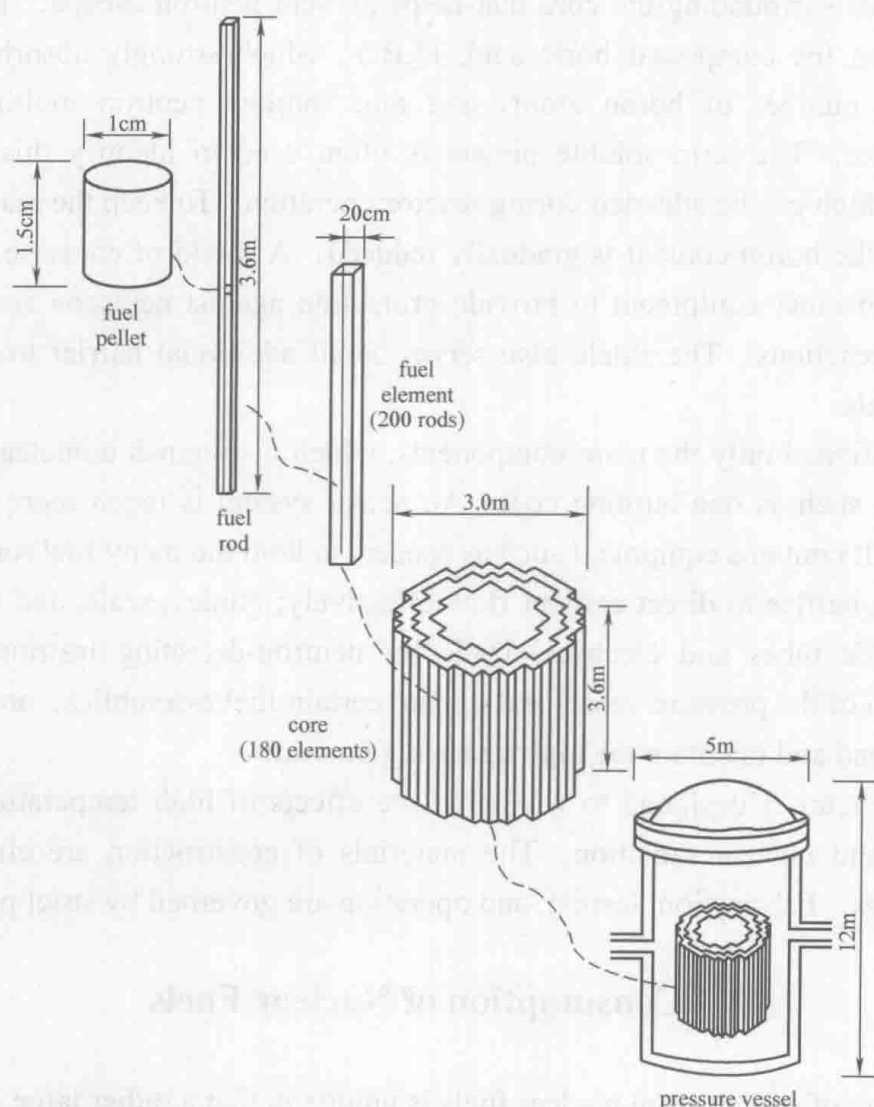


Fig.5.1 reactor construction

The fresh fuel installed in a typical PWR consists of cylindrical pellets of slightly enriched (3% ^{235}U) uranium oxide (UO_2) of diameter about 1cm and length about 1.5cm. A zircaloy tube of wall thickness 0.6mm is filled with the pellets to an “active length” of 365cm and sealed to form a fuel rod(or pin). The metal container serves to provide support for the

column of pellets, to provide cladding that retains radioactive fission products, and to protect the fuel from interaction with the coolant. About 200 of the fuel pins are grouped in a bundle called a fuel assembly of about 20cm on a side, and about 180 assemblies are assembled in an approximately cylindrical array to form the reactor core. This structure is mounted on supports in a steel pressure vessel of outside diameter about 5m, height 12m and walls up to 30cm thick. Control rods, consisting of an alloy of cadmium, silver, and indium, provide the ability to change the amount of neutron absorption. The rods are inserted in some vacant fuel pin spaces and magnetically connected to drive mechanisms. On interruption of magnet current, the rods enter the core through the force of gravity. The pressure vessel is filled with light water, which serves as neutron moderator, as coolant to remove fission heat, and as reflector, the layer of material surrounding the core that helps prevent neutron escape. The water also contains in solution the compound boric acid, H_3BO_3 , which strongly absorbs neutrons in proportion to the number of boron atoms and thus inhibits neutron multiplication, i.e., "poisons" the reactor. The term soluble poison is often used to identify this material, the concentration of which can be adjusted during reactor operation. To keep the reactor critical as fuel is consumed, the boron content is gradually reduced. A shield of concrete surrounds the pressure vessel and other equipment to provide protection against neutrons and gamma rays from the nuclear reactions. The shield also serves as an additional barrier to the release of radioactive materials.

We have mentioned only the main components, which distinguish a nuclear reactor from other heat sources such as one burning coal. An actual system is much more complex than described above. It contains equipment such as spacers to hold the many fuel rods apart; core support structures; baffles to direct coolant flow effectively; guides, seals, and motors for the control rods; guide tubes and electrical leads for neutron-detecting instruments, brought through the bottom of the pressure vessel and up into certain fuel assemblies; and bolts to hold down the vessel head and maintain the high operating pressure.

The power reactor is designed to withstand the effects of high temperature, erosion by moving coolant, and nuclear radiation. The materials of construction are chosen for their favorable properties. Fabrication, testing, and operation are governed by strict procedures.

5.2 Consumption of Nuclear Fuels

The generation of energy from nuclear fuels is unique in that a rather large amount of fuel must be present at all times for the chain reaction to continue. (In contrast, an automobile will operate even though its gasoline tank is practically empty.) There is a subtle relationship between reactor fuel and other quantities such as consumption, power, neutron flux, criticality, and control.

The first and most important consideration is the energy production, which is directly related to fuel consumption. Let us simplify the situation by assuming that the only fuel

consumed is ^{235}U , and that the reactor operates continuously and steadily at a definite power level. Since each atom “burned,” i.e., converted into either ^{236}U or fission products by neutron absorption, has an accompanying energy release, we can find the amount of fuel that must be consumed in a given period.

Let us examine the fuel usage in a PWR with initial enrichment in ^{235}U of 3%. Suppose that the thermal power is 3000MW and the reactor operates for 1 yr. Using the convenient rule of thumb that 1.3g of ^{235}U is burned for each megawatt-day of energy, the weight of fuel used is

$$3000\text{MW} \times 365\text{d} \times 1.3\text{g}/(\text{MW} \cdot \text{d}) = 1.4 \times 10^6\text{g}$$

Now each gram of ^{235}U at 3% enrichment costs around \$25. The cost of the fuel consumed is around 33 million dollars. Adding the expense of fuel fabrication and storage brings the total to around 45 million dollars. A typical efficiency of conversion of thermal energy to electrical energy is 1/3, so the electrical power is 1000MW. Over the year (8760h) the energy delivered to the customers is $8.76 \times 10^9\text{kWh}$, and thus the fuel cost is, per kilowatt hour, \$0.0051, 0.51 ¢ or 5.1 mills. These figures are rough because the effect of generation and consumption of fissile plutonium has been ignored.

5.3 Reactor Control

Since no fuel is added during the operating cycle of the order of a year, the amount to be burned must be installed at the beginning. First, the amount of uranium needed to achieve criticality is loaded into the reactor. If then the “excess” is added, it is clear that the reactor would be supercritical unless some compensating action were taken¹. In the PWR, the excess fuel reactivity is reduced by the inclusion of control rods and boron solution.

The reactor is brought to full power and operating temperature and pressure by means of rod position adjustments. Then, as the reactor operates and fuel begins to burn out, the concentration of boron is reduced. By the end of the cycle, the extra fuel is gone, all of the available control absorption has been removed, and the reactor is shut down for refueling. The trends in fuel and boron are shown in Fig.5.2, neglecting the effects of certain fission product absorption and plutonium production. The graph represents a case in which the power is kept constant. The fuel content thus linearly decreases with time. Such operation characterizes a reactor that provides a “base load” in an electrical

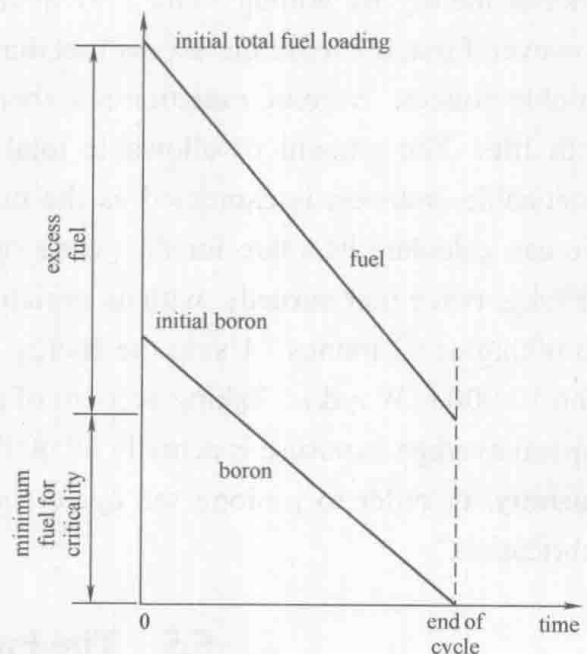


Fig.5.2 reactor control during fuel consumption in power reactor

generating system that also includes fossil fuel plants and hydroelectric stations.

5.4 Other Effects on Reactor Operation

The description of the reactor process just completed is somewhat idealized. Several other phenomena must be accounted for in design and operation.

If a reactor is fueled with natural uranium or slightly enriched uranium, the generation of plutonium tends to extend the cycle time. The fissile ^{239}Pu helps maintain criticality and provides part of the power. Small amounts of higher plutonium isotopes are also formed: ^{240}Pu , fissile ^{241}Pu (14.355 year half life), and ^{242}Pu . These isotopes and those of elements farther up the periodic table are called transuranic materials or actinides. They are important as fuels, poisons, or nuclear wastes.

Neutron absorption in the fission products has an effect on control requirements. The most important of these is a radioactive isotope of xenon, ^{135}Xe , which has a cross section at 0.025 3eV of around 2.6 million barns. Its yield in fission is high, $y=0.06$, meaning that for each fission, one obtains 6% as many atoms of ^{135}Xe . In steady operation at high neutron flux, its rate of production is equal to its consumption by neutron absorption. Hence

$$N_X\sigma_{aX} = N_F\sigma_{fY}$$

Using the ratio σ_f/σ_a for ^{235}U of 0.86, we see that the absorption rate of ^{135}Xe is around $(0.86)(0.06)=0.05$ times that of the fuel itself. This factor is about 0.04 if the radioactive decay ($t_H=9.10\text{hr}$) of ^{135}Xe is included.

It might appear from Fig.5.2 that the reactor cycle could be increased to as long a time as desired merely by adding more ^{235}U at the beginning². There are limits to such additions, however. First, the more the excess fuel that is added, the greater must be the control by rods or soluble poison. Second, radiation and thermal effects on fuel and cladding materials increase with life. The amount of allowable total energy extracted from the uranium, including all fissionable isotopes, is expressed as the number of megawatt-days per metric ton(MW • d/t). We can calculate its value for the year's operation by noting that the initial ^{235}U loading was 2800kg, twice that burned; with an enrichment of 0.03, the uranium content was $2800/0.03 = 93\,000\text{kg}$ or 93 tonnes. Using the energy yield of $(3000\text{MW})(365\text{d}) \cong 1\,100\,000\text{MW} \cdot \text{d}$, we find $12\,000\text{MW} \cdot \text{d/t}$. Taking account of plutonium and the management of fuel in the core, a typical average exposure is actually $30\,000\text{MW} \cdot \text{d/t}$. It is desirable to seek larger values of this quantity, in order to prolong the cycle and thus minimize the cost of fuel reprocessing and fabrication.

5.5 The Fast Breeder Reactor

Several fast reactors have been operated successfully throughout the world. In the United States, the Experimental Breeder Reactor was the first power reactor to generate electricity; the

Fermi reactor operated for some time; the Fast Flux Test Facility (FFTF) provides heat but no electricity. The Liquid Metal Fast Breeder Reactor (LMFBR) has been under design and construction for many years in a program entitled Clinch River Project located near Knoxville, Tenn³.

The use of liquid sodium ^{23}Na as coolant ensures that there is little neutron moderation in the fast reactor. The element sodium melts at 98°C , boils at 883°C , and has excellent heat transfer properties. With such a high melting point, pipes containing sodium must be thermally insulated and heated electrically to prevent freezing. The coolant becomes radioactive as the result of neutron absorption, producing the ^{24}Na . Great care must be taken to prevent contact between sodium and water or air, which would result in a serious fire, accompanied by the spread of radioactivity. To avoid such an event, an intermediate heat exchanger is employed. Heat is transferred in it from radioactive sodium to nonradioactive sodium.

The French fast reactor Phoenix at Marcoule is one of the most successful commercial breeder reactors. ⁴We shall describe its features in some detail. It is of the pot type, with a tank of 12.75m diameter. The reactor has a core of height 85cm and diameter 139cm, fueled with a mixture of UO_2 and PuO_2 in the form of 0.55cm diameter pellets encased in stainless steel tubes of wall thickness 0.045cm. Bundles of 217 fuel pins form a fuel element, of which there are 103 total in the core. There are five control rods and three safety rods composed of B_4C . Above and below the core and surrounding it is the “blanket” containing depleted uranium in which neutron absorption occurs to produce new plutonium. The design power rating of Phoenix is 250MWe(563MWt). The linear heat rate 430W/cm is much higher than for a PWR. Similarly, since the reactor operates with fast neutrons, the flux is high, $6.7 \times 10^{15}/(\text{cm}^2 \cdot \text{s})$. The coolant temperature rises from 400°C to 560°C on passing through the reactor at a speed of 7.4m/s, flow rate 25kg/s. The energy is supplied to a turbine with steam at 510°C and pressure $163\text{kg}/\text{cm}^2$. The reactor is shut down every 6 months for refueling. Plutonium produced in the blanket is extracted by reprocessing for use as new core fuel.

Although the principal attention throughout the world has been given to the liquid metal cooled fast breeder using U and Pu, other breeder concepts remain of interest. Included are: ①uranium and thorium fuel particles suspended in heavy water; ②fuel and fertile elements as fluoride compounds mixed with other salts in molten form; ③a high-temperature gas-cooled graphite moderated reactor containing also a compound of beryllium.

5.6 Uranium as an Energy Resource

The role which uranium can play in contributing to the world's energy needs is potentially very great, and the development of nuclear power will only be justified in the long run if uranium and other fissile fuels are able to make a substantial contribution to energy supplies for many decades to come⁵. This will depend on at least three factors: the extent of uranium resources which can be used, the way in which uranium is used in breeder and non-breeder

reactors, and public acceptance of nuclear power.

Uranium, in the form of a mixture of oxides whose approximate formula is U_3O_8 , is found quite widely in the earth's crust in very small concentrations. The richest ores contain 0.05 to 0.2 percent U_3O_8 , and those which are at present being exploited produce low cost uranium at prices less than \$80/kgU. Much poorer ores in shales and granites containing less than 0.01 percent U_3O_8 will then have to be exploited, and the price of uranium will then become much higher than the quoted above. The environmental impact of processing huge quantities of rock and shale to obtain relatively small quantities of uranium will be considerable. Uranium is also present in minute quantities in sea water, but it is doubtful if it will ever be economically justifiable to exploit this source.

Uranium resources are classified according to their production costs (\$/kgU) and the levels of confidence in the resource data. Reasonably assured resources (RAR) refer to known deposits of uranium ores that can be recovered by proven technology. Estimated additional resources (EAR) refer to quantities which are expected to exist as an extension to existing deposits (EAR-1) or in other areas of mineralization which have not been fully explored (EAR-2)⁶. Finally, speculative resources (SR) refer to deposits of uranium ores that are thought to exist on the basis of extrapolation and indirect evidence.

The total of 3.65 million tonnes of uranium which includes RAR and EAR-1 up to costs of \$130/kgU are regarded as the resource which the nuclear industry can be confident will be available for the thermal reactor programme of the next few decades.

Uranium is comparable with the world's proved reserves of oil, and slightly greater than the world's proved reserves of natural gas. These two fossil fuels will be considerably depleted within 50 years at the present rate of consumption. Similarly, the uranium in categories RAR and EAR-1 at costs up to \$130/(kgU) will be nearly used up within 50 years.

On the other hand, if breeder reactors are successfully introduced on a large scale, without posing any serious operational or safety problems, then uranium has enormous potential, and will be capable of providing energy for many centuries. The large existing stockpiles of depleted uranium, i. e. the tails from enrichment plants and spent fuel from thermal reactors, will with ^{239}Pu become the fuel for future breeder reactors.

The justification, therefore, for the development and use of nuclear power is that in breeder reactors uranium will be capable of providing energy for many centuries. This will not be without its dangers, for the widespread use of nuclear power and the production of plutonium on a large scale have several hazardous aspects as were demonstrated at the serious accidents at nuclear power stations at Windscale, UK in 1957, Three Mile Island, USA in 1979 and Chernobyl, USSR in 1986, and these hazards will be described in a later lesson⁷. As was stated at the beginning of this section, the use of nuclear power on a large scale will depend on public acceptance of this source of energy, with its several hazards, and it may well be that public opinion will have the final say in the determination of the future of nuclear energy.

Self-test

1. A thin zircaloy tube is filled with____.
A. fuel pellets and helium B. pellets
C. helium D. uranium
2. The reactor core consists of____.
A. about 200 fuel pins B. cylindrical array
C. about 180 fuel rods D. about 180 bundles
3. The flow direction of reactor coolant can be changed through____.
A. guides B. baffles C. fuel pin D. fuel elements
4. In a power reactor the energy production depends on____.
A. the scale B. the fuel consumption
C. how much the fuel pin D. moderator
5. From the text, the time of the operating fuel cycle is about____.
A. ten years B. several months C. one year D. several years
6. The breeder reactor has the ability to use____.
A. all of the uranium B. ^{235}U
C. ^{238}U D. both ^{235}U and ^{238}U
7. The fission products can____neutrons.
A. emit B. capture C. produce D. absorb
8. The liquid sodium is a good____in the reactor.
A. moderator B. fertile material C. coolant D. blanket
9. Above and below the Phoenix core is____.
A. fertile material B. reflector
C. water D. moderator
10. Uranium is____quantities in sea water.
A. not present any B. present large
C. present small D. some
11. The resource of uranium is____the world's proved reserves of oil.
A. slightly less than B. greater than
C. exact same as D. similar to
12. The depleted uranium is valuable in____.
A. PWR B. heavy water reactor
C. a breeder blanket D. tested reactor

关键字解释

Base load: That part of the total load on an electrical power system which is applied,

where possible, by the most efficient connected generating station, the remaining peak load being supplied intermittently by the more expensive stations.

基本负荷: 电力系统的总负荷中由效率最高的发电厂供给的那一部分负荷, 剩下的高峰负荷则间断地由较不经济的发电厂供给。

Breeder reactor: One which produces more fissile material than is consumed in operation. Fast reactor can be so designed.

增殖反应堆: 一种产生的裂变材料量多于消耗量的反应堆。快中子反应堆可设计成增殖堆。

Fast neutron: A neutron that has an energy level higher than 100 000 electron volts.

快中子: 能量大于 0.1MeV 的中子。

Fast reactor: One reactor without moderator in which chain reaction is maintained almost entirely by fast fission.

快中子反应堆: 没有慢化剂, 几乎全部由快中子裂变维持链式反应的反应堆。

Fission Products: Radioactive atoms result from nuclear fission. They have masses of roughly half of that of the fissile nucleus.

裂变产物: 核裂变产生的放射性原子。它们的质量大约是原裂变核的一半。

Reactor: Nuclear reactor, pile, a device in which controllable chain reactions take place to produce nuclear energy, it contains the fuel which undergoes fission when bombarded by neutrons and which then produces more neutrons to maintain the chain reaction.

反应堆: 核反应堆; 也称堆; 可控制的链式反应在其中产生核能的装置; 反应堆含有经中子轰击后发生裂变产生更多的中子的燃料从而使链式反应持续下去。

Reflector: A scattering substance surrounding the core of a nuclear reactor, used for reducing the loss of neutrons due to leakage, and thus making the critical dimensions of the reactor smaller.

反射层: 散布于核反应堆堆芯周围的一种物质, 用于减少泄漏的中子量, 因此也缩小了反应堆的临界尺寸。

Shield: Screen used to protect persons or equipments from γ -rays, neutrons etc. In a nuclear reactor the shield surrounds it to prevent the escape of neutrons and radiation into a protected area.

屏蔽: 使人和设备与 γ 射线、中子等隔开的屏障。在核反应堆中围绕它的屏蔽是为防止中子和辐射进入被保护区。

Spent fuel: Reactor fuel elements which must be replaced due to bursting, burn-up or depletion, poisoning by fission fragments, or swelling. The fissile material is not exhausted and so-called spent fuel is normally subsequently reprocessed.

乏燃料: 由于破裂、燃耗或贫化、裂变产物的中毒或肿胀而必须更换的反应堆燃料元件。在这种燃料中易裂变燃料并没有耗尽, 所谓的乏燃料通常要被再处理。

词 汇

actinides ['æktinaɪdɪz] *n.* 锕系元素

- array [ə'rei] *n.* 组列, 阵列
- assembly [ə'sembli] *n.* 组合, 组合件, 组件, 系统
- baffle ['bæfl] *n.* 隔板
- barrier ['bæriə] *n.* 屏障
- bundle ['bʌndl] *n.* 棒束, 组件
- concrete ['kɒnkri:t] *n.* 混凝土
- crust [krʌst] *n.* 地壳
- deposit [di'pɒzit] *n. & v.* 储存, 堆焊、覆盖
- diluent ['diljuənt] *n.* 稀释剂
- drive mechanism (控制棒) 驱动机构
- entitle [in'taitl] *v.* 叫作
- erosion [i'rəʊʒən] *n.* 侵蚀, 腐蚀, 削弱
- evidence ['evidəns] *n.* 迹象, 证据
- extrapolation [ekstrəpəu'leiʃən] *n.* 外推
- fluoride ['fluəraɪd] *n.* 氟化物
- granite ['grænit] *n.* 花岗岩
- hydroelectric [haɪdrəi'lektrɪk] *a.* 水电的
- inhibit [in'hɪbɪt] *v.* 防止, 防腐蚀
- insulate ['ɪnsjuleɪt] *v.* 使绝热, 隔离
- justifiable ['dʒʌstɪfaɪəbl] *a.* 合理的, 正当的
- microgram ['maɪkrəgræm] *n.* 微克
- mill [mɪl] *n.* 密尔, 千分之一美元
- mineralization [,mɪnərəlaɪ'zeɪʃən] *n.* 矿物, 矿化作用
- ore [ɔ:] *n.* 矿
- pin [pɪn] *n.* 柱, 棒
- pose [pəʊz] *n. & v.* 姿态; 造成
- pot type 池式
- reasonable ['ri:znəbl] *a.* 适当的
- shale [ʃeɪl] *n.* 页岩
- spacer ['speɪsə] *n.* 定位架
- speculative ['spekjʊlətɪv] *a.* 推测的
- stockpile ['stɒkpɪl] *n. & v.* 储存, 堆放, 储存
- substantial [səb'stænʃəl] *a.* 实质的
- subtle ['sʌtl] *a.* 微妙的, 稀薄的
- suspend [səs'pend] *v.* 悬浮
- transuranic [trænsju'rænik] *a.* 超铀的, 原子序数高于铀的
- turbine ['tɜ:bin] *n.* 透平, 汽轮机
- vacant ['veɪkənt] *a.* 空的
- xenon ['zenɒn] *n.* 氙

hold...apart 将……隔开

in the long run 从长远来看

shut down 停堆

take account of 考虑

课 文 注 释

1. If then the...were taken.

句中 excess 指超过需要量的燃料。整句译为: 如果加了超过需要量的燃料, 除非采取一些补偿, 否则反应堆会超临界。

2. It might appear...at the beginning.

句中 as long a time as...表示“像……长的一段时间”。整句译为: 从图 5.2 可以看出, 只要在开始加更多的 ^{235}U , 反应堆循环就能增加到想要达到的长时间。

3. The liquid Metal Fast Breeder Reactor (LMFBR) ...Tenn.

句中 LMFBR 是液态金属冷却的快中子反应堆; Clinch River Project 可译为“克林奇河项目”; Knoxville 是地名努克斯威里; Tenn 是 Tennessee 的缩写, 田纳西州。整句译为: 液态金属冷却的反应堆在一个称为克林奇河项目中设计和建造已有几年的历史了, 这个项目是在田纳西州的努克斯威里附近进行的。

4. The French fast...breeder reactors.

句中 Phoenix 是法国快堆的名字“凤凰”; Marcoule 是法国的地名马库勒。整句译为: 在马库勒的法国凤凰快堆是最成功的一个商业快增殖反应堆。

5. The role which uranium...to come.

句中 The role...play 指……发挥的作用; in the long run 的意思是从长远来看。整句译为: 铀在贡献世界能源需要方面能发挥的作用有很大的潜力, 从长远来看, 只有铀和其他的易裂变燃料能够对今后几十年的能量供应有很大贡献, 核动力的发展才能被认为是合理的。

6. Estimated additional...explored (EAR-2).

句中 Estimated additional resources 意思是预计的附加资源; extension to existing deposits 意思是现存储量的扩充。整句译为: 预计的附加资源 (EAR), 指的是预计可能在现存储量上的扩充, 或存在于还没有完全勘探的矿区中的储量 (EAR-2)。

7. This will not be without...later lesson.

句中 not be without 是用 not 否定了 without, 表示不是没有; Three Mile Island 是地名三里岛; Chernobyl 是切尔诺贝利。整句译为: 这样的做法不是没有危险的, 由于核动力的大规模使用并产生大量的钚, 已经产生了一些危害, 例如一些核电站发生了严重的事故, 像 1957 年英国温德斯格尔核电站事故; 1979 年美国三里岛核事故; 1986 年前苏联切尔诺贝利核电站事故, 这些事故将在后面的课文中介绍。

Unit 6 Nuclear Reactor Thermo – hydraulics

6.1 Methods of Heat Transmission

Most of the energy released in fission or fusion appears as kinetic energy of a few high-speed particles. As these pass through matter, they slow down by multiple collisions and impart thermal energy to the medium. It is the purpose of this chapter to discuss the means by which this energy is transferred to a cooling agent and transported to devices that convert mechanical energy into electrical energy.

We learned in basic science that heat, as one form of energy, is transmitted by three methods—conduction, convection, and radiation. The physical processes for each of these are different: in conduction, molecular motion in a substance at a point where the temperature is high causes motion of adjacent molecules, and a flow of energy toward a region of low temperature takes place. The rate of flow is proportional to the slope of the temperature, i. e. , the temperature gradient. In convection, molecules of a cooling agent such as air or water strike a heated surface, gain energy, and return to raise the temperature of the coolant. The rate of heat removal is proportional to the difference between the surface temperature and that of the surrounding medium, and also dependent on the amount of circulation of the coolant in the vicinity of the surface¹. In radiation, molecules of a heated body emit and receive electromagnetic radiations, with a net transfer of energy that depends on the temperatures of the body and the adjacent regions. For reactors, this mode of heat transfer is generally of less importance than the other two.

6.2 Conduction in Reactor Fuel

The transfer of heat by conduction in a flat plate (insulated on its edges) is reviewed. If the plate has a thickness x and cross-sectional area A , and the temperature difference between its faces is ΔT . The rate of heat flow through the plate, Q , is given by the relation

$$Q = kA \frac{\Delta T}{x}$$

where k is the conductivity, with typical units $\text{J}/(\text{s} \cdot \text{K} \cdot \text{cm})$. For the plate, the slope of the temperature is the same everywhere. In a more general case, the slope may vary with position, and the rate of heat flow per unit area Q/A is proportional to the slope or gradient written as $\Delta T/\Delta x$.

The conductivity k varies somewhat with temperature but for our estimates it is assumed to be constant.

This idea is applied to the conduction in a single fuel rod of a reactor, with the rate of supply of thermal energy by fission taken to be uniform throughout the rod. If the rod is long in comparison with its radius R , or if it is composed of a stack of pellets, most of the heat flow is in the radial direction. If the surface is maintained at a temperature T_s by the flow of coolant, the center of the rod must be at some higher temperature T_0 . As expected, the temperature difference is large if the rate of heat generation per unit volume q or the rate of heat generation per unit length $q_1 = \pi R^2 q$ is large. We can show that

$$T_0 - T_s = \frac{q_1}{4\pi k}$$

and that the temperature T is in the shape of a parabola within the rod. If we wish to keep the temperature low along the center line of the fuel, to avoid structural changes or melting, the conductivity k should be high, the rod size small, or the reactor power level low. In a typical reactor there is a small gap between the fuel pin and the inside surface of the cladding. During operation, this gap contains gases, which are poor heat conductors and thus there will be a rather large temperature drop across the gap. A smaller drop will occur across the cladding which is thin and has a high thermal conductivity.

The preceding equations enable the temperatures of the fuel, cladding and coolant to be determined in terms of the reactor power. From the thermodynamic point of view it is desirable that the coolant temperature should be as high as possible as the coolant acts as the heat source for the power cycle, however, limitations on the maximum temperature of certain components of the reactor impose a limitation on the coolant temperature and the reactor power². These limitations are due to the properties of the materials used in the reactor, and some examples are as following.

(1) The metallurgical phase change of pure uranium at 660°C produces changes in the crystalline structure of the metal. If this temperature is exceeded repeatedly, the uranium suffers considerable dimensional change and this distortion may rupture the cladding.

(2) The temperature of the cladding should not be high enough to allow its creep strength to drop seriously.

(3) The temperatures of all materials in the core should not be high enough to allow any chemical reactions such as corrosion or oxidation to take place on a significant scale. In carbon dioxide-cooled reactors using Magnox cladding, for example, oxidization of the Magnox limits its maximum temperature to about 525°C.

(4) In a water-cooled reactor in which boiling is to be avoided, the temperature of the water must be well below the saturation temperature corresponding to the water pressure.

The derivation of the equations for the temperatures of the components of a reactor has been based on the assumption that ideal conditions exist in which all the factors which determine the temperatures are known exactly. This is of course seldom the case in a real engineering system, and in a nuclear reactor factors such as the distortion of the neutron flux, incorrect distribution of the coolant flow in the channels or slight variations in the dimensions

of the fuel or cladding can cause calculated values of the temperatures to be incorrect. To allow for these uncertainties, "hot-channel factors" are introduced which are analogous to factors of safety in engineering. Calculated temperature drops are multiplied by appropriate hot-channel factors to determine the maximum temperatures that might exist under adverse conditions.

The thermal analysis of the central channel of a reactor determines the temperature in that channel alone. If the coolant flow is equally distributed in all channels (bearing in mind that the inlet temperature is common to all channels and the neutron flux decreases towards the sides of the core), the outlet temperature of the coolant and the maximum fuel temperature in the outer channels is less than in the central channel. The mixed temperature of the coolant flowing from the core to the heat exchangers is also less than the outlet temperature from the central channel and this is a disadvantage from the thermodynamic point of view. The mixed coolant outlet temperature can be raised if the coolant flow in the fuel element channels is controlled by means of orifices or gags. If the coolant flow in each channel is proportional to the energy release in that channel, then the coolant outlet temperature from all channels is the same, and the mixed outlet temperature is raised to this value. In this case the maximum fuel temperatures in the outer channels are still less than in the central channel, and a further improvement can be made by restricting the flow in the outer channels in such a way that the maximum fuel temperatures are the same in all channels. In this way the coolant outlet temperature in the outer channels is higher than in the centre channel, and the mixed outlet temperature is raised with a consequent improvement from the thermodynamic point of view.

A method which enables the total thermal output of a reactor to be increased without increasing the maximum fuel or coolant temperatures (assuming equal coolant flow in all channels) is to vary the enrichment of the fuel in different regions of the core. Thus, if the enrichment in the outer region of the core is greater than in the central region, the fuel rating which is proportional to the neutron flux and the fraction of ^{235}U in the fuel can be increased in the outer region and the thermal output can be increased without exceeding the limiting temperatures in the core.

A further assumption in the preceding analysis should be mentioned, namely that all the heat release due to fission takes place in the fuel. This is not the case in a thermal reactor in which neutrons are slowed down and gamma radiation is absorbed in the moderator. The fraction of the fission energy released as heat in the moderator is about 5 percent of the total. This energy is transferred to the coolant (unless special cooling of the moderator is provided), and the temperature rise of the coolant is the same as if all fission energy is released as heat in the fuel. However, the cladding and fuel temperatures are lower when only about 95 percent of the heat release takes place in the fuel, and the temperature drops through the fuel and cladding are reduced by about 5 percent. The assumption that all heat is released in the fuel overestimates the fuel and cladding temperatures and therefore errors on the safe side when determining these temperatures.

6.3 Heat Removal by Coolant

Convective cooling depends on many factors such as the fluid speed, the size and shape of the flow passage, and the thermal properties of the coolant, as well as on the area exposed and the temperature difference between surface and coolant $T_s - T_c$. Experimental measurements yield the "heat transfer coefficient" h , appearing in a working formula for the rate of heat transfer Q across a surface S

$$Q = hS(T_s - T_c)$$

The units of h are typically $\text{W}/(\text{m}^2 \cdot \text{K})$. In order to keep the surface temperature low, to avoid melting of the metal cladding of the fuel or to avoid boiling if the coolant is a liquid, a large surface area is needed or the heat transfer coefficient must be large, a low-viscosity coolant of good thermal conductivity is required, and the flow speed must be high³.

As coolant flows along the many channels surrounding fuel pins in a reactor, it absorbs thermal energy and rises in temperature. Since it is the reactor power that is being extracted, we may apply the principle of conservation of energy. If the coolant of specific heat c enters the reactor at temperature $T_c(\text{in})$ and leaves at $T_c(\text{out})$, with a mass flow rate M , then the reactor thermal power P is

$$P = cM [T_c(\text{out}) - T_c(\text{in})] = cM\Delta T$$

The temperature of coolant as it moves along any channel of the reactor can also be found by application of the above relation. In general, the power produced per unit length of fuel rod varies with position in the reactor because of the variation in neutron flux shape. For the special case of a uniform power along the z -axis with origin at the bottom, the power per unit length is $P_1 = P/H$, where H is the length of fuel rod. The temperature rise of coolant at z with channel mass flow rate M is

$$T_c(z) = T_c(\text{in}) + \frac{P_1 z}{cM}$$

which shows that the temperature increases linearly with distance along the channel. The temperature difference between coolant and fuel surface is the same at all points along the channel for this power distribution, and the temperature difference between the fuel center and fuel surface is also uniform.

If instead, the axial power were shaped as a sine function. For this case, the highest temperatures of fuel surface and fuel center occur between the halfway point and the end of the reactor. In the design of a reactor, a great deal of attention is given to the determination of which channels have the highest coolant temperature and at which points on the fuel pins "hot spots" occur. Ultimately, the power of the reactor is limited by conditions at these channels and points. The mechanism of heat transfer from metal surfaces to water is quite sensitive to the temperature difference. As the latter increases, ordinary convection gives way to nucleate boiling, in which bubbles form at points on the surface, and eventually film boiling can occur,

in which a blanket of vapor reduces heat transfer and permits hazardous melting. A parameter called “departure from nucleate boiling ratio” (DNBR) is used to indicate how close the heat flux is to the critical value⁴. For example, a DNBR of 1.3 implies a safety margin of 30 percent. Fig.6.1 indicates maximum temperature values for a typical PWR reactor.

To achieve a water temperature of 315°C requires that a very high pressure be applied to the water coolant-moderator.

Suppose that the pressure vessel of the reactor contains water at 13.8MPa and 315°C and the temperature is raised to 343°C. The result will be considerable steam formation (flashing) within the liquid. The two-phase condition could lead to inadequate cooling of the reactor fuel. If instead the pressure were allowed to drop, say to 8.4MPa, the vapor region is again entered and flashing would occur. However, it should be noted that deliberate two-phase flow conditions are used in boiling water reactors, providing efficient and safe cooling.

In gas-cooled reactor it may be necessary to take steps to promote more effective heat transfer from the fuel element to the coolant by the use of finned cladding. The function of the fins is to increase the surface area for heat transfer and also promote turbulence and better mixing of the gas stream. All these effects increase the heat transfer rate, however for the fins to be really effective they should be of a material of high thermal conductivity, and the design of the fins should aim to promote the circulation of gas between the heated surface and the main gas stream⁵.

The introduction of fins results in certain disadvantages. Firstly, the power required to circulate the coolant through the core is increased as a result of the increased turbulence, and secondly, unless the cladding material has a very low neutron capture cross-section, the additional neutron absorption in the fins may be unacceptable from the physics point of view. Heat transfer can also be improved without any increase in the amount of cladding used by roughening the surface of the cladding, and the heat transfer surface area can be increased by using a larger number of smaller diameter fuel rods. The use of small diameter fuel rods is also to be preferred for UO₂ fuel whose thermal conductivity is low.

The fins used in the Magnox clad fuel elements of British gas-cooled reactors are of several types-longitudinal, circumferential and helical. Heat transfer coefficients for such surfaces cannot be predicted from the empirical equations introduced earlier for smooth surfaces, and experimental data is necessary for each design. It is sufficient to reiterate that the

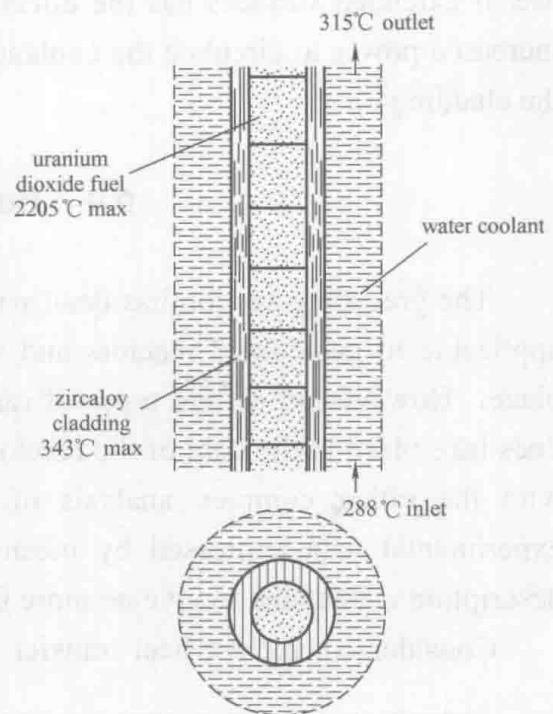


Fig.6.1 reactor channel heat removal

use of extended surfaces has the effect of increasing the heat transfer rate at the expense of increased power to circulate the coolant through the core, and increased neutron absorption in the cladding⁶.

6.4 Boiling Heat Transfer

The preceding section has dealt with single phase forced convection heat transfer, and is applicable to gas-cooled reactors and water-cooled reactors in which boiling does not take place. However, in certain types of reactors, notably boiling water reactors (BWRs) boiling does take place in the core of the reactor. It is beyond the scope of this book to deal in detail with the rather complex analysis of boiling heat transfer, much of which depends on experimental data expressed by means of empirical equations. The following qualitative description is intended to give no more than an introduction to the subject.

Consider convective heat transfer from a heated surface at temperature T_s (such as the

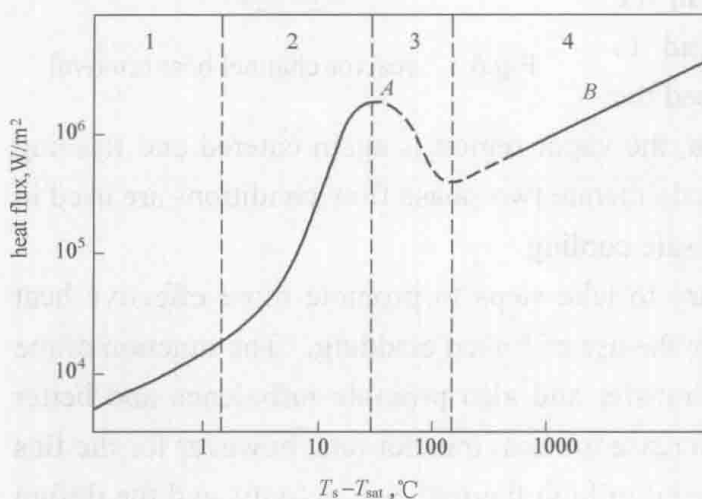


Fig.6.2 boiling heat transfer curve

surface of a fuel rod) to water whose saturation temperature is T_{sat} flowing across the surface. So long as T_s is less than T_{sat} , boiling of the water is not possible and single phase heat transfer occurs. When T_s exceeds T_{sat} it is possible that boiling may begin, dependent on the bulk water temperature T_w and the heat flux at the hot surface. So long as T_w is significantly less than T_{sat} or the heat flux is low, boiling will not occur. This is indicated by the single phase heat transfer region 1 in Fig.6.2.

However, as the heat flux increases, resulting in an increase in T_s , it is possible for boiling of the water to begin while its bulk temperature T_w is still less than T_{sat} . This is referred to as subcooled boiling. It occurs at high heat fluxes when the film temperature of the water in contact with the heated surface reaches the saturation temperature while the bulk temperature of the water is lower. As the steam bubbles formed at the heated surface move into the cooler bulk of the water they rapidly condense and collapse, but the increased turbulence caused by this bubble movement causes an increase in the heat transfer coefficient. This increasing heat transfer coefficient is indicated on Fig.6.2 by the steepening portion of the curve between regions 1 and 2 of the figure.

As the bulk water temperature rises and becomes equal to the saturation temperature, bulk boiling commences and becomes fully developed⁷. The turbulence caused by the formation of steam bubbles and their movement away from the heated surface causes the heat transfer

coefficient to increase further, resulting in the pronounced steepening of the curve in region 2 of Fig.6.2. In this region, called the nucleate boiling region, the heat flux increases rapidly with only a modest increase in the heated surface temperature.

As nucleate boiling develops, the heated surface becomes increasingly blanketed by steam bubbles which, having a thermal insulating effect, restrict heat transfer and cause a lowering of the heat transfer coefficient. The curve of Fig.6.2 begins to flatten. A heat flux is reached called the departure from nucleate boiling (DNB) at which, if the heated surface temperature is further increased, the heat transfer rate will decrease due to the increased thermal resistance of the vapour film which is spreading over the heated surface. This leads to an unstable state of partial film boiling shown as region 3 in Fig.6.2. Once the vapour film is continuous across the heated surface stable film boiling is established, and any further increase in the surface temperature will cause the heat flux to increase. However, as the heat transfer coefficient is rather low due to the vapour film, the increase in heat flux is much less pronounced than in the nucleate boiling region, see region 4 of Fig.6.2. Modest increases in the heat flux can lead to big increases in the surface temperature T_s , which may become high enough to result in damage, e.g. melting. This is called burnout, and is a condition which must be avoided in a nuclear reactor at all costs, as the melting of the fuel cladding would cause the release of fission products into the coolant and other very serious consequences.

The heat flux corresponding to the peak of the curve at point A in Fig.6.2, the departure from nucleate boiling, is also known as the critical heat flux. It is the maximum heat flux permissible in a boiling water reactor as any attempt to increase the heat flux above this value would result in a transition from nucleate boiling at A to film boiling at B, accompanied by a sudden and large rise in the surface temperature leading possibly to burnout.

It is important to be able to predict the critical heat flux in a reactor in which boiling may occur. Such reactors include not only BWRs in which nucleate boiling occurs during normal operation, but also PWRs in which boiling, either subcooled or nucleate, may occur under abnormal operating conditions. Much experimental work has been done to develop empirical correlations which involve the heat flux, surface and fluid temperatures, fluid pressure and velocity, dryness fraction (or steam quality) and the system dimensions among the relevant parameters. As a general rule it may be said that these parameters affect the critical heat flux as follows.

- (1) Increasing pressure may cause the critical heat flux to decrease.
- (2) Increasing fluid velocity causes the critical heat flux to increase.
- (3) Increasing dryness fraction (or steam quality) causes the critical heat flux to decrease.

As the dryness fraction increases along a channel in the reactor core in which boiling is taking place, the critical heat flux decreases along the channel, reaching a maximum at its upper end⁸.

The heat flux during nucleate boiling can be related to the heated surface temperature by an equation of the form:

$$\frac{Q}{A} = C (T_s - T_{\text{sat}})^n$$

Where C and n are constants. The value of n is typically about 4, so it is evident that the flux increase rapidly during nucleate boiling as the surface temperature T_s increases. This is illustrated as region 2 of Fig.6.2.

Self-test

- During operation the highest temperature in the reactor is _____.
A. in the center of fuel rod B. on the surface of fuel rod
C. on the surface of cladding D. in the fluid
- There is no limit to the rate at which _____.
A. heat is conduction B. heat is convection
C. heat is radiation D. fission can take place
- To determine the maximum temperature, the calculated temperature drops are multiplied by _____.
A. fuel temperature B. hot-channel factors
C. hot pot D. liquid temperature
- The inlet temperature _____ all channels.
A. is different for B. is changed to
C. is same for D. is mixed between
- The fuel rating is proportional to the _____ and the fraction of ^{235}U in the fuel.
A. dryness fraction B. heat conduction of fuel rod
C. neutron flux D. heat flux
- In a thermal reactor the heat release due to fission _____ in the fuel.
A. takes place mostly B. mostly
C. does not take place D. all takes place
- The power produced per unit length of fuel rod _____ in the reactor.
A. is same B. varies with the position
C. varies with length D. is different
- The maximum permissible _____ of the fuel is one of the most important factors in the thermal design of a reactor.
A. conductivity B. hot channel factor
C. temperature D. strength
- The power of the reactors is limited by the _____ channels.
A. center B. out
C. largest D. hottest
- The introduction of fins _____ certain disadvantages.
A. produces B. results in C. takes place D. gives

11. The departure from nucleate boiling is ____ critical heat flux.
A. a kind of B. same as C. different from D. similar as
12. During nucleate boiling the small increase in the heated surface temperature ____ the heat flux ____.
A. induces/decreases B. imposes/increases
C. results in/increases rapidly D. makes/decreases

关 键 字 解 释

Burnout: Sudden failure of the fuel cladding tube, caused by overheating.

烧毁: 由于过热, 导致燃料包壳突然毁坏。

Conduction: The transmission of heat from one portion of a medium to another, without visible motion of the medium, the heat energy is passed from molecule to molecule.

传导(热、电): 从媒质中一部分到另一部分的热传导, 没有可见的媒质运动, 热能是从分子到分子传递的。

Convection: The transmission of heat from one place to another, by fluid circulation between the spots of different temperature.

对流(传热、电): 通过不同温度点之间的流体循环, 热从一处到另一处的传递。

Dryness fraction: The weight proportion of dry steam in steam and water mixture.

干度分量: 在汽水混合物中干蒸汽所占的重量份额。

Empirical equation: A formula founded on experience or experimental data only, not deduced from theoretical considerations.

经验公式: 只根据经验或实验得来的公式, 不是从理论推导得来的。

Fins: The thin projecting strips formed integral with a tube to increase the heating area.

肋片: 与管子做成一体的外伸薄片, 用来增加传热面积。

Heat conductor: A body, through which heat flows by conduction.

热导体: 能够传导热流的物体。

Heat flux: The quantity of heat flows from unit heat transfer surface in unit time.

热流量: 单位时间在单位面积上传递的热量。

Heat transfer coefficient: The rate of heat transfer between two phases, it depends on the physical properties and relative motions of the two phases.

传热系数: 两相之间的传热速率, 它取决于两个相的物理性质和相对运动。

Hot pot: The point of the highest temperature on the fuel rod.

热点: 燃料棒上温度最高的点。

Hot-channel factor: A kind of safety factor in the thermal design of nuclear reactor.

热管因子: 在反应堆热工设计中的一种安全系数。

Radiation: The heat transfer by electromagnetic radiation.

辐射(传热): 通过电磁辐射进行的热量传递。

Specific heat: The heat quantity needed to raise the temperature of the unit mass of a

substance by 1°C.

比热: 使单位质量的物质温度升高 1°C 所需要的热量。

Subcooled boiling: The boiling of the water is commenced while the water bulk temperature is below the saturation temperature.

过冷沸腾、欠热沸腾: 当水的总体温度低于饱和温度时, 水中出现的沸腾。

Thermal conductivity: A measure of the flow rate of thermal energy through a material which a temperature gradient is present.

热导率: 在温度梯度下, 热能通过材料流动速率的量度。

词 汇

agent ['eidʒənt] *n.* 介质

axial ['æksɪəl] *a.* 轴向的

circumferential [sə,kʌmfə'renʃəl] *a.* 环形的, 周边的, 圆周的

collapse [kə'læps] *v. & n.* 消失, 破裂, 坍塌

commence [kə'mens] *v.* 开始

condense [kən'dens] *v.* 冷凝

correlation [,kɒrɪ'leɪʃən] *n.* 关系式

creep [kri:p] *v. & n.* 蠕变

deliberate [di'libəreɪt] *a. & v.* (仔细) 考虑的, 谨慎的; 考虑, 商讨

dryness ['draɪnis] *n.* 干度

empirical [em'pɪrɪkəl] *a.* 经验的

evident ['eɪdɪənt] *a.* 明显的

fin [fɪn] *n.* 肋片

finned cladding 带肋片的包壳

flashing ['flæʃɪŋ] *n.* 闪蒸

fusion ['fju:ʒən] *n.* 核聚变

gag [gæg] *n.* 塞盖, 压板

gradient ['greɪdɪənt] *n.* 梯度

helical ['helɪkəl] *a. & n.* 螺旋 (的)

impose [ɪm'pəʊz] *v.* 施加, 发生影响

longitudinal [lɒndʒɪ'tju:dɪnəl] *a. & n.* 纵肋 (的); 纵向 (的)

metallurgical [,metə'lɒ:dʒɪkəl] *a.* 冶金学的

notably ['nəʊtəbli] *ad.* 特别是, 显著地

nucleate boiling 核态沸腾

orifice ['ɒrɪfɪs] *n.* 孔口, 孔板

overestimate ['əʊvə'estɪmeɪt] *v. & n.* 过高估计

parabola [pə'ræbələ] *n.* 抛物线

pronounced [prə'naʊnst] *a.* 显著的

qualitative ['kwɒlɪtətɪv] *a.* 定性的
 quality ['kwɒlɪtɪ] *n.* 纯度, 干度, 质量
 rating ['reɪtɪŋ] *n.* 负荷, 额定值
 reiterate [ri:'ɪtəreɪt] *v.* 重申, 反复地说
 restrict [rɪs'trɪkt] *v.* 限制, 约束, 节流
 roughen ['rʌfən] *v.* 使变粗糙
 seldom ['seldəm] *ad.* 很少
 stack [stæk] *n.* 堆积, 许多
 steepen ['sti:pən] *v.* 变陡
 subcool ['sʌb'ku:l] *v.* 欠热, 过冷
 thermodynamic ['θə:məʊdaɪ'næmɪk] *a.* 热力学的
 turbulence ['tɜ:bjuləns] *n.* 扰动
 uniform ['ju:nɪfɔ:m] *a.* 均匀的
 at all cost 不惜任何代价, 无论如何
 deal with 处理, 对付
 so long as 只要, 因为

课 文 注 释

1. The rate of heat removal is...of the surface.

句中 the difference 是介词 to 的宾语, between 后的介词短语做 difference 的定语。整句译为: 热量输出的速率与表面温度和周围介质温度的差值成正比, 也取决于冷却剂在表面周围循环的量。

2. From the thermodynamic...reactor power.

句中 as the coolant acts as...的第一个 as 是因为的意思, 后一个 as 与 acts 搭配在一起表示起……作用的意思。全句译为: 从热力学的观点来看, 由于冷却剂是动力循环中的热源, 冷却剂的温度应尽可能的高; 然而, 对反应堆中一些部件的最高温度的限制, 使对冷却剂温度和反应堆功率产生了限制。

3. In order to keep...must be high.

句中 to avoid melting 和 to avoid boiling 都是与 In order 相联系的, 与 to keep 相并列。整句译为: 为了保持表面的低温度, 以避免燃料的金属包壳熔化, 或为了避免液体冷却剂沸腾就必须要有足够大的传热面积或足够大的传热系数, 这就要求有高导热率的低黏度的冷却剂, 同时流速还必须高。

4. A parameter called...critical value.

句中 departure from nucleate boiling ratio 直译的意思是“偏离泡核沸腾比”, 在工程中一般称为“最小燃毁比”, 是反应堆热工设计中的一个安全系数。

5. All these effects increase...gas stream.

句中 of a material...是介词短语作表语, of 引导做表语的介词短语时, 一般表示具有……的意思。整句译为: 所有这些影响会增加传热系数, 然而真正有效的肋片材料应该具有高的

热传导性, 肋片设计的重点是增进加热表面与气体主流之间的循环。

6. It is sufficient...in the cladding.

句中 sufficient to reiterate 是更强调的说明了 reiterate, 意思是“已经多次重申了”。整句译为: 已经多次重申了扩展表面有增加传热速率的效应, 但会使冷却剂循环通过堆芯所需的动力增加, 同时还增加了包壳的中子吸收量。

7. As the bulk water temperature...fully developed.

句中 bulk water temperature 是整体水温, bulk boiling 是容积沸腾。整句译为: 当整体水温升高达到饱和温度时, 容积沸腾开始并成为充分发展的沸腾。

8. As the dryness fraction...upper end.

句中 reaching a maximum at its upper end 是名词 the dryness fraction 的补语; as the...是状语从句, 主句是 the critical...。整句译为: 当通道产生沸腾时, 干度份额沿通道长度增加, 临界热流密度沿通道减少, 干度份额在通道出口达到最大值。

Unit 7 Pressurized Water Reactor

7.1 General

In the most common type of pressurized water reactor (PWR), the light water-moderated and cooled core is contained within a large pressure vessel with a removable head (see Fig.7.1). The core is cooled by water which flows downward through an annulus around the core and then upward through the core. The heated water leaves the core through one of several pipes, each of which leads to a steam generator. The cooled water leaving the steam generators is returned to the core by a large circulating pump, and the steam produced flows to a turbine, which drives the electrical generator. The steam leaving the turbine is condensed and returned to the secondary side of the steam generator by feed pumps.

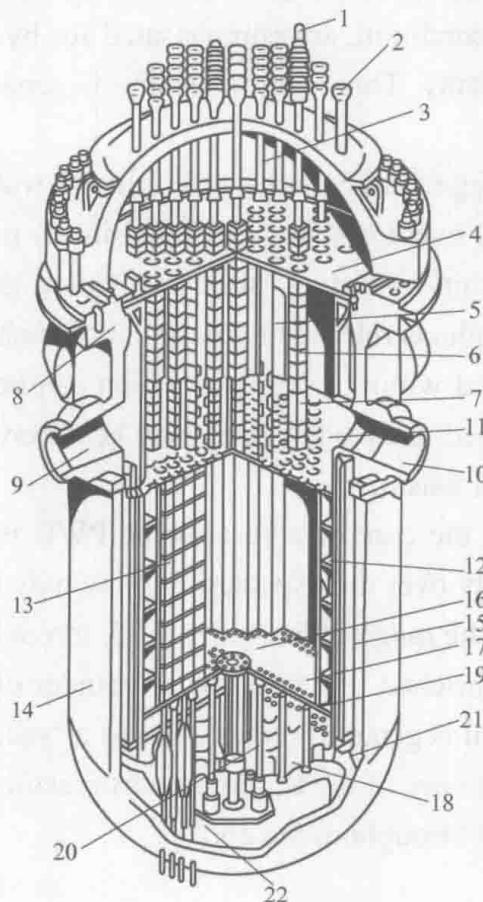


Fig.7.1 sectional view of the Sequoyah pressurized water reactor

- 1—control rod drive head adaptors; 2—instrumentation ports; 3—thermal sleeves;
 4—upper support plate; 5—support column; 6—control rod drive shaft; 7—control rod guide tube;
 8—internals support ledge; 9—inlet nozzle; 10—outlet nozzle; 11—upper core plate; 12—baffle and former;
 13—fuel assemblies; 14—reactor vessel; 15—thermal shield; 16—access port; 17—lower core plate;
 18—core support; 19—diffuser plate; 20—lower support column;
 21—radial supports; 22—instrumentation thimble guides

In a pressurized water reactor, the mixed fluid leaving the core is not allowed to boil. Net boiling is prevented by maintaining a substantial overpressure. This pressure, generally in the range of 13~16MPa, is maintained by an electrically-heated pressurizer connected to one of the circulating loops by a branch line.

PWR fuel elements consist of long, thin (approximately 6~10mm diameter and 3~5m long) hollow-wall metal tubes filled with cylindrical uranium dioxide pellets. The metal tube, generally referred to as “cladding”, retains the fission products and prevents UO_2 particles from becoming a part of the circulating coolant stream. The cladding is generally an alloy of zirconium, although stainless steel was used in the early designs.

The fuel elements are assembled into large rectangular cross-section bundles, which may contain from 150 to 300 fuel elements. The fuel assemblies are positioned vertically in the core and are placed in a configuration which approximates a right circular cylinder. The coolant thus flows parallel to the axes of the fuel elements.

Control rods, inserted through the removable head, provide sufficient control to bring the core to a hot shutdown condition. The changes in reactivity with core lifetime, and in going from the hot to cold shutdown condition, are compensated for by varying the concentration of boric acid dissolved in the coolant. This control scheme is generally called “chemical shim” control.

Provision is made for the injection of large quantities of water to the system to maintain core cooling in the very unlikely event of a break in the primary piping. In addition, the entire primary system is enclosed within an airtight structure, called the “containment”, which can contain all steam and fission products released in the event of such a piping rupture.

The reactor vessel is located within a deep pit, which can be flooded when the reactor is shutdown. When the pit is flooded, the reactor head may be removed and underwater refueling, using long-handled tools, is then feasible.

Neutron absorption within the core of a vessel-type PWR is too high to allow criticality with natural uranium. Criticality over the expected lifetime may be achieved by using slightly enriched uranium, generally in the range of 3%~5% ^{235}U , throughout the core. Alternatively, one may use a ring of highly enriched fuel with the remainder of the core natural uranium or thorium. This latter arrangement is generally referred to as a “seed and blanket”. The seed and blanket concept was used in the core of the Shipping port reactor but most later central station units have slightly enriched fuel throughout the core¹.

7.2 Pressurized Water Reactor Components

Nuclear power plants are, with some exceptions, large installations made up of a great number and variety of components. Since several manufacturers are in the business of supplying nuclear reactors internationally and since an even larger number of architect/engineer organizations collaborate with the vendor in the design, procurement, and construction of the

plant, it is not surprising that a great deal of variation exists in the details of plant layout, component selection, and general engineering approaches. Compounding these differences was the rapid development of nuclear power plant design from one generation to the next, by the same manufacture. These changes were dictated, to some extent, by necessary improvements in plant performance and the need to conform to environmental and safety regulations. Another powerful motivation was to increase the size of the plant in order to achieve economics of scale. This rapid evolution was difficult for the utility industry to absorb comfortably, and the economies of scale reached a point of diminishing returns in the 1970s. The tendency in recent years has been toward stabilization to a certain model by each vendor, and a need for standardization.

Despite the great variability in plant layout, system, and components, the basic features of the plants remain essentially the same. Even between PWRs, with their differences in the energy transport loops, there exists a great deal of similarity: A large pressure vessel containing a nuclear core of fuel bundles made of similar fuel rods, occupies the heart of the nuclear plant. Control rods are provided to control the nuclear reaction, to change power level, and to shut the reactor down. Piping provides a path for the coolant to carry away the heat produced and is complemented by valves and pumps. Turbines, feedwater heaters, condensers, and other items are used in the energy conversion cycle. All plants are provided with multiple barriers to prevent the escape of radioactivity under normal or abnormal conditions. Redundant and diverse components are installed to fulfill safety functions and to provide comfortable margins with respect to imposed requirements. Containment structures are required now in all countries to contain potential release of radioactivity from the plant and to protect against external natural or man-made forces such as earthquakes, floods, tornadoes, tsunamis, explosions, fires, and even airplane crashes.

Fig.7.2 depicts a four-loop plant such as those manufactured by Westinghouse. In the figure an additional vessel, called the “pressurizer” is also shown connected to the primary system. Its function is to maintain control of primary system pressure.

It is useful to review briefly the various conditions under which the components must be made and tested. In the United States these standards were set in the codes written by the American Society of Mechanical Engineers (ASME code) and, separately, by Committee N-18 of the American Nuclear Society². (Similar codes exist in Europe and Japan.) The categorization into safety classes done by these organizations is very similar. This categorization defines component and system classes as follows.

Safety Class 1. This is the most vital category and applies to all components of the primary coolant system whose failure could cause a loss of reactor coolant, in operating conditions III or IV. The criteria are, therefore, most stringent.

Safety Class 2. This applies to structures and components that are required to fulfill a safety function (for example, to shut down the reactor; to provide cooling for the core or the containment; to cool another safety system; to contain and control radioactivity that may be

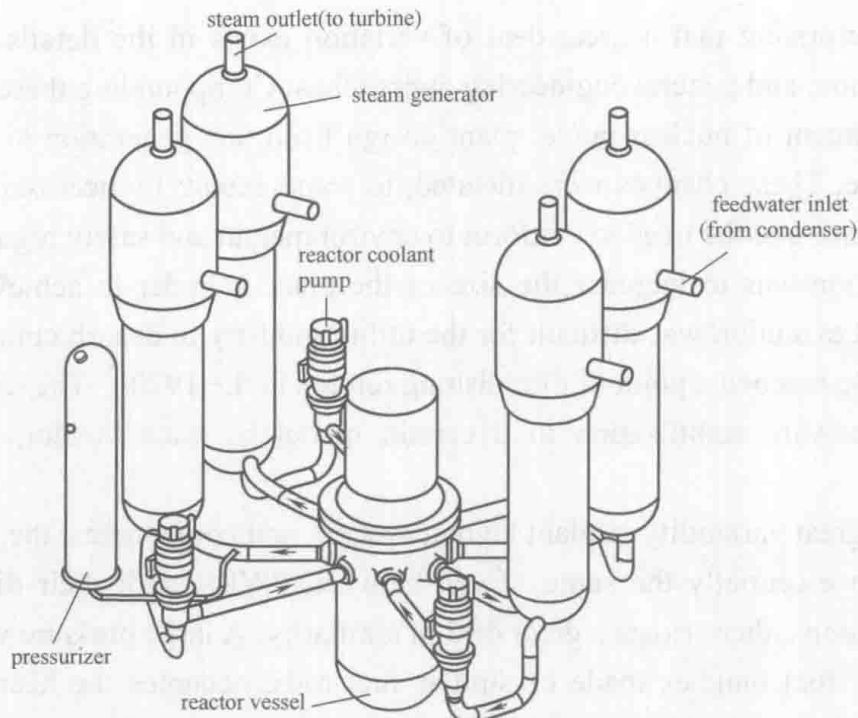


Fig.7.2 schematic of reactor coolant system for PWR

released in an accident, etc.).

Safety Class 3. This applies to systems whose failure would allow release to the environment of gaseous radioactivity that would normally be held for decay within the plant.

7.3 Introduction to the Nuclear Steam Supply System 80+

7.3.1 Reactor and Reactor Coolant System

The Reactor and Reactor Coolant System function together to provide the means for the generation and transfer of heat from the core and fuel to the steam generators. The steam generators subsequently transfer heat to the main steam system and the turbine.

The advanced System 80+ design has evolved from the system 80 reference design³. The primary objective of this evolutionary design effort has been to increase operating margin in a manner which would enhance performance and reliability while preserving the well proven configuration.

For the Reactor, the core operating margin has been increased by reducing the normal operating hot leg temperature and revising core parameter monitoring methods. The ability to change operating power level (i.e., maneuver) using control rods only (without adjusting boron concentration in the coolant system) has been provided, simplifying reactivity control during plant load changes and reducing liquid waste processing requirements.

Improvements have also been made to the RCS. For example, the reactor pressure vessel is ring-forged. The ring-forged design results in a significant reduction in the number of welds (with resulting reduction in-service inspection) and eliminates concern for pressurized thermal

shock. Also, the pressurizer volume has been increased relative to the system 80 design to enhance transient response and reduce unnecessary challenges to safety systems. Furthermore, the System 80+ steam generators include Inconel 690 tubes, improved steam dryer efficiency, and a seventeen percent increase in overall heat transfer area relative to System 80, and a ten percent margin for potential tube plugging. The steam generators have a twenty-five percent larger secondary feedwater inventory than System 80 to extend the “boil dry” time and improve system response to upset conditions. Steam generator improvements including larger and repositioned manway, a standby recirculation nozzle, and a redesigned flow distribution plate have been made to facilitate maintenance, and to maintain long term integrity.

The System 80+ reactor and RCS includes the reactor vessel, reactor internals, core, steam generators, pressurizer and associated coolant pumps, piping, and valves. There are two parallel heat transfer loops each containing one steam generator (SG) and two reactor coolant pumps (RCPs). The pressurizer is connected by a surge line to one of the reactor vessel outlet pipes.

The reactor is a light water-moderated and cooled pressurized water reactor (PWR). The fuel is slightly enriched uranium in the form of sintered UO_2 pellets enclosed in tubes. A multi-zone refueling strategy replaces between one-third and one-fourth of the core at each refueling, depending on the fuel cycle length. Fresh fuel is loaded in the core interior along with partially burned fuel. The remaining partially burned fuel is moved to the core periphery to reduce neutron leakage, thereby improving fuel economy and reducing irradiation of the pressure vessel.

Control of the core power distribution and reactivity is achieved by a combination of fuel loading pattern, burnable absorber rod depletion, chemical shim reactivity control, and selected patterns of Control Element Assembly (CEA) insertion and withdrawal.

Reactor coolant enters the reactor vessel through inlet nozzles, flows downward between the reactor vessel shell and core barrel, into the lower plenum where flow distribution is equalized, and then upward through the core, removing heat generated by the nuclear reactor core. Coolant leaves the reactor vessel shell through outlet nozzles and enters the tube side of the vertical shell economizer steam generators, where heat is transferred to the secondary system⁴. Steam generated in the shell side of the steam generator passes through moisture separators and dryers to ensure that moisture content at the steam generator outlet is minimized. After leaving the steam generators, reactor coolant is returned to the reactor vessel by limited leakage reactor coolant pumps.

The supports for each of the individual components of the reactor coolant system are designed to form an integrated support system which mitigates the effects of earthquakes, branch line breaks, and steam line breaks on the RCS. This capability has been achieved without sacrificing the ability of the reactor coolant system to expand thermally with minimum restraint.

Therefore, there are no pipe whip restraints or jet impingement shields required for the

main loop piping, and dynamic loads due to postulated main loop pipe breaks are not imposed upon component support structures or other plant equipment.

The design pressure and temperature of the reactor coolant system are 172.4 bars and 343.3°C. The system will operate at a pressure of 155.1 bars and has a design lifetime of 60 years.

7.3.2 Reactor Control Characteristics

Reactor power increases or decreases with reactivity insertion or removal. Reactivity insertion may be accomplished by Control Element Assembly (CEA) withdrawal, dilution of reactor coolant boron concentration or, to a lesser degree, by reduction of reactor coolant average temperature. Conversely, reactivity is removed (negative reactivity inserted) by CEA insertion, increasing the reactor coolant boron concentration, or increasing the reactor coolant average temperature. The reactor design provides regulatory banks, consisting of full-strength CEAs, and independent part-strength CEA (PSCEA) banks for reactivity compensation and axial power distribution control⁵.

Very slow but ultimately very large changes in reactivity may be made by adjusting the boron concentration. This means of changing reactivity is normally used to compensate for fuel burn-up or to override fission product buildup and is manually controlled by the reactor operator.

Changes in reactor power in response to changes in turbine load are accomplished automatically by means of regulating CEA movement. No change in soluble boron concentration is necessary. The Reactor Regulating System (RRS) senses reactor power, reactor coolant temperatures, and a load reference (usually turbine power) and provides signals to the controlling CEA group demanding direction and speed of movement, if required.

After a load change, the RRS brings the reactor to a new steady-state power level. Simultaneously, reactor coolant average temperature is adjusted to a value programmed with reactor power to maintain design steam conditions. Average temperature is then allowed to float about the programmed value within a dead band to take advantage of the reactor coolant temperature reactivity effect, which provides a negative feedback loop. This effect stabilizes reactor power, eliminating the need for frequent CEA movement.

7.3.3 Reactor Core Monitoring

The In-core Neutron Monitoring System is used to determine the neutron behavior within the reactor core. The system is comprised of a Fixed In-core Detector System (FICDS) which ABB-CE has provided for its pressurized water reactors over the last twenty years. The FICDS utilizes rhodium self-powered neutron detectors (SPNDs) to perform continuous on-line monitoring of the flux distribution in the reactor core which is used in the calculation of other important core parameters such as peak linear heat rates, axial power distributions, azimuthally tilts and limits to departure from nucleate boiling(DNB)⁶.

The FICDS has significant advantages for reactor core monitoring over a system which uses ex-core detectors in combination with a movable in-core detector flux mapping system.

The basic advantage of using fixed instead of movable in-core detectors is that fixed detectors allow the core power distribution to be observed on-line and can be used continuously for core monitoring. Movable detectors permit only periodic observation of the core power distribution, and allowance must be made for changes between observations. Further, by using information from in-core rather than ex-core detectors to monitor the core, greater accuracy is achieved, which yields increased margins to thermal limits. These margin gains can be translated into one or more of the following: higher allowable power levels or peaking, additional operating flexibility, and additional flexibility in fuel management.

In addition to the FICDS, an optional Movable In-Core Detector System (MICDS) is available which provides redundant and diverse core power distribution information. The MICDS includes two movable miniature fission chambers, movable detector signal conditioning equipment, and the movable drive system hardware and software.

Self-test

1. The MICDS means _____.
 - A. Movable In-Core Data System
 - B. Miniature In-Core Detectors
 - C. Movable In-Core Detectors
 - D. More In-Core Data System
2. In the Light Water Reactor, water is used as _____.
 - A. coolant
 - B. moderator
 - C. steam generator
 - D. both A and B
3. Primary system water enters the reactor vessel through the inlet nozzles, then travels downward in _____.
 - A. the tube
 - B. the annular space
 - C. the vessel
 - D. the low plenum
4. Criticality over the reactor lifetime is achieved by _____.
 - A. using slightly enriched uranium
 - B. control rods
 - C. boric acid
 - D. fission products
5. The powerful motivation to increase the size of the plant is _____.
 - A. the shortage of electricity
 - B. the economies of scale
 - C. manufacturer
 - D. plant layout
6. The reactor core in PWR is located _____.
 - A. in the containment
 - B. in the vessel
 - C. in a pool
 - D. below the inlet and outlet nozzles
7. Chemical shim control is employed as _____ reactivity control.
 - A. supplementary means of
 - B. the most important
 - C. emergency
 - D. normal

8. The pressurizer is connected by ____ to the hot leg.
A. a small pipe B. a valve C. a surge line D. a U tube
9. According to multi-zone refueling strategy, the fresh fuel is loaded in ____.
A. the core periphery B. different places
C. the core center D. half the core
10. When the CEA is inserted, the reactivity is ____.
A. changed B. removed C. increased D. withdrawn
11. The in-core neutron monitoring system is used to determine the ____.
A. neutron flux distribution B. neutron velocity
C. both A and B D. neither A nor B
12. From the text the hot leg temperature of System 80+ is ____ that of System 80.
A. higher than B. the same as C. similar as D. lower than

关键字解释

ABB-CE: ABB Combustion Engineering.

ABB 燃烧工程公司.

Boric acid: H_3BO_3 , A tribasic acid.

硼酸: H_3BO_3 , 一种三价酸.

CEA: Control Element Assembly, a bundle of neutron absorber rods.

控制棒组件: 由多个中子吸收棒组成的棒束.

Core: That part of a nuclear reactor which contains the fissile material, either dispersed or in cans.

堆芯: 核反应堆内含有易裂变材料的部分, 易裂变材料可以是弥散的或是带包壳的.

Feed pump: The pump forcing the condensate from condenser to steam generator.

给水泵: 将水由冷凝器打入蒸汽发生器的水泵.

Pressurized water reactor (PWR): The reactor cooled and moderated by pressurized light water.

压水反应堆: 用加压的轻水冷却和慢化的反应堆.

Pressurizer: A large, vertical, cylindrical vessel, connected at its bottom to one of the hot legs of the primary coolant loop. The main purpose of the pressurizer is to maintain primary system's pressure within prescribed bounds during normal operation.

稳压器: 一个大的立式圆柱形容器, 底部与主回路的一个热管段相连, 稳压器的主要目的是在正常运行情况下保持系统压力在一定范围之内.

Primary coolant: Reactor coolant. The fluid circulated through the reactor to remove heat. In direct-cycle reactors such as the boiling water reactor this drives the turbine directly but in other types the heat is passed via a heat exchanger to secondary system.

主冷却剂: 反应堆冷却剂. 指循环流经反应堆而带出热量的流体, 例如在直接循环的沸水反应堆中, 冷却剂变成蒸汽直接去驱动汽轮机; 在另一种类型的反应堆中, 冷却剂通过热

换热器将热量传给二回路系统。

Procurement: Organizational procedure for obtaining equipments.

采购: 获得设备的有组织的行为。

Reactor trip: Rapid reduction of reactor power to zero by emergency insertion of control mechanisms and (in some case) removal of liquid moderator.

快速停堆: 通过控制棒紧急插入或在有些情况下排出液体慢化剂, 使反应堆功率迅速降到零。

Reactor vessel: The vessel in which the core, moderator, coolant and control rods are situated.

反应堆容器: 装有堆芯、慢化剂、冷却剂和控制棒等的容器。

Reliability: Probability that an equipment or a component will continue to function when required.

可靠性: 当需要时, 一个设备或元件可以持续运转的可能性。

RCS: Reactor Coolant System. The system consists of reactor, steam generator, pressurizer and associated coolant pumps, piping, and valves.

反应堆冷却剂系统: 由反应堆、蒸汽发生器、稳压器以及相连接的冷却剂泵、管路和阀门组成的系统。

RRS: Reactor Regulating System. The system regulating reactor power rating.

反应堆调节系统: 调节反应堆功率的系统。

词 汇

approach [ə'prəʊtʃ] *n.* 方法, 途径

architect ['ɑ:kitekt] *n.* 建筑师

azimuthal ['æzi,mʌθəl] *a.* 方位角的

bank [bæŋk] *n.* 棒系

barrel ['bærəl] *n.* 吊篮, 桶

chemical shim 化学补偿剂

code [kəʊd] *n.* 法规, 规范

collaborate [kə'læbəreɪt] *v.* 合作

complement ['kɒmplɪmənt] *n.* 补充

comprise [kəm'praɪz] *v.* 由……组成

compounding ['kɒmpaʊndɪŋ] *n.* 复合, 综合

conditioning ['kən'dɪʃənɪŋ] *n.* 处理

configuration [kən,figju'reɪʃən] *n.* 结构, 配置, 形状, 方位

containment [kən'teɪnmənt] *n.* 安全壳

detector [di'tektə] *n.* 探测器

dictate [dik'teɪt] *v.* 支配

dilution [dai'lju:ʃən] *n.* 稀释, 冲淡

- diverse [dai'və:s] *a.* 多样的
- dynamic [dai'næmik] *a.* 动力学的, 动态的
- impingement [im'pindʒmənt] *n.* 冲击
- installation [instə'leiʃən] *n.* 装置
- internal [in'tə:nl] *n.* 堆内构件
- layout ['leiaut] *n.* 布置
- manufacturer [,mænju'fæktʃərə] *n.* 制造商
- miniature ['minjətʃə] *a. & n.* 小型(的)
- mitigate ['mitigeit] *v.* 减轻, 缓和
- motivation [,məuti'veiʃən] *n.* 动力, 刺激
- override [,əuvə'raid] *v. & n.* 超越, 克服
- periphery [pə'rifəri] *n.* 边缘, 周边
- redundant [ri'dʌndənt] *a.* 冗余的
- reed [ri:d] *n.* 弹簧
- regulatory ['regjuleitəri] *a.* 管理的, 规章(限制)人的, 调节的
- reposition [,ri:pə'ziʃən] *n.* 复原位, 改变……位置
- rhodium ['rəudiəm] *n.* 铑
- ring-forged 锻造出的环
- sacrifice ['sækrifais] *v.* 损失
- scheme [ski:m] *n.* 方案, 系统
- strategy ['strætɪdʒi] *n.* 战略
- subtract [səb'trækt] *v.* 减去
- terminology [,tə:mi'nɒlədʒi] *n.* 专门术语
- tilt [tilt] *v.* 倾斜, 偏离
- tornado [tɔ:'neɪdəu] *n.* 飓风, 龙卷风
- trip [trip] *n.* 脱扣, 停堆
- tsunami [tsju'nɑ:mi] *n.* 海啸
- vendor ['vendɔ:] *n.* 卖主, 供货商
- whip [hwip] *v. & n.* 拍打
- conform to 适应
- dead band 死区, 不需要控制系统工作的区
- make up 补充
- great deal 大量的

课文注释

1. The seed and blanket concept...throughout the core.

句中 seed 指反应堆堆芯内燃料富集度高的区域, 称点火区; blanket 是燃料富集度低的区域, 称再生区; Shipping port 是地名希平港。整句译为: 在希平港反应堆中使用了点火区

和再生区的概念，但后来的大部分中心电站都在整个堆芯使用微浓缩的燃料。

2. In the United States these...American Nuclear Society.

句中 ASME 是美国机械工程师协会的缩写；Separately 是副词，修饰动词 written。整句译为：在美国，这些标准是分别由美国机械工程师协会和美国核协会 N-18 委员会编写的规范制定的。

3. The advanced System 80+...reference design.

句中 System 80+和 System 80 都是核动力系统的型号，System 80+是 System 80 的改进型。整句译为：先进的系统 80+设计是参考系统 80 的设计发展而来的。

4. Coolant leaves the reactor vessel shell...the secondary system.

句中 economizer 是“节热器”或“节能器”。全句译为：冷却剂通过出口接管离开反应堆容器，进入立式的有节热器的蒸汽发生器管内，将热量传给二次侧系统。

5. The reactor design provides regulatory...distribution control.

句中 regulatory banks 指调节棒组件，banks 是棒束。全句译为：反应堆设计提供有调节棒束，由全强度的控制棒组件组成，还有进行反应性补偿和轴向功率分布控制的独立的部分强度控制棒组件。

6. The FICDS utilizes rhodium...nucleate boiling (DNB).

句中 self-powered 是“自给能”；on-line monitoring 表示“在线监测”。全句译为：固定的堆内监测系统使用铑自给能中子监测器，对反应堆堆芯中子通量分布进行在线监测，用测得参数确定堆芯内其他的重要参数，例如最大线功率、轴向功率分布、中子通量的方位角偏差及对偏离泡核沸腾的限制等。

Unit 8 Reactor Vessel and Internals

8.1 Reactor Vessel

The System 80+ reactor vessel is designed to contain and support the core and fuel and is essentially unchanged from the well proven System 80 design. A major improvement in manufacturing and operation has been achieved through the use of ring-forgings. The use of a forging as opposed to rolled and welded plates used in previous vessel designs reduces the number of welds and the overall complexity of the vessel and thus reduces fabrication time¹. Furthermore, the remaining welds have been relocated to areas of lower neutron flux thus enhancing the vessels resistance to brittle fracture. To further ensure vessel integrity throughout its 60 year design life an analysis regarding vessel brittle fracture is performed.

The reactor vessel is comprised of the reactor vessel assembly, supports, level indication and surveillance specimens.

In addition, the reactor pressure vessel is designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Vessels.

8.1.1 Reactor Vessel Assembly

The reactor vessel (see Fig. 8.1) is a vertically mounted cylindrical vessel with a hemispherical lower head attached to the vessel and a removable hemispherical upper closure head. The reactor vessel is fabricated from low alloy steel and the internal surfaces that are in contact with the reactor coolant are clad with austenitic stainless steel.

The reactor vessel consists of a vessel flange, three shell sections (upper, intermediate, and lower) and a bottom head. The vessel flange is a forged ring with a machined ledge on the inside surface to support the core support barrel, which in turn supports the reactor internals and the core. The flange is drilled and tapped to receive the closure studs and is machined to provide a

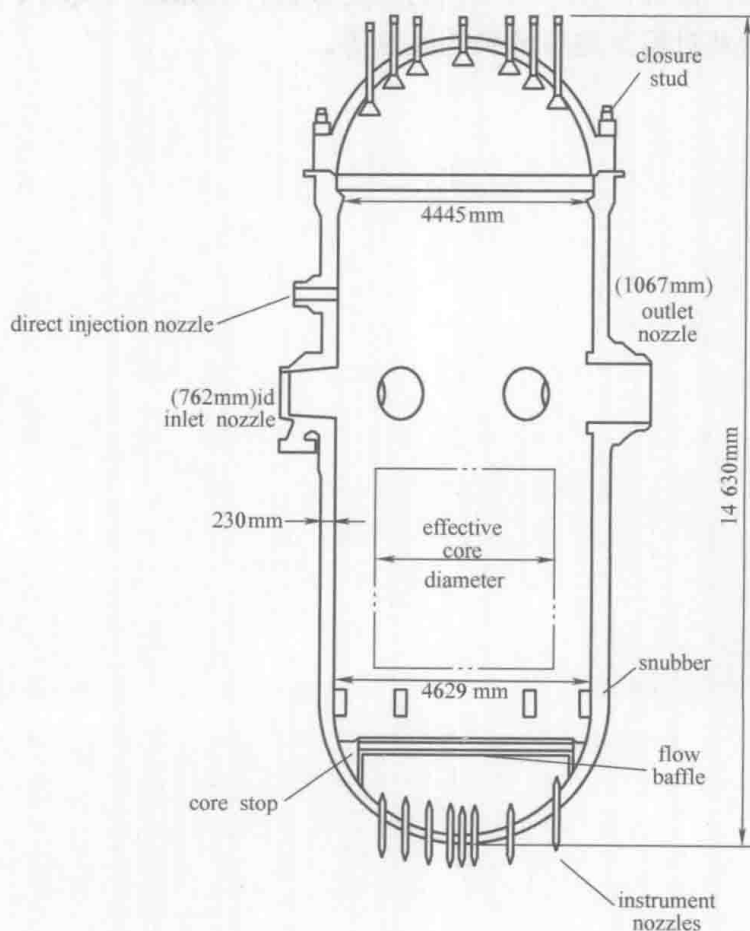


Fig.8.1 reactor vessel

mating surface for the reactor vessel closure seal rings. Each shell consists of one 360 degree forged ring. The bottom head is constructed of a single hemispherical forging. The three shell sections, the bottom head forging and vessel flange forging are joined together by welding, along with four inlet nozzle forgings and two outlet nozzle forgings to form a complete vessel assembly.

The centerlines of the coolant flow nozzles are located on a common horizontal plane in the nozzle shell course. A boss is located around the two outlet nozzles on the inside diameter of the vessel wall to provide a mating surface for the internal structure and to guide outlet flow². This boss and the mating outlet sleeve on the core barrel are machined to a common contour to control reactor coolant bypass leakage. The transition section joining the nozzle and vessel shell courses is tapered externally.

The closure head is fabricated separately since it is joined to the reactor vessel by bolting. The closure head consists of a head flange and a dome.

The head flange is a forged ring. The flange is drilled to match the vessel flange stud hole locations, and the lower surface of the flange is machined to provide a mating surface for the vessel closure seal rings. The dome is constructed of a single hemispherical forging. The dome and flange are welded together to form the closure head, and the control element drive mechanism (CEDM) nozzles are welded into the head to complete the assembly.

The studs for the closure head are tensioned using hydraulic stud tensioners. Flange sealing is accomplished utilizing silver plated Ni-Cr-Fe alloy self-energizing O-Rings.

Nozzles for direct vessel injection (DVI) of borated water from the Safety Injection System are provided near the top of the cylindrical vessel.

8.1.2 Reactor Vessel Level Indication

The water level within the reactor vessel for the System 80+ design is monitored using several methods. The first method is to monitor vessel water level inventory using the inadequate core cooling (ICC) monitoring instrumentation. The ICC instrumentation contains the reactor vessel level monitors which use heated junction thermocouples (HJTC's). The reactor vessel level monitors provide control room indication of the status of the liquid inventory in the reactor vessel.

Another method of monitoring the water level in the reactor vessel during refueling operations is the Shutdown Reactor Coolant Level Monitor (SRCLM). The SRCLM consists of the piping, valves, transmitter, indicator and other equipment required for monitoring reactor coolant level during shutdown conditions. The SRCLM displays Reactor Coolant System (RCS) level in the control room via the Discrete Indication and Alarm System (DIAS) and the Data Processing System (DPS).

8.1.3 Reactor Vessel Design-Material Selection

Design of the reactor internals and the water annulus between the active core and vessel wall is such that, for reactor operation at 105 percent rated core thermal output, and an 80 percent plant capacity factor, the vessel fluence ($E > 1$ MeV) is predicted to not exceed 6.0×10^{19}

nvt (total integrated neutron flux) over the 60-year design life of the vessel. This is an upper limit based on the most unfavorable combination of reactor vessel diameter, core power distribution, and uncertainty in the fluence calculation.

The reactor vessel is fabricated from SA-508 steel forgings with controlled copper, nickel, sulfur, and phosphorous content in the belt line region of the vessel. By proper control of these materials, maximum increase in reference nil-ductility transition temperature RT_{NDT} can be held to a value of about 9.4°C .

The cooldown time from operating temperature to refueling temperature of 27.5 hours is not limited by reactor vessel irradiation and the RT_{NDT} shift. This duration is dictated solely by the rate of change of decay heat generation, and the capacity of the shutdown cooling system.

Likewise, there is no effect on the startup sequence. Plant heatup rate is limited by the heat input from the pumps. No additional limit needs to be imposed at the end of life as a result of reactor vessel irradiation and RT_{NDT} shift.

8.1.4 Surveillance Specimen Compartment Assembly

Surveillance specimens are provided for pre-irradiation testing to establish the baseline properties, and for irradiation exposure within the reactor to determine the changes which occur in these properties as a result of neutron irradiation.

Pre-irradiation data is used as a basis for determining the shift in RT_{NDT} temperature and other fracture toughness properties. A sufficient number of specimens are provided so that unirradiated properties of reactor vessel materials can be accurately determined. The unirradiated baseline specimens are tested to establish the baseline properties of the base metal, weld metal, and the heat affected zone metal. Included in these baseline evaluations are standard and pre-cracked Charpy impact specimens, and drop weight, tensile, and compact tension specimens to determine the static and dynamic fracture toughness properties of these materials as a function of temperature³. This testing confirms that the pre-irradiation toughness properties are consistent with the requirements of 10 CFR 50.

Charpy impact (standard and pre-cracked) and tensile specimens are placed within the reactor in corrosion resistant compartments for neutron irradiation. A chain of compartments joined together for removal from the reactor at any given time is defined as a capsule assembly. These assemblies are located between the core support barrel and reactor vessel.

Three metallurgical different materials, representative of the reactor vessel, are investigated: ①base metal; ②weld metal; ③weld heat affected zone material. Specimen removal and testing is performed in accordance with 10 CFR 50 Appendix H.

Fission threshold detectors are also inserted into each capsule to monitor the fast neutron flux. Low melting point alloys are included to monitor the maximum temperature experienced by the test specimens. Correlation monitors (impact specimens made from standard heats of steel) are included to aid in analyzing results obtained from the surveillance sample tests.

In addition to test specimens, a section of base metal and a section of welded plate are provided with full documentation and identification for retention by the customer for use in

future evaluations, should the need arise⁴.

8.2 Reactor Vessel Internals

The Reactor Vessel internals provide the structural support for the core and fuel located within the reactor vessel. The internals also provide for the alignment of the core with the CEAs, CEDMs, and instrumentation, and serve as a shield to protect the reactor vessel from radiation damage. In addition, they also accommodate the expansion and contraction of the core.

8.2.1 Configuration

The reactor vessel internals are composed of support structures (e.g., core support barrel) and other structures (e.g., lower support structure and core shroud) within the reactor vessel. Core support structures are those that restrain the core. All other structures within the reactor vessel, exclusive of the fuel assemblies, in-core instrumentation, and control element assemblies, are considered as internal structures. Core support structures are designed, fabricated and stamped in accordance with the ASME boiler and pressure vessel code Section III.

The reactor vessel internals and their relationship to the rest of the reactor are shown in Fig.8.2. The reactor vessel internals and core support structures guide the primary coolant through the core. The core support structure assembly, in conjunction with the water in the downcomer annulus, provides a shield for the reactor vessel against radiation damage by the neutron flux. The core support structure assembly also provides alignment of the core with the control drives, instrumentation, and the reactor vessel.

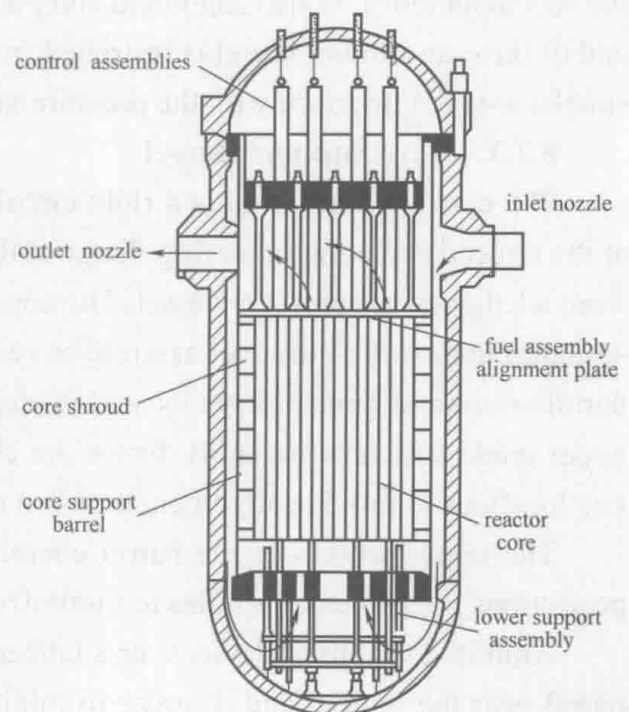


Fig.8.2 reactor arrangement

The reactor vessel internals are comprised of two basic components: ① a core support barrel assembly; ② an upper guide structure assembly. Both assemblies are designed such that maximum operating stresses are less than the values specified in Section III, Sub-section NG of the ASME pressure vessel code, for normal operating and transient plant conditions. The reactor vessel internals are fabricated from austenitic stainless steel. A high strength stainless steel holddown ring provides an axial load on the internals to prevent harmful vibration.

Welded connections are used throughout the design. Full penetration welds, where required, are designed to develop full strength of joined members.

8.2.2 Core Support Barrel Assembly

The major structural member of the reactor internals is the core support barrel assembly. The core support barrel assembly consists of the core support barrel, the lower support structure, and the core shroud. The material for the assembly is austenitic stainless steel.

The core support barrel assembly is supported at its upper end by the upper flange of the core support barrel, which rests on a ledge in the reactor vessel flange. Alignment is accomplished by means of four equally spaced keys in the flange, which fit into the keyways in the reactor vessel ledge and reactor vessel closure head⁵. The lower flange of the core support barrel supports, secures, and positions the lower support structure, and is attached to this structure by means of a welded flexural connection. The lower support structure provides support for the core by means of support beams that transmit the load to the core support barrel lower flange. The locating pins in the beams provide orientation for the lower ends of the fuel assemblies. The core shroud, which provides a flow path for the coolant and lateral support for the fuel assemblies, is also supported and positioned by the lower support structure. The lower end of the core support barrel is restricted from excessive radial and torsional movement by six snubbers which interface with the pressure vessel wall.

8.2.3 Core Support Barrel

The core support barrel is a right circular cylinder including a heavy external ring flange at the top end and an internal ring flange at the lower end. The core support barrel is supported from a ledge on the pressure vessel. The core support barrel, in turn, supports the lower support structure upon which the fuel assemblies rest. Shrink-fitted into the flange of the core support barrel are four alignment keys located 90 degrees apart. The reactor vessel, closure head, and upper guide structure assembly flange are slotted in locations corresponding to the alignment key locations to provide alignment between these components in the vessel flange region.

The upper section of the barrel contains two outlet nozzles that interface with internal projections on the vessel nozzles to minimize leakage of coolant from inlet to outlet.

Amplitude limiting devices, or snubbers, are installed on the outside of the core support barrel near the bottom end in order to minimize vibration induced by the coolant flow. The snubbers consist of six equally spaced lugs around the circumference of the barrel and act as a tongue-and-groove assembly with the mating lugs on the pressure vessel. Minimizing the clearance between the two mating pieces limits the amplitude of vibration.

During initial assembly, as the internals are lowered into the pressure vessel, the pressure vessel lugs engage the core support barrel lugs in an axial direction. Radial and axial expansion of the core support barrel are accommodated, but lateral movement of the core support barrel is restricted.

8.2.4 Core shroud

The core shroud provides an envelope for the core and limits the amount of coolant bypass flow. The shroud consists of a welded vertical assembly of plates designed to channel

the coolant through the core⁶. Circumferential rings and a top and bottom end plate provide lateral support. A small gap is provided between the core shroud outer perimeter and the core support barrel in order to provide upward coolant flow in the annulus, thereby minimizing thermal stresses in the core shroud. Four hard-facing alignment lugs, spaced 90 degrees apart, protrude vertically from the top of the core shroud and engage in corresponding hard-facing slots in the upper guide structure fuel alignment plate to ensure proper alignment among the upper guide structure assembly, core shroud, and lower support structure.

Self-test

1. The reactor vessel flange is a _____ with a ledge on the inside surface.
A. shell section B. forged ring
C. forged plate D. hemispherical forging
2. The closure head is joined to the reactor vessel by _____.
A. bolting B. welding C. forging D. joining
3. The ICC stands for the _____.
A. In-Core Cooling B. In-Core Condense
C. Inadequate Core Cooling D. Inadequate Core Control
4. The fission threshold detectors monitor the _____.
A. neutron irradiation B. neutron flux
C. temperature change D. fast neutron flux
5. Within the reactor vessel the core support barrel is a _____.
A. core shroud B. support structure
C. vessel D. guide tube
6. The core support barrel assembly is supported _____.
A. at its upper end B. by its lower end
C. by snubbers D. by lower support structure
7. The fuel assemblies are put on the _____.
A. core support barrel B. plate
C. lower support structure D. lower flange
8. The lower end of the core support barrel is restricted by _____.
A. flange B. snubbers C. bolts D. keys
9. The space between the core shroud and the core support barrel is filled with _____.
A. fuel B. helium C. neutrons D. coolant
10. The reactor vessel is a large _____, steel made container.
A. safety B. movable C. cylindrical D. long
11. A number of nozzles is protruded from the vessel _____, to provide connection to the piping of the primary coolant system.
A. waist B. top C. bottom D. outside

12. The core support barrel is used to_____.

A. guide the coolant flow

B. shield neutrons

C. contain coolant

D. support the nuclear core

关键字解释

CFR: Code of Federal Regulations

联邦法规全书(美国)

Charpy test: A notched bar impact test in which a notched specimen, fixed at both ends, is struck behind the notch by a striker carried on a pendulum. The energy absorbed in fracture is measured by the height to which the pendulum rises.

摆锤式冲击试验: 一种切口试件冲击试验, 两端固定的切口试件从切口的背后被一个装在摆锤上的锤撞击, 断裂时吸收的能量通过摆锤升高的高度来计算。

Core support barrel: A circular cylinder, including a external ring flange at the top, supports the lower support structure upon which the fuel assemblies rest.

活性区支撑吊篮: 上端部有外法兰的圆筒, 它支撑下支撑构件, 燃料组件放在下支撑构件上。

Decay heat: The heat produced by the radioactive decay of fission products in a reactor core.

衰变热: 在反应堆堆芯中, 由裂变产物的放射性衰变所产生的热量。

Detector: A device which can indicate the physical change induced by radiation.

探测器: 一种可探测到辐射引起的物理量变化的装置。

Ductility: Ability of metals or alloys to retain strength and freedom from cracks when shape is altered.

延展性: 金属或合金在形状被改变时保持强度而且不出现裂纹的能力。

Forging: The operation of shaping (hot) malleable metals by means of hammers or presses.

锻造: 用锤或压力机将可锻金属(热)成形的加工方法。

NVT: Total Integrated Neutron Flux

积分中子通量

Reactor vessel internals: The assembly composed of support structure, low support structure and core shroud within the reactor vessel.

反应堆堆内构件: 反应堆容器内由堆芯内支撑结构、下支撑件和活性区围板组成的组件。

Spacer grid: The grid consists of a series of strips joined in an egg-crate fashion and welded together, to maintain spacing between the fuel rods and restrict their movement.

定位格架: 由许多条带焊接在一起构成的蛋篮形网格架, 用来保持燃料元件之间的距离并限制燃料元件之间的相对运动。

Specimen: A sample; A part that is representative of the whole, or an individual representative of a large group.

样品: 一种样本; 能代表整体的一部分或代表群体的一份。

Thermocouple: Two wires of different metals are joined at their ends to form a loop, when their junction is heated, a current will flow round the loop.

热电偶: 把两种不同材料金属线的端点连接起来形成一回路, 当它们的连接点被加热时, 则会有电流通过。

Toughness: Ability of material to withstand disruption.

韧性: 材料抗断裂的能力。

词 汇

alignment [ə'lainmənt] *n.* 成直线

amplitude [ˈæmplɪtju:d] *n.* 振幅

appendix [ə'pendɪks] *n.* 附录

assemblage [ə'semblɪdʒ] *n.* 集合, 总体

baseline *n. & a.* 基线; 基本的

beam [bi:m] *n.* 拉杆, 梁

blind [blaɪnd] *a.* 封闭的

boss [bɒs] *n.* 凸台, 浮凸

brittle ['brɪtl] *a.* 脆的

bypass ['baɪpɑ:s] *n.* 旁通, 支路

capsule ['kæpsjʊl] *n.* 腔, 囊, 密封壳

Charpy (impact) [tʃɑ:pɪ] *n.* 摆锤冲击

contour ['kɒntʊə] *n.* 形状, 外形

discrete [dɪs'kri:t] *a.* 不连续的, 分立的

dome [dəʊm] *n.* 圆拱顶

ductibility [ˌdʌktɪ'bɪləti] *n.* 延展性, 可锻性

flange [flændʒ] *v. & n.* 带法兰; 法兰

flexural ['flekʃərəl] *a.* 挠性的, 弯曲的

fluence [fluəns] *n.* 能量密度 (J/m^2), (中子) 注量

forging ['fɔ:dʒɪŋ] *n.* 锻件

hard-facing *n.* 表面硬化

hemispherical [hemi'sferɪkəl] *a.* 半球形的

hookup ['hʊkʌp] *n.* 挂钩, 连接器

integral ['ɪntɪgrəl] *a.* 整体的

keyway *n.* 键槽, 销座

ledge [ledʒ] *n.* 突出部分

lug [lʌg] *n.* 突出部, 凸台

mating ['meɪtɪŋ] *n.* 配合件, 相连件

nil-ductility *n.* 零延性

- nozzle ['nɒzl] *n.* 接口
 phosphorous ['fɒsfərəs] *n.* 磷
 slot [slɒt] *n. & v.* 槽; 开槽
 snubber ['snʌbə] *n.* 缓冲器
 specimen ['spesimin] *n.* 样品, 试件
 stamp [stæmp] *v.* 盖章通过
 stud [stʌd] *n.* 双头螺栓
 sulfur=sulphur ['sʌlfə] *n.* 硫
 susceptible [sə'septəbl] *a.* 敏感的
 taper ['teipə] *v.* 逐渐变细
 torsional ['tɔ:ʃənəl] *a.* 扭曲的
 threshold ['θreʃhəuld] *n.* 临界点, 限度
 weld [weld] *v. & n.* 焊

课文注释

1. The use of a forging as opposed to rolled...time.

句中 rolled 是滚压的意思, 它与 welded 一起修饰 plates; as opposed to 是与……相对照的意思; as 引导的这个短语在整句中作状语。整句译为: 与原来使用的滚压板材焊接的压力容器相对照, 使用锻件减少了焊缝的数量及容器的复杂性, 这样减少了建造时间。

2. A boss is located around the two outlet...outlet flow.

句中 boss 指堆芯吊篮上冷却剂出口与压力容器壁上冷却剂出口之间的连接件, 整句译为: 在两个出口接管的压力容器的内壁侧, 沿接口圆周有凸台, 它为内部构件提供一个连接面并引导出口水流。

3. Included in these baseline evaluations are...of temperature.

句中 drop weight, tensile 和 compact tension 都是 specimens 的定语; drop weight 落锤试验。整句译为: 这些基本的测定包括标准的和裂前的摆锤冲击试验样品、落锤试验样品、拉伸和压紧拉伸试验样品, 用这些来确定这些材料作为温度函数的静态和动态断裂强度特性。

4. In addition to test specimens...need arise.

句中 should the need arise 是将 should 提前的条件语句, 说明前面的介词短语。整句译为: 除了这些试验样本, 一块基础材料和一块焊接板与完整的文件和认证一起提交给用户保留, 以便在需要时用来检测和评估。

5. Alignment is accomplished...vessel closure head.

句中 equally spaced keys 是指间隔相等的键; keyways 是键槽。整句译为: 定位是由法兰上四个等间隔的键来完成的。这些键放在反应堆容器边缘和上封头之间的键槽内。

6. The shroud consists of a welded vertical...the core.

句中 channel 是动词, 有“引导”的意思。整句译为: 围板由焊在一起的立式钢板组成, 它引导冷却剂流过堆芯。

Unit 9 Reactor Core and Fuel

9.1 Introduction

The Reactor Core is designed to generate the required thermal output. The core provides a flow path for the forced circulation of coolant to remove heat generated by the core under all power operating conditions, and for the removal of decay heat by natural or forced circulation under shutdown conditions. The core contains the fuel rods which are designed to transfer the heat generated by fission reactions in the fuel pellets to the reactor coolant. The core also contains poison rods which provide negative reactivity in the fuel throughout core life in conjunction with soluble boron distributed throughout the reactor coolant. The heat transferred from the core and fuel is circulated by the reactor coolant to the steam generator where the heat is used to generate steam.

The physical properties of the fuel lattice provide a negative power coefficient under all conditions which provides inherent protection in the event of power transients. The major contributor to the power stability is the Doppler feedback which acts promptly in opposition to fuel temperature increase.

The fuel rod design provides two barriers to the escape of fission products. The ceramic pellets operate well below the melting point, and as a result, retain most of the fission products within the structure of the fuel. The small fraction of released fission products is contained within the hermetically sealed Zircaloy-4 cladding tube. In conjunction with the reactor coolant system envelope, the containment, and reactor building, these barriers provide five barriers against the release of fission products.

The cylindrical control elements are provided with full length guidance in the core and in the area of radial flow above the core, and move downward under the influence of gravity when released¹. The unique design of the upper guide structure and control element assemblies provides exceptional mechanical simplicity, and ruggedness for withstanding and protecting all control element assembly fingers from the combined effects of seismic and blowdown loads resulting from a LOCA. This further adds to the reliability of operation of the reactor protection systems. The control element arrangement provides ample shutdown margin under all conditions and without restriction of the fuel management scheme.

9.2 Core

The reactor core consists of an array of mechanically identical fuel assemblies in an arrangement that approximates a right circular cylinder. The core consists of control element

assemblies (CEAs), in-core instrumentation assemblies, and neutron sources inserted into the fuel assemblies, together with the reactor vessel internals which support and position the core and guide the coolant flow. There are no internal shrouds or channels. This simplifies the reactor design and permits flexibility in fuel arrangement. All of the structural materials in the active core zone, including the CEA guide tubes and the spacer grids, are Zircaloy, which minimizes the enriched uranium requirements of the core.

Fig.9.1 shows the cross-sectional arrangement of the fuel assemblies as positioned in the reactor core, and the arrangement of the core within the core support structure.

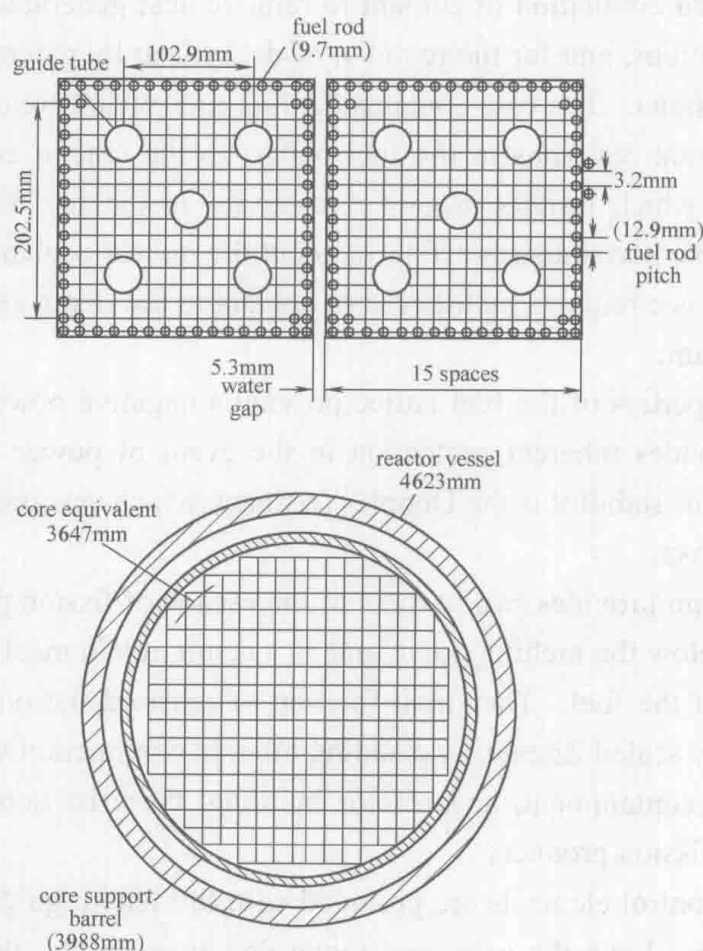


Fig.9.1 the cross-sections of fuel assemblies and reactor core

9.3 Fuel Rod

Each fuel rod consists of slightly enriched dish-ended UO_2 pellets, upper and lower spacer discs, a cladding tube, upper and lower end caps, and a helical spring². The fuel is manufactured in the form of pellets having a material microstructure and configuration which minimizes the effects of fuel densification during irradiation. The pellet column rests on an aluminum oxide spacer disc at the bottom of the cladding tube, and is held in place by a stainless steel compression spring acting on a similar spacer disc at the top of the column. The space at the top of the column allows for fuel and gas expansion. A cold diametrical gap of

sufficient size is allowed to provide for differential expansion between the cladding and fuel, which limits clad strain³. Net unrecoverable circumferential strain will not exceed one percent, as predicted by analysis considering clad creep and fuel-clad interaction effects.

The cladding tube is slightly cold-worked Zircaloy-4. The ratio of clad thickness to outer diameter is large enough to provide adequate margin in limiting the maximum short-term cladding stresses to less than the yield strength. The fuel rods are internally pressurized with helium to improve both their reliability and performance characteristics.

Pressurization results in reduced clad stresses and strains because of a smaller differential pressure across the cladding and a reduction in pellet-clad interactions. Thus, under internal pressure and with the relatively large wall thickness, the cladding tube is essentially free-standing. In addition, the high concentration of helium inside the tube improves gap thermal conductivity and thus lowers fuel pellet temperatures. Lower fuel temperatures are beneficial not only with respect to clad integrity at steady state conditions, but also in the case of transients where peak clad temperature is important.

The end caps are welded to the cladding tubes by an exclusive ABB-CE process involving magnetic force welding. This process has the advantage of producing a high degree of uniformity and consistency of weldment, thus providing a very high degree of leak tightness. There are no gas bubbles, as may be found in more conventional methods of welding, that could lead to weld porosity and leakage. All end cap welds are 100 percent helium leak tested.

9.4 Fuel Assembly

The fuel rods are arranged in a 16×16 square array to form a fuel assembly, as shown in Fig.9.2. The lower end cap of each fuel rod engages the lower end fittings of the fuel assembly, which provides lateral support of the rods. Lateral support and positioning is maintained throughout the length of the rods by spacer grids. The upper end of the fuel rod is free to expand in the axial direction.

Five Zircaloy guide tubes are welded to the spacer grids and mechanically attached to the upper and lower end fittings. These guide tubes are a key feature of the fuel assembly design. They provide channels which guide the CEAs over their entire length of travel within the core, as well as space for the in-core instrumentation and neutron sources. These guide tubes form the longitudinal structure of the assembly and offer the advantage of simple, rugged construction.

The guide tubes form a closely spaced repetitive array of water channels throughout the core, which minimizes local power peaking in adjacent fuel, while permitting the use of rugged, yet adequately flexible control elements. The use of rugged, large-size control elements simplifies the design and increases the reliability of associated reactor hardware. The low local power peaking allows fuel to operate at a high average specific power to achieve a

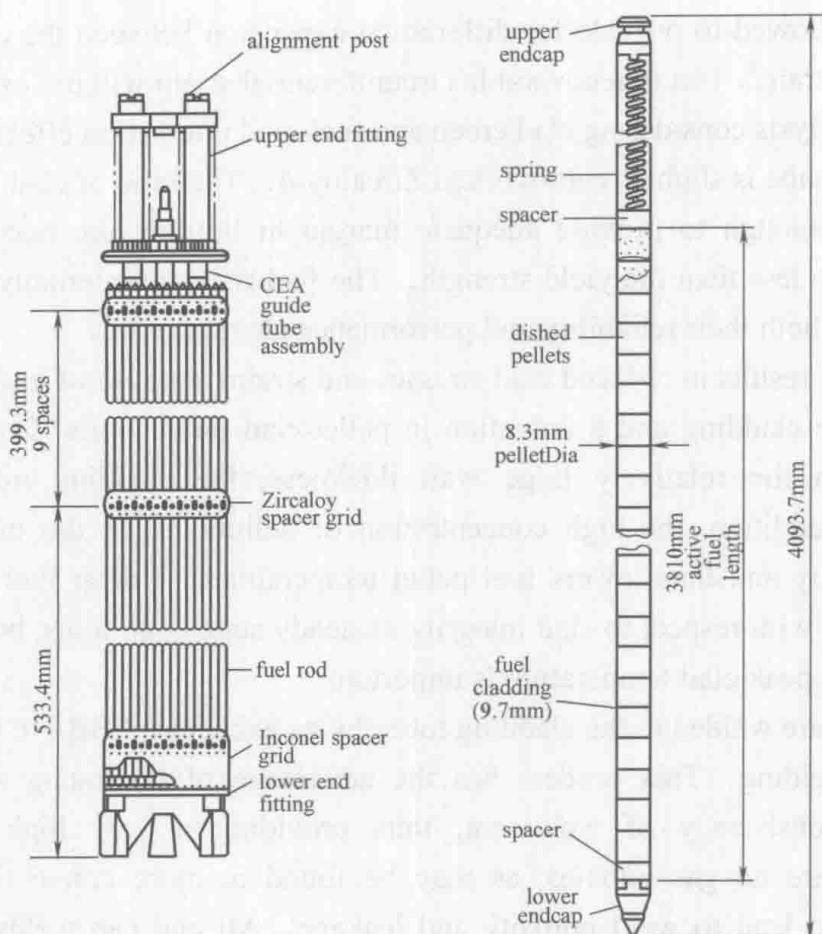


Fig.9.2 the fuel assembly and fuel rod

high burnup before discharge.

The fuel spacer grids maintain spacing between the fuel rods and restrict their movement. These grids consist of a series of preformed strips joined in an egg-crate fashion and welded together. Each fuel rod is laterally positioned by two leaf springs which are integral with two of the adjacent strips. The springs press the fuel rod against two stiff arches, which are integral with the other two adjacent strips. The grid perimeter strips have punched-out sections adjacent to the fuel rods to provide local cooling. Although the spacer grids position the fuel rods laterally, the rods are still free to expand axially.

The fuel assembly lower end fittings contain coolant flow holes. Alignment of the lower end and support of the assembly is provided by the core support structure. Lateral location of the upper end of each fuel assembly is provided by the lower ends of the control element shroud tubes which extend downward from the alignment plate which is part of the guide structure above the core⁴. Fuel assembly positioning is accomplished by the control element shroud tube extensions into which the fuel assembly extension is guided. The shroud tube extensions prevent the assemblies from moving so far upward as to disengage from the lower core support grid in the event of an accidental flow surge. Holddown springs on the upper end fitting react against the shroud tube extensions preventing fuel assembly uplift due to hydraulic forces and allowing differential expansion between the fuel assembly and the core

internals. The upper end fitting also serves as the lifting fixture for the fuel assemblies.

9.5 Control Element Assemblies and Drive Mechanisms

Control Element Assemblies (CEA) can be either four or twelve fingered as shown in Fig.9.3. The CEA fingers are fixed to a spider which serves as the central support structure for the 4-and 12-fingered CEA assemblies. The hub of the spider also couples the CEA to the drive mechanism through the Control Element Drive Mechanism (CEDM) extension shaft assembly. The grouping of CEAs, based on the core arrangement in Fig.9.2, is compatible with maximizing both the control and shutdown reactivity worth of the CEAs.

Each control element in the CEA is guided and shrouded from the main coolant flow above the core by the individual control finger shroud tubes and by the CEA shroud. In-core guidance is provided by the guide tube, which is an integral part of the fuel assembly. The guide tubes are open near the bottom to allow entry of coolant. Inlet orifices and control element fingers limit bypass flow through the tubes.

The CEAs are designed to provide a minimum service life of 15 effective full power years with an objective of 20 effective full power years, utilizing absorber and cladding materials which have proven successful performance based on extensive operating experience in ABB-CE plants. All CEAs used in the core design contain neutron absorbing (poison) materials which extend over the active length when fully inserted. The 12-finger full-strength CEA contains boron carbide as the strong neutron absorbing material. The 4-finger full-strength CEA contains silver-indium-cadmium as the strong neutron absorbing material. The part-strength CEA contains Inconel 625, a relatively weak neutron absorbing material, which is well suited for the purpose of reactivity and axial shape control for daily load-follow maneuvering.

The boron carbide in the CEA fingers has a density of approximately 73 percent of the theoretical density. Each tube is sealed by welded end caps. A gas expansion space is provided above the poison column to limit maximum tube stress due to the internal pressure developed by the release of helium gas and free moisture in the boron carbide. The bottom portion of the 12-element CEA, which is subject to irradiation due to proximity to the active core, contains

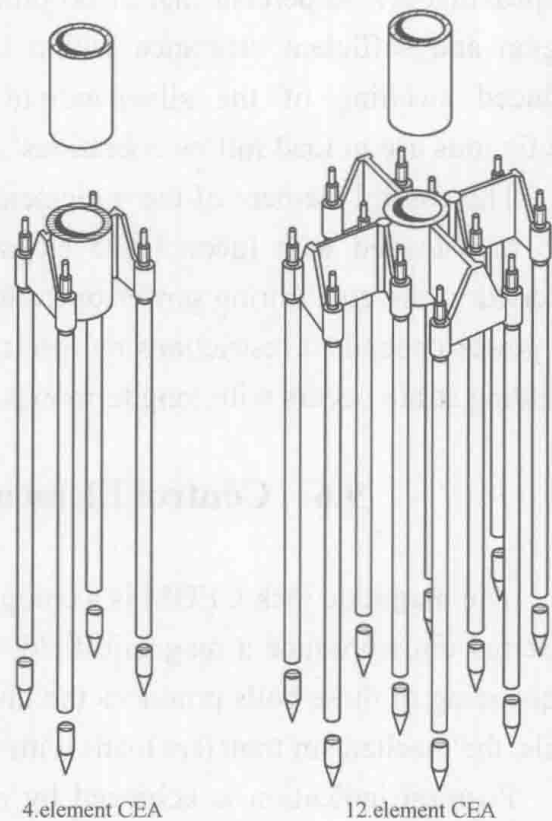


Fig.9.3 control element assemblies

reduced diameter boron carbide pellets which are wrapped in a sleeve of Type 347 stainless steel. This design accommodates the long term swelling of the boron carbide pellets by compressing the felt metal sleeve and thus limiting the induced cladding strain.

The 4-element full-strength CEA design is utilized in the regulating CEAs, which may be inserted in the core during normal power operation under automatic or manual control. Silver-indium-cadmium is used over the active length to provide strong neutron absorber worth (approximately 90 percent that of boron carbide). The absorber material has a central hollow region and sufficient clearance within the cladding to accommodate long term irradiation induced swelling of the silver-indium-cadmium under maximum CEA duty, including continuous use in load follow operations⁵.

The control element of the 4-element part strength CEA (PSCEA), consists of an Inconel 625 tube loaded with Inconel 625 slugs over the full active length. The PSCEA, may be inserted in the core during power operation and, unlike the full strength regulating CEAs, has no power dependent restrictions on insertion. Because of the use of Inconel 625 absorber, no cladding strain occurs with long term exposure of the PSCEA to core operating conditions.

9.6 Control Element Drive Mechanisms (CEDM)

The magnetic jack CEDM is a completely sealed, magnetically operated linear actuator. External coils produce a magnetic field, which operates driving and holding latches. Proper sequencing of these coils produces the linear motion of the driveshaft. Using external driving coils, the mechanism transfers loads with no sliding wear on latching members.

Position indication is achieved by means of redundant reed switch position transmitter assemblies (two per CEDM) attached to the upper pressure housing, and also by counting power pulses which actuate the CEDM driving CEAs up or down. Each reed switch assembly is operated by a magnet in the upper end of the driveshaft.

All electrical connections are made at the top of the mechanism with pin type stainless steel electrical connectors. A shroud surrounds the upper pressure housing enclosing the reed switch position transmitter assemblies, providing additional structural support for the CEDM, and serving as a lifting device for removal of the coil stack assembly.

9.7 Neutron Sources

The neutron sources are a combination type consisting of Californium-252 as the startup portion and Antimony/Beryllium as the sustainer. The californium is contained in stainless steel capsules which are then encapsulated in an outer stainless steel clad having an O.D. of 20.6 mm.

The Antimony/Beryllium pellets are also encapsulated in the outer stainless steel cladding. The active portion of the source assembly is shipped to the site in an approved

shipping container where it undergoes final assembly and installation. The source is positioned in an empty fuel assembly CEA guide tube prior to fuel loading of the reactor core⁶. The fuel assemblies containing the sources are placed in core locations adjacent to the start-up detectors. However, during refueling it is necessary to relocate the source to other fuel assemblies.

9.8 Burnable Absorber Rod Assembly

The fuel assembly is designed to incorporate integral burnable absorber rods in the fuel lattice, which remain in the assembly throughout its lifetime and require no separate handling or storage. Burnable absorber rods are provided in the initial core and in higher enrichment UO_2 feed fuel to enable extended burnup of the fuel in reload cycles. The purpose of the burnable absorber rods is to provide a more negative core moderator temperature coefficient at beginning-of-cycle, and to improve core thermal margin by flattening the radial power distribution over cycle.

The standard burnable absorber rod design has $\text{Al}_2\text{O}_3\text{-B}_4\text{C}$ absorber pellets contained within Zircaloy-4 tubing with welded end caps. The burnable absorber rod assembly is similar to that of the fuel rod, and is designed conservatively to provide integrity of the cladding and pellets throughout the fuel assembly lifetime. The active length of burnable absorber is approximately 3 454 mm. The absorber pellets consist of a relatively small fraction of B_4C particles dispersed in a continuous Al_2O_3 matrix with a bulk density greater than 93% of the theoretical density. The design of the burnable absorber rod assembly thus provides the capability for strong initial absorber strength, and low residual parasitic absorption at the end of the irradiation cycle. The $\text{Al}_2\text{O}_3\text{-B}_4\text{C}$ burnable absorber rod design provides performance capability for longer cycle lengths (e.g., 18 to 24 months) and is well proven in operating ABB-CE plants.

Self-test

1. The heat is transferred by the reactor coolant from the core to the ____.
 - A. secondary loop
 - B. outlet nozzles
 - C. steam generators
 - D. pressurizer
2. The burnable absorber rod assembly provides the capability of high absorber strength ____.
 - A. in all its life
 - B. at the beginning of refueling
 - C. at the end of refueling cycle
 - D. during operation
3. The fuel rod provides ____ to the escape of fission products.
 - A. two barriers
 - B. three barriers
 - C. seals
 - D. a barrier

4. The fuel assemblies used in a reactor are _____.
A. similar B. different C. three types D. same
5. The CEA guide tubes are made of _____.
A. stainless steel B. ceramic C. zircaloy D. B₄C
6. In the fuel rod the spring is set on the _____.
A. pellet column B. spacer disc C. end cap D. top
7. The helium inside the fuel cladding improves the _____.
A. characteristics B. gap heat conductivity
C. gap heat convection D. clad strain
8. The CEA guide tubes are welded to the _____.
A. spacer grids B. lower end fittings
C. control rods D. end cap
9. The CEA guide tubes are arranged _____.
A. in the fuel assembly B. around the fuel assembly
C. in the center of fuel assembly D. outside the fuel assembly
10. The fuel spacer grids are made of _____.
A. stiff arches B. springs C. plates D. zircaloy strips
11. The coolant can _____ through the guide tubes.
A. not flow B. be C. flow D. pass
12. After long time radiation the boron carbide will _____.
A. decrease in mass B. be swelling
C. increase in mass D. be dissipated

关 键 字 解 释

Absorber rod: Neutron absorber rod consists of boron or cadmium, moving in and out of reactor core to vary reactivity.

吸收棒(控制棒): 由硼或镉等组成的中子吸收棒, 插入或抽出反应堆堆芯以控制反应性。

Burnable absorber rod: A rod similar to control rod introduced into reactor core to reduce initial reactivity but becoming progressively less effective as burn-up proceeds. This helps to counteract the fall in reactivity as the fuel is used up.

可燃吸收棒: 类似于控制棒的元件棒, 插入反应堆的堆芯内, 可减少初始反应性, 随着燃料的增加, 其有效性逐渐降低。它可用来补偿由于燃料消耗而造成的反应性下降。

CEDM: Control Element Drive Mechanism.

控制棒驱动机构

Creep: Continuous deformation of metals under steady load.

蠕变: 金属在稳定载荷下的连续变形。

Curtain: Neutron-absorbing shield, usually made of cadmium.

屏蔽: 通常指用镉制成的中子吸收屏蔽层。

Fuel assembly, Fuel bundles: A group of nuclear fuel elements forming a single unit for purposes of charging or discharging a reactor.

燃料组件: 由一些燃料元件组成的、为反应堆装料和卸料的单元。

ICIs: In-Core Instrumentation assemblies.

堆内仪器组件

Lattice: Regular geometrical pattern of discrete installation of fuel rods in a nuclear reactor.

栅格: 在核反应堆中, 燃料棒分开而有规律的几何布置。

Neutron source: One source giving a high neutron flux.

中子源: 一种可以给出高中子通量的源。

Plug: Piece of metal used to close the passage of a channel through which the coolant passing.

阻力塞: 在反应堆中用来堵住冷却通道的金属部件。

Porosity: The occurrence of small pores throughout the mass of a substance.

多孔性: 物体上有许多遍布的小孔。

PSCEA: Part Strength Control Element Assembly.

部分强度控制棒组件

Reactivity worth: Change of reactivity of reactor caused by the addition or removal of a material.

反应性价值: 由于添加或移去材料而引起反应堆的反应性变化。

Specific Power: The ratio of total energy released to initial fuel load, usually expressed in megawatts per tone.

比功率: 反应堆释出的总能量与原装燃料的比值, 一般用(兆瓦/吨)来表示。

Swelling: Change of volume of fuel rods which may occur during irradiation.

肿胀: 燃料棒受辐照时可能发生的体积变化。

词 汇

actuator ['æktʃueɪtə] *n.* 传动机构、执行机构

airtight ['eətaɪt] *a.* 气密的

antimony ['æntiməni] *n.* 锑

cadmium ['kædmiəm] *n.* 镉

californium [kæli'fɔ:niəm] *n.* 锿

ceramic [si'ræmɪk] *a.* 陶瓷的

clearance ['kliərəns] *n.* 间隙

cold-worked *a.* 冷加工的

conservative [kən'sə:vətɪv] *a.* 保守的

consistency [kən'sɪstənsi] *n.* 一致性

contaminate [kən'tæmɪneɪt] *v.* 污染

Doppler ['dɒplə] *n.* 多普勒

- elevation [ˌeli'veiʃən] *n.* 高度
 encapsulate [in'kæpsjuleit] *v.* 封装
 fitting [ˈfitɪŋ] *n.* 配件, 装配部件
 grid [ɡrɪd] *n.* 栅格架
 helical [ˈhelɪkəl] *a. & n.* 螺旋(的)
 Inconel [in'kɒnəl] *n.* 因科镍合金
 indium [ˈɪndiəm] *n.* 铟
 jack [dʒæk] *n.* 传动
 latch [lætʃ] *n.* 凸轮
 longitudinal [lɒndʒɪ'tjuːdɪnəl] *a.* 纵向的
 microstructure ['maɪkrəʊ'strʌktʃə] *n.* 微观结构
 midplane ['mɪd,pleɪn] *n.* 中平面
 parasitic [ˌpærə'sɪtɪk] *a.* 附加的
 protrude [prə'truːd] *v.* 凸出
 proximity [prɒk'sɪmɪti] *n.* 邻近
 punched-out *n.* 切口, 凸起
 reed switch 弹簧开关
 reliance [ri'laɪəns] *n.* 信赖, 依赖
 repetitive [ri'petɪtɪv] *a.* 重复的
 ruggedness ['rʌɡɪdnɪs] *n.* 强度, 坚固性
 significance [sɪɡ'nɪfɪkəns] *n.* 意义, 含义
 sleeve [sli:v] *n.* 套管
 spacer grid 定位格架
 spider ['spaɪdə] *n.* 星形接头
 stiff [stɪf] *a.* 刚性的
 sustainer [səs'teɪnə] *n.* 二次中子源, 维持者
 thimble ['θɪmbl] *n.* 套管
 inherent safety 固有安全性

课 文 注 释

1. The cylindrical control elements are provided with...released.

句中 in the area of radial flow above the core 是指堆芯燃料组件上方冷却剂径向横流的区域。整句译为: 圆棒形的控制元件在堆芯内和燃料上方冷却剂径向横流区全长度导向, 放开后可靠重力插入堆芯。

2. Each fuel rod consists of slightly enriched...helical spring.

句中 dish-ended 指燃料芯块两端有碟形凹窝; spacer disc 是两端防止热量轴向传递的隔热垫片。整句译为: 每个燃料棒由带有碟形凹窝的微浓缩 UO_2 芯块、上、下隔热垫片、包壳管、上、下端塞和螺旋弹簧组成。

3. A cold diametrical gap of sufficient size...clad strain.

句中 cold diametrical gap 是冷态径向间隙; limits clad strain 指减少了包壳的应力。整句译为: 一个足够的冷态径向间隙, 可以允许燃料和包壳有不同的膨胀。这可以减少包壳应力。

4. Lateral location of the upper end of...above the core.

句中 control element shroud tubes 是位于堆芯上方的控制元件导向筒。整句译为: 每个燃料组件的上方定位是由位于堆芯上方的控制元件导向筒提供的。导向筒从定位板向下伸出, 是堆芯上方导向构件的一部分。

5. The absorber material has...follow operations.

句中 maximum CEA duty 指最大的控制组件负载。整句译为: 控制材料有中心孔, 并与包壳之间有足够的间隙, 这样可以缓解银—铟—镉在最大负载下, 包括在跟踪负荷下连续运行时, 长期辐照造成的肿胀问题。

6. The source is positioned...the reactor core.

句中 prior to...是在……之前, 它与后面的成分一起在句中作状语。整句译为: 在燃料装入反应堆之前, 中子源放在空的控制棒导向管中。

Unit 10 The Main Components in PWR Coolant System

10.1 Steam Generators

The two vertical, U-tube steam generators used in the System 80+ NSSS provide the means of transferring heat from the primary system to the secondary system (i.e., main steam, turbine and feed and condensate systems)¹.

The System 80+ steam generators have been improved relative to the currently operating System 80 units. The operating hot leg temperature has been reduced from 327°C to 324°C to

improve core operating margin. Consequently, the steam generator steam pressure has been reduced from 73.8 bars to 69 bars, and the overall heat transfer area has been increased by 17%. A tube plugging margin of 10% is afforded by the increased number of tubes. The total liquid inventory on the secondary side has been increased by 25% to provide a significantly greater margin in terms of “boil dry time” in the event of a loss of all feedwater.

Each steam generator is a recirculating, vertical, U-tube heat exchanger with an integral axial flow economizer as shown in Fig.10.1.

Both primary tube and secondary shell sides of the steam generator are designed and fabricated in accordance with ASME Boiler and Pressure Vessel Code Section III, Rules for Construction of Nuclear Vessels.

Reactor coolant enters the steam generator inlet plenum via the primary inlet nozzle, flows up through the tube sheet and U-bend heat transfer tubing, returns through the tube sheet to the steam generator outlet plenum and exits via two outlet nozzles. Vertical divider plates separate the primary inlet and outlet plenums of the steam

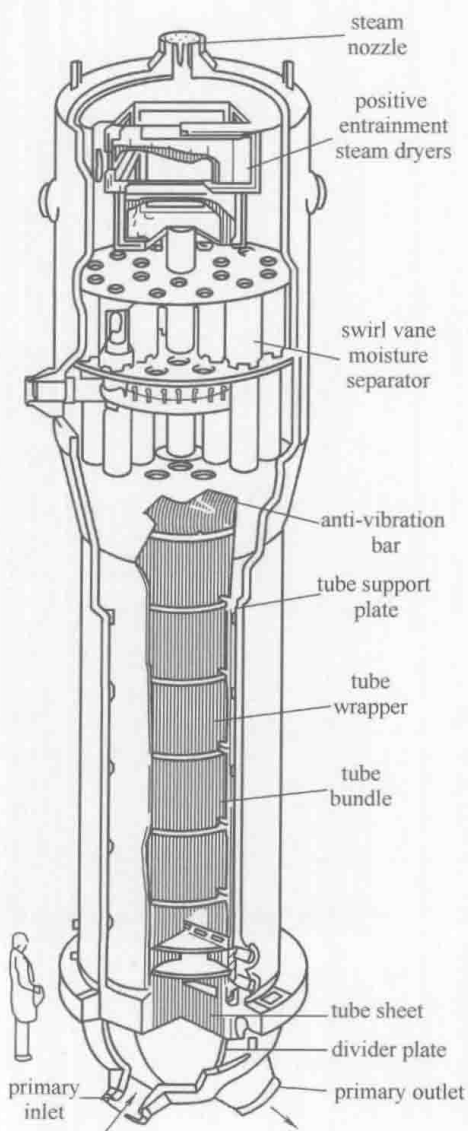


Fig.10.1 inverted U-tube type steam generator

generator. All surfaces in contact with reactor coolant are either Ni-Cr-Fe alloy or clad with stainless steel or Ni-Cr-Fe alloy. Specifically, the heat transfer tubes are made with thermally treated Alloy 690. The tube sheet cladding is weld deposited to obtain the strongest type of metallurgical bond. The center support cylinder minimizes bending moments on the tube sheet, and provides a solid assembly for distribution of support loads. Manway are provided in the two plenums.

The Alloy 690 heat transfer tubes are connected to the tube sheet by first being seal welded to the primary side cladding and then expanded (i.e., explosively expanded) into the tube sheet by detonating an explosive contained in a plastic sheath inserted into the tube². The detonation applies a uniform force throughout the thickness of the tube sheet. In contrast to mechanical rolling processes, there is no scoring or local thinning and minimal springback of the tube material. Expansion through the full tube sheet thickness also minimizes the potential for solids collection and crevice corrosion by eliminating the gap between tube and tube sheet on the secondary side.

The integral axial flow economizer is shown in Fig.10.2. The economizer increases the cold leg side temperature difference for heat transfer by bringing relatively cold feedwater into contact with the primary outlet or cold leg side of the tube bundle. Feedwater enters the economizer region via the two main feedwater nozzles. This feedwater is introduced into a distribution box of rectangular cross section which forms a half ring around the cold leg portion of the tube bundle. Feedwater leaves the distribution box through uniformly distributed holes at the bottom and then flows radially inward across the tube sheet beneath a flow distribution baffle. The flow distribution baffle is designed to provide a uniform axial mass velocity of feed water at the economizer inlet.

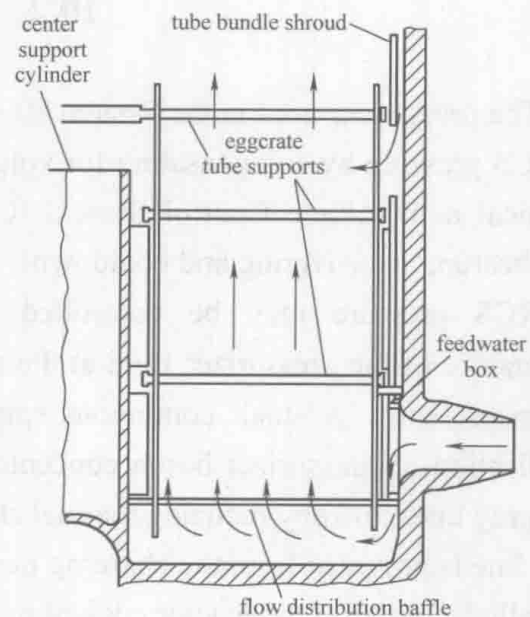


Fig.10.2 the economizer

Water removed by the steam separators recirculates into the evaporator section through an annular downcomer formed by the pressure shell and the tube bundle shroud³. The downcomer and tube bundle shroud are designed so that approximately 60 percent of the recirculating water enters the bundle at the tube sheet on the hot leg side and approximately 40 percent enters through an opening in the shroud on the cold leg side at the top of the economizer section. In this manner, density differences between the hot leg (evaporator) side and the cold leg (economizer) side are reduced at the level of the economizer discharge, and the potential for lateral density wave instabilities is minimized. At full power approximately 10 percent of feedwater flow is admitted through the downcomer feedwater nozzle into the downcomer

section. With partial feed flow to the downcomer, a degree of subcooling is obtained in the downcomer, promoting an increased recirculation ratio. Even without subcooling in the downcomer, recirculation ratios of 3.5 to 1 are expected in the System 80+ steam generators. Introduction of feedwater into the downcomer will also retard boiling of the recirculating flow entering the evaporator region, further improving hydraulics in this region⁴.

Blowdown fluid enters the duct through holes located at the outside radius of the curved portion located adjacent to the center support cylinder. This is the region of lowest horizontal velocity across the tube sheet where settling of heavier solid particles is most likely to occur. The duct terminates before reaching the generator shell and flow is directed through drilled passages in the tube sheet exiting at the blowdown nozzles. The steam generator internal blowdown ducts and passages are sized to accommodate up to a maximum blowdown rate of approximately 10 percent of the individual steam generator's maximum steaming rate⁵. The connecting steam generator blowdown system provides the means to quench, collect, and purify the blowdown fluid.

10.2 Pressurizer

The pressurizer used in the System 80 + Reactor and RCS provides a means of maintaining the RCS pressure by compensating for volumetric changes in the RCS in conjunction with the Chemical and Volume Control System (CVCS) during various modes of system operation (e.g., heatup, maneuvering and cooldown).

RCS pressure may be controlled automatically or manually by maintaining the temperature of the pressurizer fluid at the saturation temperature corresponding to the desired system pressure. A small continuous spray flow is maintained to the pressurizer to avoid stratification of pressurizer boron concentration and to maintain the temperature in the surge and spray lines, thereby reducing thermal shock as the spray control valves open. An auxiliary spray line is provided from the charging pumps to permit pressurizer spray during plant heatup, or to allow cooling if the reactor coolant pumps are shutdown.

During load changes, the pressurizer limits pressure variations caused by expansion or contraction of the reactor coolant. The average reactor coolant temperature is programmed to vary as a function of plant power level. Reduction in RCS load is followed by a decrease in the average reactor coolant temperature to the programmed value for the lower power level. The resulting contraction of the coolant lowers the pressurizer water level, causing the RCS pressure to decrease. This pressure reduction is partially compensated by flashing of pressurizer water into steam. All pressurizer heaters are automatically energized on low system pressure, generating steam and further limiting any pressure decrease. Should the water level in the pressurizer drop sufficiently below its setpoint, the letdown control valves close to a minimum value, and additional charging pumps in the chemical and volume control system are automatically started to add coolant to the system and restore pressurizer level⁶.

When steam demand is increased, the average reactor coolant temperature is raised in accordance with the coolant temperature program. The expanding coolant from the reactor coolant piping hot leg enters the bottom of the pressurizer through the surge line, compressing the steam and raising system pressure. The increase in pressure is moderated by the condensation of steam during compression and by the decrease in bulk temperature in the liquid phase. Should the pressure increase be large enough, the pressurizer spray valves open, spraying coolant from the reactor coolant pump discharge (cold leg) into the pressurizer steam space. The relatively cold spray water condenses some of the steam in the steam space, limiting the system pressure increase.

A number of the heaters are connected to proportional controllers, which adjust the heat input to account for steady state losses and to maintain the desired steam pressure in the pressurizer. The remaining heaters are connected to on-off controllers. These heaters are normally deenergized but are automatically turned on by a low pressurizer pressure signal or a high level error signal. This latter feature is provided since load increases result in an in-surge of relatively cold coolant into the pressurizer, thereby decreasing the bulk water temperature⁷. The CVCS acts to restore level, resulting in a transient pressure below normal operating pressure. To minimize the extent of this transient, the backup heaters are energized, contributing more heat to the water. Backup heaters are deenergized in the event of concurrent high-level error and high-pressurizer pressure signals. A low-low pressurizer water level signal deenergized all heaters before they are uncovered to prevent heater damage.

10.2.1 Configuration

The RCS pressurizer is a vertically mounted, bottom supported, cylindrical pressure vessel (Fig.10.3). Replaceable direct immersion electric heaters are installed vertically in the bottom head. The pressurizer is furnished with nozzles for spray, surge, safety valves, and pressure and level instrumentation. The pressurizer surge line is connected to one of the reactor coolant hot legs and the spray lines are connected to two of the cold legs at the reactor coolant pump discharge.

The pressurizer heaters are single unit, direct immersion heaters that protrude vertically into the pressurizer through sleeves welded in the lower head. Each heater is internally restrained from high amplitude vibrations and can be individually

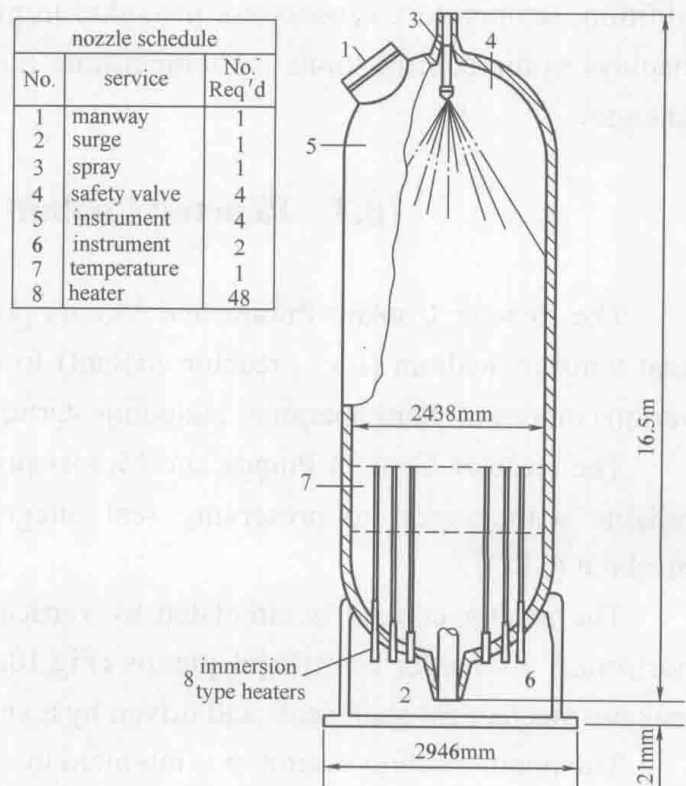


Fig.10.3 the pressurizer

removed for maintenance during plant shutdown. Protective devices for pressurizer heaters are provided to prevent damage by overloads and short circuits. These devices are usually located in the heater switchgear.

10.2.2 Pressurizer-RCS Overpressure Protection

Overpressure protection of the RCS is provided in accordance with the requirements set forth in the ASME Boiler and Pressure Vessel Code, Section III. Four spring-loaded safety valves installed in piping connected to the top of the pressurizer are provided. The valves have an enclosed bonnet and a balanced bellows to compensate for backpressure. The valves are sized to pass sufficient pressurizer steam to limit the reactor coolant system pressure. The pressurizer safety valves discharge to the in-containment refueling water storage tank (IRWST) where the steam is released under water to be condensed and cooled. If the steam discharge exceeds the capacity of the IRWST, it is relieved to the containment atmosphere via vents installed in the tank.

10.2.3 Safety Valve Position Indication

Indirect indication of pressurizer safety valve leakage is provided by a decrease of pressurizer pressure and pressurizer level, monitored by safety-grade instrumentation.

Positive indication of pressurizer safety valve position by acoustic monitoring of flow is provided in the control room. The sensing instrumentation is environmentally qualified to function in a post-LOCA environment in accordance with Regulatory Guide 1.89⁸. A plant annunciator alarm is provided to indicate valve opening. The valve position instrumentation is powered from a reliable instrument bus with Class 1E backup power. In addition, temperature sensors are provided in piping downstream of the safety valves with displays in the control room. A temperature increase will indicate pressurizer safety valve leakage.

10.3 Reactor Coolant Pumps and Motors

The Reactor Coolant Pumps and Motors provide the motive force for circulation of the heat transfer medium (i. e. , reactor coolant) from the reactor to the steam generators during various modes of plant operation including startup, maneuvering and shutdown.

The Reactor Coolant Pumps and Motors provided for the System 80+ design are highly reliable with respect to preserving seal integrity during various events including station blackout (SBO).

The reactor coolant is circulated by vertical, single stage, single bottom suction, single horizontal discharge, centrifugal pumps (Fig.10.4). The pump shaft is sealed by controlled leakage mechanical shaft seals and driven by a vertical AC induction motor.

The pump rotating assembly is mounted in a diffuser type pump casing. The pump casing is a one-piece design in accordance with applicable sections of Section III, ASME Boiler & Pressure Vessel Code. The one piece casing reduces the ASME Section XI examination

requirements to only the welds between the pump casing and the suction and discharge nozzle extension pieces and other minor items⁹. The pump impeller is face spliced and locked to the pump shaft. The lower portion of the pump rotating assembly is supported by a self-aligning water lubricated bearing mounted in the pump cover and above the impeller. Additional shaft support is provided by oil lubricated thrust bearings mounted on the pump shaft and in the motor. The motor is coupled to the pump by a flexible diaphragm type coupling.

The temperature of the water in the seal assembly is maintained within acceptable limits by externally supplied seal injection water. Water cooled heat exchangers are also furnished to provide the necessary cooling should seal injection fail. The pump is capable of operating continuously without seal injection or without cooling water to the pump seal water coolers. If both seal injection and component cooling to the pump seal water coolers are lost, the pump must be shutdown. If component cooling water to the

oil lubricated bearing cooler is lost, but seal cooling is available by either seal injection water or seal component cooling water, the pump can operate for up to ten minutes without damage before shutdown is required. Once shutdown, the seals are capable of withstanding the effects of a loss of all cooling water without creating a small break LOCA.

The motors are designed to start and accelerate to operating speed under full load with an initial drop to 80 percent of rated voltage at the motor terminals. Each motor is provided with an anti-reverse rotation device, designed to withstand each of the following conditions: motor starting torque resulting from reversed power leads; reactor coolant flow through the pump in the reverse direction of up to 52 percent of rated capacity with the motor deenergized; and reverse flow due to LOCA of sizes up to the largest remaining pipe break.

The motor is furnished with a flywheel to provide sufficient coast down flow for core cooling following a loss of power to the pumps. The flywheel design and manufacture is based on applicable requirements including an over speed test at 125 percent of rated speed. The

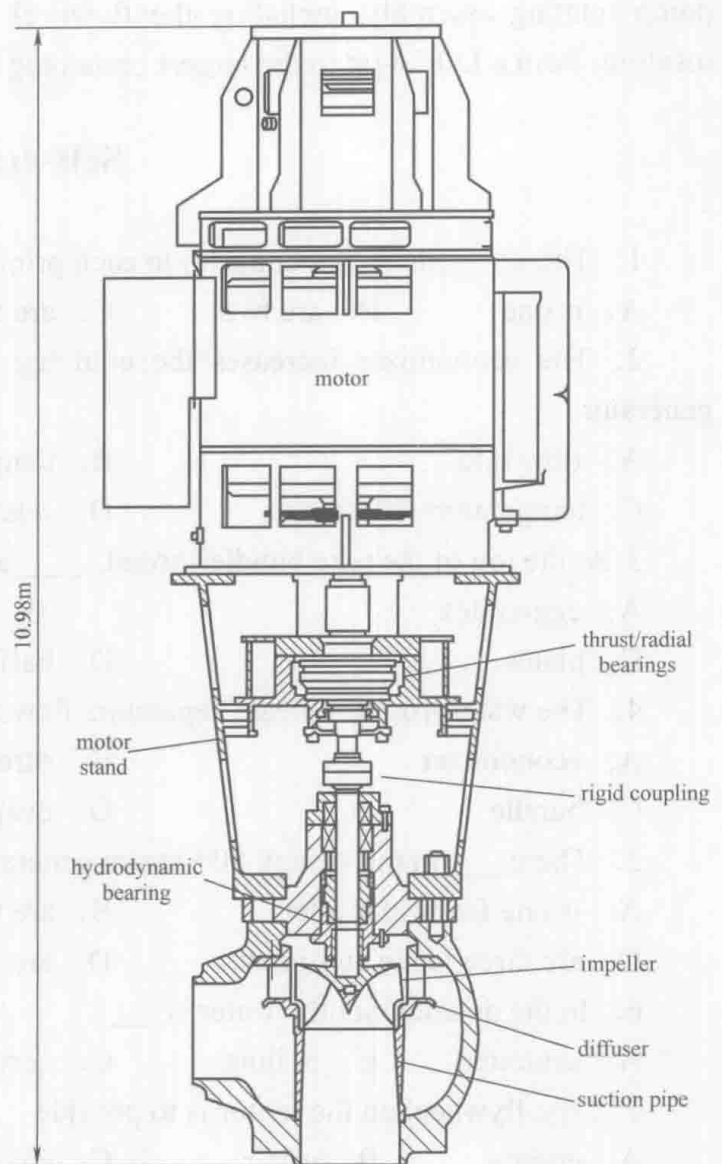


Fig.10.4 the centrifugal pump

resulting from a LOCA up to the largest remaining pipe break.

Self-test

1. There ____ steam generator(s) in each primary loop.
A. is one B. are two C. are three D. is no
2. The economizer increases the cold leg side ____ before coolant leaving the steam generator.
A. flow rate B. temperature
C. temperature difference D. relative flow
3. At the top of the tube bundle shroud, ____ are mounted.
A. egg-crates B. steam-water separators
C. plates D. baffles
4. The water from the steam separators flows down into the ____.
A. economizer B. shroud
C. bundle D. evaporator section
5. There ____ in the system 80+ steam generator.
A. is one feedwater inlet B. are two feedwater inlets
C. are three feedwater inlets D. are many feedwater inlets
6. In the downcomer the water is ____.
A. saturated B. boiling C. very cold D. subcooled
7. The flywheel on the motor is to provide ____.
A. energy B. water C. coast down flow D. torque
8. The pump is connected with the motor by a ____.
A. diaphragm B. tube C. coupling D. casing
9. The temperature of the water in the seal assembly is maintained within acceptable limits by ____.
A. water B. oil C. steam D. water and oil
10. ____ pressurizer heaters are connected to proportional controllers.
A. A few B. The remaining
C. Most of the D. Many
11. In accordance with the coolant temperature program, the average reactor coolant temperature is ____ when steam demand is increased.
A. constant B. fall C. raised D. high
12. The fluid from economizer section is ____.
A. two-phase flow B. boiling water
C. subcooled water D. saturated water

关键字解释

Backup: A secondary device designed to take over a specific task if the primary device malfunctions.

备用：当主要的设备失灵时，用来接替某一特定任务的辅助设备。

Blowdown: Draining some water from second side of steam generator bottom for minimizing the solids content of the water.

排污：从蒸汽发生器底部的二次侧引出一部分水，以减少二次侧水中的固体杂质含量。

Boil dry time: The time for evaporating all water in the steam generator.

蒸干时间：将蒸汽发生器内所有的水全部蒸发成蒸汽所用的时间。

Economizer: The region of tube bank, placed on the tube sheet, here the feed water is heated by the primary system water before it leaving the steam generator.

节热器：在蒸汽发生器底部管板上一回路出口处的一段管排，给水在这里被离开蒸汽发生器前的一回路水所加热。

Coupling: A connection between two co-axial shafts, conveying a drive from one shaft to another.

联轴节：连接两个同轴线轴的连接器，使驱动力从一轴传到另一轴。

Diffuser: A chamber surrounding the impeller of a centrifugal pump, in which part of the kinetic energy of the fluid is converted to pressure energy by a gradual increase in the cross-sectional area of flow.

扩压段：离心泵叶轮周围的管腔，在此腔内由于流道横截面积逐渐扩大，流体的部分动能转化为压力能。

Feedwater: The water, previously treated to remove air and impurities, which is supplied to a steam generator.

给水：事先经过处理，以消除空气和杂质而后送入蒸汽发生器以供蒸发的水。

Impeller: The rotating member of a centrifugal pump or blower, which imparts kinetic energy to the fluid.

转子、动叶轮：离心泵或鼓风机中给流体以动能的转动部件。

NSSS: Nuclear Steam Supply System

核蒸汽供应系统

Quality: In a saturation condition the ratio of the vaporized portion to the total weight of liquid and vapor.

蒸汽干度：在饱和蒸汽中被蒸发的部分与液体及蒸汽总重量的比值。

RCS: Reactor Coolant System

反应堆冷却剂系统

Safety valve: A valve, spring loaded, fitted to a steam generator, to allow fluid to escape to the atmosphere when the pressure exceeds the maximum safe value.

安全阀：一种弹簧加载的阀，装在蒸汽发生器上，当压力超过最大安全值时，允许流体

溢出到大气中。

Thermal shock: That resulting when a body is subjected to sudden changes in temperature.

热冲击: 当物体受到温度突然变化时造成的冲击。

Switch gear: The generic name for that class of electrical apparatus whose sole function is to open and close electric circuits.

开关设备: 具有断开和闭合电路单一功能的电器的总称。

词 汇

acoustic [ə'ku:stik] *a.* 声音的

annunciator [ə'nʌnʃieɪtə] *n.* 信号器

blackout ['blækaut] *n.* 停电, 关闭

bonnet ['bɒnɪt] *n.* 阀盖

chloride ['klɔ:raɪd] *v.* 氯化物

crevice ['krevis] *n.* 裂缝, 缝隙

curved *a.* 弯曲的

deposit [di'pɒzɪt] *v.* 堆焊、覆盖

detonate ['detəuneɪt] *v.* 爆炸、爆燃

diaphragm ['daɪəfræm] *n.* 隔板, 垫圈

elevation [eli'veɪʃən] *n.* 正面图, 纵剖图, 标高

ferritic [fə'ritɪk] *a.* 铁素(体)的

inventory ['ɪnvəntɔ:ri] *n.* 总装量

letdown ['letdaʊn] *n.* 排出, 下降

maneuvering *n.* 操纵, 机动

manually ['mænjuəli] *ad.* 手动地

plugging *n.* 堵塞

preclude [pri'klu:d] *v.* 预防

quench [kwentʃ] *v.* 急冷

restrain [ris'trein] *v.* 约束

retard [ri'tɑ:d] *v.* 延迟

rolling ['rəʊlɪŋ] *n.* 滚压

scoring *n.* 划痕

sheath [ʃi:θ] *n.* 外壳, 套管

splice [splais] *v.* 镶接

split [split] *v.* 分开, 分解

spring-back *n.* 回弹, 弹性后效

stratification [ˌstrætifi'keɪʃən] *n.* 分层

surge [sə:dʒ] *n.* 波动

surge line 波动管

tube sheet *n.* 管板

verify ['verɪfaɪ] *v.* 验证

volatile ['vɒlətaɪl] *a. & n.* 挥发性的; 挥发物

课 文 注 释

1. The two vertical, U-tube steam generators...condensate systems.

句中 80+ 是核动力装置的型号; feed and condensate systems 是凝给水系统。全句译为: 80+核蒸汽供应系统有两个立式 U 形管蒸汽发生器, 它们将一回路系统(即反应堆及主冷却剂系统)的热量传递给二回路系统(即主蒸汽系统、汽轮机和凝给水系统)。

2. The Alloy 690 heat transfer tubes are connected...into the tube.

句中 explosive 是名词, 意思是炸药; contained 是过去分词, 它引导的分词短语作 explosive 的定语。全句译为: 690 合金传热管与管板相连接, 首先在一回路侧的堆焊层处密封焊接, 然后胀接(爆炸膨胀)到管板上, 其方法是将炸药装在塑料包壳中插入管内引发爆炸。

3. Water removed by the steam separators...bundle shroud.

句中 formed 引导的分词短语作 downcomer 的定语。全句译为: 从分离器分离出来的水通过环形的下降段再循环进入蒸发段, 下降段是由蒸汽发生器的压力壳和管束围板构成的。

4. Introduction of feed water into the downcomer will...in this region.

句中 entering 引导的现在分词短语作 flow 的定语, 这里 entering 有正在进入的意思。全句译为: 给水引入下降段会延迟进入蒸发段再循环流体的沸腾, 可以进一步改善这个区的水力学特性。

5. The steam generator internal blowdown ... maximum steaming rate.

句中 size 的意思是尺寸确定, accommodate 的意思是接纳。全句译为: 蒸汽发生器的内部排污导管和通道尺寸的确定, 要以接纳蒸汽发生器最大蒸汽产量的大约 10% 的排污流量为依据。

6. Should the water level in the pressurizer drop...restore pressurizer level.

句中 should 提前, 构成一个条件语句, 这里 should 的语法作用与 if 相类似; charging pumps 是“上充泵”。全句译为: 如果稳压器内的水位降低于它的设定点, 下泄控制阀关到最小值, 化学和容积控制系统的上充泵会自动启动, 将冷却剂打入系统, 使稳压器水位回复。

7. This latter feature is provided...the bulk water temperature.

句中 in-surge 指向稳压器内波动。全句译为: 负荷增加会造成相对冷的冷却剂向稳压器内波动, 使总体水温降低, 因此要提供后者的特性。

8. The sensing instrumentation is environmentally...Regulatory Guide 1.89.

句中 post-LOCA 指失水事故后; Regulatory Guide 1.89 是指规程导则第 1.89 条。全句译为: 根据规程导则第 1.89 条, 信号传感仪器要能满足在失水事故后的环境条件下仍能工作。

9. The one piece casing reduces the...other minor items.

句中 to 是与 reduces 相呼应的。全句译为: 这种整块的壳体, 使 ASME 第 XI 部分的检验要求减少, 只有泵壳与吸入口和排出口延伸段之间的焊缝以及其他小的项目上有检验要求。

Unit 11 Pressurized Water Reactor Systems and Containment

11.1 Plant Heat Balance and the Steam Cycle

The nuclear chain reaction in the reactor core produces energy in the form of heat as the fission fragments slow down and dissipate their kinetic energy in the fuel. This heat must be removed efficiently and at the same rate it is being generated in order ① to prevent overheating of the core and ② to transport the energy outside the core where it can be converted to a convenient form for further utilization. A fluid is used for this purpose. In the light-water reactor (LWR) case, water is used as both moderator and coolant. The energy transferred to the coolant, as it flows past the fuel elements, is stored in it in the form of temperature and pressure, and is called the ‘enthalpy’ of the fluid. In an electric power plant the energy stored in the fluid is further converted into kinetic energy through a device called ‘a prime mover’ which in the case of nuclear reactors is predominantly a steam turbine. In high temperature gas-cooled reactors (HTGR), the enthalpy of the coolant gas could be converted into kinetic energy, in one of the proposed schemes, through direct expansion in a gas turbine. Another conversion takes place in the electric generator where kinetic energy is converted into electric power as the final energy form to be distributed to the consumers through the power grid and distribution system.

The transport of thermal energy from the core to the turbine takes place either through one circuit or loop as in the BWR, or through two separated but interlocking loops, connected through a steam generator.

After the steam expands in the turbine, it exits into the condenser where the remaining amount of heat is removed to produce condensate water which is then fed back either to the core or to the steam generator, and the cycle is repeated. The condenser is a very large and important part of the plant. Roughly, twice the amount of heat that has been converted in the turbine for the production of electric power is removed in the condenser for further rejection to the environment due to the second law of thermodynamics which limits the fraction of useful energy that can be attained by a thermal engine¹. Large amounts of water circulate through the condenser to carry the waste heat to its ultimate sink, which may be the sea, a lake, or the atmosphere itself through cooling towers.

The energy conversion part of the power plant is often called Balance of Plant (bop). It incorporates a large number of components and systems which will be described in detail in this lesson. It represents the largest portion of the plant’s total cost, and its importance for the

efficient and safe operation of the plant is great.

In the following sections, we will focus on a description of plant systems in the pressurized water reactors.

11.2 PWR Primary and Secondary Loops

The name of the PWR derives from the fact that the primary circuit is kept at a very high pressure, typically around 15 MPa, such that at the maximum prevailing temperature in the system, no boiling can occur. At 15 MPa, the boiling temperature of water is about 343°C, well above the maximum temperature of 332°C allowed in the primary water. The following components and systems are mainly included in PWR plant.

- (1) The reactor pressure vessel, which contains the core.
- (2) The primary coolant circuit. These constitute the primary pressure boundary which contains water circulating at high pressure to remove fission-produced heat from the core.
- (3) A heat exchanger or steam generator, in which heat is transferred to a secondary loop, also containing circulating water, where steam is generated.
- (4) The energy conversion system where steam is used to make electricity.

The primary system with all its piping and components, along with the steam generators, is enclosed in a specially designed structure called the containment. It is provided to isolate from the environment any radioactivity that may escape from the fuel into the primary water and from the primary water into the surroundings via a potential leak. Primary system water, at about a pressure of 15 MPa and a temperature of about 300°C enters the reactor vessel through the inlet nozzles near the top. It then travels downward in the annular space (called the downcomer) between the vessel and the core barrel and enters the reactor, it picks up heat, and its temperature reaches about 332°C as it exits into the upper plenum of the reactor vessel. In a typical PWR producing an electrical output of 1244 MWe (which requires the production of about 3600MW of thermal power), a water flow on the order of 150 million lb/hr (18.8 Mg/sec) carries away the heat released in the core and holds the cladding temperature below a comfortable 370°C.

The hot primary fluid exchanges its heat with the secondary fluid in the steam generator and then returns to the core through the primary coolant pumps. Two, three, or four such primary loops have been used in various designs. Some current PWR designs are equipped with primary loop isolation valves. These enable continued operation of the plant in the case of a fault in one of the loops.

The secondary system is maintained at considerably lower pressure, typically 5 MPa and a temperature of 265°C. A turbine bypass line has the capacity to take steam from the main steam line directly to the condenser. It is used to minimize transient effects on the reactor coolant system during events such as start-up, hot shutdown, and cooldown and abrupt reductions or total shedding of the generator load. It consists of pipes, headers and valves. The

bypass line, combined with the ability of the reactor coolant system to accept a 10% abrupt load change, enables the plant to accept up to 50% load shedding of the turbine-generator rated load, without reactor trip and actuation of the relief and safety valves². These, however, could not be avoided for a load rejection of more than 50%.

High-pressure and low-pressure heat exchangers are used to gradually heat the condensed water as it makes its way back to the steam generator by drawing (extracting) small amounts of steam from various stages of the turbine. Thermodynamic theory indicates that by this process the thermodynamic efficiency of the cycle is improved by a few percentage points.

Temperature limitations, imposed by the requirements for the safe operation of the plant, have considerable effect on plant efficiency. Total plant thermal efficiency is the product of the theoretical thermal efficiency, also called Carnot efficiency and the conversion (or engine) efficiency:

$$\eta_{\text{overall}} = \eta_c \eta_{\text{conversion}}$$

The Carnot efficiency is given by the simple formula:

$$\eta_c = \frac{T_1 - T_2}{T_1}$$

where T_1 — the temperature of the hot reservoir;

T_2 — the temperature of the cold reservoir.

Both in absolute degrees Kelvin. The conversion efficiency relates to the various losses during heat transfer and transport losses from friction and other similar processes; it can be improved, but not much beyond a certain point. It is obvious that in order to increase the plant's overall thermal efficiency it is desirable to increase the temperature T_1 to as high a point as possible. The limitations in T_1 originate from the capacity of various materials to perform under conditions of stress. This capacity is normally reduced at high temperatures.

Because the primary coolant must be kept below a certain temperature limit, dictated by primary system pressure and materials behavior, the secondary fluid temperature must necessarily be kept at a still lower temperature. Thus, the overall thermal efficiency of PWRs is a modest 32%~33%, which is considerably lower than the thermal efficiency achievable in modern fossil-burning power stations (about 40%), where higher temperature steam can be used.

11.3 Auxiliary System for Primary Loop

11.3.1 Chemical and Volume control system

During power operation a continuous feed and bleed stream is maintained to and from the reactor coolant system. The bleed stream or let-down flow passes to the chemical and volume control system (CVCS), which performs a number of functions: it initially fills the reactor coolant systems, provides a source of high-pressure water for pressurizing the reactor coolant system when cold, maintains the water level in the pressurizer when the system is hot, reduces

the concentration of corrosion and fission products in the system, adjusts the boric acid concentration in the primary coolant, and provides high-pressure water for the reactor pump seals.

11.3.2 Residual Heat-removal System

A low-pressure (28 bar) low-temperature (177°C) residual heat removal system is provided for the purposes of removing the decay heat from the core during plant shutdown and refueling operations. It consists of dual heat exchangers and circulating pumps together with associated piping, valves, and controls.

11.3.3 Emergency Core-cooling System

In addition to the equipments needed for normal operation of the plant, a number of systems and devices are provided with the express purpose of protecting the plant against damage during accident or fault conditions. Since even when the reactor is shut down the radioactive decay of the fission products still provides substantial quantities of heat, a system must be provided to cool the core and prevent the fuel pins from melting if normal cooling is interrupted. The emergency core cooling system (ECCS) is thus one of the more important protective systems.

The ECCS differs in detail from plant to plant but, in general, consists of several independent sub-systems characterized by equipment and flow path redundancy which assures a very high reliability of operation and continued core cooling even in the event of a failure of any one component to carry out its design functions. The first sub-system is an accumulator injection system. It consists of a number of large tanks (one per loop in the Westinghouse design) that are connected through non-return valves and piping to the main coolant loops between the pump and the reactor vessel. These tanks contain cool borated water stored under nitrogen gas at a pressure of 45 bar. If a large break were to occur followed by a rapid depressurization when the circuit pressure dropped below the accumulator gas pressure, the non-return valves would open to discharge rapidly a large volume of water into the reactor vessel and core. The accumulator injection system is termed a passive system because it functions automatically without any activation of pumps, motor-driven valves, or other equipment.

Two active sub-systems are also incorporated in the ECCS. One is a low-pressure system for use in the event of large breaks which would result in rapid depressurization and discharge of the primary coolant. This low-pressure system injects water taken from the refueling water storage tank into the system over long time periods. Recirculation from the containment sump via residual heat exchangers is also possible for longer-term cooling of the core after an accident.

The other active sub-system is a high head system which will supply borated cooling water if the break is small and the primary system pressure remains high. Use is initially made of the chemical and volume control system charging pumps to force highly concentrated boric acid solution into the cold legs of each loop³. Subsequently, the high-pressure safety injection

pumps are activated, drawing water from the refueling water storage tank, and these supplement the charging pumps. The high-pressure ECCS is activated when the primary circuit pressure falls to 110 bar.

11.3.4 Fuel-handling system

The reactor is refueled with equipment which handles spent fuel under water from the time it leaves the reactor vessel until it is placed in a transport flask for shipment from the site. The reactor cavity is flooded for refueling and is connected to the spent fuel pond by a transfer canal. Refueling normally requires the disassembly of all the components mounted on the vessel head. Typically the refueling outage occupies about 27 days and is the dominant item during the annual outage. New methods have, however, been introduced for rapid refueling. These include the replacement of the conventional studs and nuts on the pressure vessel by quick-acting studs and the development of a two-lift refueling concept. In this latter concept the refueling operations have been simplified to just two activities: removal of the upper package, which consists of the missile shield connected to the vessel head, together with the cooling system for the control rod drive mechanisms, and the removal of the reactor upper internal assembly. This improvement reduces time of the refueling outage to about 11 days.

11.4 Containment

11.4.1 General

The term "containment" refers to the technical means provided to contain radioactivity, primarily fission products, and prevents its escape to the environment. In a LWR there are four successive barriers to such escape: the first barrier is provided by the fuel matrix; the second barrier is the fuel cladding; the third barrier consists of the primary system (pressure vessel, pumps, steam generator primary side, pressurizer, etc.); the fourth barrier is provided by the reactor containment proper⁴ (some of the early, smaller plants built in the Soviet Union had no containment as such, but used a large building to contain radioactivity). It consists of a large building, usually about 42m in diameter (but occasionally up to 64m) and about 60m in height built as a pressure vessel and capable of withstanding an internal pressure as high as 0.44MPa. The pressure at which failure would occur is, however, quite higher. There are two basic reasons for the use of a containment building. Under normal operating conditions, the primary water contains some radioactivity through activation (corrosion products or ^{16}N produced from oxygen in water) and from any leaky fuel. If a leak developed in the primary system (e.g., through a faulty pump seal) radioactivity would be released. The second reason is guard against the likelihood of an accident such as a major break in a primary loop pipe which could lead to a Loss of Coolant Accident (LOCA). The possible scenarios for such accidents vary widely, ranging from relatively mild to quite serious. The most serious scenario could be a core meltdown in which fission products contained in the fuel could escape from the primary system. The containment structures with their associated systems are designed to withstand a

set of postulated internal and external loads and to effectively contain radioactivity that may be released from the primary system. They also provide a shielding function, namely protection of personnel from exposure to radioactivity contained inside containment.

11.4.2 PWR Containment Design

Containment buildings vary considerably in design. Many are vertical cylindrical structures covered with a hemispherical or a shallow domed roof and with a flat foundation slab as in the typical arrangement shown in Fig.11.1 (A cylindrical wall has roughly double the strength of a flat wall of equivalent thickness. Also, a cylindrical building provides a smaller target for missiles for the same volume, similarly, for spheres.). One of the main factors influencing containment design is economic: since the containment is a major cost item the plant designer tries to minimize its size (within the constraints of safety and maintainability) to save money.

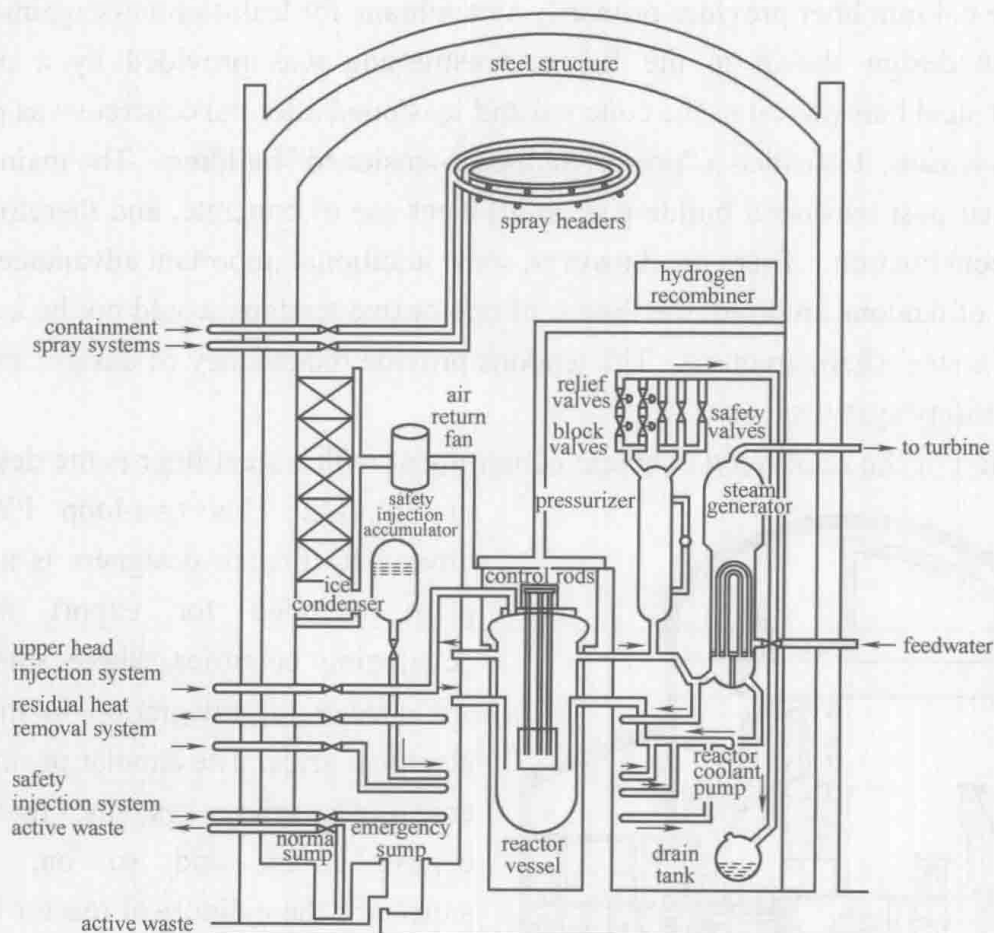


Fig.11.1 the containment with ice condenser

A brief review of the historical development of containment structures provides interesting insight. The early reactor facilities were experimental, low-power reactors installed in water pools. The large amounts of water present were considered adequate to absorb the energy generated in any power excursion and relatively small amounts of steam were expected. Accordingly, reactor buildings were made of reinforced concrete (concrete with a network of criss-crossed steel bars imbedded in them) and designed for an inside pressure of 5

to 8 psig (pounds per square inch gauge, i.e. above atmospheric pressure). An elastomer coating of the inner walls was enough to provide an impermeable membrane. With the construction of the first commercial PWR plant at Shipping port, Pennsylvania in 1957, the total energy stored in the primary coolant jumped by orders of magnitude, giving rise to much higher pressures and temperatures that needed containment. Plain reinforced concrete was no longer adequate to fulfill containment requirements economically and designers turned toward steel. A thick steel shell was used to provide the necessary strength and a layer of concrete was used to provide shielding. Later, as plant size grew to 1000 MWe, the design shifted again, as designers turned to prestressed reinforced concrete for structural strength using the steel liner primarily as an impermeable membrane.

Fig.11.1 shows such a reinforced concrete containment with such a steel liner. The approximately 1m thick wall is made of reinforced concrete to provide mechanical strength. The 6.4 mm liner provides primarily a membrane for leak-tightness against escape of gases. In the design shown in the figure, prestressing was provided by a network of reinforcement steel bars placed in the concrete and tensioned after the concrete was poured and set⁵. For this reason, it is called a “prestressed, post-tensioned” building. The main advantage of a prestressed post tensioned building is an efficient use of concrete, and therefore a leaner and cheaper construction. There are, however, some additional important advantages. Since a large number of tendons are used, the failure of one or two tendons would not be as serious as the failure of a steel shell structure. The tendons provide redundancy of design, an important feature of all safety systems.

A variation of the reinforced concrete containment with a steel liner is the design shown

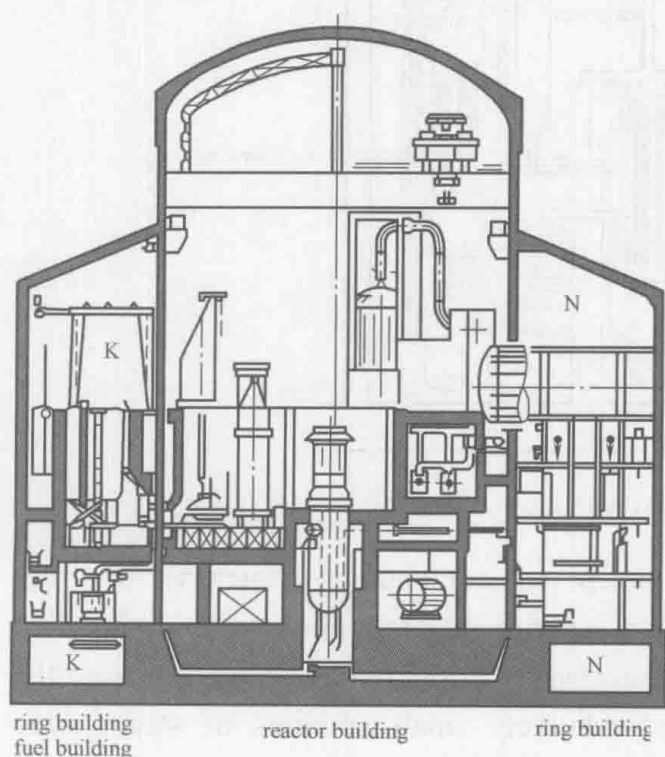


Fig.11.2 the containment with annular building

in Fig.11.2. This two-loop PWR design offered by French designers is a 600 MWe plant intended for export markets in developing countries where smaller plants are needed for integration in their smaller electrical grids. The annular or ring building, containing safety systems, fuel handling, control room, and so on, completely surrounds the cylindrical reactor building on a common mat and provides compactness and, it is hoped, economy of construction⁶.

A different design of PWR containment called the ice-condenser type. It consists of a freestanding inner steel shell, about 45 mm thick around which an outer containment of reinforced concrete is built, with an air space between the two. There are also spherical

shell designs with the sphere eccentricity placed within the concrete building by a German designer of nuclear plants. A similar spherical shell design, but with a concentric arrangement of inner shell and outer concrete building, has been adopted for the latest design of one of the U.S. vendors (Systems 80 by Combustion Engineering).

The containment buildings enclose the entire primary system and have a number of compartments for the housing of auxiliary equipment, safety systems (of which there must be two redundant and physically separate trains), accumulators, electrical equipment, and ventilation equipment, as well as facilities for the handling of spent and fresh fuel.

11.4.3 PWR containment Function

In some PWR plants, the pressure inside containment is kept below atmospheric so that any leakage (through penetrations) would be inward. Other plants use double barriers to prevent any release of radioactive material to the environment.

In case of an accident that causes the pressure inside containment to rise above a certain level, a command will be given to isolate the containment by closing all valves on lines that lead to the outside world. At the same time the blowers will be started to collect leakage from the reactor building, pass it through filters and absorption beds to remove radioactive materials, and discharge it through the station stack. The suction for the blowers is located at the penetration room located outside the reactor building, where leakage is most likely to occur. For this reason, penetrations are grouped together and penetration rooms are constructed around them.

In order to mitigate the effects of large releases of steam (and potentially of radioactivity) in the containment two full capacity independent safety systems are provided: the reactor building spray system and the reactor building emergency coolers. The systems are designed to provide cool water to condense discharge steam and to prevent containment pressure from reaching its design limit. Individual systems differ considerably but a typical system may be described as follows: The initial capacity of the systems in removing heat from the containment atmosphere is typically 253 GJ/h.

When a containment pressure of 4 psig is reached, the emergency coolers of the reactor building are actuated. In their post accident mode, the system consists of three units each with a fan and an emergency cooler. As the reactor building air is circulated across a tubular heat exchanger, a portion of steam is condensed. These coolers alone would be capable of returning the containment pressure to near atmospheric within 24h after an accident. When the pressure reaches a level of 10 psig, the second safety system, the reactor building spray system, is automatically actuated. It consists of a pump, piping, headers, and spray nozzles arranged uniformly under the containment dome. It can spray borated water into the reactor building at a rate of $11.35 \text{ m}^3/\text{min}$. A sodium hydroxide additive is also provided in the spray water to increase the retention of iodine, and hence, to reduce its concentration in the containment atmosphere in the event of a sizable breach of fuel cladding⁷.

An alternative method for condensing steam is provided by the ice-condenser design

introduced by Westinghouse in 1967. The containment space is divided into two compartments by concrete walls and partitions. The two spaces communicate through a vertical, annular space in which a number of ice packs are stored, at about 10°C below freezing. Since all heat sources are located in the lower compartment, any produced steam would flow through the annular space coming in contact with the ice packs. The large sensible and latent heat of ice is used in this design to condense the steam. The design requires a refrigeration system and an antifreeze circulation system to cool the ice machine and maintain low temperatures in the cold storage compartment. With the high heat capacity of the stored ice, a considerably lower design pressure results. Thus, containment buildings of this design are smaller and lighter, with a design pressure of 10~15 psig. The ice contains boron poison (needed to ensure a subcritical core when the water is used for core cooling) and caustic (sodium hydroxide) for the same reasons as in the containment spray designs.

Self-test

1. In pressurized water reactors water is used as _____.
A. moderator B. coolant
C. both A and B D. neither A nor B
2. The amount of heat that has been converted in steam turbine is _____ that removed in condenser.
A. more than B. less than C. same as D. similar to
3. The thermal efficiency of PWRs is _____.
A. more than 40% B. about 30%
C. less than 10% D. about 70%
4. The hot primary fluid exchanges its heat with the secondary fluid in the _____.
A. pressurizer B. condenser
C. header D. steam generators
5. The space between the reactor vessel and the core barrel is the _____.
A. fuel rods B. control rods
C. downcomer D. plenum
6. Residual heat-removal system remove the decay heat from the core _____.
A. during plant operation B. after plant shutdown
C. during start-up D. during transient condition
7. The ECCS is provided to _____.
A. cool the core B. control reactivity
C. control volume D. drive pump
8. The _____ is connected to the spent fuel pond by transfer canal.
A. containment B. pressure vessel
C. pipe D. reactor cavity

9. There are ____ successive barriers in PWR to prevent radioactivity escape.
A. two B. three C. four D. five
10. One of the important factors influencing containment design is ____.
A. the size B. the pressure inside
C. LOCA D. economic
11. The containment encloses the ____.
A. reactor
B. reactor and pressurizer
C. primary system and secondary system
D. entire primary system
12. A sodium hydroxide additive is provided to reduce ____.
A. the iodine concentration B. the pressure
C. the radioactivity D. vapor

关 键 字 解 释

Auxiliary equipment: A term used in power plant to cover the pumps, heat exchangers, valves and other equipment, which accompany with the primary system.

辅助设备: 在动力装置中用的术语, 主要包括泵、热交换器、阀门和其他辅助主系统的设备。

Containment: A building which retains radioactivity and fission products released from reactor, and prevents them escaping to the environment.

安全壳: 用来防止从反应堆漏出的放射性和裂变产物泄漏至环境中去的建筑物。

Enthalpy: Thermodynamic property of a working substance defined as $H = U + pV$ where U is the internal energy. p the pressure and V the volume of a system. The unit of SI is the joule.

焓: 工质的热力学性质, 定义为 $H = U + pV$; 其中 U 为热力学能; p 为压强; V 为系统体积。焓的国际单位是焦耳。

Header: A box or manifold supplying fluid to a number of tubes or passages, or connecting them in parallel.

联箱: 一个箱体或集管, 供给流体到许多小管道, 或将这些小管道或通道并联。

Latent heat: The heat which is required to change the state of unit mass of a substance from solid to liquid, or from liquid to gas, without change of temperature.

潜热: 单位质量的物质, 在温度不变时从固体到液体或从液体到气体的状态变化所需的热量。

Matrix: The phase or phases which form the continuous skeleton of a powder body.

基体: 构成粉末物体连续结构的一个相或多相。

Power grid: The system for electrical power generation, transmission and, to a lesser extent, distribution system.

电网: 产生和传输电能的系统, 有时还包括配电系统。

Prime mover: An engine or other device by which a nature source of energy is converted into mechanical power.

原动机: 将自然能源转变为动力所用的发动机或其他装置。

Rated load: The power output of a power plant, which can continue in conformity with a criterion.

额定负荷: 动力装置按某一规范连续的动力输出值。

Redundancy: The provision of extra components or equipments, in parallel, so as to ensure continued operation after one fails.

冗余: 并行地准备额外的部件或设备, 以保证当某一个设备出故障时系统能继续工作。

Start-up: That beginning process when bringing a nuclear reactor into operation.

起动: 反应堆投入运行的开始过程。

Reinforced concrete: Concrete work in which steel bars or wires are embedded to increase strength.

钢筋混凝土: 埋有钢筋或钢丝以加强其强度的混凝土结构。

词 汇

accumulator [ə'kju:mjuleitə] *n.* 储存器

active ['æktiv] *a.* 主动的, 有源的

actuation [æktju'eɪʃən] *n.* 动作, 触发

antifreeze ['ænti,fri:z] *n.* 防冻液

borated ['bɔ:reitɪd] *a.* 用硼酸处理过的, 加硼的

boundary ['baundəri] *n.* 边界

canal [kə'næl] *n.* 通道

caustic ['kɔ:stɪk] *a.* 苛性的

cavity ['kævɪti] *n.* 内腔, 坑

concentration [kɒnsen'treɪʃən] *n.* 浓度

concentric [kɒn'sentrik] *a.* 同心的

conduit ['kɒndɪt] *n.* 导管

constraint [kən'streɪnt] *n.* 制约

derive [dɪ'raɪv] *v.* 导出, 从……得出

disassembly [dɪsə'sembli] *n.* 拆下

dissipate ['dɪsɪpeɪt] *v.* 消散

domed [dəumd] *a.* 圆顶的

drawing *n.* 抽出, 引出, 绘图

eccentricity [ˌɛksen'trɪsɪti] *n.* 偏心

elastomer [ɪ'læstəmə] *n.* 弹性体

exclude [ɪks'klu:d] *v.* 除去

- flask [flɑːsk] *n.* 储罐
- friction ['frikʃən] *n.* 摩擦
- giga *n.* 十亿, 千兆, 10^9
- GJ=Giga Joule 十亿焦耳
- hydroxide [hai'drɒksaid] *n.* 氢氧化物
- impermeable [im'pə:miəbl] *a.* 不渗透的
- incipient [in'sipiənt] *a.* 开始的
- interlocking [intə'lɒkiŋ] *a.* 可联动的, 内锁的
- iodine ['aiədi:n] *n.* 碘
- kelvin ['kelvin] *n.* 绝对温度
- lean [li:n] *a.* 瘦的
- maintainability [menteinə'biliti] *n.* 可保养性
- mat [mæt] *n.* 底板
- membrane ['membrein] *n.* 薄膜
- mild [maild] *a.* 轻微的, 温和的
- missile ['misail] *n.* 导弹
- mitigate ['mitigeit] *v.* 减轻, 减缓
- mount [maunt] *v.* 安装
- nut [nʌt] *n.* 螺帽
- package ['pækidʒ] *n.* 组件
- passive ['pæsiv] *a.* 被动式的, 无源的
- pick up 吸收, 加速
- pond [pɒnd] *n.* 池
- post-tensioned *a.* 后张力的
- postulate ['pɒstjuleit] *v.* 假设
- prevailing [pri'veiliŋ] *a.* 主要的, 占优势的
- proper ['prɒpə] *a.* 本身, 正文, 本体
- psig=pounds per square inch gauge, 1psig=0.07kg/cm²
- reservoir ['rezəvwa:] *n.* 储存器, 水池
- retention [ri'tenʃən] *n.* 保留
- scenario [si'na:riəu] *n.* 情况
- shallow ['ʃæləu] *a.* 浅的
- shedding *n.* 卸下, 脱开负载
- sink [siŋk] *n.* 散热器, 热井
- slab [slæb] *n.* 平板
- steam line *n.* 蒸汽管路
- sub-system ['sʌbsistəm] *n.* 辅助系统
- sump ['sʌmp] *n.* 储水池, 水坑
- supplement ['sʌplimənt] *v.* 补充

tendon ['tendən] *n.* 筋

tension ['tenʃən] *n.* 张力

tubular ['tju:bjulə] *a.* 由管组成的, 管形的

课文注释

1. Roughly, twice the amount of heat that has been...thermal engine.

句中 **that** 引导的是状语从句, 说明原因。全句译为: 由于热力学第二定律限制了热机中能量转换的份额, 因此, 大约在蒸汽轮机中转换成生产电能所用热量的两倍, 通过冷凝器排到环境中去了。

2. The bypass line, combined with the ability of the reactor...safety valves.

句中 **The bypass line** 是主语, 意思是旁通管路; **actuation** 是动作或起跳的意思。全句的意思是: 旁通管路和反应堆冷却剂系统所能承受的 10% 的负荷变化, 能使动力装置承受汽轮发电机 50% 的甩负荷变化, 在这种情况下反应堆不会紧急停堆, 也不会出现释放阀及安全阀的启跳。

3. Use is initially made of the chemical...cold legs of each loop.

句中 **cold legs** 指冷却剂系统中从主泵出口至反应堆入口的管段, 称冷管段, 也称“冷腿”。全句译为: 开始时, 使用化学和容积控制系统的排出泵, 将高浓缩的硼酸溶液注入每条回路的冷管段。

4. The fourth barrier is provided by the reactor containment proper.

句中 **containment proper** 是指安全壳本身; 类似的表示方法还有 **Shanghai proper**, 是上海市区。

5. In the design shown in the figure...was poured and set.

句中过去分词 **placed** 和 **tensioned** 都是 **steel bars** 的定语; **after** 引导的介词短语作状语。全句译为: 在图中所示的设计中, 预应力是由预应力钢筋提供的, 钢筋放在混凝土内, 当混凝土浇灌、凝固后使钢筋拉紧。

6. The annular or ring building, containing...of construction.

句中 **common mat** 指公共使用的地基座。全句译为: 环形或圈形的建筑内部放置安全系统、燃料处理间和控制间等, 这些建筑将圆柱形的反应堆大厅全部围在中间, 大厅与环形建筑使用公用的地基座, 这样提供一个紧凑式布置, 希望是一个经济性建筑。

7. A sodium hydroxide additive is also provided...fuel cladding.

句中 **additive** 是添加剂, **retention of iodine** 是碘的抑制。整句译为: 在喷淋水中也提供氢氧化钠添加剂, 来增加对碘的抑制, 从而在燃料包壳大面积破损的情况下可以减少安全壳空气中碘的含量。

Unit 12 The Steam Turbine

12.1 Introduction

The production of power in a turbine differs from the engine process since it is steady and continuous instead of intermittent in character, as in the latter. Thus, the turning effort on the shaft of the turbine is even and regular at constant load and a flywheel is unnecessary. Essentially, there are two steps in the turbine process. The first is to utilize the thermal energy stored in the steam to give it a high velocity as it issues from a nozzle or orifice in which its pressure is reduced. The second is to remove as large a proportion as possible of the kinetic energy of this jet of steam in blades or buckets by means of which work is delivered to the turbine shaft¹.

Although the turbine principle was known and applied over two thousand years ago, the application to the manufacture of commercial power is a comparatively recent development. Throughout the eighteenth and nearly all of the nineteenth centuries the steam engine monopolized the field. One of the earliest patents on a steam turbine was granted to De Laval, a Swedish engineer, in 1883. Consistent development of the principle followed, due to the work of which men as Parsons of England, Rateau of France, and Curtis of the United States, until, by the beginning of the twentieth century, the turbine was beginning to be an important factor in the development of the modern power plant.

12.2 Classification of Turbines

The principal classification of turbines into: ① impulse; ② reaction; ③ combined impulse-and-reaction types is based on the method of utilizing the kinetic energy of the steam to produce a turning effort on the turbine rotor.

In the steam engine the conversion of energy from the thermal form to shaft work was effected by the static mass of expanding steam exerting pressure on a piston; however, for the same energy transformation in the turbine, two processes are required. During the first step, steam is expanded to a lower pressure (and temperature) by causing it to flow in a properly shaped stationary passageway; due to this expansion, thermal energy is converted into kinetic energy and the steam moves with a greater velocity. In the next step, this rapidly moving stream enters a passageway on the rotating part of the turbine (rotor and blade assembly) where it undergoes a change in direction of flow. This change of velocity which produces an increase of momentum will exert a force on the rotor and thus cause it to rotate. In this manner is kinetic energy converted into shaft work, thus completing the two energy transformations.

The device used to produce the first energy conversion in turbines is called a nozzle and the two types are: ①the converging design for low ratios of expansion; ②the converging-diverging type for high ratios of expansion.

Turbine classification depends on the method used to effect the second energy transformation. The force which can be exerted by a jet produced by expansion in the nozzle is utilized in one of two ways. If the nozzle in Fig.12.1(b) is stationary and the jet of steam is directed against the bucket, an impulsive force created by the change in direction of flow of the jet will push the bucket forward and lift the weight W . Thus kinetic energy of the jet has been transformed into mechanical work and this is the principle of action of the impulse turbine.

The other method of utilizing the kinetic energy is by the use of a moving nozzle as shown in Fig.12.1(a). A jet of steam issuing from the nozzle produces a reactive force which pushes against the opposite wall of the box and lifts the weight. This principle of reaction may be understood better by again referring to the container in Fig.12.2(a). Suppose there is no nozzle opening and the box is filled with steam as before. The pressure exerted by this fluid will be the same on all the walls so the box will be at rest since there will be no unbalanced force to move it. However, if an opening is made in one of the walls for a nozzle, a jet of steam will be released and the pressure in the nozzle will be less than at the corresponding point on the opposite wall. This unbalanced force causes the box to move in the direction opposite to the jet. Thus the reaction is caused by the expansion of steam in a moving passageway and this is the principle upon which the design of the reaction turbine is based.

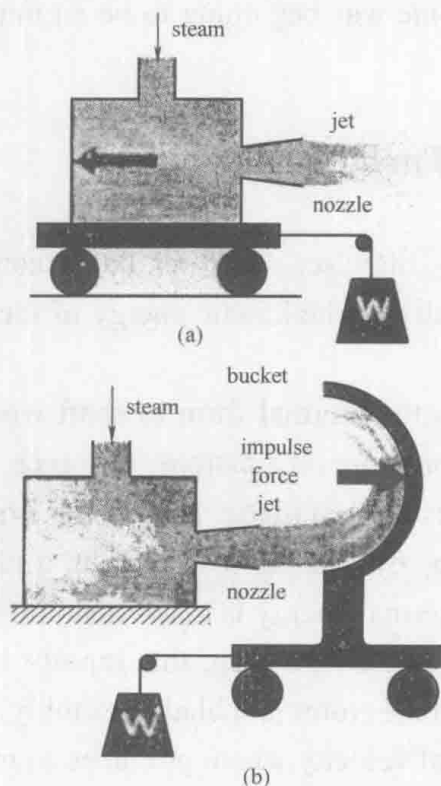


Fig.12.1

(a) reaction force; (b) impulse force

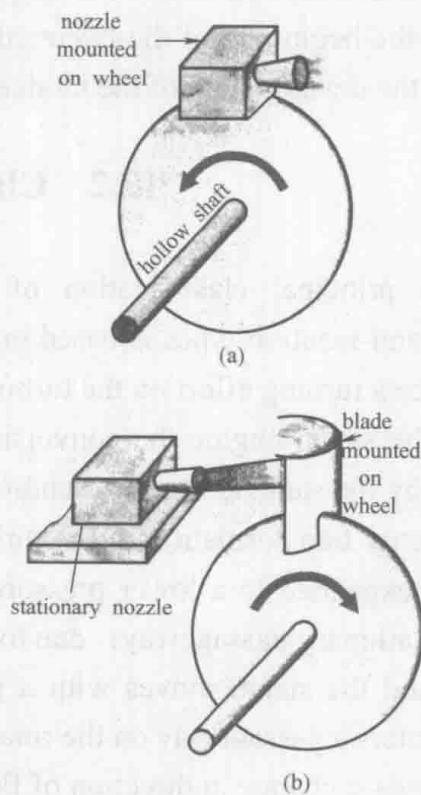


Fig.12.2

(a) reaction application; (b) impulse application

12.3 Impulse and Reaction Blading

The steam turbine takes two steps to change heat energy to mechanical work. First, steam expands in a nozzle, discharging at high velocity. In this way the total heat (enthalpy) of the steam partly changes to velocity (kinetic energy, the energy of motion). The forces of the jet produced by this transformation in the nozzle can be used in either of two ways. In each case, heat energy of the steam is transformed into velocity energy in the nozzle, and then into mechanical work.

In an actual turbine, the stationary nozzles of a reaction stage are made by using blade sections shaped to form nozzle walls². These are arranged in a ring around the shaft or rotor. An adjacent set of moving blades mounted on the turbine rotor receives steam flow from these stationary nozzles. Some expansion occurs in both moving and fixed blades and approximately the same pressure drop occurs across each. By definition this is a 50 percent reaction stage, and the acceleration of the steam is divided equally between the stationary and rotating rows. The absolute velocity rises in the fixed row of blades and falls in the moving row.

There may seem to be a contradiction between the principle that expansion in a nozzle produces increased velocity and the prior statement that acceleration of the steam is divided equally between stationary and rotating rows. While the absolute velocity (referenced to the earth) decreases in the moving blades, expansion in the moving blades does boost velocity in relation to the moving blades that is, if we had viewpoint on a blade the steam would seem to speed up.

A pure impulse stage is one where the entire stage pressure drop occurs across the stationary row and no pressure drop across the rotating row. Thus, the acceleration of the steam takes place entirely in the stationary row or nozzle. The nozzles arranged in a ring at one side feed steam at an angle to blade travel. The crescent-shaped blades permit free entry and discharge and produce the change of speed and direction which gives the rotational force. The steam expansion (pressure drop) occurs only in the nozzles, and pressure remains essentially constant in blades. The velocity rises in nozzles but falls in the blades.

A force is exerted in changing either the speed or direction of a body in motion and the amount of force depends on the extent to which speed or direction is changed. For this reason, and since all of the pressure drop occurs across the stationary row in a pure impulse stage, the steam velocity leaving the stationary row is quite high generally twice that of the blade speed. In addition, since the change in direction is important in the impulse blade to optimize the energy conversion, the turning angle is quite large.

In the case of the reaction stage, the pressure drop is divided over both the stationary and rotating row, thus the steam velocity leaving the stationary row is lower. The steam velocity relative to the rotating blade is also lower. A lower turning angle is required through

the rotating blade of the reaction stage than is required through the rotating blade of an impulse stage.

12.4 Velocity Compounding Curtis Stage

An ideal Rateau stage³ is simply the pure impulse stage previously discussed. Rateau was one of the early builders to successfully use this type.

When the pressure drop across a Rateau stage is large, the steam velocity is relatively high and the high speed steam jet gives up only part of its kinetic energy in the first row of moving blades. If the direction of the steam is reversed, the remainder of the energy can be utilized without the necessity for a row of nozzles. Thus, in the Curtis stage⁴ the steam jet gives up part of its energy in the first row of moving blades. Then it passes through a row of stationary reversing blades which redirect the steam into the second row of moving blades, where most of its remaining energy is absorbed (see Fig.12.3). The Curtis stage is also referred to as a velocity compounded impulse stage.

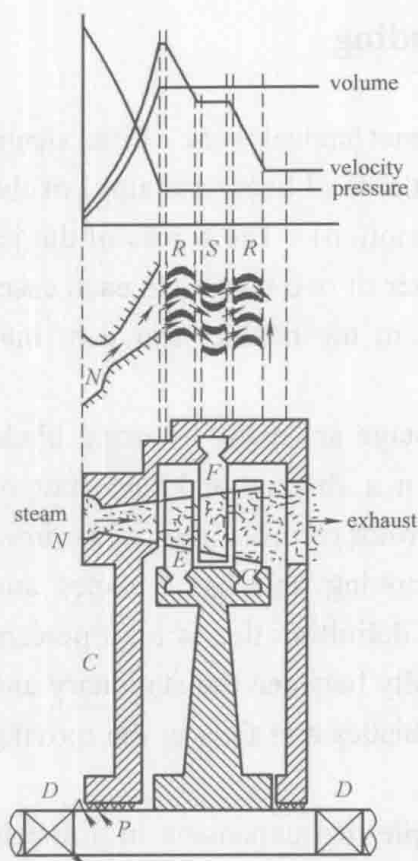


Fig.12.3 velocity staging

12.5 The Reaction-Turbine Principle

In the reaction turbine the moving blades are, in effect, traveling nozzles and expansion is secured by making a close working fit between the rotor and casing and by providing convergent passages between adjacent blades. The fixed ring of blades discharges steam into the moving ring the same as in the impulse turbine, but the subsequent expansion between the moving blades increases the magnitude of the relative velocity, and this alteration in addition to the changed direction of steam flow (as in the impulse turbine) causes a reaction on the blades.

Fixed blades of a reaction turbine are, in effect, rings of vane-formed converging nozzles. The outer ends of the nozzle partitions are fastened to the casing and the inner ends are brought as near to the surface of the revolving drum as is practicable in order to reduce leakage between the stationary and rotating parts. The inner ends of the moving nozzles or blades are fastened to the drum and the other ends are brought as near to the inside surface of the casing as is practicable⁵.

The degree of reaction a turbine attains is determined by the relation between the area of the passage at inlet and that at the outlet of the blade. If the exit area is made small enough, a pressure difference will be built up across the passage and for this reason the term choke is sometimes used to denote reaction or drop in pressure in the passage.

The diaphragm type of construction used in the impulse turbine is not employed in the reaction turbine because of the pressure difference across each row of blades which would produce an excessive axial thrust on the disks. Therefore the reaction turbine rotor is generally a drum on which the rows of blades are mounted.

12.6 Reaction Blading

Modern reaction blades have a cross section tending towards a blunt nose teardrop shape. They are usually made from rolled sections and are attached to the rotor or casing by serrations. The advantage of this form of attachment is that the blade can be inserted into the groove and turned into position and, by varying the width of the groove, various blade angles may be produced from a common rolled section. Short blades are uniform in width while long blades are tapered and have a warped or nonuniform section. Thus the blade angles are changed according to the difference of peripheral speed between the root and tip of the blade. Fig.12.4 shows several views of reaction blades and assemblies. Since the effects of leakage are more serious in the reaction type, seal strips are provided.

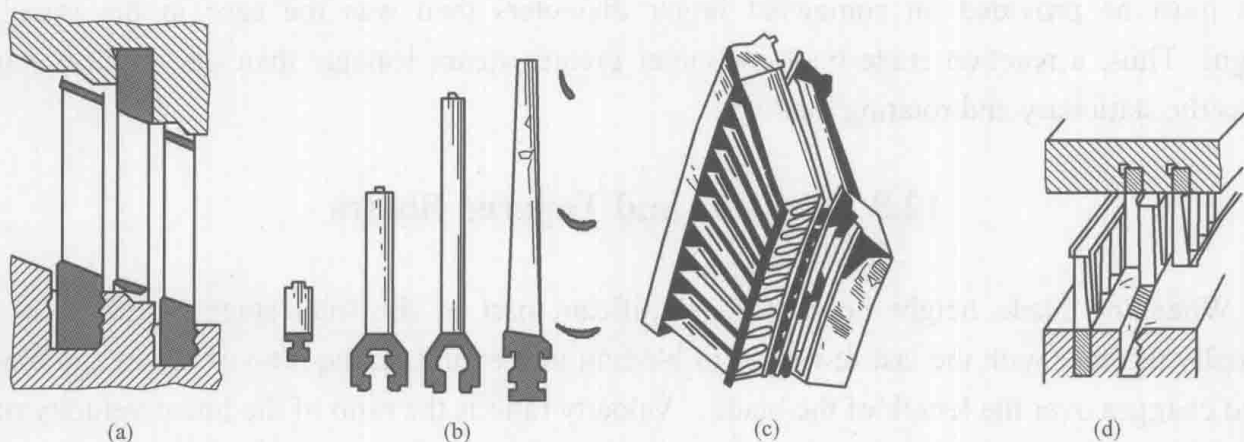


Fig.12.4 reaction blades

- (a) fixed and moving reaction blading shown in section; (b) blades of various sizes;
(c) segments of typical fixed and moving reaction-blade rings; (d) assembled reaction blading

In turbines of the reaction type there is commonly no flat surface on the nozzle partition which may be used to define an angle; instead, the blades or partitions have only curved surfaces.

12.7 Sealing Arrangement

The effect of leakage also plays an important role in the selection of stage type.

Where the leakage areas become an appreciable percentage of the total steam flow area, leakage effects will outweigh the greater aerodynamic passage efficiency of reaction blading and one would select an impulse stage⁶.

However, where leakage areas are a small percentage of the total steam flow area, the

greater aerodynamic quality of reaction stages outweighs the leakage losses and the reaction stage should be selected.

Impulse turbines are normally constructed with the moving blades on wheels or discs that are shrunk on and keyed to the shaft. An alternate construction uses a solid forging which has been machined out to form an integral disc and shaft arrangement. Nozzles are carried in diaphragms that have center holes through which the shaft passes. Seals at these openings limit the steam leakage which bypasses the nozzles. Since steam pressure drops in passing through the fixed nozzle, the pressure is higher on the inlet side of the diaphragm and it is important that effective seals be maintained to minimize the quantity of steam bypassing the turbine blades.

This construction permits arranging seals on the shaft of the machine on relatively small diameters. However, as turbine ratings increase, the shaft diameter must also increase so the impulse stage no longer has the small diameter on which to provide sealing arrangements.

Reaction turbines are usually built with a rotor machined from a solid forging. The blades are mounted directly on the rotor which increases in diameter toward the low pressure end where steam volume is greater. This requires that steam seals to prevent leakage between blade rows must be provided on somewhat larger diameters than was the case in the impulse design. Thus, a reaction stage may encounter greater steam leakage than the impulse stage across the stationary and rotating blading.

12.8 Twisted and Tapered Blades

When the blade height becomes a significant part of the total stage diameter, as is normally the case with the last few rows of blading in the turbine, the ratio of steam and blade speed changes over the length of the blade. Velocity ratio is the ratio of the linear velocity of a given point on the length of a blade to the absolute velocity of the steam jet at entrance to the blade.

Ideally, steam enters the nozzles in an axial direction and it leaves in a circumferential direction, forming a vortex flow or eddy that is contained by the turbine casing before the steam enters moving blades. It is characteristic of the vortex flow leaving the nozzles that the steam velocity at the inner radius is higher than that at the outer radius. However, the linear velocity of the blade increases with radius. Therefore, the velocity ratio is steadily increasing from root to tip. The ideal velocity ratio for a pure impulse design would be approximately 0.5; whereas the ideal ratio for a pure reaction blade would be higher.

Since the velocity ratio at the root of the blade favors the impulse design, it is typical to design the root of the blade for impulse flow, equivalent to 0 percent reaction (no pressure drop). As we said, the steam velocity in the ideal impulse design should be double the blade velocity. The entrance angle of the blade is determined by the angle of approach of the steam so that the steam will slide smoothly over the blade.

At the tip of this long blade, the blade velocity is higher than the approaching steam velocity. This means the steam approaches from a direction almost opposite to the blade's motion. For this reason the blade section must be twisted in order to receive the steam smoothly.

Due to the vortex flow, the steam pressure is higher at the tip than that at the root, and there will be some pressure drop through the blade. Thus, the conditions favor the use of a reaction blade section since the pressure ratio is in the range suitable for reaction design and since there will be pressure drop through the blade, there will be velocity increase at the blade exit which also conforms to the reaction design. Thus, while a pure impulse force exists at the blade root, a pure reaction force is acting at the blade tip.

The principal factor that controls the amount of twist is the distribution of pressure between the stationary and rotating blade. The smaller the change in pressure between the base and tip, the smaller the change in velocity and thus the smaller the difference in flow angle to the rotating blade. Relatively short blades exhibit only a small pressure variation along the length of the blade and as a result, the rotating blade inlet flow angle variation becomes negligible.

12.9 Erosion Control and Moisture Removal

The longer last rows of blades are fitted with stellite strips which are silver soldered to the leading edge. Stellite is a very hard alloy a binary alloy of cobalt and chromium which is highly resistant to erosion. In the last rows of blading in the turbine, there is moisture which would be quite erosive to the blade material without this protection. This minimizes erosion in this section of the turbine and increases blade life, maintains blade efficiency, and reduces maintenance requirements.

Moisture removal and erosion control are major considerations in the design of steam turbines applied with nuclear steam supply systems that generate dry and saturated or low superheat steam. Moisture particles are removed with the extraction steam and also at specially designed moisture removal zones.

Increased axial spacing between stationary and rotating blades also decreases erosion. This allows sufficient time for the large moisture droplets which stream from the trailing edges of the stationary blades, to be broken up into smaller particles, which more easily pass through the rotating blades without impingement.

An additional moisture removal device at the inlet to the last rotating row in the low pressure turbine has proved particularly effective in minimizing erosion in the blade tip area. This device consists of a narrow circumferential slot just ahead of the inlet to the tip of the last rotating blade. A small quantity of blade path steam flows through this slot directly to the exhaust, and by eductor action removes a considerable amount of entrained moisture from the steam at the tip area.

12.10 Other Blade Features

There are other blade features which are required for mechanical or a combination of mechanical and thermodynamic reasons. One example of a requirement which serves as a dual role is the blade shrouds. These shrouds supply several important functions as follows.

- A sealing surface to prevent steam leakage. Steam that bypasses the blade represents a direct loss in efficiency and must be carefully controlled.

- Controls the level of stress at the base and root of a blade.

- Controls the blade frequency to a limited degree. Blading design must include consideration of the various modes of vibration which will occur. Stress levels which will result from vibration must be a factor in the blade design.

In addition to shrouds, some low pressure blades are also provided with integral lashing wires which are welded together to provide additional frequency control⁷. Shrouds, as well as lashing wires, may not be continuous but are used to cause the blades to act in groups. This grouping provides additional frequency of vibration control. Such lashing wires are provided on unshrouded last row blading and are primarily a mechanical means of providing frequency control and maintaining stress levels within design limits.

12.11 Low Pressure Turbine

Construction of the low pressure nuclear turbines for light water reactor applications is essentially the same as that of 1800 rpm elements in many fossil fired plants. To minimize possible distortion effects of large fabricated stationary parts on seal clearances, the temperature drop from inlet to exhaust is taken in steps across multiple walls, utilizing a combination of separate inner cylinders and separate blade carrier rings. In addition, thermal shields are attached to the outside of the inner cylinder to reduce temperature gradients that cause distortions. The rotors are built up of alloyed discs shrunk and keyed onto an alloy shaft.

Water is removed at extraction points and specially designed moisture removal zones. The effectiveness of moisture removal was determined from a series of laboratory and field tests. Since the size of the water droplets varies inversely with the pressure level, a greater amount of water can be removed at the lower pressure extraction zones.

12.12 Turbine Control

Control of the turbine speed and load is provided by a combination stop-throttle valve and control valve assembly, which is placed ahead of the system. Four stop-throttle and control valve combinations are normally used.

The stop-throttling valve provides positive shut-off the steam supply to the main

The various methods of controlling the flow and load for turbine generator units utilize either area or pressure control or combinations of the both.

D. for

9. In the reaction stage, steam expansion____.
- A. occurs in rotating blading
B. is divided by stationary and rotating blading
C. occurs in nozzles
D. occurs in blade section
10. The rotor of reaction turbine____ that of impulse turbine.
- A. is different from
B. is same as
C. is longer than
D. is smaller than
11. In the last stage of a turbine the degree of reaction____.
- A. is a constant
B. is the same
C. is even
D. increases from root to tip
12. The output power of the turbine depends on____.
- A. the rotating revolution
B. the size of the blade
C. the steam flows through it
D. how many stages it has

关 键 字 解 释

Casing: The part of a machine which encloses the working portions.

机壳: 把工作部分封闭起来的机器部件。

Clearance: The distance between two objects, or between a moving and stationary part of a machine.

间隙: 两物体之间或机器的运动部件同静止部件之间的距离。

Distortion: Any departure from the original shape because of applied stress or release of residual stress in the material.

变形: 材料由于施加应力或解除残余应力而对原有形状发生的任何偏离。

Impulse turbine: A steam turbine in which steam is expanded in nozzles and directed on blades carried by a rotor, in one or more stages, there being no change in pressure as the steam passes the blades.

冲动式汽轮机: 一种蒸汽轮机, 其中蒸汽在喷嘴内膨胀并直接引向转子上的叶片, 分一级或多级, 蒸汽通过叶片时压力不变。

Leading edge: The edge of a streamline body or blade which is forward in normal motion.

前缘: 流线体或叶片向前作常规运动时其前端的边缘。

Nozzle: In impulse turbines, specially shaped passages for expanding steam, thus creating kinetic energy of flow with minimum loss.

喷嘴: 在冲动式汽轮机中, 为蒸汽膨胀所设的特殊形状的通道, 以此在最小的损失下增加气流的动能。

Reaction turbine: A turbine in which the steam expands progressively in passing alternate rows of fixed and moving blades. The kinetic energy developed by moving blades.

反动式汽轮机：在这种汽轮机中，蒸汽在通过交替排列的定叶片和动叶片中逐渐膨胀，动叶片中也产生动能。

Shroud: An outer or peripheral strip used to strengthen turbine blading.

围带：用来加强透平叶片强度的外侧或周边的围板。

Stop-throttle valve: A governor-controlled steam valve in steam engines and turbines, usually a double beat valve.

节流阀：蒸汽机或汽轮机中调速器控制的蒸汽阀，一般是双座阀。

Torque: The turning moment exerted by a tangential force acting at a distance from the axis of rotation. It is measured by the product of the magnitude of the force and the distance.

转矩：由作用于转子轴线一定距离处的切向力所产生的转动力矩，由力的大小与到轴距离的乘积来度量。

Trailing edge: The rear edge of an aerofoil or blade.

后缘：机翼或叶片的后部边缘。

词 汇

aerodynamic [æərəʊdaɪ'næmɪk] *a.* 空气动力学的

appreciable [ə'pri:ʃiəbl] *a.* 可观的

barely ['beəli] *ad.* 几乎没有

binary ['baɪnəri] *a.* 二元的

blunt nose *n.* 钝头

bucket ['bʌkɪt] *n.* 叶片

choke [tʃəʊk] *n.* 阻塞

chromium ['krəʊmjəm] *n.* 铬

circumference [sə'kʌmfərəns] *n.* 圆周

circumferential [sə'kʌmfə'renʃəl] *a.* 周边的，圆周的

compound ['kɒmpaʊnd] *v. & n.* 复合

converging [kən'veɪdʒɪŋ] *n.* 渐缩的

crescent-shaped *a.* 月牙形的

diaphragm ['daɪəfræm] *n.* 隔板

diverging [daɪ'veɪdʒɪŋ] *a.* 扩张的

eductor [i'dʌktə] *n.* 排放装置

entrain [ɪn'trein] *v.* 携带

flywheel ['flaɪwi:l] *n.* 飞轮

grant [grɑ:nt] *v. & n.* 给予

HP=High Pressure 高压

impart [ɪm'pɑ:t] *v.* 给予

key [ki:] *v.* 用键固定

lashing ['læʃɪŋ] *n.* 捆索

locomotive [ləʊkə'məʊtɪv] *n.* 机车

LP=Low Pressure 低压

monopolize [mə'nɒpəlaɪz] *v.* 独占, 垄断

optimize ['ɒptɪmaɪz] *v.* 优选

orifice ['ɒrɪfɪs] *n.* 孔口

outweigh [aʊt'wei] *v.* 重于, 胜过

rev/s=revolutions per second 每秒多少转

rigidity [rɪ'dʒɪdɪtɪ] *n.* 刚性

rpm=revolutions per minute 每分钟多少转

serration [se'reɪʃən] *n.* 锯齿形

slot [slɒt] *n.* 狭缝

solder ['sɒldə] *v.* 焊

stage [steɪdʒ] *n.* 级(由一组喷嘴叶片和一组转动叶片组成的)

stellite ['stelait] *n.* 斯太立特硬质合金

strip [stri:p] *n.* 带, 条

tapered ['teɪpəd] *a.* 变截面的, 锥形的

teardrop *n.* 泪珠

throttle ['θrɒtl] *n.* 节流

twist [twɪst] *v.* 扭转

warp [wɔ:p] *v.* 扭曲

free from 没有……的, 去除……的

课文注释

1. The second is to remove as large a proportion as...turbine shaft.

句中 as large a proportion as possible 是尽可能大比例的, 注意该句型中不定冠词 a 的位置是放在形容词的后面。整句译为: 第二步是在转动叶栅中耗去汽流内尽可能大比例的动能, 这样就可以在汽轮机轴上输出功。

2. In an actual turbine, the stationary nozzles...nozzle walls.

这里 shaped 是过去分词, 做 blade sections 的定语。此句的意思是: 在实际的汽轮机中, 反动级的静止喷嘴是由成型的叶片部分构成喷嘴壁而形成的。

3. Rateau stage

拉托级。拉托是法国科学家, 他首先发明了喷嘴内产生高速汽流冲击转动叶片做功的冲动式汽轮机, 因此把纯冲动级称为拉托级。

4. Curtis stage

寇迪斯级。寇迪斯是美国科学家, 他在 1897 年发明了复合式冲动式汽轮机, 这种汽轮机在一个轮盘上有两排转子叶片, 中间是导向叶片, 喷嘴产生的高速汽流在两级转动叶片中转换成功。

5. The inner ends of the moving nozzles...is practicable.

句中 **drum** 指反动式汽轮机的轴，其直径较大，一般称转鼓；**as near...as is practicable** 是尽可能接近的意思。整句译为：转动喷嘴或叶片的根部固定在转鼓上，另外一端尽可能接近壳体的内表面。

6. Where the leakage areas become an appreciable...impulse stage.

句中 **Where** 引导的是状语从句，**outweigh** 是大于或重于的意思。整句译为：当泄漏面积占总蒸汽流通面积很大一部分时，泄漏的影响大于反动叶栅空气动力学效率高的优势，这时应选择冲动级。

7. In addition to shrouds, some...frequency control.

这里 **shroud** 指固定在叶片端部起加强和固定作用的围带，一般用在冲动级的较短的转动叶片上。**Lashing wire** 是拉筋，一般用在汽轮机后几级的长叶片上，在叶片中部打孔将拉金穿过并将叶片和拉金焊在一起，起加固叶片和控制叶片振动频率的作用。整句译为：除围带外，一些低压级叶片还使用整体拉金，它们与叶片焊接起来，以提供附加的频率控制。

Unit 13 Main Steam, Feed and Condensate Systems

13.1 Introduction

The steam system serves to convey the generated steam from the nuclear steam supply system to the turbine generator, turbine driven feedwater pumps, steam dump, reheaters, gland steam system, and air ejectors. Main steam is also provided for the turbine driven auxiliary feedwater pump to ensure steam generator water level is maintained in case of a station blackout. A typical main steam system simplified diagram is shown in Fig.13.1.

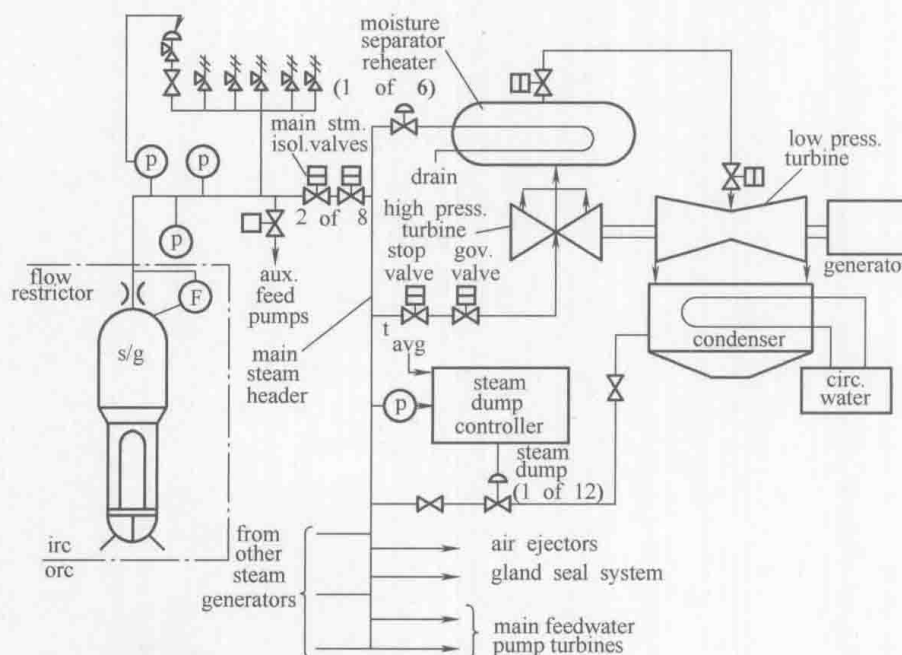


Fig.13.1 Main steam system (simplified)

Steam from each steam generator flows through a flow restrictor and two main steam isolation valves to the main steam header. The flow restrictor acts to limit the steam flow from a steam break accident but has little resistance to normal flow. The isolation valves are provided to isolate the steam generator with a series of relief and safety valves. From the main steam header, the steam is directed to the various steam loads.

The condensate and feedwater systems return the condensed steam from the turbine condenser, and drains from the regenerative feed heating cycle to the steam generators while maintaining the water inventory throughout the cycle. These systems must also automatically maintain the steam generator water level during steady-state and transient operation.

The steady state flow varies with unit load. The feed and condensate systems recover the

steam used for auxiliary steam loads as well as steam to the main turbine. They also make up for losses due to steam generator blowdown and system leakage.

For a pressurized water reactor, steam pressure varies with load. The pressure in the steam generators drops with increased thermal output. This characteristic, combined with the lower feed system frictional resistance at lower flow, results in a small increase in total pumping head required from full-load to no-load¹. Sufficient margin must be allowed above the system-required head/flow for feed control, pump wear, design and testing, and system fouling to ensure adequate flow to the steam generators during steady-state and transient operation.

Uniform feed water temperature to all steam generators must be ensured under all operating conditions. A continuous steady feed flow should be maintained at all loads. In addition, sufficient feed water storage capacity must be maintained within the condensate feed systems to accommodate the mass transfer of fluid due to the expansion and contraction arising from the thermal and pressure effects on steam generator fluid inventory and condensate feed systems during load changes. This is also necessary to compensate for loss of fluid from the system during load changes or plant cooldown by atmospheric steam dump.

13.2 Main Steam and Feedwater Piping

Steam piping design pressure and temperature matches the steam generator shell side design values. The arrangement and bore of the steam lines between the steam generators and turbines ensure balanced steam flow from the steam generators so that the steam generator pressures are essentially equal under all operating conditions. This is accomplished by interconnecting the main steam lines downstream of the main steam isolation valves. The feedwater piping is arranged so that feedwater is supplied to all steam generators at essentially equal temperatures.

The main steam and feedwater piping is designed so that the following failure of any line or malfunction of a valve installed therein or any consequential damage will not happen.

- (1) Reduce flow capability of the auxiliary (emergency) feedwater system below minimum required;
- (2) Render inoperable any engineered safeguard service (i. e. , control, electric cables, containment cooling piping, etc.);
- (3) Initiate a loss-of-coolant accident;
- (4) Cause uncontrolled flow from more than one steam generator;
- (5) result in the containment pressure exceeding 90% of the design value (80% for ice condenser containments) or impair its impermeability and integrity;
- (6) Result in a loss of safety function due to fatigue or overstress.

13.3 Main Steam Piping System Flow Restrictor

A flow restriction is provided in each main steam line inside the containment building to

limit flow in the event of a steam line break accident. It is located as close as feasible to the steam generator outlet nozzle in order to minimize piping preceding the restriction, thereby reducing the probability of a steam line break upstream of the restrictor. The preheat steam generator and later model ring riser steam generators have a flow restrictor which is an integral part of the outlet steam nozzle². For these steam generators a separate flow restrictor in the steam piping is not required.

The flow restriction is designed with the following features.

(1) Provides plant protection in the event of a steam line break downstream of the restriction. Should a break occur, the flow restriction reduces the flow rate of steam from the steam generator, thus reducing the cooling rate of the primary system. Limiting the steam flow rate limits the core cooldown rate which reduces the reactivity excursion and the probability of fuel clad damage.

(2) Provides minimum unrecovered pressure loss across the restrictor during normal operation.

(3) Reduces the potential thrust forces on the main steam piping caused by the steam jet effect in the event of a steam line break downstream of the restriction, thus minimizing pipe whip.

(4) Reduces back flow into containment from the other steam generators in the event of a pipe break between the flow restriction and the steam generator, until the non-return and/or main steam isolation valves close, thus minimizing containment pressurization.

(5) Provides a portion of the pressure differential employed for steam flow measurement.

13.4 Safety Requirements

13.4.1 General Safety Oriented Design Provisions

The safety aspects of the main steam system are concentrated on the portions of main steam lines from the steam generators, out through the containment, and up to and including the main steam line isolation valves. The following description identifies the general safety oriented areas.

(1) The steam lines and the shell side of the steam generator are considered basically as an extension of the containment boundary and, as such, must not be damaged as a consequence of reactor coolant system damage. This requires that the steam generator shell and steam lines within the containment are to be classified and designed for the reactor coolant system missile-protected category. The reverse is also true in that a steam break is not to cause damage to the reactor coolant system.

(2) The measured steam flow has both functional and safety related requirement. Functionally, the flow signal is used by the three-element feedwater controller and as a load index signal for plants using variable speed main feedwater pumps³. It is used for safety in the development of a steam flow, feed flow mismatch signal.

(3) The portion of the main steam system outlined above is also necessary to the safe

shutdown of the plant and is accordingly designated safety class.

(4) Uncontrolled steam release as a result of a steam line failure must be limited to the contents of one steam generator in order to ① limit the related effect upon the reactor core within specified fuel design limits; ② limit containment pressure to a value less than the prescribed fraction of design pressure.

13.4.2 Other Safety Oriented Design Provisions

These design provisions include the following parts.

(1) The steam generator safety valves.

(2) The steam supply to the turbine-driven auxiliary feedwater pump has a safety classification because of safeguard requirements for auxiliary feedwater. The supply lines are connected up-stream of the steam line isolation valves to not less than two steam generator steam lines of units which are not equipped with reactor coolant loop stop valves have three steam supply lines to the turbine, each from an individual main steam line, to insure redundancy of supply in the event one loop is isolated. Isolation and non-return valves in each supply line insure the independence of each main steam line by preventing interconnecting backflow.

(3) For a loss-of-coolant accident, we cannot prevent initial discharge of main steam to the atmosphere due to turbine trip. If tube leaks were present prior to the accident, some radioactivity accumulated in the shell side water would be discharged through the safety valves. Before, during and after the initial blow through the safety valves, radioactivity leakage is restricted by the barrier at the steam generator tubes, shell, steam lines, blowdown lines, and sample lines within the containment structure. The total plant releases associated with a loss-of-coolant accident, including steam side leakage, must be within 10 CFR 100 criteria.

Releases associated with other incidents, such as main steam line break and steam generator tube ruptures, also must be within the criteria specified. If tube leaks are present, the releases associated with the steam generator will consist of the steam flow through the main steam safety and relief valves. Isolation of the release is again effected at the steam generator tubes. For a main steam line break, a limit is placed on maximum tube leakage depending on primary activity, meteorology, dilution, etc. For a steam generator tube rupture, a limit is placed on reactor coolant activity level.

The safety aspects of the main feed system are concentrated from the steam generators back to and including the first stop and non-return valve outside the containment. This portion of the feed system is an integral part of the auxiliary feed system.

13.5 Feed and Condensate System Description

The feed and condensate system is composed of a condenser, three 50 percent capacity condensate pumps and three 50 percent capacity condensate booster pumps, two 50 percent

capacity variable-speed turbine-driven feed pumps, two 50 percent capacity heater drains pumps, three strings of feedwater heaters, three sets of steam jet air ejectors, and one turbine gland steam condenser (Numbers of pumps and their capacities may vary considerably from one plant to the next).

Condensate pumps take suction from the condenser hotwell and pump through the air ejector condensers and gland seal condenser to the condensate booster pump suction. The air ejector system is used to remove air and non-condensable gas from the condenser. The gland seal system prevents air from leaking into the condenser from around the turbine shaft. A recirculation line is provided back to the main condenser to insure adequate cooling flow through the air ejector and gland seal condensers during startup. Additional lines are taken from the condensate pump discharge to provide cooling spray to the flexible exhaust connection between the low pressure turbines and the condenser, and to provide quenching spray inside the condenser during steam dump operation⁴.

The condensate booster pump provides the required net positive suction head to the main feedwater pumps. It pumps the water through the low pressure feedwater heaters to the feed pump suction. The heater drains, together with the moisture separator drains, are collected and pumped into the feed pump suction. Extraction steam used to preheat the feedwater along with drains from the moisture separator reheaters are collected in the heater drain tank. The heater drain pumps pump this water to the feed pump suction. About one third of the feedwater comes from the heater drain tank.

The feed pump discharges through one stage of high pressure feed heating into a common header. Each of the four steam generators is supplied through a single line taken from the common header. Each steam generator is supplied with a feed regulating valve, a feed regulating bypass valve and a flow element. Outside the containment building in each line are stop and non-return valves. To allow plant maintenance, bypasses are provided around feed heaters. The bypass around the final feed heater joins the common steam generator feed header to ensure adequate mixing and equalized temperatures in the feed to the individual steam generators.

The most common steam plant design is condenser steam dump only with 50 percent load rejection capacity without reactor trip. The feed pump turbines normally exhaust into the main condenser. On loss of the condenser vacuum or main feed system, the plant will trip, and maintaining steam generator inventory is no longer a requirement for the main feed system. Under these circumstances the auxiliary feed system takes over. Sufficient water is stored in the condenser hotwell to make up system fluid inventories on a 50 percent load rejection. The makeup pipe between the condenser and the condensate storage tank is sized to provide flow to make up fluid inventory during a ramp load change of five percent per minute. The overflow pipe between the condensate system and the condensate storage tank is sized for the ramp load change of five percent per minute.

It is assumed the heater drain pumps would stop due to cavitation on a step load rejection greater than 10 percent. To ensure adequate flow to the feed pumps, the condensate pumps are

made to increase their capacity by reducing the dynamic resistance of the condensate system with a bypass around the low pressure heaters. A control valve opens on a pressure differential signal sensing the drop in turbine load. The overflow valve to the condensate storage tank is tripped closed when the bypass opens. The drain pumps are tripped at their minimum flow and the drains are returned to the condenser on high tank level. Even feed temperature to all steam generators is assured by the length and routing of the common feed header after the high pressure heater bypass.

The feed piping to each steam generator, including the stop and non-return valves, is supported, restrained, protected, manufactured, and inspected in accordance with the requirement of the safety systems. Controls are provided in the control room for the condensate, heater drain and feed pumps, low pressure heater bypass valve, condensate makeup and overflow valves, and feed pump discharge valve. Indicators are provided for these controls.

13.6 Transients

The Nuclear Steam Supply System can be designed to accept various types of load changes. Each load rejection capability and type of steam dump will have the following effect on the condensate-feed system.

- (1) Atmospheric steam release results in loss of fluid from the system.
- (2) Condenser steam dump has no loss of fluid from the system.
- (3) Ten percent step load or five percent per minute ramp changes is achieved without steam dump or major effect on the feed system. Fluid inventory transfer within system is accommodated by the storage capacity of the system.
- (4) A step reduction of 50 to 85 percent in load will result in steam dump, loss of drains due to flashing at reduced pressures throughout the regenerative feed heating system, and reduction of fluid inventories throughout the cycle.
- (5) Loss of condensate flow by the loss of regenerative feed heating drains or process steam recovery will result in reduced feed flow.
- (6) Trip of plant without loss of feed system or condenser will result in condenser steam dump, normal feed system operation, and reduction of fluid inventory in cycle.
- (7) Trip of plant with loss of feed system and condenser will result in steam release to atmosphere, no feed through normal feed system, and shrinkage of fluid in the cycle. Auxiliary feedwater will automatically operate.
- (8) Trip of plant with loss of condenser will result in atmospheric steam release, normal feed system operation, and reduction of fluid inventory in cycle.

13.7 Controls

Feed flow is controlled automatically above 15 percent load by a three-element controller

using steam generator water level, steam flow and feed flow to control the feed regulating valve to each steam generator. This valve is pneumatically operated and is designed to fail closed on loss of air. With turbine-driven feed pumps, the speed is varied to reduce the pressure drop across the regulating valve at low loads. In addition, a bypass around the main feed valve is provided for fine control at low loads. The bypass is sized for 15 percent of the plant design steam flow and is capable of remote manual control from the control room. Both valves have handwheels for local emergency control and provisions for automatic closure upon receipt of signals from the reaction system.

To assure stable valve control of the feed system, using variable speed pumps, the design pressure drop across the valve at guaranteed flow must equal the dynamic losses in the feed system from feed pump discharge into the steam generators⁵.

Simulation and operational experience show that a large load reduction is the governing transient in the design of the feed control system. The closing of the turbine governor valves results in a pressure increase and a level decrease in the steam generators. To limit this level decrease, the feed flow should be increased. The increased pressure results in a decreasing flow rate from the feed pumps as long as the valve position and pump speed remain unchanged.

To minimize the duty of the valve, the control system will increase the speed of the pump so that the pump discharge pressure increases at least as much as the pressure in the steam generator. To achieve this, the pressure difference between feed header (after the last heater) and steam header is used as a control variable, to be compared with a setpoint in order to generate a pump speed signal.

Normal controls, indication and alarms are provided throughout the system. Additional controls are provided in the area of Nuclear Steam Supply Systems plant safety. As an example, under certain conditions the feedwater control and bypass valves are closed within five seconds from receipt of signal and the feed pumps are tripped simultaneously. In addition, one of the following must occur.

(1) A shutoff valve in the feedwater line (in addition to the feedwater control and bypass valves), located between the feed pump discharge and auxiliary feedwater piping connection will close within five seconds from receipt of signal.

(2) All condensate booster, heater drain pumps which deliver flow to the feedwater system, will be tripped coincident with the signal to close the feedwater control valves.

Self-test

1. Auxiliary feed water pump is driven by_____.

A. electricity B. main steam C. turbine D. motor

2. At lower loads the_____of feed pump is reduced.

A. speed B. power

C. head D. steam

3. When the power output from the PWR system increased, the pressure in the steam generators____.
A. drops
B. maintains constant
C. increases
D. changes
4. At different loads the feed flow should be____.
A. changed
B. continued
C. accommodated
D. maintained continuous steady
5. The pressures of steam generators are____under all operating conditions.
A. different
B. equal
C. maintained
D. changed
6. The restrictor is installed____.
A. on the steam generator
B. in the nozzle
C. in main steam line
D. in containment
7. Which one is not the function of the restrictor?
A. provides plant protection in the event of a steam line break
B. reduces steam flow rate
C. reduces the potential thrust forces
D. provides protection on reactor directly
8. If tube leaks were present____the accident, some radioactivity accumulated in the shell side water would be discharged through the safety valves.
A. prior to
B. after
C. in
D. during
9. The air in the condenser is removed by____.
A. pump
B. ejector
C. fan
D. compressor
10. The condensate pumps pump the water from the____to the____.
A. condenser/steam generator
B. condenser/ejector
C. condenser hotwell/condensate booster pump suction
D. condenser hotwell/heater
11. To allow plant maintenance, ____are provided around feed heaters.
A. pumps
B. valves
C. tanks
D. bypasses
12. A control valve opens on a pressure differential signal sensing the____in turbine load.
A. change
B. drop
C. increase
D. trip

关键字解释

CFR: Code of Federal Regulations

(美国) 联邦管理法规

Condensate: The liquid obtained as a result of removing the latent heat of evaporation from vapor.

冷凝液：从蒸汽中排出其中所含的汽化潜热所得到的液体。

Condenser: A chamber into which the exhaust steam from a steam turbine is delivered, to be condensed by the circulation or the introduction of cooling water.

冷凝器: 容纳来自蒸汽轮机排出的蒸汽, 通过冷却水的循环或引入冷却水使蒸汽冷凝的容器。

Discharge head: Vertical distance between intake and delivery of a pump.

出口压头: 泵的排出水位与吸入水位之间的竖直距离。

Dump: Discharge steam from steam generator to be bypassed from the turbines in a nuclear power station if the electrical load is suddenly taken off.

蒸汽排放: 当核电站突然失去电负荷时, 使蒸汽不通过汽轮机直接由蒸汽发生器排放。

Ejector: A device for exhausting a fluid by entraining it with a high velocity steam or air jet.

喷射器: 一种用来排出流体的装置, 以高速蒸汽或空气的喷射把流体夹带排出。

Excursion: The phenomenon of rapid increase of reactor power above set operation level, either deliberately caused for experimental reasons or accidental.

功率骤增: 由于事先安排的或由于事故造成的反应堆功率迅速增加至设置的水平之上的现象。

Fatigue: The phenomenon of the failure of metals under the repeated application of a cycle of stress.

疲劳: 在循环的应力作用下金属的失效现象。

Feed: Water pumped into a vessel, e.g. feed water to a steam generator.

给水: 泵入压力容器的水, 例如加给蒸发器的水。

Hotwell: The tank below the steam condenser, into which the condensate from the condenser is stored, and from which it is returned by the feed pump to the steam generator.

热井: 在凝汽器下的一个容器, 从凝汽器来的冷凝液存在这里, 然后由给水泵打回蒸汽发生器。

Regulating valve: The valve for regulating or adjusting flow rate.

调节阀: 调节或调整流量的阀门。

Gland steam system: The steam system for sealing glands and shafts.

轴封蒸汽系统: 用来密封法兰和密封轴的蒸汽系统。

词 汇

auxiliary [ɔ:g'ziljəri] *a.* 辅助的; *n.* 辅机

bore [bɔ:] *n.* 孔径, 孔

cavitation [kævi'teɪʃən] *n.* 气蚀

coincidence [kəu'insidəns] *n.* 吻合, 同时发生

dilution [dai'ljʊ:ʃən] *n.* 稀释

downstream ['daunstri:m] *a.* 在下游的

drain [dreɪn] *v. & n.* 排出, 排水管

ring riser *n.* 环形冒口

- flexible ['fleksəbl] *a.* 挠性的
- impair [im'peə] *v.* 削弱
- impermeability [impə:mjə'biliti] *n.* 不可渗透性
- index ['indeks] *n.* 指示器, 标志, 指标
- inoperable [in'ɒpərəbl] *a.* 不能操作的
- interconnect [ˌintəkə'nekt] *v.* 互连
- malfunction [mæl'fʌŋkʃən] *n.* 工作不正常
- meteorology [ˌmi:tjə'rɒlədʒi] *n.* 气象状态, 气象学
- overflow [ˌəʊvə'fləʊ] *v. & n.* 横流, 溢流
- pneumatically [nju:'mætikəli] *ad.* 靠压缩空气
- preheat [pri:'hi:t] *v. & n.* 预热
- ramp [ræmp] *n.* 斜面, 上升
- regenerative [ri'dʒenərətiv] *a.* 再生的
- remote [ri'məʊt] *a.* 远程的
- render ['rendə] *v.* 提出, 表现
- restrictor [ris'triktə] *n.* 节流阀, 限流器
- sensing ['sensɪŋ] *a. & n.* 信号传感
- shrinkage ['ʃrɪŋkɪdʒ] *n.* 收缩, 减少
- shutoff ['ʃʌtɔ:f] *n.* 关闭
- simulation [ˌsimju'leɪʃən] *n.* 模拟, 仿真
- simultaneously [siməl'teɪnjəsli] *ad.* 同时地
- string [striŋ] *n.* 串排
- therein [θeər'in] *ad.* 其中
- thrust [θrʌst] *v.* 冲入
- transient ['trænzɪənt] *n.* 瞬态
- turbine trip 汽轮机脱扣 (甩负荷)

课文注释

1. This characteristic, combined with...to no-load.

句中 total pumping head 是总泵压头。整句译为: 这个特性与在低流速下给水系统摩擦阻力小两个因素一起, 造成了从满功率至零功率总泵压头有小幅的增加。

2. The preheat steam generator...steam nozzle.

句中 preheat steam generator 预热蒸汽发生器, 指的是有节热器 (economizer) 的蒸汽发生器; later model 是新型号。整句译为: 预热式蒸汽发生器和新型号环形进水的蒸汽发生器都有流量限制器, 它与蒸汽的接口管成一体。

3. Functionally, the flow signal is used...feedwater pumps.

句中 using 引导现在分词短语, 作 plants 的定语。整句译为: 在功能上, 流动信号被三冲量给水控制器所使用, 并作为使用可变速主给水泵电站动力装置的负荷指示信号。

4. Additional lines are taken from the condensate pump...dump operation.

句中 flexible exhaust connection 是指汽轮机与凝汽器之间的排汽接管, 由于它连接两个大的刚性体, 为了解决热膨胀问题, 这个接管有热膨胀补偿, 因此这段管称为挠性接管。整句译为: 另一个管路是从凝结水泵的出口引出, 为低压汽轮机和凝汽器之间的挠性接管提供冷却喷淋, 在蒸汽排放运行时为凝汽器内部提供冷却喷淋。

5. To assure stable valve control of the feed system, using...steam generators.

句中 across 是介词, 后面的短语作 pressure drop 的定语。整句译为: 为了保证给水系统阀控制的稳定, 使用变速泵, 在保证的流量下, 阀两端的设计压降必须等于从给水泵出口到蒸汽发生器之间的给水系统的动力损失。

Unit 14 Operation of a Nuclear Power Plant

14.1 Introduction

The usual measure of nuclear power's operational efficiency is the total amount of electricity produced and the cost of producing that electricity. A small number of plants may have somewhat different criteria; for example, some reactors are designed to produce both process heat for industrial companies and electricity for the utility grid¹. In any case because the capital cost associated with a nuclear power plant far exceeds its operating costs and because its power is often the least expensive on the electric power grid, utilities try to run their nuclear units as close to 100% power level as long as possible. This is called base load operation. When a nuclear plant is out of service, replacement power from a fossil fuel plant will often have an incremental cost in excess of \$5 750 000 per day for a 1000 MWe plant. With such a large cost penalty, even the start-up rate of a nuclear unit can have significant financial impact. Therefore, an attempt is usually made to come to full power (from 0 to 100% power) as fast as possible without compromising the safe efficient operation of the unit.

14.2 Plant Safety

Safety requirements that restrict the operations of a nuclear plant necessarily are a judgment between the benefits and potential hazards of nuclear power. At one end, the safest nuclear power plant would not be allowed even to approach criticality; of course, no public benefit would be derived from the plant. At the other end, a nuclear plant allowed to operate without operable safety systems would be an undue risk to the public. Unacceptable operations are ones that threaten the safety of the public or might lead to plant damage. In the United States, criteria protecting the former are contained in 10CFR Parts 20, 50, and 100, while the latter is the subject of engineering codes. This lesson examines how safe operation is achieved, how a plant is operated, and what is necessary for high reliability.

14.3 Plant Operations

Normal plant operation refers to planned maneuvers needed to run a plant without significant incidents. Operations during an incident are not considered normal until the plant returns to pre-incident status. A plant operates according to prescribed guidelines called procedures. There are many volumes of these procedures in a plant's control room. They specify the steps required to go from one operating state to another and the plant's operating limits.

Only two operating states are desired for a nuclear plant: refueling outage and full power operation. The former should be as short as possible. Occasionally, plants must shut down because of equipment failure, or for safety inspections mandated by the NRC. Many planned maintenance operations and periodic safety inspections required by the NRC take place during a refueling outage.

In the United States, NRC licensing procedures overlay a degree of formality on all nuclear plant operations. Operational requirements restrict both the plant's condition and the range of operability of plant systems. Such requirements must be observed during all modes of plant operation (not just while the plant is at power) to assure safe operation. Two kinds of requirements are placed on plant equipment.

(1) Performance criteria while the reactor is operating.

(2) Surveillance requirements to assure that the system is capable of performing its essential functions.

Table 14.1 shows the nuclear safety criteria used to set operational requirements. These criteria reflect the NRC approach to protecting the public health and safety by assuring a nuclear plant operates only in a safe way. These criteria are reinforced by technical operating specifications and limits that are a part of the licence for every nuclear plant.

Table 14.1

Nuclear Safety Operational Criteria

Applicability	Criteria
Planned operation, normal operational transients, and accidents	The plant shall be operated in a safe manner at all times
Abnormal operational transients and accidents	The plant shall be operated so that no single active component failure can prevent vital safety actions. However, this requirement is not applicable during system repair if the availability of the safety action is maintained

The emphasis on safety has resulted in all operations being conducted according to written procedures. Modern instrumentation, by reducing the complexity of monitoring and controlling the plant, has greatly simplified these procedures, the procedures are intended to keep the plant running evenly and to allow smooth transitions between operating states.

Steady-state operation implies that the plant is not undergoing any change; we are also concerned with plants undergoing changes of state (i.e., from one power level to another), referred to as transients. Both steady-state and transient conditions are classified according to mode: normal, abnormal, and off-normal. The following three sections describe these basic modes of plant operations.

14.4 Normal Mode

Those operations that routinely performed in producing the electric power, including starting up and shutting down the plant, are considered normal. Also included is dealing with

random changes in output which are not planned but which happen frequently enough that handling them is a routine matter. An example of the latter is a sudden change in demand for power caused by transmission lines downed in a storm.

The five following transients are considered normal conditions:

(1) Heatup and Cooldown. Heatup and cooldown are limited by plant operating specifications to rate $55^{\circ}\text{C}/\text{h}$. In actual practice, the rate of $55^{\circ}\text{C}/\text{hr}$ will not be usually attained because of: ① criteria for preventing brittle failure of components in the primary coolant system, which establish lower permissible temperature rates; ② slower initial rates achieved when energy from the coolant pumps is used to heat the system; ③ interruptions in the heatup and cooldown cycles. These interruptions might be due to various operating procedures such as drawing a pressurizer steam bubble, control rod withdrawal, or coolant sampling.

Ideally, heatup and cooldown would occur only before and after refueling. In practice, additional unscheduled plant cooldowns may be necessary for plant maintenance.

(2) Unit Loading and Unloading. The loading and unloading of a unit's electrical generator on the utility's grid cause power changes. A load swing of $5\%/\text{min}$ is the maximum that the automatic reactor control system for most reactors can handle. The reactor coolant temperature will vary with the load during such a transient.

(3) Step Increase and Decrease of 10%. A $\pm 10\%$ change in load demand is a transient which occasionally occurs. If a turbine control valve were to open, such a transient would be possible. It might also be caused by disturbances in the electrical network into which the plant output is tied. Nuclear plants have reactor control systems that restore plant equilibrium without a reactor trip. In effect, during load change conditions, the reactor control system attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized and gradually restored to its desired value. For example, following a decrease in turbine load, the steam pressure and temperature in the primary system increase since the nuclear power does not decrease as fast as the turbine load. In a PWR, this power mismatch between the turbine and reactor cause the control system to insert the control rods partially to reduce core power. In a BWR, the expanding steam fraction inside the core automatically reduces the power. Following a load increase in turbine load, the reverse situation occurs.

(4) Large Decrease in Load. This transient is caused by a major decrease in turbine load from full power. The resultant rapid increase in reactor coolant temperature initiates the steam dump system. This prevents a reactor shutdown or lifting of the coolant system safety valves. For example, in a 95% load decrease from full power to 5% power, the steam dump system accepts 85% of the turbine load. The remaining 10% of the total change is handled by the rod control system (pressurized water reactor, PWR) or void expansion (boiling water reactor, BWR). If the steam dump system could not cope with the transient, the reactor would trip and the safety valves would lift.

(5) Steady-State Fluctuations. According to commonly used operating specifications, the reactor coolant temperature may not increase or decrease more than 3°C in 1 min. These temperature changes are assumed to fluctuate around the average temperature value ($T_{\text{avg}} \pm 1.5^{\circ}\text{C}$). Temperature changes lead to pressure changes according to the equation of state for the coolant. The reactor coolant pressure, therefore, fluctuates up and down proportionally to the system's (absolute) temperature.

14.5 Abnormal Mode

Abnormal operation can result from unplanned transients. Such transients are more severe than those encountered in the previous section. There are eight conditions that are considered potentially serious.

(1) Nuclear system pressure increase.

(2) Moderator temperature decrease.

(3) Positive reactivity insertion.

(4) Coolant inventory decrease.

(5) Coolant flow decrease in the core.

(6) Coolant flow increase in the core.

(7) Coolant temperature increase.

(8) Excess coolant.

These variations, if uncontrolled, could result in damage to the reactor or the fuel, or both. A pressure increase threatens the reactor piping and will cause the safety/relief valves to open. In a BWR, a pressure increase also collapses voids in the moderator, causing an insertion of positive reactivity which threatens fuel damage as a result of overheating. On the other hand, a temperature decrease of reactor vessel water (i.e., the moderator) in either BWRs or PWRs also results in an insertion of reactivity as density increases, again leading to fuel overheating. A coolant temperature increase also threatens the integrity of the fuel; such a variation could be the result of a heat exchanger or steam generator malfunction during operation. Moreover, reactivity insertions are possible from causes other than moderator temperature or pressure changes. Such reactivity insertions are obviously undesirable.

Both a coolant inventory decrease and a reduction in coolant flow through the core are more serious in a BWR than in a PWR since they both reduce the void content of the moderator, resulting in an increased power. Another serious condition, an excess of coolant inventory, could be the result of malfunctioning water level control equipment; this can result in a turbine trip which causes an increase in nuclear system pressure and an increased power level.

The above abnormal conditions could result from either equipment failure or an operator mistake. Examples of the latter are opening the wrong valve or withdrawing the control rods in the wrong sequence. To prevent damage to the plant, detailed procedures are available to the

operators to handle such abnormal system states.

14.6 Off-Normal (Upset) Mode

The following transients are considered as upset conditions.

(1) Loss of Load Without Immediate Turbine or Reactor Trip. This transient is caused by a rapid decrease in turbine load from full power. It represents the most severe transient on the reactor coolant system. The reactor and turbine will eventually trip; however, before any serious damage to the system occurs in a PWR, the trip would be caused by a high pressurizer level.

In a BWR, the situation is more serious, since the reactor would continue to operate, although at much lower power (about 20%). The safety/relief valves would open, but the steam continuing to be generated would eventually lead to a core meltdown due to depletion of the cooling water if not terminated². This scenario is referred to as an ATWS (Anticipated Transient Without Scram) in safety analysis reports. Since redundant means of tripping all reactors are provided as a part of the reactor protection system, transients of this nature are very rare.

(2) Loss of Power. This transient might result because of an area-wide electrical blackout around the plant resulting in the loss of outside electrical power. Under these circumstances, the reactor coolant pumps are without power will “coast down”. Following the coast down of the coolant pumps, natural circulation will permit removal of core residual heat (i.e., through the steam generators in a PWR). Diesel generators are also available to provide emergency power to the plant.

(3) Loss of Flow. This type of transient could be caused by a reactor coolant pump inadvertently shutting down, for instance, due to a faulty signal, resulting in partial loss of coolant flow. The reactor and turbine will trip because of low reactor coolant flow, terminating the transient. The steam dump system will subsequently open, helping to control pressure in the system.

(4) Reactor Trip from Full Power. A reactor trip from full power may occur for a variety of reasons, such as a problem on the electrical grid. A plant trip results in temperature and pressure transients in the reactor coolant system. The transients continue until the reactor coolant and the heat sink temperatures are in equilibrium. A continued supply of feedwater to the core will remove any residual heat in the system and prevent the safety valves from lifting.

(5) Inadvertent Auxiliary Spray (PWR). In a PWR, a transient will occur if the pressurizer auxiliary spray valve is opened inadvertently. This will introduce cold water into the pressurizer, a very sharp pressure decrease will result. The reactor trip that would occur would only accelerate the pressure decrease. The pressure is finally stabilized at the hot leg saturation pressure. This transient is more severe on a two-loop PWR plant than on a three-loop or four-loop plant.

(6) Earthquakes. Mechanical stresses on the plant and equipment can occur from the deflections caused by earthquakes. The plant components, especially those that perform safety functions, are engineered to handle such stress. Their design incorporates margins according to the seismic conditions where the plant is located.

14.7 Control Rooms and Instrumentation

The actual operation of a nuclear plant is usually no more difficult than operating a modern fossil fuel power station. In some sense it may be easier, since the operational strategy calls for constant full power operation, if possible. As a result, very little load following (power maneuvering to keep up with daily fluctuations in the demand for power) occurs. Also, much of the actual control of the plant is automatic. Consequently, a few trained people could actually run a nuclear plant, aided, of course, by the automatic instrumentation and process computers³.

Reliance on plant process computers has grown with the demands for improved availability, efficiency, and safety. In the early plants, process computers were frequently designed as peripheral elements. The newer plants use techniques that improve plant operations.

Advanced control rooms and instrumentation in nuclear plants are currently being used to assist operators with the analysis and integration of plant status information. These new designs incorporate the concept of functional modularity at the hardware level. This simplifies validation and modification, thereby minimizing obsolescence. Such systems help overcome the increasing complexity of plant control required to maximize availability and still meet stringent safety and environmental requirements. Almost all reactor control rooms now have a "safety panel" to inform the operator of the status of the plant during an accident⁴. The panel groups together all relevant information, presenting it more clearly. It helps find the cause of the accident, and determines the action needed. The safety panel emphasizes the validation of system parameters. Thus, values recorded by the sensor are compared, to assure the information is consistent. The safety panel computer functionally integrates the information, verifies it, and interprets it. As an example, coolant level, density, temperature, and pressure information are processed to yield the coolant inventory, void fraction, and subcooling.

During the last decade, there has been a dramatic escalation in costs of plant construction, operation, and fuel, as well as a growing public concern about conservation. As a result, utilities are committed to the improvement of plant performance. This commitment, the increasing size and complexity of power plants, the greater stringency of regulatory requirements, and the rapid advances in the electronics industry are all contributing to the introduction of new instrumentation and control technology⁵.

Consequently, in a power-generating facility there are thousands of monitoring and control points that must be individually hard wired between the field and the control

building. Hard-wired requires that hundreds of miles of cable be strung, costing millions of dollars in materials and labor. Cable separation and fire protection requirements compound the complexity of plant wiring.

Recent advances in electronics make it possible to reduce instrumentation and control wiring through the use of remote multiplexing. Remote Multiplexing Systems (RMSs) are able to transmit many signals on a small number of wires. The individual field sensors and actuators are wired to remote terminals that are strategically distributed throughout the plant and sequentially sampled for transmission by the remote multiplexer⁶. In the control building, the signals are separated and distributed on short cables to their final destinations.

14.8 System Parameters

During normal operation of a nuclear reactor, the temperature and pressure of the coolant are carefully controlled. In a PWR, the operating pressure is maintained by the pressurizer as is the coolant volume. The latter is adjusted by a water level control system. The water level is designed to change with the operating conditions of the reactor. In contrast, a BWR maintains coolant inventory by changing the rate at which the feedwater pumps operate.

System temperature is controlled by the nuclear heat produced in the core, that is, by the regulation of the core reactivity by the reactor control system. At low power levels, the control system is operated manually. Above a predetermined percentage of full power, the reactor control system operates automatically, thereby regulating the system temperature.

The actual power in the reactor is monitored by neutron level detectors which are sensitive to the wide range of power levels that the reactor might produce. To cover so wide a range of power, three sets of neutron detectors are used: source range, intermediate range, and power range. Any change in core reactivity can be inferred by the rate of change of the neutron flux seen by these monitors, while the core power level is proportional to their signal.

The long-term control of core reactivity is usually accomplished by means of chemical "shims". This means use of boric acid in the coolant for PWRs and burnable poisons (usually gadolinia, although europia and hafnium are sometimes considered) in the fuel for BWRs. In the case of a boron shim technique, the concentration of the boric acid must be adjusted downward as the nuclear fuel builds up fission products throughout core lifetime. This is referred to as boron "letdown". The boron concentration is usually dropped quite rapidly during the first 200 MWd/t of fuel burnup after refueling due to the effects of the new fuel elements. Thereafter, the boron concentration is more slowly adjusted downward until the next loading of new fuel. The burnable poisons are routinely used in BWRs to control excess reactivity, especially in the first fuel cycle. This reduces power peaking in the core and leads to better fuel utilization. However, more recently, the use of such materials in PWRs is being contemplated. The main disadvantage of burnable poisons seems to be the more difficult design and manufacture of the fuel assemblies.

Self-test

1. The nuclear power station provides_____for the utility grid.
A. expensive electricity B. base load
C. peak load D. full load
2. From the text the safest nuclear power plant may be a _____.
A. expensive one B. cheap one C. big one D. small one
3. _____specify the steps required to go from one operating state to another.
A. The codes B. The NRC
C. The operators D. The procedures
4. Following a decrease in turbine load, the water pressure and temperature in the reactor_____.
A. will increase B. increase
C. decrease D. will be unchanged
5. When large decrease in turbine load the steam from the steam generators is sent to the_____.
A. steam condenser B. atmosphere
C. steam dump system D. heat exchanger
6. A temperature decrease of reactor vessel water results in an_____.
A. insertion of control rods B. power decrease
C. power increase D. increase of reactivity
7. An excess of coolant inventory can result in a _____.
A. power increase B. temperature increase
C. reactor trip D. turbine trip
8. Loss of power is a_____.
A. upset condition B. abnormal condition
C. turbine trip D. reactor shutdown accident
9. Following the coast down of the coolant pumps, the core residual heat will be removed by_____.
A. condensers B. natural circulation
C. boiling heat transfer D. emergency power
10. Inadvertent auxiliary spray will result in_____.
A. temperature decrease B. power decrease
C. pressure decrease D. abnormal condition
11. The operation of a nuclear plant is_____than operating a modern fossil fuel power station.
A. easier B. more difficult
C. rather D. more complex

12. The long-term control of core reactivity means adjustment of ____.

A. control rods

B. boric acid

C. power output

D. coolant

关键字解释

ATWS: Anticipated Transient Without Scram.

未能停堆的预期瞬态。

Diesel generator: An electric generator using internal combustion engine as power source.

内燃发电机：以内燃机为动力源的发电机。

Heat sink: A substance or device for the absorption or dissipation of unwanted heat.

散热器、热井：将不需要的热量吸收或散去的物质或装置。

Mode: Situation or method of performing a specified task.

模式：完成指定任务的方式或方法。

Multiplex: Use of one channel for several messages by time-division multiplex or frequency division.

多路通信：利用时分复用通信或频分复用通信，在一个通道上传递多个信息。

NRC: Nuclear Regulatory Commission

核管会（美国）

Sampling: The survey of a small but representative part of water or gas, with the intention of obtaining an estimate of some characteristics of the whole.

采样：检测水或气体中有代表性的一部分，从而估算出整体特征。

Scram: General term for emergency shutdown of a nuclear power plant, especially of a reactor when the safety rods are automatically and rapidly inserted to stop the fission process.

事故停堆：核电站事故停堆的通称，特指由于快速和自动地把安全棒插入反应堆造成裂变过程停止。

词汇

abnormal [æb'nɔ:məl] *a.* 不正常的

actuator ['æktjueitə] *n.* 制动器，执行机构

anticipate [æn'tisipeit] *v.* 预先考虑

availability [ə'veilə'biliti] *n.* 可利用性

coast down ['kəustdaun] *n.* 惰走，下降

collapse [kə'læps] *v.* 破裂，坍塌

commit [kə'mit] *v.* 委托，使承担任务

commitment [kə'mitmənt] *n.* 许诺，保证

compromise ['kɒmprəmaiz] *v.* 妥善处理，折中，妥协

- conservation [ˌkɒnsə'veɪʃən] *n.* 自然资源保护, 保守
- consistent [kən'sistent] *a.* 一致的
- contemplate ['kɒntempleɪt] *v.* 考虑, 预期
- encounter [ɪn'kaʊntə] *v.* 遇到
- escalation [ˌeskə'leɪʃən] *n.* (逐步) 升级
- europia [ju'rəʊpiə] *n.* 氧化铯
- fluctuation [ˌflʌktju'eɪʃən] *n.* 起伏, 波动
- formality [fɔ:'mælɪti] *n.* 手续
- hardwired *a.* 硬连线的
- inadvertent [ˌɪnəd'vɜ:tənt] *a.* 偶然的
- incremental [ɪnkri'mentəl] *a.* 增加的
- infer [ɪn'fɜ:] *v.* 推断
- licensing *n.* 批准, 执照, 申请
- mandate ['mændeɪt] *v.* 托管
- modularity [ˌmɒdju'lærɪti] *n.* 积木性
- multiplexer ['mʌltɪpleksə] *n.* 多路调制(转换)器
- obsolescence [ˌɒbsə'lesəns] *n.* 逐渐过时
- off-normal *a.* 超常的, 越界的
- overlay [ˌəʊvə'leɪ] *v.* 放上, 盖上
- peripheral [pə'rɪfərəl] *a.* 外围的(设备)
- pre-incident *n.* 事故前
- random ['rændəm] *a.* 随机的
- scenario [si'nɑ:riəu] *n.* 情景, 方案
- strategically [strə'tɪ:dʒɪkəli] *ad.* 战略上, 策略地
- stringency ['strɪndʒənsɪ] *n.* 紧急, 迫切
- swing [swɪŋ] *n.* 摆动
- terminate ['tɜ:mɪneɪt] *v.* 终止, 结束
- unacceptable [ˌʌnək'septəbl] *a.* 不能接受的, 不合格的
- validation [væli'deɪʃən] *n.* 有效, 确认, 证实
- vital ['vaɪtl] *a.* 极重要的

课文注释

1. A small number of plants may have somewhat...utility grid.

句中 process heat 是工业用热; utility grid 是公用电网。整句译为: 少量装置有些不同的标准, 例如有些反应堆设计为工业公司生产工业用热并为公用电网输电。

2. The safety/relief valves would open, but...not terminated.

句中 core meltdown 指堆芯熔化, 是反应堆最严重的事故之一; due to 引导的短语作状语。整句译为: 安全阀或释放阀将打开, 但蒸汽连续产生, 使水不断蒸发, 如果这一过程不

终止，将由于冷却水耗尽而导致堆芯熔化。

3. Consequently, a few trained people could actually...process computers.

句中 of course 是插入成分，它将句子断开，从语义上讲应该是 aided by...。整句译为：因此，少量的经过训练的人员就能实际操作核动力装置了，当然需借助于自动化仪表和过程计算机的辅助。

4. Almost all reactor control rooms...an accident.

句中 safety panel 是安全性操纵屏。整句译为：现在差不多所有的反应堆控制室都有一个安全性操纵屏，来告知操纵人员在事故状态下装置的状态。

5. This commitment, the increasing size and complexity...control technology.

句中 the increasing size and complexity of power plants 和 the greater stringency of regulatory requirements 都是 commitment 的同位语，进一步说明 commitment 的内容。整句译为：这种承诺，随着动力装置尺寸和复杂性的增加，更严格的标准要求，以及电子工业的迅速发展，都意味着要引入新的仪器和控制技术。

6. The individual field sensors...remote multiplexer.

句中 field sensors 是现场传感器；sampled 是与 distributed 相并列的。整句译为：各个独立的现场传感器和执行机构按策略分布，用电缆连接到远程终端上，并用远程多路调制器进行有序取样。

Unit 15 Radiation Hazards and Shielding

15.1 Introduction

One of the most important characteristics of nuclear radiation such as alpha, beta, gamma radiation and neutrons is that these radiations and particles are a hazard to health of human beings and animals due to their ionizing effect in body tissue. Gamma radiation and neutrons, which have a high penetrating power, are an external hazard, and sources of these radiations must be shielded. Alpha and beta particles have a low penetrating power and are only dangerous if a substance emitting these particles is breathed or ingested into the body¹.

In view of this hazard it is important in the construction of a nuclear reactor to provide adequate shielding so that neutrons and gamma radiation originating in the reactor core are prevented from escaping into the reactor's surroundings where people are working. The same problem arises to a smaller extent with radioactive isotopes which are used on a large scale in science, engineering and medicine. In this case the size and strength of the radioactive source are very much less than the core of a nuclear reactor, but it is nevertheless necessary to reduce the radiation escaping from such a source to an acceptable low level.

In this lesson we will discuss first the effects of radiation on living tissue, the units by which these effects are measured, and the presently accepted limits of radiation dose. The shielding of simple radioactive sources will be studied to establish the relationship among source strength, shield thickness and dose rate. The problem of reactor shielding, which involves complex sources of both neutrons and gamma radiation, is beyond the scope of this book and will only be described qualitatively. Finally, the heating of shields due to the absorption of radiation will be discussed.

15.2 Health Physics

The study of the effect of nuclear radiation on human beings is known as health physics, and it has become an increasingly important branch of science in recent years as a result of the expanding use of nuclear power as a source of energy, and the widespread use of radioactive isotopes in science, medicine and industry. As a result of these developments there has been a growing public concern about the health effects of the radiation levels which are now present in the environment. Only with a good understanding of the effects of radiation can adequately safe standards of exposure be set, and public fears allayed. Several different sources of radiation exist in the environment to which the public may be exposed. These include the following:

(1) Natural background radiation, which has three components. ①Cosmic radiation, the stream of charged particles (protons and alpha particles) originating in space which bombards the earth's atmosphere and interacts to produce gamma radiation. This gamma radiation reaches the earth's surface, being more intense at high altitude (where the atmosphere is "thinner") than at low altitude. ②Naturally occurring radioactive isotopes in the earth's crust, such as ^{238}U and ^{232}Th , and their radioactive decay products. The intensity of this source is greater than average in regions of granite rocks, as these contain uranium, and also at one or two places, such as Kerala in India, where the Monazite sands contain quite large quantities of thorium². The most important of the decay products is radon, an inert and heavy gas produced by the decay of both ^{238}U and ^{232}Th . Radon in very small quantities diffuses from uranium and thorium bearing rocks and tends to accumulate in closed spaces, such as unventilated mines, or poorly ventilated buildings. The decay products of radon include radioactive isotopes of lead, bismuth and polonium, and when breathed these can lodge in the lungs and pose a health hazard. Many deaths among early uranium miners were due to this effect. ③Naturally occurring radioactive isotopes, principally potassium40, in the human body.

(2) Medical X-rays and gamma-rays used for diagnostic purposes and the treatment of certain types of disease. Radiation which can cause diseases such as cancer can also be used, in carefully controlled doses, to cure cancer. This is the largest man-made contribution to the radiation dose of the population, but it does not affect the entire population.

(3) Miscellaneous natural sources, including the burning of coal which releases very small quantities of uranium and its daughter products into the atmosphere.

(4) Fallout from the atmospheric testing of nuclear weapons. This effect is related in terms of time to the atmospheric testing of nuclear weapons. In the months following tests the radiation level is quite high, and in years when there is no testing the level drops.

(5) The discharge of radioactive effluents and the disposal of radioactive waste from nuclear reactors and reprocessing plant. Gaseous fission products, principally krypton, discharged into the atmosphere may be widely disseminated, and liquid waste discharged into the sea may be carried to other places by tides and currents.

(6) Occupational exposure, that is the radiation received at work by workers in nuclear power plant, reprocessing plant and other places where radioactive materials are used and handled.

15.3 The Biological Effects of Radiation Exposure

The biological effects of exposure to radiation have been studied since the nineteen-twenties when it first became obvious that radiation could cause serious damage to health, and in extreme cases, death. The international Commission on Radiological Protection (ICRP) was established in 1928, and since then it has been the one internationally recognized, body responsible for setting standards relating to radiation levels.

In the seventy years of existence the ICRP has gathered data from many sources, including early workers using radioactive materials in a way which would now not be permitted, survivors of the two atomic bomb explosions over Japan in 1945, people who have received radiation diagnosis and therapy in hospital, workers in the nuclear industry and people living either near to a nuclear plant, such as power stations, or in areas of high natural background radiation. More recently, as a result of the serious reactor accident at Chernobyl in the former USSR, another large group of exposed persons, those living near to that reactor at the time of the accident, will provide scientists with much data for years to come on the long-term effects of a reactor accident which resulted in a massive discharge of radioactive material into the environment³.

During these seventy years the ICRP has continuously reviewed radiation dose levels that in the light of existing knowledge were regarded as having a negligible probability of causing serious illness. These levels have from time to time been reduced as more evidence has become available, and in 1977 the ICRP recommendation replaced the use of a maximum permissible dose by a more comprehensive system of dose limitation.

The 1977 recommendations may be summarized as the following items.

- (1) No practice shall be adopted unless its introduction produces a positive net benefit.
- (2) All radiation exposures shall be kept as low as reasonably achievable (ALARA), economic and social factors being taken into account.
- (3) The dose equivalent to individuals shall not exceed the limits recommended for the appropriate circumstances by the Commission.

The present whole body dose equivalent limit recommended by the ICRP for members of the public (not including the dose due to natural background radiation and medical sources) is 5 millisieverts per year (mSv/y). In the case of workers undergoing occupational exposure, the corresponding dose limit recommended is 50 millisieverts per year (mSv/y).

The health effects of radiation exposure can be divided into two categories, namely acute and delayed effects. Acute effects are commonly called radiation sickness. The symptoms depend on the magnitude of the dose received. In order of increasing dose the symptoms are fever, internal bleeding, vomiting and diarrhoea, tremors, cramps and coma. The dose threshold for the onset of radiation sickness is about 1 Sv, and for a dose of 4 Sv about 50 percent of the victims would probably die. Few people could survive an exposure of more than 6Sv. Radiation sickness may not be evident for up to two weeks following exposure, depending on the dose, and death, if it occurs, may not happen until tight weeks after exposure⁴. In cases of very severe exposure, however, death may follow within two days.

Delayed effects take the form of cancers and hereditary effects, and in discussing them the following terms should be noted.

- (1) Somatic effects: those effects in which the radiation induced sickness appears in the irradiated person.
- (2) Hereditary effects: those effects which are evident only in the children of the irradiated

person as a result of radiation damage to the reproductive organs.

(3) Stochastic effects: those effects for which the probability or occurrence is regarded as a function of the dose of radiation received.

15.4 The Effects of Inhaled and Ingested Radioactivity

Studies of the effects of accidents involving the release of radioactivity show that, with the exception of those very close to the scene of the accident at the time it happens, the main hazard to the population as a whole arises from the inhalation or ingestion of air or food which is contaminated by the radioactive material from the accident. For example, the accident at the former USSR nuclear power station at Chernobyl in April 1986 produced severe acute somatic effects in a few hundred people, namely the power plant operators and firemen exposed to radiation at the time of the accident and the subsequent fire⁵. In addition, many thousands of people may suffer in years to come from delayed stochastic effects from the radioactive debris of the accident (principally fission products) which was carried in the atmosphere for hundreds of miles from Chernobyl and deposited on the ground by rain, where it entered the food chain of humans and animals. Fallout from nuclear weapons testing produces a similar hazard.

In order to make calculations about the risks of such exposure, it is necessary to consider the following points.

- (1) How much of each radioactive isotope is inhaled or ingested.
- (2) The organs of the human body in which each isotope may become concentrated.
- (3) The dose which each organ receives from a given quantity of inhaled or ingested radioactive material.
- (4) The dose-risk coefficient, known as the weighting factor, for each organ.

The first point depends on the amount of radioactivity released and its subsequent dispersion in the atmosphere and return to ground level. The second and third points have been the subject of medical research, and Table 15.1 shows the organs affected by each of three important radioactive fission products, and the dose (Sv) which results from the inhalation or ingestion of 1 GBq of the fission product concerned.

Table 15.1 The dose due to certain fission products

Isotope	Principal organ (s) affected	Inhalation factor (Sv/GBq)	Ingestion factor (Sv/GBq)
⁹⁰ Sr	(1) Red bone marrow	330	190
	(2) Bone	730	420
¹³¹ I	Thyroid	290	480
¹³⁷ Cs	Whole body	8	13

The weighting factors mentioned above are the ratios of the stochastic risk resulting from radiation damage to an individual organ to the total risk when the body is uniformly irradiated.

15.5 Reactor Shielding

The problem of reactor shielding is great deal more complicated than that of shielding a simple source of gamma radiation. In a reactor the principal source of radiation, neutrons and gamma radiation, is in the core itself, however radioactive coolant flowing in ducts and heat exchangers, and irradiated fuel elements in storage pits may provide additional sources of gamma radiation.

Neutrons and gamma radiation produced in the core have a wide spectrum of energies, and neutrons passing from the core into the shield produce more gamma radiation as a result of (n, γ) reactions in the shield. Beta radiation is produced by fission product decay, but only high energy beta radiation is important because as it is slowed down, gamma radiation known as Bremsstrahlung (braking radiation), is produced.

The important sources of radiation in a reactor can be summarized as follows.

- (1) Neutrons.
- (2) Prompt gamma radiation. This is the gamma radiation emitted at the instant of fission; the average number of photons emitted per fission is 8 with an energy spectrum up to about 7 MeV. The average energy per photon is about 1 MeV.
- (3) Fission product gamma radiation. This is the gamma radiation emitted during the decay of radioactive fission products. The radiation from short-lived fission products is important during reactor operation and rapidly decays when the reactor is shut down. The radiation from long-lived fission products may be of more importance after shut-down, particularly in the cooling of fuel in the core and the shielding of irradiated fuel elements.
- (4) Capture gamma radiation. This radiation is emitted as a result of (n, γ) reactions in the core and (to a lesser extent) in the shield. It provides a source of radiation in the shield itself.
- (5) Activation product gamma radiation. If the product of an (n, γ) reaction is radioactive, it will provide a further source of gamma radiation as it decays. Reactor coolants which become radioactive as a result of (n, γ) reactions in the core are a source of radiation in external ducts and heat exchangers both during operation and after shutdown. Sodium is an important example of such a coolant.

Less important sources of radiation are as follows.

- (1) Gamma radiation from the inelastic scattering of fast neutron.
- (2) Photo-neutrons resulting from (n, γ) reactions in reactors containing significant quantities of heavy water and beryllium. (^2H and ^9Be are the isotopes responsible for this reaction.)
- (3) Annihilation radiation. This is the name given to the 0.51 MeV gamma radiation which results from the annihilation of positrons with electrons. The positrons may originate

from high energy gamma radiation undergoing pair production processes.

(4) Bremsstrahlung. This gamma radiation is produced by the deceleration of high energy beta particles. It may be of importance if ${}^7\text{Li}$ is used as a reactor coolant because neutron capture in ${}^7\text{Li}$ produces ${}^8\text{Li}$ which decays by the emission of high energy beta particles.

The main radiation shield of a reactor is known as the biological shield. Its function is to reduce the intensity of neutrons and gamma radiation escaping from the reactor to a level which is acceptable from the health physics point of view. The design of the biological shield involves the following steps.

(1) Determine the neutron flux distribution in the core of the reactor, and hence determine the distribution of sources of gamma radiation in the core, and the energy spectrum of these sources.

(2) Determine the current and energy of neutrons leaking out of the core into the shield, and hence the rate of slowing down and absorption of neutrons in the shield. This establishes the sources of gamma radiation in the shield.

(3) The neutron current through the outer surface of the shield is determined.

(4) Once the sources of gamma radiation in the core and shield are known, the gamma flux and energy in the shield, and the current through its outer surface are determined.

(5) The dose rate at the outer surface of the shield is calculated from the neutron and gamma currents through it. The shield thickness is designed so that this dose rate is within acceptable limits.

(6) The rate of energy absorption of gamma radiation and neutrons in the shielding must be determined. This energy absorption results in the heating of the shield, and in the case of power reactors this heating may lead to unacceptable temperatures and thermal stresses.

To reduce the heating effect in the biological shield, a comparatively thin thermal shield is placed between it and the core of the reactor. The function of the thermal shield, which is cooled by a suitable flow of coolant, is to reduce the neutron and gamma currents from the core into the biological shield, and thereby reduce the rate of energy absorption and the temperature rise in this shield. The thermal shield should have a good thermal conductivity, and be able to withstand thermal stresses. Steel is commonly used as the material of the thermal shield.

The biological shield should contain some hydrogen compound to slow down fast neutrons, and be dense enough to attenuate gamma radiation effectively. Concrete satisfies both these requirements fairly well and is suitable for land-based reactors. Barytes concrete, containing the heavy element barium, and steel-shot concrete have been used for biological shields⁶. They are more dense than ordinary concrete, with improved shielding properties, however their higher cost offsets this advantage. The biological shield for a marine reactor, which is usually a fairly compact pressurized water reactor, must satisfy a minimum space and weight requirement. This leads to a shield design which consists typically of alternate layers of water (for fast neutron slowing) and steel (for gamma ray attenuation).

Self-test

1. The biological shields for a marine reactor are usually _____.
A. thermal shields B. barytes concrete
C. water and steel D. lead and water
2. The hazard of nuclear radiation to human beings is due to the _____ of radiations and particles.
A. high power B. ionizing effect
C. penetrating power D. high energy
3. If someone receives a little dose, the symptom is _____.
A. fever B. internal bleeding
C. both A and B D. neither A nor B
4. The person received 8 Sv exposure will be _____.
A. internal bleeding B. fever
C. both A and B D. dead
5. The ICRP investigated the people from _____.
A. different places B. Japan
C. nuclear plant D. nuclear industry
6. The delayed stochastic effects come after _____.
A. two weeks B. a month
C. long time D. several years
7. In a reactor the principal radiations are _____.
A. beta and gamma radiations B. α and β radiations
C. neutrons and gamma radiations D. fission products
8. The radiation from long-lived fission products is important _____.
A. during reactor operation B. long time
C. after reactor shutdown D. several years later
9. Capturing gamma radiation is a result of _____.
A. bremsstrahlung B. (n, γ) reaction
C. fission D. radiation decay
10. The main radiation shield of a reactor is _____.
A. biological shield B. thermal shield
C. concrete shield D. neutron shield
11. The heat of the thermal shield is cooled by _____.
A. water B. gas C. liquid D. coolant
12. Cosmic radiation consists of _____.
A. neutron and gamma radiation B. protons and alpha particles
C. heavy particles D. radon

关键字解释

Biological shield: Screen made of material highly absorbent to radiation used as a protection.

生物屏蔽: 用吸收辐射能力强的材料制成的屏蔽, 用于防护。

Bremsstrahlung: Electromagnetic radiation emitted when a charged particle changes its velocity.

韧致辐射: 带电粒子速度改变时发射的电磁辐射。

Cosmic radiation: Strongly penetrating ray coming from beyond the earth's atmosphere and containing particles that move at extremely high speed.

宇宙射线: 来自地球大气以外的穿透能力很强的射线, 含有高速运动的粒子。

Daughter product: A nuclide that originates from the radioactive disintegration of another parent nuclide.

子代产物: 从核素的放射性蜕变中产生的核素。

Effluent: Liquid sewage after having passed through any stage in its purification.

废水: 净化处理的任何阶段排出的废液。

Fallout: Airborne radioactive contamination resulting from nuclear explosion, inadequately filtered reactor coolant, failure of reactor containment after an accident.

放射性坠尘: 因核爆炸、未经充分过滤的反应堆冷却剂、事故后反应堆安全壳故障等造成的空气放射性污染。

ICRP: International Commission of Radiological Protection.

国际辐射防护委员会

Natural background: In the detection of nuclear radiation, the radiation due to natural radioactivity and to cosmic rays, enhanced by contamination and fall-out.

天然本底: 在探测核辐射中, 由于天然放射性和宇宙射线引起、因污染和降尘增加的辐射。

Shielding: Use of dense matter to attenuate radiation when it might be harmful to operator or measuring systems.

防护屏蔽: 在可能伤害操作人员或测量仪器的情况下, 用高密度材料使辐射衰减。

词 汇

acute [ə'kju:t] *a.* 急剧的

adequately ['ædikwitli] *ad.* 充分地

allay [ə'lei] *v.* 缓和

altitude ['æltitju:d] *n.* 高度

barite(s) [bə'reit(s)] *n.* 重晶石 (混凝土)

barium ['beəriəm] *n.* 钡

- bismuth ['bizməθ] *n.* 铋
bombard [bɒm'bɑ:d] *v.* 轰击
coma ['kəʊmə] *n.* 昏迷
cramp [kræmp] *n.* 绞痛
crust [krʌst] *n.* 壳
debris ['debri:] *n.* 碎片, 残渣
diagnostic [ˌdaɪəɡ'nɒstɪk] *a.* 诊断的
diarrhea [daɪə'riə] *n.* 腹泻
disseminate [di'semineɪt] *v.* 传播
evident ['evidənt] *a.* 明显的
granite ['grænit] *n.* 花岗岩
hereditary [hi'redɪtəri] *a.* 遗传性的
inhale [in'heɪl] *v.* 吸入
krypton ['kriptɒn] *n.* 氪
lodge [lɒdʒ] *n.* 射入, 沉淀
marrow ['mæərəʊ] *n.* 骨髓
millisievert *n.* 毫希伏
miscellaneous [mɪsi'leɪniəs] *a.* 杂项的, 各种的
pit [pɪt] *n.* 坑
polonium [pə'ləʊniəm] *n.* 钋
potassium [pə'tæsjəm] *n.* 钾
radon ['reɪdɒn] *n.* 氡
somatic [səʊ'mætɪk] *a.* 身体的
spectrum ['spektrəm] *n.* 谱
stochastic [stəʊ'kæstɪk] *a.* 随机的
survivor [sə'vaɪvə] *n.* 幸存者
symptom ['sɪmptəm] *n.* 征兆
therapy ['θerəpi] *n.* 治疗
threshold [ˌθreʃ'həʊld] *n.* 临界点, 限度, 阈值
thyroid ['θaɪərɔɪd] *n.* 甲状腺
tissue ['tɪʃu:] *n.* (细胞) 组织
tremor ['tremə] *n.* 颤抖
vomit ['vɒmɪt] *v.* 呕吐

课文注释

1. Alpha and beta particles have...into the body.

句中 only dangerous if...表示只有当……时才是危险的。整句译为: α 和 β 粒子的穿透能力低, 只有当发射这些粒子的物质吸入或饮入人体内才有危险。

2. The intensity of this source is...of thorium.

句中 Kerala in India 是印度的喀拉拉邦，它位于印度的西南部，西临阿拉伯海。Monazite sands 是莫纳斯特沙滩。整句译为：因为这里含有铀，这里源的强度大于花岗岩区的平均值；另外，还有一些地方，例如印度的喀拉拉邦的莫纳斯特沙滩，含有大量的钍。

3. More recently, as a result of the serious...the environment.

句中 Chernobyl in the former USSR 是前苏联的切尔诺贝利核电站，这里曾发生过严重的核泄漏事故。整句译为：最近在前苏联的切尔诺贝利核电站发生了严重的反应堆事故，使当时生活在附近的很多人受到辐射，这一核事故造成大量放射性物质排放到环境中去，在未来的几年中，这些将为科学家研究反应堆事故后的长期影响提供了丰富的数据。

4. Radiation sickness may not be evident...after exposure.

句中 if it occurs 是插入在主语 death 和谓语之间的状语；tight weeks 是紧接着的几个星期内。整句译为：辐照病变在受到辐照后的两个星期内一般不太明显，这与受到辐照的剂量有关；如果出现死亡的话，一般在受到辐照后的近几星期内不会出现。

5. For example, the accident at...the subsequent fire.

句中 namely 是副词“就是”的意思，它后面的短语是 people 的补语，说明是哪一些人。整句译为：例如前苏联切尔诺贝利核电站 1986 年 4 月发生的事故，对几百人造成了严重的急剧身体影响。这些人员是核电站的运行人员以及参加核电站救火的消防人员。

6. Barytes concrete, containing the heavy...biological shields.

句中 Barlytes concrete 是重晶石混凝土，steel-shot concrete 是钢珠混凝土。全句译为：含有重元素钡的重晶石混凝土和钢珠混凝土已经被用来作为生物屏蔽。

Unit 16 Nuclear Safety

16.1 Safety and Hazards in Nuclear Power

From the earliest years of nuclear reactor development it has been realized that an accident in which a reactor becomes supercritical by a substantial amount (i. e. greater than the delayed neutron fraction), resulting in a sudden rise in the fission rate, power and temperatures, is an event that can have very serious consequences. These consequences might include fuel overheating and meltdown, release of radioactive fission products into the coolant and possibly eventually into the atmosphere, and very rapid increases in temperatures and pressures resulting in possible rupture of pressure vessels and other containment structures.

As a result reactor designers have paid great attention to the inherent safety of reactors which can be achieved by negative temperature and power coefficients and fail-safe control systems. It can be said with some confidence that present-day thermal reactors are safe in the sense that under no conceivable circumstance can they explode like a bomb, and control systems have been designed which can, in the event of any malfunction on the part of the reactor or its associated plant, automatically and rapidly shut down the reactor, i. e., make it subcritical by a substantial amount, in a very few seconds¹.

Fast reactors, with their greater concentration of fissile fuel, would appear to be inherently more hazardous than thermal reactors, and it is conceivable that if the core of a fast reactor melted the fuel might slump into a supercritical configuration which would result in a very sudden energy release, but hardly an explosion. Needless to say, fast reactors are designed in such a way as to eliminate this possibility, and a negative temperature coefficient is an important characteristic for a fast reactor.

Many reactor-years of operating experience have shown that it is not the fission chain reaction in the reactor core that is the most likely source of malfunction and accidents. but the ‘conventional’ components of the power plant such as pumps, valves, switches, relays and parts under stress such as pressure vessels or pipework. Human error on the part of operating and maintenance staff has also been proved to be a rather frequent source of trouble in nuclear power plant.

These factors are not peculiar to nuclear power plant, but they assume great importance because of the hazardous nature of nuclear reactors. Designers have to ensure that all systems should as far as possible be fail-safe and redundant, i.e. if one system fails to function correctly, another is available to fulfill the same function.

As stated above, nuclear reactors cannot explode like nuclear bombs. This is primarily because of the fast acting negative thermal feedback due to Doppler broadening of the ^{238}U absorption resonances². In addition in thermal reactors where neutrons are moderated, the

prompt neutron lifetime L_p is the order of 10^{-4} seconds; in a bomb, since the neutrons are unmoderated, the prompt neutron lifetime is of the order of 10^{-8} seconds. Finally, reactor fuel consists typically of 2 to 3 percent ^{235}U , whereas nuclear weapons contain almost pure ^{239}Pu . The net effect of these differences is that, even in a reactor which is totally out of control and gone prompt critical (as happened at Chernobyl), the reactor period will not be much less than a second or so. In a nuclear bomb, the period is of the order of nanoseconds.

16.2 Reactor Accidents

Serious reactor accidents, which might lead to a significant release of radioactivity, can only arise following events which may be described under four broad categories, as described below. In a well-designed system, each accident route requires a number of coincident failures for a release of radioactivity to occur; the probability of any given accident is thereby minimized. Two serious reactor accidents which have occurred in the last decade, at Three Mile Island, USA in 1979 and at Chernobyl, USSR in 1986 will be described in this section.

16.2.1 Reactivity Insertion Accident (RIA)

This can occur if a significant amount of reactivity is added to a reactor core. Such an event could arise due to an uncontrolled withdrawal of control rods from the reactor due to faults in their control systems, with a coincident APS failure. This event could also arise in a reactor design which has fast-acting positive reactivity coefficients (leading to instability), with a coincident APS failure and/or a poor fast shutdown capability (The last scenario occurred at Chernobyl). The probability of such accidents is minimized by ensuring that all fast-acting reactivity coefficients are negative, and that reliable, fast-acting, diverse shutdown systems exist. These accidents cannot lead to nuclear explosions.

16.2.2 Loss-of-coolant Accident (LOCA)

This arises as a result of a breach of the primary coolant circuit, with a coincident failure of the emergency core cooling system (ECCS). In particular, the term LOCA usually relates to water-cooled reactors, where a complete loss of coolant will also give rise to a loss of decay heat rejection capability, and hence fuel damage. The Three Mile Island accident was such an event. Gas-cooled reactors are not subject to LOCAs, but to the less severe event of a depressurization of the primary circuit, decay heat rejection would normally be sustained in such accidents.

The term anticipated transient without scram (ATWS) is used in water-cooled reactor systems to refer to event which would follow a 'routine' plant failure, a turbine trip or a feedwater pump trip when the reactor trip systems fail to insert the control rods into the core. Such events should not lead to any activity release in any well-designed systems; the negative reactivity coefficients should lead to a rapid fall in reactor power, and in any case, most power reactors are fitted with secondary shutdown systems.

Certain external events could conceivably lead to reactor accidents, say an earthquake or

collision from a crashing aircraft. Reactors are commonly designed to resist the maximum magnitude of earthquake that is foreseeable in the district where the plant is to be sited. Similarly, reactors are often designed to resist light aircraft crashes and they are usually sited away from civil airline or airforce training routes. Hence the risk from external events can be minimized.

16.3 The Accident at Three Mile Island

The Three Mile Island Power Station is situated on an island in the Susquehanna River ten miles from the town of Harrisburg in Pennsylvania³. The layout of reactor and its associated plant are shown in Fig.16.1.

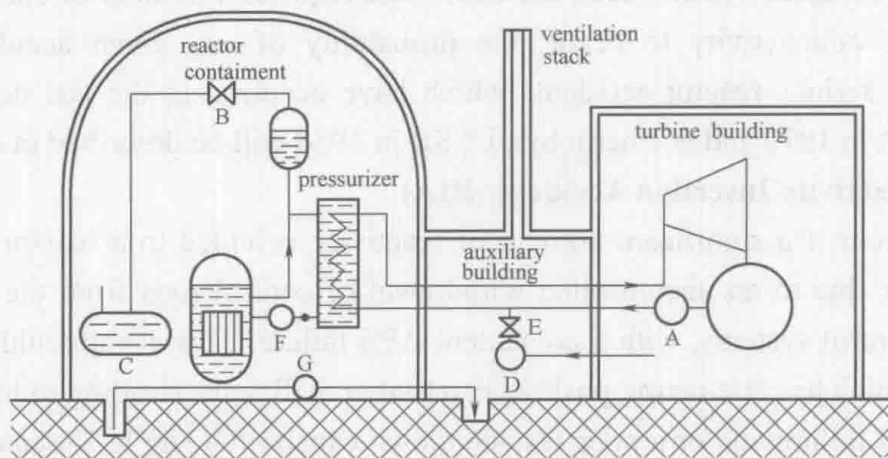


Fig.16.1 layout of the three mile island PWR

The accident, which affected only one of the two reactors, originated with a failure of the feedwater system which caused the feedwater pump A to cut out. The turbine and the reactor shut down automatically, but with the feedwater pump out of action, fission product decay heat was not transferred from the reactor via the heat exchangers. As pressure in the reactor and the primary system increased, the pressure relief valve B opened and primary coolant was released to the pressurizer relief tank C. At the same time the standby feedwater pumps D started up, but were unable to deliver water to the heat exchangers because valves E were closed, contrary to operating regulations. About 8 minutes after the start of the accident the operators opened valves E and flow was established in the secondary circuit; at the same time the pressure in the primary system continued to drop because the safety valve B was stuck in the open position. As the reactor pressure dropped, the ECCS was automatically started (as it is designed to do) and cold water was injected into the reactor. A few minutes later an instrument indication of rising water level in the pressurizer led the operators to shut off the ECCS.

An hour or two later severe vibration in the primary circuit pumps F led the operators to switch off these pumps and circulation in the primary systems stopped. Meanwhile with decay heat still being produced, the water in the core boiled and part of the core was uncovered for a

few hours, with the steam environment unable to provide adequate cooling.

The resulting high temperature caused damage to the fuel elements resulting in release of certain volatile fission products into the primary system, and these fission products found their way via the pressurizer, the still open pressure relief valve, the pressurizer relief tank (which overflowed) into the sump in the reactor containment floor. From there the contaminated water was transferred by the pump G into the auxiliary building and there the radioactive volatile fission products vented from the water and escaped into the atmosphere via the ventilation stack. Thus significant quantities of xenon 135 and iodine 131, with smaller amounts of krypton, escaped into the atmosphere; the iodine release, which was the most significant in radiological terms, amounted to some 16 curies.

Meanwhile, back in the reactor the very high fuel temperatures in the uncovered part of the core caused the zirconium-steam reaction and hydrogen was formed, collecting in the top of the pressure vessel and causing a restriction to the flow of primary coolant. It was thought at the time that this hydrogen presented an explosion risk, but this is unlikely as oxygen was not present in the pressure vessel. Hydrogen did, however, escape into the containment building, and ten hours after the beginning of the accident there was a hydrogen 'burn' which caused a rise in the pressure within the containment building.

About 16 hours after the start of the accident the primary coolant pumps were restarted and water flow through the reactor was re-established to cool the core. Gradually during the next few days the hydrogen in the pressure vessel was removed and the accident was brought under control. There can be no doubt that, although there were no immediate disastrous consequences, this was a very serious accident resulting in the release of considerable quantities of radioactivity into the atmosphere and the destruction of the reactor itself, which is now out of action.

The lessons to be learned from this accident are many. It certainly illustrates how a relatively simple fault can lead through a sequence of events to a major accident, and how this sequence of events can occur as a result of a combination of malfunction of everyday engineering components such as pumps and valves, human error and misjudgment and inadequate design.

It has been estimated that this accident may lead to one additional cancer death among the nearby residents in the years to come. The financial cost of one destroyed reactor and the subsequent clean-up operation has been huge.

16.4 The Accident at Chernobyl

In April 1986, the No.4 reactor of the Chernobyl power station, situated on Pripyat River north of Kiev in the former USSR, suffered a major accident which caused a significant release of radioactivity to the environment⁴. The reactor was of the graphite-moderated, water-cooled, pressure-tube design, which is a design unique to the former USSR. The layout of the plant is

shown in Fig.16.2.

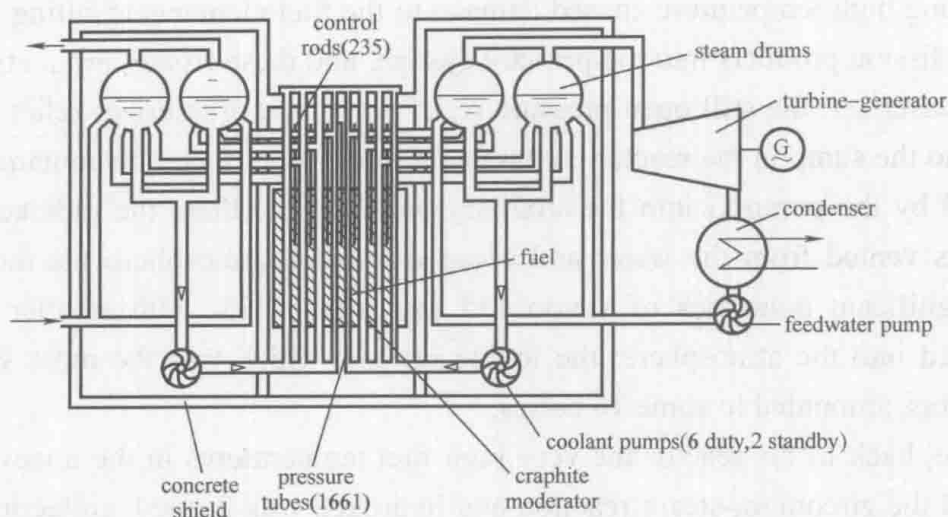


Fig.16.2 Layout of the reactor at Chernobyl

The reactor has many features which are not found in other reactor systems, and which raised concern about its safety. A team of British engineers visited the former Soviet Union in the 1970s and reported the following concerns about the reactor design (amongst others).

(1) The reactivity worth of the control rods was inadequate to meet all possible shutdown requirements.

(2) The reactor had a positive void coefficient and a positive power coefficient at low powers. This arose because the design was over-moderated which, in turn, was apparently because a decision had been made to use fuel that was only about two per cent enriched, since the USSR was short of enrichment capacity.

(3) There was no secondary shutdown system.

(4) The graphite moderator normally operated at temperatures in excess of 700°C, at which it spontaneously ignites on exposure to air.

(5) The containment structure was less robust than would normally be the case for Western water-cooled reactors.

Furthermore, we now know that the design allowed greater freedom of action for the operators than might be considered normal in a Western design. For example, the operators were apparently able to override trip systems at the flick of a switch; in Western designs, key interlock systems prevent such precipitate action. Also it was essential for the safe operation of the plant that the control rods should never be withdrawn beyond the point at which the control rod "reactivity margin" became dangerously low; this vital aspect was left entirely to the operators, with no automatic trip system.

Finally, it would appear that reactor emergency shutdowns were normally carried out by motoring the rods into the core, and not by disconnecting electromagnetic clutches. Because the rods were driven-in instead of being allowed to drop, shutdown could take 20 seconds to occur.

Ironically, the accident occurred during a test to check whether the emergency core cooling systems were adequate. The test was to be performed during a routine shutdown. The main events were as follows.

(1) Power was to be reduced in stages from 3200 MW (th) to 700 MW (th). Operator error caused the power to undershoot to 30 MW, and the consequent increase in xenon poisoning meant that the operators could only manage to raise the power back to 200MW. This power level was lower than stipulated in the test instructions, the operators should have abandoned in the test instructions.

(2) In violation of the test procedure, the operators started up the standby coolant pumps. This meant that core power was 7 percent normal and coolant flow was 120 percent normal. This made the whole core virtually isothermal.

(3) Difficulties in steam drum water level control at this juncture led the operators to override the reactor trip signals generated by low drum level. They then topped up the drum water level under manual control, with relatively cold feedwater. This caused a fall in reactor temperature, and hence (via the positive void coefficient) the control rods had to be withdrawn yet further to maintain reactor power level at 200 MW (th).

(4) The operators noticed that the control rod reactivity margin was too low. Indeed, the reactivity margin was at a level where operating rules stipulated that the reactor should have been shut down. No such action was taken.

(5) The test was to be initiated by tripping the turbine. Normally, this would have tripped the reactor also. However, it would appear that the operators wanted to reserve their options in case the test of the emergency core cooling system was unsatisfactory, hence they overrode the reactor trip. This meant (they thought) that they would be able to repeat the test if necessary.

(6) At 1: 23'10" on the 26th April 1986, the turbine was tripped. The operators had contrived, unwittingly, to put an already unstable reactor into a highly dangerous condition. The reactor was operating at low power, and was almost isothermal due to the high coolant flow-rate. By tripping the turbine the only significant heat sink had been removed. In addition, the control rods were too far out of the core to have any significant immediate effect in the event of a reactor trip.

(7) The coolant temperature now began to rise steadily until it approached the saturation temperature, at which point bulk boiling occurs. Because of the virtually isothermal state of the reactor, the value of the positive void coefficient had been maximized, i.e. bulk boiling would begin throughout the core more-or-less simultaneously.

(8) At 1: 23'40", a rise in power was noted and a reactor trip was initiated manually, by starting to drive the control rods into the core. Because the rods were so far out of the core, however, they had no significant immediate effect on reactivity, which continued to rise.

(9) Coolant bulk boiling led to a rapid rise in power, which Doppler broadening could not counteract.

(10) The power rose to 530 MW (th) at 1: 23'43", and thereafter rose exponentially with a

period of a fraction of a second. The reactor went prompt critical; the fuel (uranium oxide ceramic) shattered due to the thermal shock of the sudden power rise, and the cladding melted; the white hot fuel fragments came in contact with the cooling water, and there was a steam explosion at 1: 23' 48". This was followed a few seconds later by a hydrogen explosion; the hydrogen was generated by zirconium-water and graphite-water reactions. There was no nuclear explosion, for reasons stated earlier. The explosions ruptured the containment.

(11) The hot graphite moderator caught fire upon exposure to air.

(12) Truly heroic efforts by firemen, helicopter pilots and engineers led to the fire being extinguished, and the radioactive release being stopped, by 5 May, ten days after the accident began. In the interim, it is estimated that 20 percent of the iodine inventory and 12 percent of the caesium inventory was released to the atmosphere, together with practically all of the noble gas fission products.

Many significant conclusions may be drawn from the accident. However, few, if any, of the following conclusions are likely to affect Western reactor design⁵.

First and foremost, reactors should have no fast-acting positive power coefficients under normal or foreseeable operating conditions. The reactor design is unique in this respect among all of the world's commercial reactor designs. Secondly, reactor protective systems should shut down the reactor automatically when important safety parameters exceed normal margins. It is insufficient to leave such action to the operators. The protective system should not be easily overridden. Thirdly, reactor shutdown systems should be fast-acting and highly reliable (There is no real need for a fast-acting shutdown system if fast-acting reactivity coefficients are negative). To obtain a sufficient degree of reliability, it may be necessary to have an alternative (secondary) shutdown system. This is common practice with Western designs. Finally, water-cooled reactor designs need sound containment buildings for the reasons previously stated.

The consequences of the accident have been that 31 people died either at the time of accident or from radiation sickness. In addition, 135 000 people living within 30km of the plant had to be evacuated from their homes, because of the doses they received. It is possible that 200 additional cancer deaths may occur amongst them. Soviet data suggest that some 10 percent of the population dies from cancer in any case. Hence, this excess mortality may just be observable with statistical confidence, when the results of epidemiological surveys become known in years to come.

Those further from the plant did not receive significant doses of radiation; the doses measured are of the order of a few millisieverts. If the (pessimistic) linear doses-risk hypothesis is assumed valid at such small doses, then it is possible that several thousand additional cancer deaths may arise in Eastern Europe over the next few decades. However, the natural incidence of the disease will mean that this additional mortality, if it occurs at all, will probably not be observable. In other words, there is no significant increase in the individual risk of death due to cancer as a result of the accident, except among those living close to

Chernobyl.

The Chernobyl accident is about as bad a reactor accident as can be envisaged. The reactor core burned without any containment for the days; it is difficult to postulate an accident which could lead to a greater release of radioactivity to the environment.

Self-test

1. Inherent safety of reactors can be achieved by____.
 - A. negative temperature and power coefficients
 - B. fail-safe control system
 - C. both A and B
 - D. neither A nor B
2. In terms of inherent safety the fast reactors are more____than thermal reactors.
 - A. safe
 - B. hazardous
 - C. fast reaction
 - D. sound
3. The reactor period is about____.
 - A. one second
 - B. much less than a second
 - C. several nanoseconds
 - D. several seconds
4. If a significant amount of reactivity is added to a reactor core it means a(n)____.
 - A. LOCA
 - B. APS
 - C. ECCS
 - D. RIA
5. The Three Mile Island Accident was a____.
 - A. RIA
 - B. LOCA
 - C. APS
 - D. both A and B
6. The accident at Three Mile Island originated with the____.
 - A. water level in reactor
 - B. lower pressure
 - C. feed water system
 - D. ECCS
7. During Three Mile Island accident high temperature in the core is caused by____.
 - A. decay heat
 - B. fission
 - C. boiling
 - D. fission products
8. During the accident the hydrogen come from the____.
 - A. reactor
 - B. coolant
 - C. fuel
 - D. zirconium-steam reaction
9. The lesson from Three Mile Island accident is that____can lead to a major accident.
 - A. a small accident
 - B. a simple fault
 - C. hydrogen
 - D. a bad operator
10. The reactor in the Chernobyl Power Station is cooled by____.
 - A. gas
 - B. graphite
 - C. water
 - D. sodium
11. The void coefficient in the reactor of Chernobyl Power Station is____.
 - A. negative
 - B. a constant
 - C. positive
 - D. positive at lower power
12. From the text the containment of Chernobyl Power Station____.
 - A. was not sound
 - B. is a small building
 - C. is a good building
 - D. was fired

关键字解释

APS: Automatic Protective System

自动保护系统

Curie: Unit of radioactivity. 1 Curie is defined as 3.7×10^{10} decays per second, roughly equal to the activity of 1 gram of ^{226}Ra . Abbrev. Ci. Now replaced by the Becquerel (Bq). $1\text{Bq} = 2.7 \times 10^{-11}\text{Ci}$.

居里: 放射性单位。1 居里定义为 3.7×10^{10} 衰变 / 秒, 约等于 1 克 ^{226}Ra 的放射性, 缩写为 Ci。现已改用贝可 (Bq), $1\text{Bq} = 2.7 \times 10^{-11}\text{Ci}$ 。

Delayed neutrons: Those neutrons arising from fission but not released instantaneously. Fission neutrons are always prompt neutrons, those apparently delayed arise from breakdown of fission products, not primary fission. Such delay eases control of reactors.

缓发中子: 裂变产生的, 但未瞬即释放的中子, 裂变中子常为瞬发中子, 明显缓发的中子是裂变产物破裂产生的, 不是原裂变产生的, 这种延迟使反应堆的控制较容易。

Doppler broadening: Frequency spread of radiation in single spectral lines, because of Maxwell distribution of velocities in the molecular radiators. This also broadens the resonance absorption curve from atoms or molecules excited by incident radiation.

多普勒增宽: 在分子放射器中, 麦克斯韦速度分布引起的单独一条光谱辐射频率的扩展, 这也使在入射辐射激发下原子或分子共振吸收曲线增宽。

ECCS: Emergency Core Cooling System.

应急堆芯冷却系统

Fail-safe: Design in which power supply, control or structure is able to return to a safe condition in the event of failure or mal-operation, by automatic operation of protective devices or others.

故障安全: 在发生故障或操作失误时, 通过保护装置的自动投入或其他措施, 仍能使供电、控制或结构回复到安全状态的一种设计。

Prompt critical: Condition in which a reactor could become critical solely with prompt neutrons. In this situation the reactor is difficult to control because the reactor period is much reduced.

瞬发临界: 反应堆处于只靠瞬发中子就能临界的状态, 此时反应堆周期变得极短, 反应堆很难控制。

Prompt neutron: Those neutrons released in a primary nuclear fission process with practically zero delay.

瞬发中子: 在初级核裂变过程中释放的中子, 其实际延迟时间为零。

词 汇

abandon [ə'bəndən] v. 放弃

clutch [klʌtʃ] n. 离合器

- conceivably [kən'si:vəbli] *ad.* 可以想象地
 consequence ['kɒnsɪkwəns] *n.* 后果
 contrary ['kɒntrəri] *v.* 与……相反
 contrive [kə'traɪv] *v.* 竟然弄到……地步, 设法
 conventional [kən'venʃənəl] *a.* 常规的, 传统的
 counteract [kauntə'rækt] *v.* 减少, 反作用
 envisage [ɪn'vɪzɪdʒ] *v.* 想象
 epidemiological [epɪ'dɪ:mɪə'lɒdʒɪkəl] *a.* 流行病的
 exponentially [ˌɛkspəʊ'nənʃəli] *ad.* 按指数规律地
 flick [flɪk] *n.* 轻按
 foreseeable *a.* 可预见的
 helicopter ['helɪkɒptə] *n.* 直升机
 inherent safety 固有安全性
 interim ['ɪntərɪm] *n.* 中间
 ironically [aɪ'rɒnɪkəli] *ad.* 令人啼笑皆非
 isothermal [aɪsəu'θə:məl] *a. & n.* 等温 (的)
 juncture ['dʒʌŋktʃə] *n.* 连接
 mortality [mɔ:'tælɪti] *n.* 致命性
 nanosecond ['nænəʊ,sekənd] *n.* 毫微秒, 纳秒, 10^{-9} 秒
 peculiar [pi'kju:ljə] *a.* 特有的
 pessimistic [pesɪ'mɪstɪk] *a.* 悲观的
 precipitate [pri'sɪpɪteɪt] *n. & a.* 突然 (的)
 radiological [reɪdɪəu'lɒdʒɪkəl] *a.* 放射学的
 relay ['ri:leɪ] *n.* 继电器
 robust [rəu'bʌst] *a.* 坚固的
 shatter ['ʃætə] *v.* 破裂, 松落
 slump [slʌmp] *n. & v.* 坍塌
 staff [stɑ:f] *n.* 工作人员
 stuck *v.* stick 的过去分词; 卡住
 sump [sʌmp] *n.* 排水沟
 supercritical [ˌsju:pə'krɪtɪkəl] *a.* 超临界
 undershoot ['ʌndəʃu:t] *v. & n.* 下冲
 unwittingly [ʌn'wɪtɪŋli] *ad.* 无意地
 volatile ['vɒlətaɪl] *a.* 挥发性的

课文注释

1. It can be said with some...very few seconds.

句中 under no conceivable circumstance can...表示在可想象的条件下不能……。整句译

为:可以确信地说,现代的热中子反应堆是安全的,在可想象的条件下都不能像原子弹那样发生爆炸;在反应堆及有关装置误动作的情况下,控制系统能自动并迅速的使反应堆停堆,即在几秒之内使它达到足够的次临界。

2. This is primarily because of the fast acting...resonances.

句中 negative thermal feedback 是负热反馈,指当温度升高后 ^{238}U 共振吸收增加会使反应性下降。全句译为:这主要是由于 ^{238}U 共振吸收的多普勒增宽产生的快速负热反馈。

3. The Three Mile Island Power Station...Pennsylvania.

句中 Three Mile Island Power Station 是三里岛核电站,它在 1979 年发生了严重的反应堆事故,使反应堆堆芯部分熔化; Susquehanna River 是萨斯魁哈娜河; Harrisburg in Pennsylvania 是宾夕法尼亚的哈里斯堡。

4. In April 1986, the No.4 reactor...to the environment.

句中 Chernobyl Power Station 是前苏联的切尔诺贝利核电站,于 1986 年 4 月发生严重的核反应堆事故并引起大火,使大量放射性物质外漏,造成大面积的环境污染; Pripyat river 是普里普塔河; Kiev 是基辅。

5. However, few, if any, of the...reactor design.

句中 few, if any 表示即使有也是很少一点。全句译为:然而只有下面很少一些结论可能会对西方的反应堆设计有影响。

Unit 17 Westinghouse AP1000

Advanced Passive Plant

17.1 Introduction

The Westinghouse AP1000 Program is aimed at making available a nuclear power plant that is economical in the US deregulated electrical power industry in the near-term. The AP1000 is a two-loop 1000MWe pressurizer water reactor (PWR). It is an uprated version of the AP600. Passive safety systems are used to provide significant and measurable improvements in plant simplification, safety, reliability, investment protection and plant costs. The AP1000 uses proven technology, which builds on over 35 years of operating PWR experience. The AP1000 received Final Design Approval from the United States Nuclear Regulatory Commission in September 2004; the AP1000 has also received Design Certification by the USNRC in December 2005. The AP1000 and its predecessor AP600 are the only nuclear reactor designs using passive safety technology licensed anywhere in the world. The safety performance of AP1000 has been verified by extensive testing, safety analysis and probabilistic safety assessment. AP1000 safety margins are large and the potential for accident scenarios that could jeopardize public safety is extremely low.

Simplicity is a key technical concept behind the AP1000, which makes the AP1000 easier and less expensive to build, operate, and maintain. Simplification also provides a hedge against regulatory driven operations and maintenance costs by eliminating equipment subject to regulation¹. The AP1000's greatly simplified design complies with NRC regulatory and safety requirements and the EPRI advanced light water reactor (ALWR) utility requirements document.

Plans are being developed for implementation of the AP1000 plant. Key factors in this planning are the economics of AP1000 in the de-regulated US electricity market, and the associated business model for licensing, constructing and operating these new plants.

In December 1999, the Nuclear Regulatory Commission granted Design Certification to the AP600. The AP600 plant meets the US utility requirements (ALWR, Volume III) including the cost goals. Westinghouse recognized that the current estimate of $4.1 \sim 4.6 \text{ ¢}/(\text{kW} \cdot \text{h})$ for the AP600 is not competitive in the US market. It, therefore, embarked on the development of the AP1000, which applies economies of scale to passive safety plants to reduce the cost per $\text{kW} \cdot \text{h}$ to about $3.0 \sim 3.5 \text{ ¢}/(\text{kW} \cdot \text{h})$.

Simplicity is a key technical concept behind the AP1000 plant. It makes the AP1000 easier and less expensive to build, operate, and maintain.

The AP1000's greatly simplified design complies with all of the NRC regulatory and

safety requirements and EPRI advanced light water reactor (ALWR) utility requirements document (ALWR Utility Requirements Document, 1999). A minimum number of changes have been made to the AP600 to realize a significant increase in power in AP1000. The large safety margins of the AP600 have been maintained. The AP1000 plant footprint and auxiliary systems remain unchanged from AP600. The AP1000 design continues to use proven components, and the inherent safety and simplicity of the AP600 has been retained.

17.2 Major Equipment Description

AP600 and AP1000 are based on proven Westinghouse PWR technology that has been successfully operated in hundreds of reactor-years of operation. The reactor coolant system (RCS) consists of two heat transfer circuits, with each circuit containing one steam generator, two reactor coolant pumps, and a single hot leg and two cold legs for circulating coolant between the reactor and the steam generators. The number of welds in the main loop piping system is reduced by about 67%. The system also includes a pressurizer, interconnecting piping, and the valves, and instrumentation necessary for operational control and the actuation of safeguards. The RCS arrangement is shown in Fig.17.1 and selected plant parameters are shown in Table 17.1.

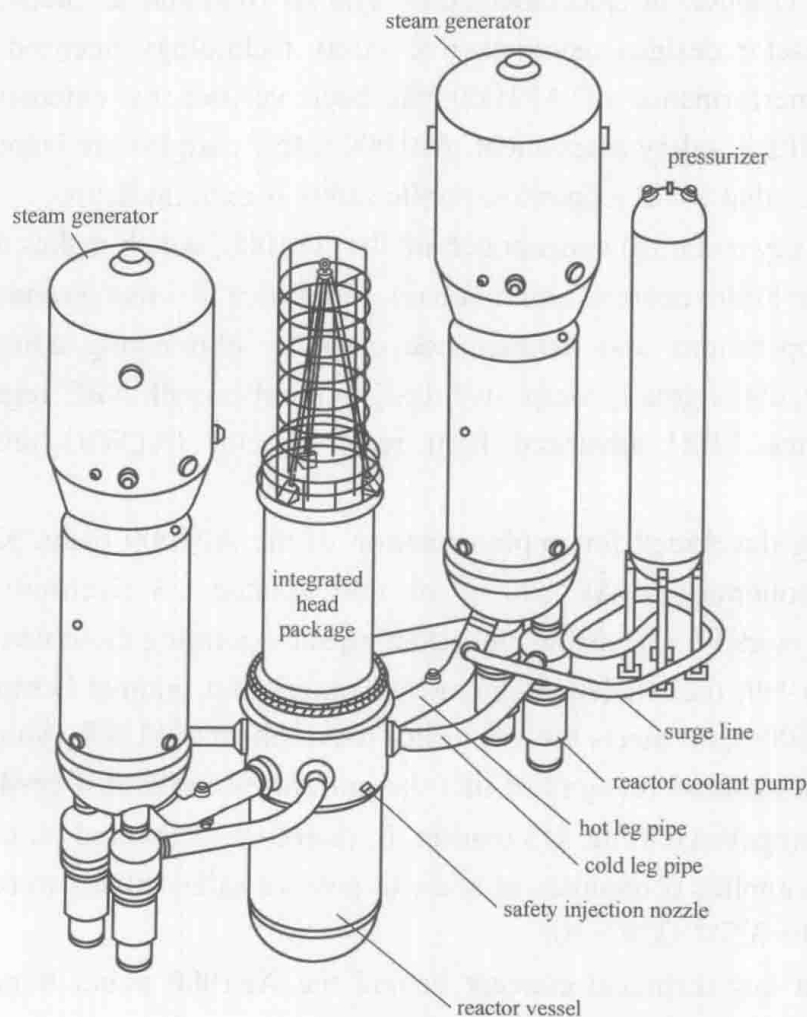


Fig.17.1 AP1000 reactor coolant system

Table 17.1 Select AP1000 RCS parameters

Parameter	Doel4/Tihange3	AP600	AP1000
Net electric output, MWe	985	610	1117
Reactor power, MWt	2988	1933	3400
Hot leg temperature, C (F)	330 (626)	316 (600)	321 (610)
Number of fuel assemblies	157	145	157
Type of fuel assembly	17×17	17×17	17×17
Active fuel length, m (ft)	4.3 (14)	3.7 (12)	4.3 (14)
Linear heat rating, kW/ft	5.02	4.1	5.71
Control rods/gray rods	52/0	45/16	53/16
R/V I.D., cm (in.)	399 (157)	399 (157)	399 (157)
Vessel flow (thermal) 10 ³ m ³ /h (10 ³ gpm)	67.1 (295)	44.1 (194)	68.1 (300)
Steam generator surface area, m ² (ft ²)	6320 (68 000)	6970 (75 000)	11600 (125 000)
Pressurizer volume, m ³ (ft ³)	39.6 (1400)	45.3 (1600)	59.5 (2100)

NSSS equipment is located in the reactor containment. All safety-related equipment is located in containment or in the auxiliary building. These two buildings are on common, seismically qualified basement, greatly reducing the plant's seismic footprint². All major components of both AP600 and AP1000 have been proven in operating reactors under similar flow, temperature, and pressure conditions, except for the AP1000 reactor coolant pump. It is a modest extension of proven pump designs.

Nomenclature

ADS: automatic depressurization system

ALWR: united states advanced light water reactor

CMF: core melt frequency

CMT: core makeup tank

COL: combined operating license

EPRI: electric power research institute

ESP: early site permit

I&C: instrumentation and control

IRWST: in-containment refueling water storage tank

LRF: large activity release frequency

LOCA: loss of coolant accident

PCS: passive containment cooling system

PRHR HX: passive residual heat removal heat exchanger

PRA: probabilistic risk assessment

SGTR: steam generator tube rupture accident

USNRC: United States Regulatory Commission

17.2.1 Reactor design

Although different from each other, the core, reactor vessel, and internals of both the AP600 and AP1000 are essentially those of conventional Westinghouse PWRs. For both, the reactor vessel is the same as that for a standard Westinghouse three-loop plant, with nozzles adjusted to accommodate the AP600/AP1000's two loops. The internals are also standard, with minor modifications.

Several important enhancements, all based on existing technology, have been used to improve the performance characteristics of the design. For example, there are fuel performance improvements, such as Zircaloy grids, removable top nozzles, and longer burnup features. This optimized fuel is currently used in approximately 120 operating plants worldwide. Both plants use a standard 17×17 fuel assembly. AP600 has a 145 assembly low power density core, while AP1000 has a 157 assembly higher power density core; there are Westinghouse PWRs successfully operating with higher power densities. Compared to Table 17.1.

The AP600 has a 3.66m (12ft) long core, AP1000 has a 4.27m (14ft) core. This makes the AP1000 core very similar to that in Doel 3 and Tihange 4. Both AP600 and AP1000 have more than 15% margin to the departure from nucleate boiling (DNB) limit for non loss-of-flow accidents.

A core shroud similar to Waterford 3 is employed; this proven design eliminates bolts from the barrel/baffle region. In addition, movable bottom mounted in-core instrumentation has been replaced by fixed top mounted instrumentation. Inconel 600 is not used in the reactor vessel welds.

17.2.2 Steam generators

Two model Delta-75 steam generators are used in AP600. Two model Delta-125 steam generators are used in AP1000. Although larger, they can still be used within the AP600 containment diameter of 39.6 m (130ft). Both steam generator models are based on standard Westinghouse Model-F technology. There are many 75 Model F-type units in commercial operation, with the highest level of reliability achieved by any steam generator worldwide. This reliability record is due to such enhancements as full-depth hydraulic expansion of the tubes in the tube sheets; stainless steel broached tube support plates; thermally treated corrosion-resistant Inconel 690 (In-690) tubing; upgraded anti-vibration bars to reduce wear; upgraded primary and secondary moisture separators; and a triangular tube pitch³. Two steam generators that are very similar to the Delta-125 model were installed at the Arkansas station in the US. The steam generators at Palo Verde are larger than the AP1000 steam generators.

17.2.3 Reactor coolant pumps

Both plants use canned motor pumps to circulate primary reactor coolant throughout the reactor core, piping, and steam generators. Canned motor reactor coolants pumps have been very successfully employed in naval reactors with over 1500 pumps in service. Two pumps are mounted directly in the channel head of each steam generator.

The AP1000 reactor coolant pump motors are rated on the less dense hot water at

operating temperature in lieu of the more dense ambient temperature water for the AP600. This provides the required increase in reactor coolant flow with only a small increase in the physical size of the canned motor. A variable speed controller is used in AP1000 for cold operation to compensate for the higher water density. At power the variable speed controller is disconnected and the pumps operate at constant speed, like AP600. Elimination of the pump shaft seals greatly simplifies the auxiliary fluid systems that support a canned motor pump, eliminates planned maintenance, eliminates possible accidents involving seal failures and eliminates the use of oil (and the potential for fires). They also require no planned maintenance.

The integration of the pump suction into the bottom of the steam generator channel head eliminates the crossover leg of coolant loop piping; reduces the loop pressure drop; simplifies the foundation and support system for the steam generator, pumps, and piping; and eliminates the potential for uncovering the core during a small LOCA.

17.2.4 Pressurizer

The AP600 pressurizer is essentially the Westinghouse design used in approximately 70 operating plants worldwide. The AP1000 pressurizer is larger with a volume of 59.5 m^3 (2100 ft^3). This same size pressurizer is used in the South Texas plants. Since the AP1000 reactor vessel is only slightly longer than the AP600 vessel and the primary loop piping sizes are the same; the AP1000 pressurizer provides similar enhanced margins and operating bands as the AP600 pressurizer.

17.2.5 Containment vessel

Both plants utilize a 39.6m (130 ft) diameter freestanding containment vessel. The AP1000 containment cylindrical section is 7.77m (25.5 ft) higher than provide additional free volume. The AP 1000 containment design pressure has been increased from 3.10bar (45 psig) to 4.07bar (59 psig); as a result, the AP1000 post accident pressure margins are slightly larger than those of the AP600.

17.3 Safety Through Simplicity

The safety systems for both AP600 and AP1000 include passive safety injection, passive residual heat removal, and passive containment cooling. All these passive systems meet the NRC single-failure criteria and addresses. Three Mile Island lessons learned, unresolved safety issues, and generic safety issues. These passive systems have been proven through extensive testing and computer code analysis at two different power levels (AP600 and AP1000).

Passive systems and the use of experience-based components do more than increase safety, enhance public acceptance of nuclear power, and ease licensing—they also simplify over-all plant systems, equipment, and operation and maintenance⁴. The simplification of plant systems, combined with large plant operating margins, greatly reduces the actions required by the operator in the unlikely event of an accident. Passive systems use only natural forces such as gravity, natural circulation, and compressed gas, simple physical principles we rely on every

day. There are no pumps, fans, diesels, chillers, or other rotating machinery required for the safety systems. This eliminates the need for safety-related AC power sources. A few simple valves align the passive safety systems when they are automatically actuated. In most cases, these valves are “fail safe”. They require power to stay in their normal, closed position. Loss of power causes them to open into their safety alignment. In all cases, their movement is made using stored energy from springs, compressed gas or batteries.

Simple changes in the safety-related systems from AP600 to AP1000 allow accommodation of the higher plant power without sacrificing design and safety margins. Since there are no safety-related pumps, increased flow was achieved by increasing pipe size. Additional water volumes were achieved by increasing tank sizes. These increases were made while keeping the plant foot-print unchanged. This ensures that the designs of other systems are not affected by layout changes. Note that detail design of a significant portion of AP600 is complete. Enforcing a rigorous “no unnecessary change policy” makes that portion of the detail design of AP1000 also complete.

17.4 Passive Safety Systems

Passive systems provide plant safety and protect capital investment. They establish and maintain core cooling and containment integrity indefinitely, with no operator or AC power support requirements. The passive systems meet the single-failure criteria and probabilistic risk assessments (PRA) used to verify reliability. The passive safety systems are significantly simpler than typical PWR safety systems. They contain significantly fewer components, reducing required tests, inspections, and maintenance. The passive safety systems have one-third the number of remote valves as typical active safety systems, and they contain no pump. Equally important, passive safety systems do not require a radical departure in the design of the rest of the plant, core, RCS, or containment. The passive safety systems do not require the large network of active safety support systems needed in typical nuclear plants. These include AC power, HVAC, cooling water, and the associated seismic buildings to house these components.

This simplification applies to the emergency diesel generators and their network of support systems, air start, fuel storage tanks and transfer pumps, and the air intake/exhaust system. These support systems no longer must be safety class, and they are either simplified or eliminated. For example, the essential service water system and its associated safety cooling towers are replaced with a non-safety-related service water cooling system.

Non-safety-related support systems and passive safety systems are integrated into the plant design. Licensing safety criteria are satisfied with a greatly simplified plant.

The passive safety systems have been sized to provide increased safety margins, especially for more probable events. Table 17.2 illustrates the improved margins. Both AP600 and AP1000 have the same passive safety-related systems and they include:

Table 17.2 AP1000 safety margins

	typical plant	AP600	AP1000
loss flow margin to DNBR limit	1%~5%	about 16%	16%
feedline break °C(°F) subcooling margin	>0(>0)	about 94(about 170)	about 78(about 140)
SG tube rupture	operator actions required in 10 min	operator actions NOT required	operator actions NOT required
small LOCA	3" LOCA	<8" LOCA	<8" LOCA
PCT °C(°F)	about 815 (about 1500)	NO core uncover	NO core uncover
large LOCA PCT °C(°F) with uncertainty	1093~1204 (2000~2200)	913 (1676)	<870* (<1600)
ATWS, Pres bar(psig)	221(3200)	221 (3200)	193 (2800)
(% core life)	90%	90%	100%

*Based on ASTRUM analysis of AP1000 which is more realistic than the "bounding" large LOCA analysis performed for AP600.

17.4.1 Emergency core cooling system

The passive core cooling system (PXS), shown in Fig.17.2, protects the plant against RCS leaks and ruptures of various sizes and locations. The PXS provides core residual heat removal, safety injection, and depressurization. Safety analyses (using NRC-approved computer codes) demonstrate the effectiveness of the PXS in protecting the core following

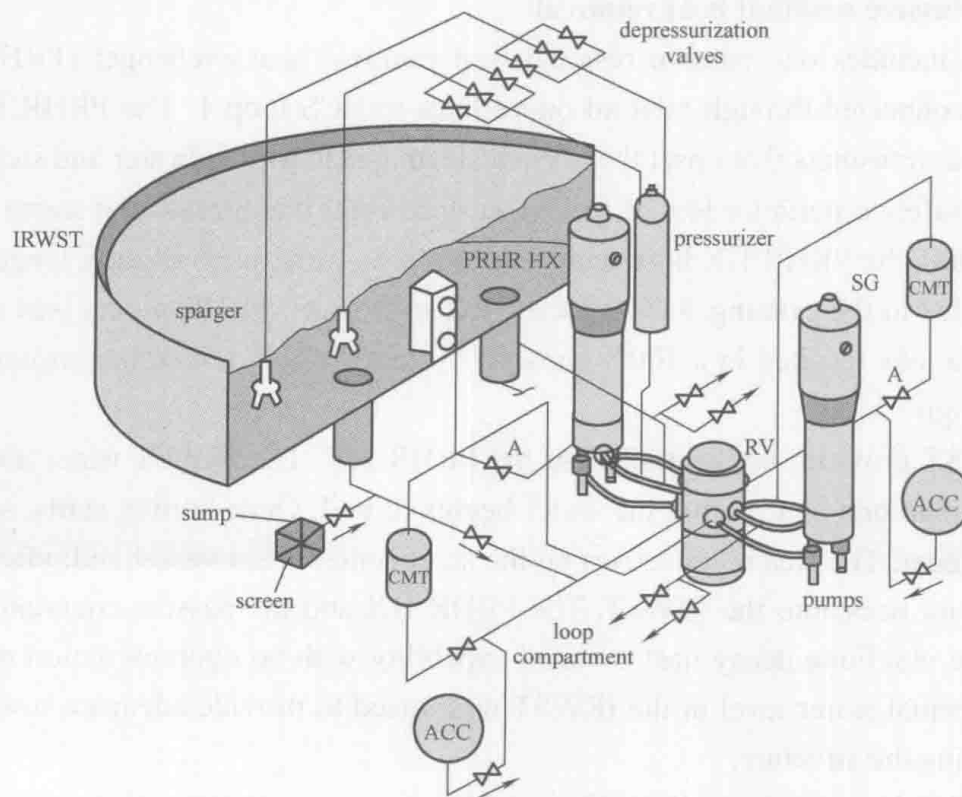


Fig.17.2 AP1000 RCS and passive core cooling system

various RCS break events. Even for breaks as severe as the 20.0 cm (8 in.) vessel injection lines, there is no core uncover for either AP600 or AP1000. Following a double-ended rupture of a main reactor coolant pipe, the PXS cools the reactor with ample margin to the peak clad temperature limit.

17.4.2 Safety injection and depressurization

The PXS uses three sources of water to maintain core cooling through safety injection. These injection sources include the core makeup tanks (CMTs), the accumulator, and the in-containment refueling water storage tank (IRWST). These injection sources are directly connected to two nozzles on the reactor vessel so that no injection flow can be spilled in case of larger breaks.

Long-term injection water is provided by gravity from the IRWST, which is located in the containment just above the RCS loops. Normally, the IRWST is isolated from the RCS by squib valves and check valves. This tank is designed for atmospheric pressure. The RCS must be depressurized before injection can occur. The RCS automatically controlled to reduce pressure to about 0.83bar (12psig), at which point the head of water in the IRWST overcomes the low RCS pressure and the pressure loss in the injection lines. The PXS provides depressurization using the four stages of the automatic depressurization system (ADS) to permit a relatively slow, controlled RCS pressure reduction.

To maintain similar margins for accidents requiring safety injection, a few lines in the PXS were made larger for AP1000. In addition, the CMTs were enlarged to provide adequate margin without requiring redesign of adjacent piping and structure.

17.4.3 Passive residual heat removal

The PXS includes one passive residual heat removal heat exchanger (PRHR HX). The PRHR HX is connected through inlet and outlet lines to RCS loop 1. The PRHR HX protects the plant against transients that upset the normal steam generator feedwater and steam systems. It satisfies the safety criteria for loss of feedwater, feedwater line breaks, and steam line breaks.

For AP1000, the PRHR HX horizontal tube portions were made slightly longer and a few tubes were added to the existing AP600 PRHR HX tube sheet. PRHR piping was made larger. These modifications resulted in a 100% capacity system without affecting surrounding piping and layout design.

The IRWST provides the heat sink for the PRHR HX. The IRWST water absorbs decay heat for more than one hour before the water begins to boil. Once boiling starts, steam passes to the containment. The steam condenses on the steel containment vessel and, after collection, drains by gravity back into the IRWST. The PRHR HX and the passive containment cooling system provide indefinite decay heat removal capability with no operator action required. For AP1000 the normal water level in the IRWST was raised to provide adequate water inventory without changing the structure.

17.4.4 Passive containment cooling system

The passive containment cooling system (PCS) provides the safety-related ultimate heat

sink for the plant. The PCS cools the containment following an accident so that the design pressure is not exceeded and the pressure is rapidly reduced, dropping to $\sim 40\%$ of the design pressure in less than 5h. The steel containment vessel provides the heat transfer surface that removes heat from inside the containment and transfers it to the atmosphere. Heat is removed from the containment vessel by the continuous, natural circulation of air. During an accident, air cooling is supplemented by water evaporation. The water drains by gravity from a tank located on top of the containment shield building.

The containment for AP1000 has the same diameter as that for AP600. The height has been increased to provide additional free volume. This additional free volume, with a change of material for the vessel shell, provides increased margin to vessel design pressure from accident pressure over AP600. Analysis shows that during severe accidents the AP1000 containment is likely to remain intact and not to be bypassed. As a result, the plants have a significantly reduced frequency of release of large amounts of radioactivity following core damage in an accident. The PCS cooling capability is very reliable with its three way redundant (and diverse) water drain valves. In addition, even with failure of water drain, air-only cooling is capable of maintaining the containment below the predicted failure pressure. Containment isolation has been improved by eliminating about 50% of the penetrations and all of the ECCS lines that circulate highly radioactive water outside containment after an accident. Note that all of the penetrations are routed directly from the containment to the auxiliary building such that any leakage would enter the auxiliary building⁵. The potential for containment bypass sequences has been significantly reduced including those involving steam generator tube ruptures (SGTR). This enhanced containment performance supports the technical basis for simplification of offsite emergency planning.

17.5 Probabilistic Risk Assessment

PRA has been used interactively as a part of the design process since the beginning of the AP600 program in 1985. Seven major PRA quantifications have been performed on the AP600. Two major quantifications have been performed on the AP1000. During each of these quantifications, the PRA results were reviewed for potential modifications. Many design and operation changes have been made based on these PRA insights, especially during the earlier AP600 quantifications.

As a result, the AP1000 PRA (AP1000 Probabilistic Risk Assessment, 2004) results show very low core melt and large release frequencies that are significantly below those of operating plants and well below the NRC safety goals. Table 17.3 shows the core melt frequency (CMF) and large release frequency (LRF) per reactor year. The AP1000 frequencies include shutdown events and external events. Shutdown events have typically not been quantified for operating plants.

Table 17.3

CMF and LRF per reactor year

	Operating	NRC	AP1000
CMF	$\sim 1 \text{ E-4}$	1 E-4	4.2 E-7/year
LRF	$\sim 1 \text{ E-5}$	1 E-6	3.7 E-8/year

A major safety advantage of passive plants versus conventional PWRs is that long-term accident mitigation is maintained without operator action and without reliance on offsite or onsite AC power sources. The passive safety systems provide long-term core cooling and decay heat removal without the need for operator actions or without reliance on active safety-related systems. For limiting design basis accidents, the core coolant inventory in the containment for recirculation cooling and boration of the core is sufficient to last for at least 30 days, even if inventory is lost at the design basis containment leak rate. PRA sensitivity studies illustrate this improvement. Table 17.4 shows the CMF for at-power internal events with and without operator action:

Table 17.4

CMF for at-power internal events with and without operator action

	Operating	AP1000
CMF with operator action	$\sim 4 \text{ E-5}$	2.4 E-7/year
Without operation action	$\sim 2 \text{ E-3}$	1.8 E-5/year

Severe accident phenomenon has been addressed with AP1000 design features. The highly redundant and diverse ADS prevent high pressure core melt sequences which can challenge the containment through direct containment heating and steam explosions. Core concrete interactions are prevented by in-vessel retention of core melt debris. In-vessel retention has been shown to be applicable to the AP1000 through testing and analysis that was reviewed by the US NRC (Scobel, 2005). Hydrogen ignitors and passive autocatalytic recombiners prevent hydrogen explosions. As a result, the AP1000 can maintain the integrity of the containment following a core melt for an indefinite time.

17.6 Simplification

The AP1000 incorporates significant simplifications through the use of passive safety-related systems and other advanced technologies. The use of passive safety-related systems eliminate many safety-related components including pumps, valves, pipe and their associated buildings as shown in Fig.17.3. In addition, many other safety-related components have been reduced to non-safety. These changes result in great simplifications in procurement, construction, startup, and operation including in-service inspection/testing and maintenance.

Other advanced technologies have also been used to gain additional simplifications, including the use of digital instrumentation systems, canned motor reactor coolant pumps and ion exchange based radioactive waste processing.

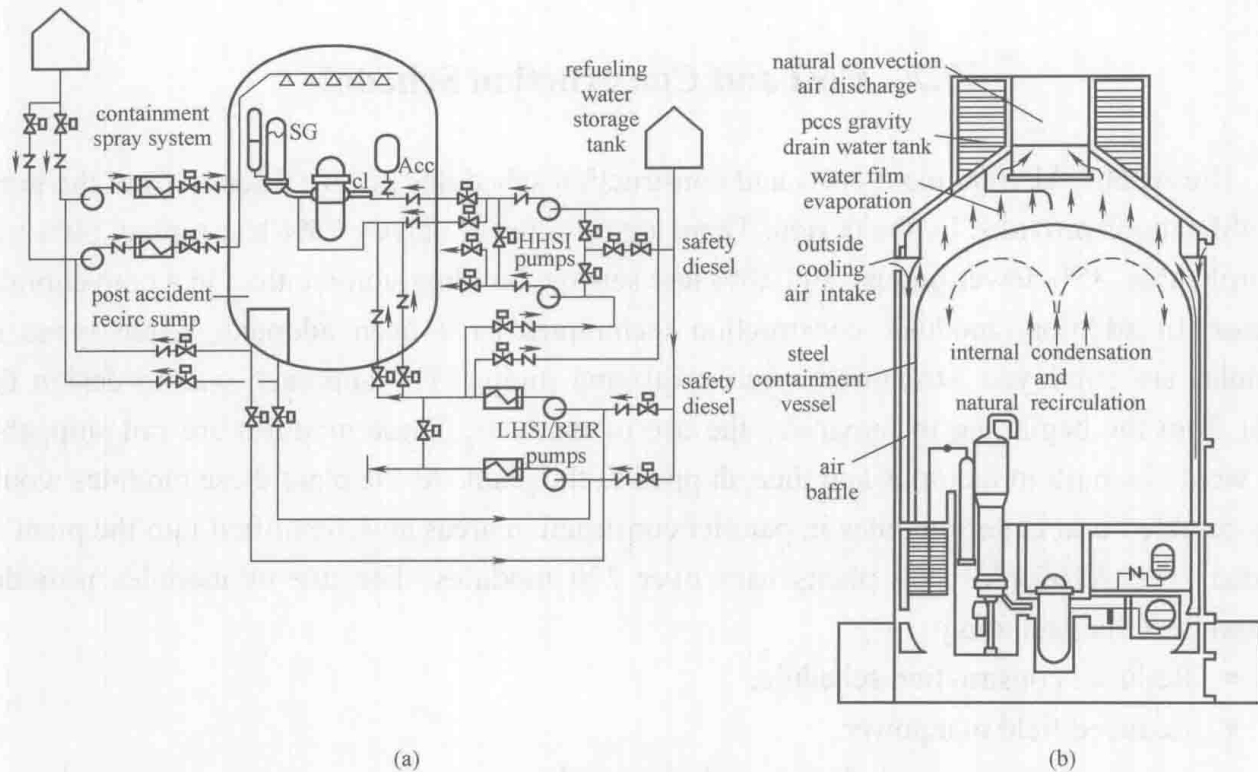
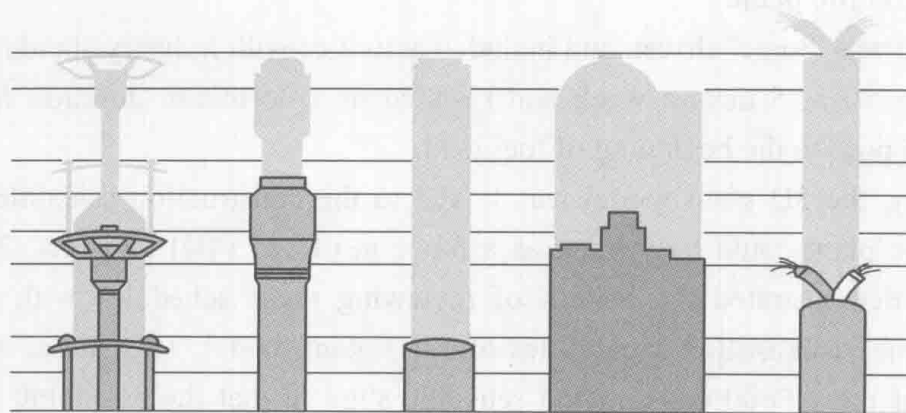


Fig.17.3 AP1000 passive safety system simplifications

(a) current PWR; (b) AP1000

Fig.17.4 shows the quantitative simplifications that have been achieved by the AP1000 as compared to currently operating PWRs. The savings are even greater when compared with advanced evolutionary PWRs using safety systems with greater redundancy (four-trains) and diversity (four large safety diesel generators and two medium station blackout diesel generators).



reduced number of components:

	1000 MW reference	AP 1000	reduction
safety valves	2844	1400	51%
pumps	280	194	34%
safety piping	11×10^4 feet	1.9×10^4 feet	83%
cable	9.1mil feet	1.2mil feet	87%
seismic building volume	12.7mil ft ³	5.6mil ft ³	56%

Fig.17.4 AP1000 quantitative simplifications

17.7 Cost and Construction Schedule

The AP600/AP1000 plant costs and construction schedules benefit directly from the great simplifications provided by the design. There are 60% fewer valves, 75% less piping, 80% less control cable, 35% fewer pumps, and 50% less seismic building volume than in a conventional reactor. In addition, modular construction techniques have been adopted. Three types of modules are employed: structural, mechanical, and piping. The approach was to design the plant from the beginning to maximize the use of modules. These modules are rail shippable and would be built in factories and then shipped to the plant. At the plant these modules would be assembled into larger modules in parallel construction areas and then lifted into the plant as needed. The AP600/AP1000 plants uses over 270 modules. The use of modules provides several benefits, including:

- Reduced construction schedule.
- Reduced field manpower.
- Increased factory work (better quality control).
- Reduction in site congestion.

Westinghouse has developed an extensive, detailed 3D computer model of the AP600 nuclear reactor plant. This model was developed over 8 years, using input from a number of design participants from a variety of countries. Westinghouse also led the effort by Morrison-Knudsen (now a part of the Washington Group) to develop a construction schedule using Primavera for AP600 over the same period as the 3D-model development. MK used its construction experience and a detailed knowledge of the plant to create a detailed schedule for construction of the entire plant.

This schedule was “logic” driven and included activities with industry standard durations. It is based upon a 50 h, 5 days a week and resulted in a 36-month duration from start of basement concrete pour to the beginning of fuel load.

More recently, the 3D plant model was linked to the construction schedule so that the construction of the plant could be viewed as a function of time(4D) (Winters, 2001). These evaluations have demonstrated the benefit of reviewing these schedules with construction specialists using the visualization capabilities of a 4D-plant model. This initial evaluation of the first portion of the AP600 construction schedule showed that the 36-month construction schedule could be reduced by at least 4 months. This study also increased the confidence of potential investors concerning the viability of the schedule and the plant’s ultimate cost.

The AP1000 uses power producing components that have been used before. As a result, there is a track record for the reliability of these components and the AP1000 availability can be predicted based on experience instead of conjecture. With proven components, design simplifications, margins, and lessons learned, the AP600/AP1000 plants are expected to exceed 93% availability since current operating plants are exceeding this value.

The total operating costs (including capital, O&M, and fuel) for the AP600 is calculated to be $4.1 \sim 4.6 \text{¢}/(\text{kW} \cdot \text{h})$. The AP1000, with its small capital cost increase and large power increase, results in a cost of about $3.0 \sim 3.5 \text{¢}/(\text{kW} \cdot \text{h})$ for a twin unit plant.

17.8 Licensing

In June 1992, AP600 safety analysis and probabilistic risk assessment reports were submitted to the NRC. The Commission documented its acceptance of the AP600 plant in the final design approval (FDA) on September 3, 1998. In December 1999, the NRC issued the Design Certification for AP600 as Appendix C of 10CFR Part 52. This made the AP600 the only licensed passive safety system nuclear power plant in the world.

Licensing discussion on the AP1000 began with the US NRC with a pre-certification review phase in September 2001, which successfully established the applicability of the AP600 test program and AP600 safety analysis codes to the AP1000 Design Certification.

On March 28, 2002, Westinghouse submitted to US NRC an application for Final Design Approval and Design Certification of the AP1000 standard plant (AP1000 Design Control Document, 2004). The application includes the AP1000 Design Control Document Standard Safety Analysis Report and Inspections, Tests, Analysis and Acceptance Criteria (ITAAC) and Probabilistic Risk Assessment (AP1000 DCD, AP1000 PRA). The NRC formally docketed the application on June 25, 2002 signifying its acceptance as a complete safety case. Because of the few design changes from AP600, approximately 80% of the AP600 Standard Safety Analysis Report remains unchanged for AP1000.

The NRC reviewed the DCD and PRA documents. They issued ~ 700 requests for additional information (RAI) by September 30, 2002. Westinghouse provided responses to the NRC requests by December 2, 2002. When the NRC issued the draft safety evaluation report in June 2003, there were ~ 180 open items from the ~ 700 RAIs issued for the AP1000. These DSER open items were resolved and the NRC issued their FDA on September 13, 2004. In December 2005, the NRC issued the Design Certification for AP1000 as Appendix C of 10CFR Part 52.

As shown in the Table 17.5, Westinghouse provided very detailed information for the NRC to review and the NRC review was very extensive and complete. The NRC performed independent safety analysis and testing to confirm the AP1000 safety performance. Schulz (2005) provides additional discussion.

This milestone, together with the review and approval of the AP600, provides a high certainty for the license ability of the AP1000 in international markets. Note that the design information in the AP1000 Design Control Document (DCD) is final and will not be re-reviewed by the NRC in the future. The AP1000 DCD includes a small number of items that require additional work to address. These items are clearly defined. They were not addressed during the AP1000 FDA/DC because they require information from the owner/operator (training, procedures), on as-procured equipment, or on fast changing technologies (I&C).

Table 17.5 The Information for the NRC to review of AP1000 and AP600

	US NRC review of AP1000 and AP600	
	AP1000	AP600
Level of review	Final design approval	Final design approval
Material reviewed ^①		
SSAR	~6500pg ^②	~6500pg ^②
PRA	~4500pg	~4500pg
Dates of review	2 1/2years(3/02—9/04)	6+years(6/92—9/98)
US NRC effort	31man-year	110man-year

① This only considers the information submitted initially for review. This does not include reports and answers to questions.

② The SSARs are consistent with US NRC Regulatory Guide 1.70 plus a Chapter 18 on human factors and a Chapter19, which summarizes the PRA.

Westinghouse and the utilities interested in the AP1000 have begun to address these items in the AP1000 DCD using a standard design approach. The approach is to submit technical reports to the USNRC during the 2006 (March through December) and get their review and concurrence. This approach reduces the resources and time required to bring these items to closure.

17.9 AP1000 Deployment

In 2002, the US Department of Energy (DOE) launched a major initiative, called Nuclear power 2010, to work with the nuclear industry on projects that would lead to the construction of new nuclear plants in the US by 2010. During the first year, DOE awarded three contracts to US utilities to cost-share in applications to the US Nuclear Regulatory Commission (NRC) for Early Site Permits (ESPs). Under NRC's 10CFR52 regulation, these utilities will be able to pre-license plant sites in advance of deciding on the type of nuclear plant to be built or when it would be built. It is expected that these US power companies will receive ESPs by 2005.

At the end of 2003, DOE issued a solicitation to cost-share with utilities and consortiums for projects that would ①complete Design Certification of standardized nuclear plants under 10CFR52; ②prepare and submit applications for Combined Construction and Operating Licenses (COLs) for specific nuclear plant projects under 10CFR52; ③complete the detailed design of the standard plants; and ④provide the utilities with the engineering and financial information on the designs necessary for them to make decisions on whether to actually proceed to construction of the plants.

Westinghouse joined with nice utilities and another reactor supplier in a consortium called NuStart Energy, to respond to DOE's solicitation. The nine utilities involved in the project are Exelon, Entergy, Constellation, Duke, Florida Power and Light, Progress, Southern, EDF of

North America, and Tennessee Valley Authority. The NuStart proposal has been accepted by DOE. Expected funding is about 100 million USD over next 4 years. Westinghouse's AP1000 design and the other supplier's design were chosen by the utilities to be a part of the NuStart project because they believed that the designs' simplified passive safety features are important to reducing the construction and operating costs of new nuclear plants—while assuring the highest levels of safety achievable with proven technologies. NuStart has selected the Bellefonte site for application of the AP1000.

Other utilities have announced their intention to individually submit COL applications based on the AP1000. These utilities include Duke Power, Southern Company, Progress Energy and South Carolina Electric and Gas. The number of the plants involved with these COL applications is about 11 AP1000's. Interest is also being expressed by other US power companies.

Westinghouse is preparing additional AP1000 design information to support these applications. In addition, Westinghouse will prepare information needed by these utilities to support making commercial decisions to proceed with these applications. By 2007, the AP1000 COL application will be complete and ready to submit to NRC. It is expected that the NRC will be able to issue The COL before 2010.

In the Peoples Republic of China bid invitations have been extended for four advanced, Generation III, PWRs. These plants would be built at two sites, Sanmen and Yangjiang. Westinghouse will offer the AP1000 in response to this bid invitation. It is expected that there may be as many as 25 additional advanced generation III PWRs built in China by 2020.

17.10 Conclusions

The AP1000 is a simple plant, which is licensed, mature design, using proven components in an innovative and elegant approach to safety. The simplifications incorporated into the AP1000 provide real and quantified benefits in construction, operation and safety. The AP1000 has a very attractive operating cost per kW • h in the range of electricity prices today [3.0~3.5¢/(kW • h)]. The US NRC completed its review of the AP1000 and issued its FDA in September 2004 and DC in December 2005. The Westinghouse AP1000 represents a nuclear power plant that is economical in the US deregulated electrical power industry that is ready for deployment now.

Self-test

1. The AP1000 uses____technology, Which builds on over 35 years of operating PWR experience.
A. new B. proven C. passive D. design
2. ____makes the AP1000 easier and less expensive to build, operate, and maintain.

- A. Passive safety B. new technology
C. New concept D. Simplicity
3. Movable bottom mounted in-core instrumentation has been ____ by fixed top mounted instrumentation.
A. moved B. removed C. placed D. replaced
4. A variable speed controller is used in AP1000 for cold operation ____ the higher water density.
A. to compensate for B. compensate
C. to compensate D. compensate for
5. Enforcing a rigorous “no unnecessary change policy” makes that portion of the detail design of AP1000 also complete. In this sentence, “no unnecessary change” means ____.
A. no change B. a little change
C. all change is necessary D. little change
6. Following a double-ended rupture of a main reactor coolant pipe, the PXS cools the reactor with ____ to the peak clad temperature limit.
A. water B. margin C. ample margin D. enough water
7. The RCS must be ____ before injection can occur.
A. depressurized B. pressurized
C. pumped D. boiled
8. For AP1000 the normal water level in the IRWST was raised to provide adequate ____ without change the structure.
A. water B. water inventory
C. tank D. inventory
9. The steel containment vessel provides the heat transfer surface that removes heat from inside the containment and transfers it to ____.
A. water B. air C. the outside D. the atmosphere
10. For limiting design basis accidents, the core coolant inventory in the containment for recirculation cooling and boration of the core is sufficient ____ at least 30 days.
A. to last for B. for C. duration D. for duration
11. This approach reduces the resources and time required ____ these items to closure.
A. for B. to make C. to bring D. to bring for
12. Westinghouse ____ nine utilities and another reactor supplier in a consortium called Nustart Energy.
A. joined B. joined with C. with D. together with

关键字解释

ADS: Automatic Depressurization System.

自动减压系统。

ALWR: United States Advanced Light Water Reactor.

美国先进轻水反应堆。

Ambient temperature: The term used to denote temperature of surrounding air.

环境温度：这一技术术语表示周围空气的温度。

CMF: Core Melt Frequency.

堆芯熔化概率。

CMT: Core Makeup Tank.

堆芯补水箱。

COL: Combined Operating License.

联合运行执照。

EPRI: Electric Power Research Institute.

电力研究院。

ESP: Early Site Permit.

前期场地许可。

I&C: Instrumentation and Control.

仪表和控制。

Inherent safety: The Safety present naturally as a system or an equipment.

固有安全性：一个系统或设备自然存在的安全性。

IRWST: In-containment Refueling Water Storage Tank.

安全壳内换料水储存箱。

Licensing: An official paper or card show that permission has been given to do some thing.

许可证：一个正式的文件或证书，表明允许做某种事情。

Moisture separator: A trap in a flow system containing vapor and droplets, which can remove droplets to produce dry steam.

汽水分离器：在含有蒸汽和水滴的流动系统中的一种收集器，这种收集器能分离出水滴而产生干蒸汽。

Passive Safety: Keeping the safety without an effective source or outside power.

非能动安全：指不需要有效的能源和外界动力而维持安全。

PCS: Passive Containment Cooling System.

非能动安全壳冷却系统。

Deployment: To spread out or arrange for effective action.

推广：指推广或安排一些事情的有效实施。

PRHR HX: Passive Residual Heat Removal Heat Exchanger.

非能动余热排除系统热交换器。

SGTRA: Steam Generator Tube Rupture Accident.

蒸发器管破裂事故。

USNRC: United States Nuclear Regulatory Commission.

美国核管理委员会。

词 汇

- actuation [æktju'eɪʃən] *n.* 动作
address [ə'dres] *vt.* 交, 寻址, 委托
adequate ['ædikwɪt] *a.* 足够的
align [ə'laɪn] *v.* 与……一起(合作)
ambient ['æmbiənt] *a.* 周围的, 环境的
ample ['æmpl] *a.* 足够的
assessment [ə'sesmənt] *n.* 估计, 评定
approval [ə'pru:vəl] *n.* 认可, 批准
bid [bɪd] *v. & n.* 出价, 投标, 表示
broach [brəʊtʃ] *v.* 钻孔
check valve 止回阀
chiller [tʃɪlə] *n.* 冷却装置
concurrence [kən'kʌrəns] *n.* 同意
congestion [kən'dʒestʃən] *n.* 拥挤
conjecture [kən'dʒektʃə] *n. & v.* 推测
consortium [kən'sɔ:tʃəm] *n.* 合伙, 联合
conventional [kən'venʃənl] *a.* 传统的
crossover ['krɒs,əʊvə] *n.* 跨越
debris ['debri:] *n.* 碎片
deregulate [di:'regjuleɪt] *vt.* 撤销对……的管理规定
diverse [daɪ'vɜ:s] *a.* 多样的
docket ['dɒkɪt] *n. & v.* 概要, 附加提要
embark [ɪm'bɑ:k] *v.* 开始, 投资
evolutionary [i:və'lu:ʃənəri] *a.* 进化
generic [dʒi'nerɪk] *a.* 通用的
hedge [hedʒ] *n.* 障碍(物)
ignitor [ɪg'naitə] *n.* 点火器
Inconel [ɪn'kənəl] *n.* 因科镍合金
indefinite [ɪn'defɪnɪt] *a.* 不明确的
interactively *ad.* 相互作用地
inherent [ɪn'hɪərənt] *a.* 固有的
innovative [ɪ'nəuveɪtɪv] *a.* 革新的
intact [ɪntækt] *a.* 未受损的, 无扰动的
integrity [ɪn'tegɪrɪti] *n.* 完整, 一体化
investor [ɪn'vestə] *n.* 投资者
issue ['ɪʃju:] *v.* 颁发

- jeopardize ['dʒepədaɪz] *vt.* 危及, 使受危害
 lieu ['liu:] *n.* 场所
 logic ['lɒdʒɪk] *n.* 逻辑
 modular ['mɒdʒulə] *a.* 模块化的
 near-term 目前的
 offsite 厂外
 optimize ['ɒptimaɪz] *v.* 优选
 participant [pɑ:'tɪsɪpənt] *n.* 参与者
 passive ['pæsɪv] *a.* 被动的, 非能动的
 probabilistic [ˌprɒbəbi'lɪstɪk] *a.* 概率的
 procure [prə'kjuə] *vt.* 取得, 采购
 procurement [prə'kjuəmənt] *n.* 采购
 quantitative ['kwɒntɪtətɪv] *n.* 定量
 regulatory ['regjələtəri] *a.* 管理的, 规章(限制)人的
 reliance [rɪ'laɪəns] *n.* 依靠, 信赖
 retention [rɪ'tenʃən] *n.* 保存
 scenario [si'nɑ:riəu] *n.* 情况
 seismic ['saɪzmɪk] *a.* 地震的
 severe [si'viə] *a.* 严重的
 shippable 可装运的
 signify ['signɪfaɪ] *v.* 表示, 意味
 solicitation [sɒ,lɪsɪ'teɪʃən] *n.* 请求, 要求
 spill [spɪl] *v.* 溢出
 supplement ['sʌplɪment] *vt.* 补充
 track [træk] *n.* 轨迹, 记录
 triangular [traɪ'æŋɡjʊlə] *a.* 三角(形)的
 utility [ju:'tɪlɪti] *n.* 公用事业公司
 version ['vɜ:ʃən] *n.* 型(式), 种类
 versus ['vɜ:səs] *prep.* 与……比较
 viability [vaɪə'bɪlɪti] *n.* 生命力, 生存能力
 in lieu of 代替
 heat sink 热井

课文注释

1. Simplification also provides a hedge against regulatory driven operations and maintenance costs by eliminating equipment subject to regulation.

句中 regulatory driver operation and maintenance costs 是指由法规要求的增多而使运行和保养的费用提高, subject to regulation 是状语, 指在符合规范的情况下。全句译为: 通过

简化,在符合规范条件下取消了一些设备,这也防止了由于法规要求的增多而使运行和保养的费用提高。

2. These two buildings are on common, seismically qualified basement, greatly reducing the plant's seismic footprint.

句中 common 表示共同或公用的, basement 是指建筑的地基, footprint 这里表示占地面积。全句译为:这两个建筑是建在一个共用的满足地震要求的地基上,大大地减小了装置防地震建筑的占地。

3. This reliability record is due to such enhancements as...tube pitch.

句中 such enhancements as...表示如……这些强化; corrosion-resistant 表示抗腐蚀的; upgrade 表示升级或提高质量。全句译为:可靠性的记录是由于如下的措施而得到强化,这些措施包括:在管板内全长度上管子的液压胀管;不锈钢钻孔的管支撑板,采用热处理的抗腐蚀因科镍 690 管;升级后的防振动拉杆可以减小磨损;升级后的一次和二次汽水分离器;以及三角形布置的管间距。

4. Passive systems and the use of...and operation and maintenance.

句中 experience-based 表示经过实践检验的。全句译为:非能动安全系统及使用经过实践检验的设备不仅增强了安全性,增强了核动力的公众接受程度及使执照申请容易,它们也简化了整个装置系统、设备、运行和保养。

5. Note that all of the penetrations...the auxiliary building.

句中 penetrations 指从安全壳大厅向外穿透的管路, such that...表示是这样以致……。全句译为:注意到所有的穿透都是从安全壳直接到辅助厂房,这样致使任何泄漏将进入辅助厂房。

Unit 18 European Pressurized

Water Reactor (EPR)

18.1 Introduction

The EPR project started in 1989 when the French company Framatome and the German firm Siemens created a joint company called Nuclear Power International(NPI). NPI was established to market the existing products of the two companies for export markets and more importantly to develop the EPR design so that it could pass a common regulatory approval process for resources of two large nuclear engineering firms the new NPI could claim involvement in more than 25% of global nuclear power generation, a figure higher than any competitor.

In 1999 following several years of successful collaboration between Framatome and Siemens in NPI and the design work on EPR, the two companies decided to fully merge their nuclear businesses to form Framatome ANP. Today, Framatome ANP is a joint activity of the Areva group and Siemens, providing a range of energy services, but with a particularly strong base in nuclear power systems.

The EPR is not characterized by the adoption of the types of passive safety systems favored by Westinghouse for the AP1000¹. Rather the EPR is an active reactor design building upon experience gained with the N4 and Konvoi models. Safety enhancements include a designed-in capacity to minimize adverse consequences arising from a core melt. Debris arising from a major and unmitigated loss of coolant accident would be prevented from reaching the bottom of the reactor pressure vessel by a core catcher. The base of the reactor pressure vessel would be fitted with a melt plug and in the event of a core melt this plug would give way and allow the molten core pieces to flow down a special channel into a prepared spreading compartment slightly displaced from the reactor. It is this compartment which is designed to catch the core. Such core catching concepts are not completely new, but they form a key part of the enhanced safety credentials of the EPR. The reactor also has enhanced containment compared with its precursors, and this gives far greater control over any gaseous fission product leaks from the core. The EPR design does incorporate one passive emergency feature, which is a capacity to flood the core in the event of serious over-heating.

From the start the proposed 1750 MWe units have been designed to use mixed oxide (MOX) fuels. The original proposal was for a 1500MWe EPR fuelled with up to 100% MOX, but more recently designs have moved to a slightly higher power level operating with a maximum 50% MOX fuel contribution. If not using MOX, the uranium fuel would be low enriched at a level of approximately 5%.

As with many designs proposed for the first wave of a nuclear renaissance, economics has been a key concern of the EPR project managers. The original 1500MWe design failed to match the projected lifetime costs of combined cycle gas turbine electricity generation and for this reason the designers were instructed to optimize the economic efficiency of the EPR proposal². This was the main driver in favor of a higher power (1750 MWe) design. In addition the reactor has a design life of 60 years rather than the 40 years of previous European PWR systems. The EPR is also designed so as to require fewer refueling interruptions and cycles of between 18 and 24 months are planned. The EPR will use instrumentation and control based upon the N4 series of reactors.

Although established as a clear Franco-German project to be matched to the needs and regulatory concerns of those two countries, Framatome ANP is particularly conscious of the possibility of orders arising from any new nuclear building programmed in other European countries. In December 2003 Framatome was able to announce a breakthrough that some believe heralds the start of a long-awaited European nuclear renaissance. The Finnish electricity company Teollisuuden Voima Oy (TVO) agreed to purchase an EPR plant for their Olkiluoto site in western Finland. The announcement of the Framatome ANP success followed the earlier Finnish parliamentary decision in May 2002 that paved the way for the building of a fifth Finnish power reactor, on economic, energy security, and environmental grounds.

Framatome ANP is following closely political developments in Eastern Europe. As several Eastern European countries continue on the path towards full European Union membership then so increases the pressure on these countries to replace their Soviet designed RBMK and VVER reactors with more modern designs.

In addition to Finland, Areva has recently started to stress business opportunities for the EPR in China and the United States. Framatome ANP already has almost two decades of experience of doing business in China and has recently extended the range of technical cooperation with that country. The United States is a potential large market in a nuclear renaissance and Areva have highlighted that fact. Despite that, however, Areva and Framatome ANP did not file for pre-licensing of their design with the US Department of Energy under the 2004 test licensing procedure for new nuclear power plants. Three consortia have proposed the AP1000, the ACR and the GE Advanced Boiling Water Reactor, but despite Framatome ANP's enthusiasm for the US market, no such pre-licensing has so far been sought for the EPR in that country³.

Any optimism in Framatome ANP from the Finnish success and potential in other markets must have been tempered somewhat by the political trends in Germany. The last two German governments have been so-called "red-green" coalitions between the Social Democratic Party and the Green Party. The German Green Party has a long-standing hostility to nuclear power generation, particularly the new nuclear power plant construction. In France there are more grounds for optimism about the possibility of new nuclear build based upon the EPR. A recent French government study showed that the EPR would represent a more economic source of

electricity generation than either coal or gas under a range of scenarios concerning interest rates. The EPR option is the most cost effective even in the absence of any carbon costs for the fossil-fuelled technologies. In 2004 the French utility Electricite de France (EdF) has been bringing forward plans to develop a demonstration RPR in France. While the board of the company has decided upon the strategy, no specific site for the new power plant has yet been identified.

Table 18.1 summarizes the main technical characteristics of the EPR design. For a detailed assessment of how the EPR matches the needs of European utilities the reader is recommended to consult the work of Holger Teichel and Xavier Pouget-Abadie.

Table 18.1 Technical characteristics of the European Pressurized Water Reactor (EPR)

Thermal power	4250/4500MW
Electrical power	In the 1600MW range
Efficiency	36%
No. of primary loops	4
No. of fuel assemblies	241
Burnup	60GWd/t
Secondary pressure	78 bar
Seismic level	0.25g
Service	60 years

The EPR is not without its critics. On 15 December 2003 Greenpeace activists occupied the offices of TVO in Helsinki objecting to the plans for the construction of a new nuclear power plant in that country. In France the independent scientific commentators Global Chance have produced a lengthy report. The English translation of the title of their report is The EPR Nuclear Reactor: An Useless and Dangerous Project. Thus far most of the criticism of the EPR plan seems to be an objection to any form of new nuclear build rather than a criticism of this particular technology. It will be interesting to see in due course how the economics of EPR with its particular technology choices compares with the economics of AP1000 and ACR with a more distinctively modular approach emphasizing simplicity of design.

The EPR was designed to meet the requirements of European utilities set out in a "European Utility Requirements" document, as well as a "Utility Requirements Document" issued by the US's Electric Power Research Institute. It also complies with the recommendations (1993) and positions on major issues (1995) that were set up jointly by French and German safety authorities.

The technical guidelines covering the EPR design were validated in October 2000 by France's standing group of experts in charge of reactor safety (the advisory committee for reactor safety to the French safety authority) supported by German experts.

The EPR is the direct descendant of the N4 and Konvoi reactors, developed by Framatome and Siemens KWU respectively, which are themselves derivatives of standard US-type PWRs.

Thanks to an early focus on economic competitiveness during its design process, the vendor believes that the EPR has significantly reduced power generation costs, estimated to be 10 percent lower than those of the most modern nuclear units currently in operation.

This high level of competitiveness would be achieved through:

- Unit power in the 1600MWe range (the highest unit power to date).
- A 36-37 percent overall thermal efficiency, depending on site conditions (presently the highest thermal efficiency for any light-water reactors).
- A shortened construction time.
- A 60-year design life.
- Enhanced and more flexible fuel utilization.

18.2 Construction Schedule

Provisions have been made in the design, construction, erection and commissioning methods to shorten the EPR construction schedule. For example, the general layout of the main safety systems in four trains housed in four separate buildings simplifies, facilitates and shortens performance of the erection tasks for all work disciplines ⁴.

Location of electromechanical equipment at low levels means that it can be erected very early on in the program, thus shortening the critical path of the construction schedule.

Three main principles are applied to the EPR construction and erection: minimization of the interfaces between civil works and erection of mechanical components; modularization; and piping pre-fabrication. Optimizing the interfaces between civil and erection works results in the implementation of a construction methodology “per level” or with “grouped levels”. That means that it is possible to carry out equipment and system installation work at level “N” where civil construction is finished, while at the same time finishing construction works at the level above ($N+1$) and carrying out major construction work at the two levels above ($N+2$ and $N+3$).

18.3 Operation, Maintenance and Services

The key to ensuring an economic return from the nuclear reactor is keeping it in operation: that means that scheduled outages (for refueling and maintenance) should be minimized and unscheduled outages should be reduced, if possible, to zero.

The EPR and its equipments and systems have been designed to allow for efficient refueling outages and to simplify and optimize inspection and maintenance in order to increase plant availability and reduce maintenance costs, two major objectives of plant

operators.

The EPR is designed to reach 92 percent availability over the entire 60 years of its design lifetime. This is made possible by short scheduled outages for fuel loading or unloading and for in-service inspections and maintenance, and also through reduced downtime attributable to unscheduled outages.

Moreover, the reactor building is designed to be accessible, under standard safety and radiation protection conditions, while the reactor is at power. This development enables outage and maintenance operations to be prepared before the scheduled outage begin, reducing the length of the outage. Access to the reactor building during power operation allows preventive maintenance and refueling tasks to begin up to seven days before reactor shutdown and can be continued for up to three days after reactor restart.

The possibility of gaining access throughout the reactor building with the reactor online also makes it easier to carry out servicing and some maintenance if they are required outside scheduled outage periods.

The duration of the plant shutdown phase is reduced by decreasing the time required for reactor coolant system cooldown, depressurization and vessel head opening. Similarly the length of the restart is reduced: the time needed to run the beginning-of-cycle core physics tests have been reduced. Durations of about 70 and 90h are scheduled for the shutdown and restart phases respectively. For the fuel loading or unloading operations, a period of about 80h is scheduled.

Thanks to these changes, the duration of a regular outage for preventive maintenance and refueling should be reduced to 16 days, while a refueling outage can be reduced 12 days. Ten-year outages for main equipment in-service inspection, turbines overhaul and containment pressure test are planned to last 38 days.

18.4 Operational Flexibility

In terms of operation, the EPR is designed to offer the utilities a high level of flexibility. It can be operated over the long term at any power level between 20 percent and 100 percent of its nominal power, with the primary and secondary frequency controls in operation.

This allows it to respond to scheduled and unscheduled power grid demands for load variations, help manage grid perturbations and mitigate grid failures.

The EPR's core design and higher overall efficiency, compared to the reactors in operation today, also claims the following advantages:

- 17 percent saving on uranium consumption per produced megawatt hour.
- 15 percent reduction on long-lived actinides generation per megawatt hour.
- 14 percent gain on the "electricity generation" versus "thermal release ratio" (compared to 1000 MWe-class reactors).
- The ability to use mixed oxide fuel.

18.5 Safety Systems

The safety design of the EPR builds on the experience of other French and German reactors.

The design is simplified by separating operating and safety functions. Four-fold redundancy is applied to the safeguard systems and to their support systems. This architecture allows one system to undergo maintenance during plant operation without requiring the plant to be shut down, as there is still three-fold redundancy, thus helping to ensure a high plant availability factor.

The four trains of the safety systems are located in four different buildings in which strict physical separation is applied. Each has systematic functional diversity.

The emergency power supply system is designed to meet the requirements of the $N + 2$ concept (i. e. assuming a single failure on one train and a maintenance operation on another).

Buildings and components

The reactor building houses the main equipments of the nuclear steam supply system (NSSS) and the in-containment refueling water storage tank (IRWST). Its main function is to ensure the protection of the environment against internal and external hazard consequences under all circumstances. It consists of a cylindrical pre-stressed inner containment with a metallic liner surrounded by an outer reinforced concrete shell. The main steam and feedwater valves are housed in dedicated reinforced concrete compartments adjacent to the reactor building.

The primary system contents the following components:

- The pressurizer is located in a separate area.
- There are concrete walls between the loops and between the hot and cold legs of each loop.
- There is a concrete wall (the secondary shield wall) around the primary system to protect the containment from missiles and to reduce the spread of radiation from the primary system to the surrounding areas.

The fuel building, located on the same common basement as the reactor building and the safeguard buildings, houses the fresh fuel, the spent fuel in an interim fuel storage pool and associated handling equipment⁵. Operating compartments and passageways, equipment compartments, valve compartments and the connecting pipe ducts are separated within the building. Areas of high activity are separated from areas of low activity by means of shielding. The mechanical floor houses the fuel pool cooling system, the emergency boration system and the chemical and volume control system. The redundant trains of these systems are physically separated by a wall into two building parts.

Two diesel buildings house the four emergency diesel generators and their support systems and supply electricity to the safeguard trains in the event of a complete loss of

electrical power. The physical separation of these two buildings provides additional protection.

To withstand a major earthquake, the entire nuclear island stands on a single thick reinforced concrete basement. The building height has been minimized and heavy components and water tanks are located at the lowest possible level.

To withstand a large airplane crash, the reactor building, spent fuel building and two of the four safeguard buildings are protected by an outer shell made of reinforced concrete. The other two safeguard buildings are protected by geographical separation. Similarly, the diesel generators are located in two geographically separate buildings to avoid common failures.

18.6 Primary System

The EPR primary circuit is a four-loop design (i.e. it has four coolant loops each with a steam generator and associated pumps).

The reactor pressure vessel, pressurizer and steam generators feature larger volumes than similar components from previous designs to provide additional benefit in terms of operation and safety margins.

The increased free volume in the reactor pressure vessel, between the nozzles of the reactor coolant lines and the top of the core, provides a higher water volume above the core and thus additional margin with regard to the core “dewatering” time in the event of a postulated loss of coolant accident.

This increased volume would also be beneficial in shutdown conditions in case of loss of the residual heat removal system function.

Larger water and steam phase volumes in the pressurizer smooth the response of the plant to normal and abnormal operating transients, allowing the operator more time to counteract accident situations and extend equipment lifetime.

The larger volume of the steam generator secondary side results in increasing the secondary water inventory and the steam volume, which offers several advantages. Due to the increased mass of secondary side water in case of an assumed total loss of the steam generator feedwater supply, the dry-out time would be at least 30 min, sufficient time to recover a feedwater supply or to decide on other countermeasures. In addition, the primary system design pressure has been increased in order to reduce the actuation frequency of the safety valves, which is also an enhancement in terms of safety.

18.6.1 Pressure vessel

To minimize the number of large welds, and consequently reduce their manufacturing cost and time for in-service inspection, the upper part of the RPV is machined from one single forging and the flange is integral to the nozzle shell course. The reduced number of welds and the weld geometry decrease the need for in-service inspection facilitate non-destructive examinations and reduce inspection duration as well. Nozzles of the set-on type facilitate the welding of the primary piping to the RPV and the welds’ in-service inspection as well.

The lower part of the RPV consists of a cylindrical part at the core level, a transition ring and a spherical bottom piece. As the in-core instrumentation is introduced through the closure head at the top of the RPV there is no penetration through the bottom of the vessel. The elimination of any penetration through the RPV bottom head strengthens its resistance in case of postulated core meltdown and prevents the need for in-service inspection and potential repairs.

The RPV has been designed for non-destructive testing during in-service inspections. In particular, its internal surface is accessible to allow 100 percent visual and/or ultrasonic inspection of the welded joints from the inside.

Consistently with the EPR's 60-year design life, there is an increased margin with regard to reactor pressure vessel embrittlement. The ductile-brittle transition temperature of the vessel material remains lower than 30°C at the end of the design life, achieved by the choice of the material and its specified low content in residual impurities, and also thanks to a reduced neutron fluency due to the inclusion of a neutron reflector surrounding the core.

18.6.2 Steam generators

The steam generator is an enhanced version of the axial economiser steam generator implemented on N4 plants. The axial economiser allows for higher steam pressure output compared to a conventional design, without impairing access to the tube bundle for inspection and maintenance. The very high steam saturation pressure at tube bundle outlet (78 bar) is a major contributor to the high efficiency of the EPR (37 percent).

The increase in steam volume and the set pressure of the secondary safety valves prevent any liquid release to the environment in case of steam generator tube rupture. Particular attention was given during the design of the EPR steam generator to cancel out secondary cross-flows to protect the tube bundle against vibration risks.

The steam drum volume has been augmented. This feature, plus a safety injection pressure lower than the set pressure of the secondary safety valves, would prevent the steam generators from filling up with water in case of steam generator tube rupture to avoid liquid release.

18.6.3 Reactor coolant pumps

The reactor coolant pumps provide forced circulation of water through the reactor coolant system. This circulation removes heat from the reactor core to the steam generators, where it is transferred to the secondary system.

A reactor coolant pump is located between the steam generator outlet and the reactor vessel inlet of each of the four primary loops. The pump design is an enhanced version of the model used in the N4 reactors. The pump capacity has been increased to comply with the EPR operating point. In addition, a new safety device, a standstill seal, has been added as shaft seal back-up. It ensures that even in case of total station blackout or failure of the main seals no loss of coolant would occur.

18.6.4 Pressurizer

The pressurizer has a larger volume to smooth operating transients for the following

reasons:

- To ensure the equipment's 60-year design life.
- To increase the time available to counteract an abnormal operating situation.
- To make maintenance and repair easier and reduce radiological doses.

A dedicated set of valves for depressurizing the primary circuit is installed on the pressurizer, in addition to the usual relief and safety valves, to prevent the risk of high-pressure core melt accident.

18.6.5 Safety injection system

The safety injection system (SIS/RHRS) performs a dual function both during the normal operating conditions in residual heat removal mode and in the event of an accident. The system consists of four separate and independent trains, each providing the capability for injection into the coolant system.

During normal operating conditions, the system should have the following capabilities.

- Transfers heat from the reactor to the Component Cooling Water System (CCWS) when heat transfer via the steam generators is no longer sufficiently effective.
- Transfers heat continuously from the coolant system or the reactor refueling pool to the CCWS during cold shutdown and refueling shutdown, as long as any fuel assemblies remain inside the containment.

In the event of an assumed accident and in conjunction with the CCWS and the Essential Service Water System (ESWS), the SIS in RHR mode maintains the RCS core outlet and hot leg temperatures below 180°C following a reactor shutdown.

The four redundant and independent SIS/RHRS trains are arranged in separate divisions in the safeguard buildings. Each train is connected to one dedicated coolant loop and is designed to provide the necessary injection capability required to mitigate accident conditions. This configuration greatly simplifies the system's design.

The design also makes it possible to have extended periods available for carrying out preventive maintenance or repairs. For example, preventive maintenance can be carried out on one complete safety train during power operation.

Within the containment the in-containment refueling water storage tank (IRWST) contains a large amount of borated water and collects water discharged inside the containment. Its main function is to supply water to various systems and to flood the spreading area in the event of a severe accident.

The EPR complies with the safety objectives set up jointly by the French and German safety authorities for future PWR power plants. They include further reduction of core melt probability, practical elimination of accident situations which could lead to large early release of radioactive materials and the requirements for very limited protective measures in the case of a core melt, that is, no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering and no long-term restriction in the consumption of food⁶.

The protection and safeguard actions needed in the short term in the eventuality of an incident or accident are automated. Operator action is not required before 30 min for an action taken in the control room, or 1h for an action performed locally on the plant.

The increase in the volumes of the major components (reactor pressure vessel, steam generators, pressurizer) gives the reactor extra inertia which helps to extend the time available to the operators to initiate the first actions. The choices taken for the installation of the safeguard systems and the civil works minimize the risks arising from the various hazards (earthquake, flooding, fire, aircraft crash).

The safeguard systems are designed on the basis of a quadruple redundancy, both for the mechanical and electrical portions and for the I&C. This means that each system is made up of four sub-systems, or "trains", each one capable by itself of fulfilling the whole of the safeguard function. The four redundant trains are physically separated from each other and geographically shared among four independent divisions (buildings). Each division includes the same systems as followed.

- A low-head injection system and its cooling loop, together with a medium-head injection system, for borated water safety injection into the reactor vessel in the case of a loss of coolant accident.
- A steam generator emergency feedwater system.
- The electrical systems and I&C linked to these systems.

The building housing the reactor, the building in which the spent fuel is interim-stored and the four buildings corresponding to the four divisions of the safeguard systems are given special protection against externally generated hazards such as earthquakes and explosions.

This protection is further strengthened against an airplane crash. The reactor building is covered with a double concrete shell: an outer shell made of 1.30m thick reinforced concrete and an inner shell made of pre-stressed concrete and also 1.30m thick which is internally covered with a 6mm thick metallic liner. The thickness and the reinforcement of the outer shell on its own have sufficient strength to absorb the impact of a military or large commercial aircraft. The double concrete wall protection is extended to the fuel building, two of the four buildings dedicated to the safeguard systems, the main control room and the remote shutdown station which would be used in a state of emergency.

18.6.6 Limiting severe accident consequences

In response to the new safety model for the future nuclear power plants, introduced as early as 1993 by the French and German safety authorities, the plant design must be such that a core melt accident, although highly unlikely, causes only very limited off-site measures in time and space.

In addition to the usual reactor coolant system depressurization systems on the other reactors, the EPR is equipped with valves dedicated to preventing high-pressure core melt in the eventuality of a severe accident. These valves would then ensure fast depressurization, even in the event of failure of the pressurizer relief lines.

Their relieving capacity guarantees fast primary depressurization down to values of a few

bars, precluding any risk of containment pressurization through dispersion of corium debris in the event of vessel rupture.

The high mechanical strength of the reactor vessel is sufficient to rule out its damage by any reaction, even high-energy, which could occur on the inside between corium and coolant.

The portions of the containment with which the corium would come in contact in the eventuality of a core melt exacerbated by ex-vessel progression—namely the reactor pit and the core spreading area—are kept “dry” (free of water) in normal operation. Only when it is spread inside the dedicated area, therefore already partially cooled, surface-solidified and less reactive, would the corium be brought into contact with the limited water flow intended to cool it down further.

In the case of a severe accident, hydrogen would be released in large quantities inside the containment. This would happen first of all by reaction between the coolant and the zirconium which is part of the composition of the fuel assembly claddings, then, in the event of core melt and ex-vessel progression, by reaction between the corium and the concrete of the corium spreading and cooling area.

For this reason, the pre-stressed concrete inner shell of the containment is designed to withstand the pressure which could result from the combustion of this hydrogen. Further, devices called catalytic hydrogen recombiners are installed inside the containment to keep the average concentration below 10 percent at all times to avoid any risk of detonation.

18.6.7 Collection of inter-containment leaks

In the eventuality of a core melt leading to vessel failure, the containment remains the last of the three containment barriers: this means that provisions must be taken to make sure that it remains undamaged and leak-tight. For the EPR, the following measures have been adopted:

- A 6mm thick metal liner internally covers the pre-stressed concrete inner shell.
- The internal containment penetrations are equipped with redundant isolation valves and leak recovery devices to avoid any containment bypass.
- The architecture of the peripheral buildings and the sealing systems of the penetrations rule out any risk of direct leakage from the inner containment to the environment.
- The space between the inner and outer shells of the containment is passively kept at slight negative pressure to enable the leaks collect there.
- These provisions are supplemented by a containment ventilation system and a filter system upstream of the stack.

Self-test

1. The EPR is not characterized by the adoption of the types of ____ favored by Westinghouse for the AP1000.

- A. active safety system
- B. passive safety system

- C. advanced CCW system
D. new RHR system
2. This was the main driver in_____of a higher power (1750MW) design.
A. need B. a lot C. favor D. the start
3. While the board of the company has_____the strategy, no specific site for the new power plant has yet been identified.
A. decided upon B. decided to C. agreed with D. approved
4. Optimizing the_____between civil and erection works results in the implementation of a construction methodology “per level” or with “grouped Level”.
A. design B. work C. interfaces D. middle
5. This development enables outage and maintenance operations_____before the scheduled outage begins.
A. to be done B. going C. worked D. to be prepared
6. The duration of the plant_____is reduced by decreasing the time required for reactor coolant system cooldown, de-pressuring and vessel head opening.
A. operation time B. shutdown phase
C. shutdown times D. outage
7. The design is simplified by_____operating and safety functions.
A. combining B. facilitating
C. optimizing D. separating
8. The four trains of the safety systems are located in four different buildings in which_____separating is applied.
A. strict physical B. no C. strict D. good
9. It consists of a cylindrical pre-stressed inner containment with a metallic linear_____an outer reinforced concrete shell.
A. in B. surrounded by
C. near D. around
10. The building height has been_____and heavy components and water tanks are located at the lower possible level.
A. reduced B. reinforced
C. minimized D. increased
11. The primary system design pressure has been increased in order to reduce the_____of the safety valves, which is also an enhancement in terms of safety.
A. actuation frequency B. activity
C. size D. pressure
12. In the case of_____, hydrogen would be released in large quantities inside the containment.
A. normal B. LOCA
C. abnormal D. severe accident

关键词解释

core catcher: a special compartment below the reactor vessel. In the event of a core melt this compartment can catch the corium and cool it.

堆芯收集器: 布置在反应堆压力容器下方的容器, 当出现堆芯熔化事故时, 它能收集堆芯熔融物并冷却它。

EPR: (European Pressurized Water Reactor) a Franco-German collaboration that has developed a new pressurized water reactor system ready for immediate deployment.

EPR: 欧洲压水反应堆是法国和德国联合研制的一种新型的压水反应堆系统, 准备推广使用。

MOX Fuel: a kind of fuel mixed-oxide uranium and plutonium, which combines uranium and plutonium oxides and utilizes the thermal neutron fissile properties of the uranium-235 and plutonium-239 isotopes.

MOX 燃料: MOX 燃料是混合的氧化铀和氧化钚燃料, 它把铀和钚的氧化物结合在一起利用 U-235 和 Pu-239 同位素的热中子易裂变特性。

RHRS, Residual Heat-removal System: During the reactor shutdown, this system provide cooling water to remove the decay heat from the core, which consists of heat exchangers and circulating pumps together with associated piping, valves, and controls.

余热排出系统: 该系统在反应堆停堆时提供冷却水从堆芯带走衰变热, 它由热交换器、循环水泵以及相关的管路、阀门和控制组成。

SIS, Safety Injection System: During the accident, this system provides water to cool the core and prevent the fuel pins from melting if normal cooling is interrupted.

安全注入系统: 在事故情况下该系统提供水冷却堆芯, 防止在正常冷却中断时燃料棒熔化。

词 汇

accessible [æk'sesəbl] *a.* 可以接近的

actinides ['æktinaɪdz] *n.* 锕系元素

attributable [ə'tribjutəbl] *a.* 基于……的

augment [ɔ:g'ment] *v.* 增大, 扩增

board [bɔ:d] *n.* 管理局委员会

catalytic [kætə'litik] *a.* 催化的

civil ['sivil] *a.* 土木, 民用的

claim [kleɪm] *v. & n.* 要求, 申请

commissioning 试运行, 调试

comply [kəm'plai] *v.* 同意, 遵守

consistently [kən'sistentli] *ad.* 一贯地

- consortium [kən'sɔ: tʃəm] *n.* 合作, 联合
- counteract [ˌkauntə'rækt] *vt.* 抵抗, 阻碍
- countermeasure ['kauntəmeɪʒə] *n.* 干扰
- course [kɔ:s] *n.* 一层, 一行, 过程
- debris ['debri:] *n.* 碎片
- deployment [di'plɔimənt] *n.* 推广
- destructive [dis'trʌktiv] *n.* 破坏性的
- detonation [detəu'neiʃən] *n.* 爆炸
- dewatering *n.* 排水, 降低水位
- discipline ['disiplin] *n.* 规定, 科目
- diversity [dai'və:siti] *n.* 多样性, 差异
- downtime *n.* 停机, 不工作
- ductile ['dʌktail] *a.* 延性的
- economiser *n.* 节热器, 省煤器
- embrittlement *n.* 脆性, 脆裂
- enthusiasm [in'θju:ziæzm] *n.* 热心, 积极性
- erection [i'rekʃən] *n.* 设备安装, 装配
- evacuation [ivækju'eɪʃən] *n.* 排空, 抽气
- eventuality [iventju'æliiti] *n.* 偶然事件
- exacerbate [eks'æsə:beɪt] *v.* 加重, 使恶化
- file [faɪl] *v.* 提出申请, 把……归档
- fluence [fluəns] *n.* 能量密度, 流量
- geographically *a.* 地理上的, 地区的
- herald ['herəld] *n.* 预言者 *vt.* 宣布的
- hostility [hɒs'tiliti] *n.* 敌视, 战争
- immediate [i'mi:diət] *a.* 直接的, 立即的
- inertia [i'nə:ʃiə] *n.* 惯性, 惰性
- instruct [in'strʌkt] *v.* 通知, 指导, 指示
- interim-stored 临时存储的
- involvement [in'vɒlvmənt] *n.* 包含, 牵连
- modularization *n.* 模块化
- molten corium *n.* 堆芯熔化物
- optimism ['ɒptimizəm] *n.* 乐观主义
- overhaul ['ouvəhɔ:l] *n.* 大修
- parliamentary ['pɑ:ləməntəri] *a.* 国会(议会)的
- performance [pə'fɔ:məns] *n.* 实施, 运转
- peripheral [pə'rɪfərəl] *a.* 周边的
- perturbation [pə:tə'beɪʃən] *n.* 扰动
- preclude [pri'klu:d] *v.* 预防, 阻碍

- precursor [pri'kə:sə] *n.* 先驱, 前身
 preventive [pri'ventiv] *a.* 预防的
 project ['prɒdʒekt] *v.* 计划, 设想; *n.* 方案
 proposal [prə'pəuzəl] *n.* 计划, 提出
 quadruple ['kwɒdrupl] *a.* 四倍……的
 radiological [reidiəu'lɒdʒikəl] *a.* 放射学的
 recommendation [rekəmen'deiʃən] *n.* 建议
 redundancy [ri'dʌndənsi] *n.* 冗余
 renaissance [rə'neisəns] *n.* 复兴
 set-on 确定, 安置, 调节
 sheltering 掩蔽, 保护
 stack [stæk] *n.* 堆积
 standstill 停滞, 间歇
 supplement ['sʌpliment] *v.* 补充
 temper ['tempə] *v.* 锻炼, 调和
 train [trein] *n.* 系列, 将不同设备串起来的系统
 ultrasonic ['ʌltrə'sɒnik] *a.* 超声的
 unmitigated [ʌn'mitigeitid] *a.* 没缓和的, 绝对的
 upstream ['ʌp'stri:m] *a.* 上游的
 versus ['və:səs] *prep.* 作为……的函数, 与……关系

课文注释

1. The EPR is not characterized ...for the AP1000.

句中 passive safety system 是非能动安全系统, Westinghouse 是美国的西屋公司。全句的意思是: EPR 的特点不是采用西屋公司在 AP1000 中所推崇的非能动安全系统。

2. The original 1500 MWe design ...the EPR proposal.

句中 fail to 表示没能做到; projected lifetime costs 表示计划的寿期费用。全句译为: 原来的 1500 MWe 的设计无法与燃气轮机联合循环发电寿期成本相竞争, 由于这个原因, 设计者被告需知优化 EPR 的经济性。

3. Three consortia have ...for the EPR in that country.

句中 consortia 是 consortium 的复数形式, 表示合作或联合, ACR 是 Advanced CANDU Reactor, 表示先进重水堆; that country 在这里是指美国。全句译为: 推荐的三个合作是 AP1000、先进重水堆和 GE 公司的先进沸水堆, 尽管法玛通 ANP 对美国市场的热心, 但至今为止 ERP 并没有申请到在美国的先期执照。

4. For example, ...for all work disciplines.

句中 train 指连接设备和管路的系统, simplifies、facilitates 和 shortens 是三个并列的谓语动词。全句译为: 例如将四套由设备和管路组成的主要安全系统放在四个分隔开的建筑中。这样使所有的安装工作更加简单、便利, 缩短了工程实施的时间。

5. The fuel building ...handling equipment.

句中 the same common basement 是指同一个共用地基, spent fuel 是乏燃料。全句译为: 燃料厂房与反应堆大厅和安全大厅建在同一个共用地基上, 燃料大厅内放置新燃料, 将燃料放在中间储存水池, 与相关的处理设备一起。

6. They include further ...in the consumption of food.

句中 limited 表示有限的、不多的, no permanent relocation 是指熔融物不存在位置不断改变的情况。全句译为: 它们包括了进一步减小堆芯熔化的概率, 消除了导致大的早期放射性释放的可能性, 在堆芯熔化后需要很有限的保护手段, 也就是熔融物不存在位置不断变化的情况, 不需要电站周围厂外的应急疏散问题, 有限的屏蔽, 没有长期食品消费的限制。

Unit 19 The Heavy Water Moderated Reactor

19.1 General Characteristics of the Heavy Water Reactor

The incentive to use heavy water as moderator arises from the excellent neutron economy made possible by the low neutron cross sections of both deuterium and oxygen (0.5 and 0.27 mb, respectively). In combination with its slowing-down power of 0.175cm^{-1} , the low cross section of D_2O leads to a moderating ratio of 6000, making heavy water, by a wide margin, the most desirable moderator on purely theoretical grounds. The fast and slow leakages from a heavy water reactor tend to be roughly equal, in contrast to the situation for a light water system, where the fast leakage is much greater.

The neutron lifetime is also greatly different from that of a light water reactor. It can be as large as 50ms, in comparison with a typical light water value of about 0.1ms. The prompt response to a change of reactivity is therefore much slower for the heavy water system. Another phenomenon which affects the kinetic response of a heavy water reactor is the production of photo-neutrons by γ radiation from fission product decay. γ rays of energies in excess of 2.23 MeV can interact with deuterium to produce neutrons through the (γ, n) reaction. Since the photo-neutron periods, determined by the β -decay periods of their γ -emitting precursors, are generally much longer than the delayed neutron periods, the longer-term kinetic behavior of a heavy water reactor can be considerably more sluggish than that of other types¹.

The excellent neutron economy associated with the D_2O moderator leads to a highly efficient use of the ^{235}U content of the fuel; in comparison with a light water reactor, twice as much energy is generated per unit of ^{235}U consumed with typically some 50% of this coming from ^{239}Pu generated from ^{238}U . The discharged fuel also has around twice the ^{239}Pu content of that discharged from a light water system.

The economics of a D_2O moderated reactor are basically a matter of balancing the better fuel utilization against the higher capital cost of the system, due in part to the relatively high cost of the moderator².

Because of its good neutron economy, the heavy water reactor is an obvious candidate for the role of thermal near-breeder or advanced converter, using the ^{233}U - ^{232}Th fuel cycle. The value of the cycle depends on the high thermal η value of U^{233} (approximately 2.29) and the greater absorption cross section of ^{232}Th as compared with ^{238}U . While the production of plutonium in a natural uranium D_2O reactor is about 2.7g of ^{239}Pu per kilogram of uranium, the equilibrium production of ^{233}U is typically 16g per kilogram of thorium.

The distinction between the pressure tube and the pressure vessel design is shown

schematically in Fig.19.1. In the former, the fuel elements are located in individual tubes through which the high-pressure coolant is pumped; these tubes pass through a separate tank containing the D_2O moderator, which can thus be maintained at a relatively low pressure and temperature. In the pressure vessel design, the heavy water moderator is within the high-pressure containment (although it may be physically separated from the coolant by relatively thin tubes which do not have to withstand a pressure differential).

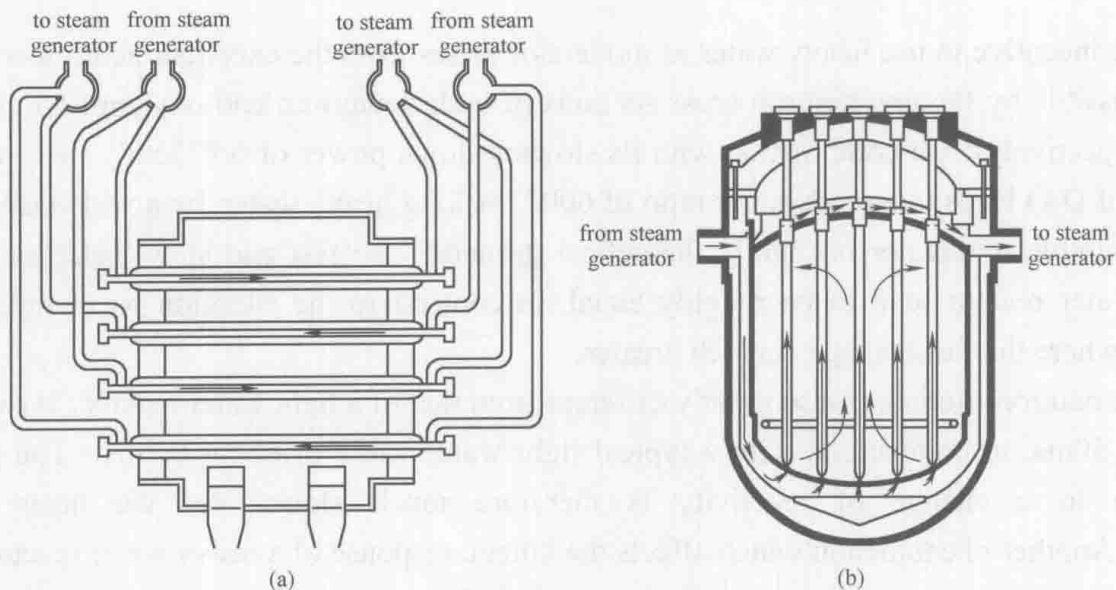


Fig.19.1 Comparison of pressure tube and pressure vessel designs of heavy water reactor

(a) pressure tube; (b) pressure vessel

One of the main advantages of the pressure vessel design is the avoidance of the highly complicated pipework associated with the large number of individual cooling pipes of the pressure tube reactor. The main disadvantage is the requirement for a heavy pressure vessel to contain the pressurized moderator. This may well impose an economic limitation on the size of pressure vessel reactor that is practicable with natural uranium as fuel. Among the advantages of the pressure tube system are the following: flexibility in choice of a coolant other than D_2O ; readier demonstration of safety in the event of pressure system rupture, arising from the subdivision of the pressure circuit; simpler scaling-up to larger reactor sizes on the basis of experience, since all that is basically involved is the addition of further fuel channels; simpler access for fuel changing on account of the individual channel arrangement facilitating on-load refueling, which is particularly advantageous for natural uranium reactors³. On the other hand, the complexity of the individual channels layout involves a significantly higher capital cost for the pressure tube system, and one is also faced with the problem of providing pressure tubes which combine the property of low neutron absorption with the assurance of an adequately long life under irradiation in the reactor core. This problem has been approached by the development of various alloys of zirconium.

Similar advantages and disadvantages can be cited for the various coolants which may be used in a heavy water reactor. The use of D_2O itself for coolant has the advantage of retaining

the optimum neutron economy in the system, at the expense of maximizing the total D_2O inventory, which is a significant factor (typically of the order of 15% for a CANDU-PHW) in the overall capital cost. The control of leakage, both of expensive D_2O and of radioactive tritium gas, demands high standards of engineering for all joints and seals in the high-pressure primary circuit. Employment of D_2O as coolant may be considered to imply an indirect cycle, with the associated capital cost of steam generators, since the attempt to operate on a direct cycle has not been encouraging.

The use of H_2O as coolant in the pressure tube reactor has the advantage of direct cycle operation, thus eliminating the complication of heat exchangers and, in addition, further lowering capital cost by the reduction of D_2O inventory. (In the U.K. steam-generating heavy water reactor, for example, approximately 30% of the moderation takes place in the light water coolant.) Leakage from the cooling circuit becomes less of a problem, although there will be a higher activity level at the turbine, which is now coupled to the reactor directly, rather than through a heat exchanger. Among the disadvantages of operation with boiling coolant is the limitation in system power which is imposed by the requirement to maintain an adequate margin against dryout in the fuel channels. A further operational and safety consideration is the high positive void coefficient of reactivity which can arise with natural uranium fuel in an H_2O -cooled pressure tube reactor; this may be avoided by the use of enriched fuel and a relatively close fuel spacing to give a degree of under-moderation, as a SGHWR, where the void coefficient is close to zero⁴. Another consideration with light water cooling, which has to be balanced against the capital cost savings due to the direct cycle, is the lower burn-up attainable owing to the poorer overall neutron economy.

Owing to the restrictions on steam temperature and pressure imposed by the need to minimize the thickness of pressure tube material for reasons of nuclear absorption, the thermal efficiency is relatively low when either H_2O or D_2O is used as coolant. A significant improvement in thermal efficiency may be obtained by the use of gas cooling with CO_2 . The high fuel temperatures necessary to achieve this improvement impose more rigorous demands on the cladding and primary circuit materials, necessitating the use of elements which have higher absorption cross sections than the zirconium alloys acceptable for the cladding and pressure tubes of the water-cooled reactors. In addition, limitations on fuel and cladding temperatures make it difficult to obtain a power density comparable to that achieved with water cooling. The advantage of higher temperature operation to the overall system efficiency is partially offset by the higher auxiliary power demand for gas circulation.

High pressures are required with either light or heavy water cooling in order to attain temperatures high enough to give adequate thermal efficiency. An alternative method of improving efficiency is the use of a kind of coolant of sufficiently low volatility to allow the reactor to operate at high temperature but moderate pressure. A kind of suitable coolant of this type is an organic liquid such as a hydrogenated terphenyl oil, which allows the coolant outlet temperature to be raised to around $400^\circ C$, compared with the $300^\circ C$ temperature characteristic

of a CANDU-PHW, where the coolant circuit has to operate at a pressure of about 87atm. High power density and a thermal efficiency in the range of 36%~40% are obtainable with organic coolant. A further potential advantage is the low coolant activity, since neutron irradiation of the hydrocarbon coolant gives rise to a negligible amount of radioactivity. Another benefit is a great reduction in the D₂O inventory, and consequently in the size of the reactor, owing to the good moderation characteristics of the organic coolant; it is estimated that the total D₂O inventory per kWe for an advanced organic-cooled CANDU would be reduced by a factor of 5, relative to standard designs, as a result of changing from D₂O to organic coolant.

The use of organic coolant (terphenyl mixtures) in a D₂O-moderated system has been investigated in the Canadian WR-1 research reactor at Whiteshell, Manitoba⁵. Experience has confirmed that zirconium-based alloys are suitable for extended operation in organic coolant provided careful control of the coolant chemistry to prevent excessive hydrogen uptake and fuel tube fouling.

For economic operation, the organic-cooled, D₂O-moderated reactor requires fuel of higher density than that of UO₂. Encouraging results have been obtained in WR-1 with uranium carbide(UC)and triuranium silicide(U₃ Si). The consensus of opinion at the present time, however, appears to be that the cost advantage of organic cooling is not sufficient to justify diversion of the CANDU natural uranium reactor on to this type. The combination of organic cooling with the thorium fuel cycle, however, appears promising as a longer-term development of the CANDU system.

19.2 The CANDU Pressure Tube Heavy Water Reactor

The CANDU heavy-water-moderated and cooled pressure tube reactor represents the main line of development of the Canadian nuclear program. Following experience with the Nuclear Power Demonstration Reactor(NPD), which began operation in 1962, and the prototype 200-EWe CANDU reactor at Douglas Point, commissioned in 1967, the major application of the concept has been in the four-unit station in Ontario. This station, which is operated by the Ontario Hydro electrical utility⁶, is one of the largest nuclear stations constructed to date, with a total gross capacity of 2160 MWe.

Apart from minor differences of detail, the four Pickering reactors are of identical design(see Table19.1). All share the following features: natural uranium fuel, heavy water moderator, pressurized heavy water coolant, indirect steam cycle, and on-load bidirectional refueling for the horizontal pressure tubes. The heavy water moderator is contained in a horizontal cylindrical tank (calandria) of about 8-m diameter, constructed of austenitic stainless steel. A total of 390 zircaloy-2 calandria tubes run through the calandria from one end face to the other. The zircaloy-2 or $\text{Zr-2}\frac{1}{2}\%\text{Nb}$ pressure tubes, in which the fuel bundles are located, are separated from the calandria tubes by a sealed insulating annulus containing nitrogen. Each

pressure tube is attached at the ends by rolled joints to stainless steel end fittings, which are supported in sliding bearings at the end faces of the calandria.

Table 19.1 Data for Pickering CANDU Reactor

Reactor type: CANDU-PHW	Length of fuel bundle: 49.5cm
Moderator: Heavy water	Fuel pellet diameter: 14.33mm
Coolant: Heavy water	Thickness of zircaloy sheath: 0.41mm
Fuel: Natural uranium (UO_2)	Total mass of UO_2 in core: 105t
Diameter of calandria: 8.1m	Average fuel irradiation: 8300 MW d/t
Overall length of calandria: 8.25m	Average linear rating of fuel pin: 37.6W/cm
Total D_2O content of moderator circuit: 284t	Total D_2O content of coolant circuit: 158t
Number of fuel channels: 390	Coolant temperature at channel inlet: 249°C
Lattice spacing: 28.6cm	Coolant temperature at channel outlet: 293°C
Core radius: 318.5cm	Mean pressure at outlet headers: 90 kg/cm ²
Core length: 595cm	Total heat output: 1744 MWt
Fuel bundles per channel: 12	Net electrical output: 508MW
Pins per fuel bundle: 28	Overall station efficiency: 29.1%

The heavy water coolant, which, as in a PWR, is pressurized to prevent boiling, is circulated through the pressure tubes and thence to the heat exchangers. The heat transport system is divided into two identical loops, each containing six tube-in-shell heat exchangers, where the heat is transferred to light water to produce steam. In addition to some engineering convenience, the division into two circuits has advantages from the safety viewpoint, as it effectively halves the voidage which would be produced by a gross failure of the primary containment system. The coolant circuit contains some 160 tonnes of heavy water, the inlet and outlet temperatures being at about 250°C and 293°C, respectively.

The 280 tonne of heavy water moderator in the calandria is maintained nominally at atmospheric pressure and is continuously circulated through an external cooler to maintain its temperature at around 60°C. Inside the calandria there are a number of spray nozzles which provide a drenching spray to cool any parts not covered by the moderator.

The Pickering reactors are fueled with natural uranium fuel in the form of cold-pressed, sintered UO_2 pellets of diameter 14.3mm, clad in collapsible zircaloy sheaths of 0.4mm wall thickness. Each pellet is dished at one end to allow for axial expansion. The short fuel element cluster, of length 495mm, is made up of 28 fuel pins (See Fig.19.2). Interpin and bundle-to-pressure tube spacing is provided by zircaloy spacers brazed to the sheaths. Each pressure tube contains 12 fuel bundles placed end-to-end. The fuel bundle is designed to give maximum neutron economy, and only the minimum of structural material is included. All related components, such as orifices, fuel handling hardware, and control and monitoring facilities, are parts of the reactor capital equipments. As a result, the non-fuel component of the

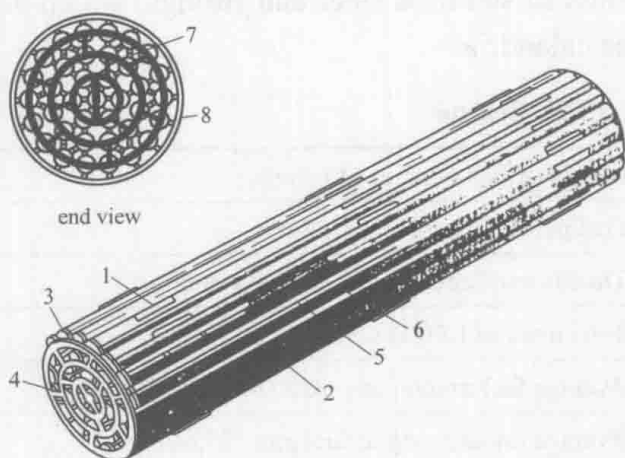


Fig.19.2 Fuel bundle for CANDU reactor

- 1—Zircaloy bearing pads; 2—Zircaloy fuel sheath;
 3—Zircaloy end cap; 4—Zircaloy end support plate;
 5—uranium dioxide pellets; 6—canlub graphite interlayer;
 7—interelement spacers; 8—pressure tube

bundle accounts for only some 0.7% of its total neutron absorption. The CANDU system, like the natural uranium Magnox reactor, requires on-load refueling for optimum economy of operation. Remotely operated refueling machines are located at both ends of a given channel for insertion and removal of the fuel assemblies. A fuel shuffling scheme is used to equalize burn-up of the clusters, the two identical fueling machines being arranged to move fuel in opposite directions in adjacent channels, thus maintaining an axially symmetric neutron flux distribution. The burn-up attained is 8000 MWd/t or better, and the fuel residence time is only about 20%~30% of that which is typical for the fuel of a light-water-moderated reactor. The increased complexity associated with on-load refueling is partly compensated, therefore, by the less stringent requirements on the material of the fuel bundle, the sheathing for which is only about half the thickness necessary in a BWR or PWR. Despite the reduced thickness, failure rates due to corrosion or fretting are negligible. The thin sheath has the advantage of deforming under the coolant pressure to rest on the fuel pellets, thereby providing excellent heat transfer. Fuel densification effects are avoided, since the reduced need for voidage provision allows the use of high-density fuel in the first place.

The arrangements for reactivity control and shutdown of the reactor are as follows.

(1) Once an equilibrium cycle has been achieved, the reactivity is maintained at a suitable value by on-load refueling. The average rate of refueling required is nine fuel bundles per day, to compensate for a reactivity decline due to burn-up of about 0.04% per day.

(2) Normal operating control of reactivity is provided by 14 zone control compartments located throughout the core. These are tubes which can be filled with ordinary water, thus reducing reactivity by increasing the neutron absorption. The requirement for stabilization of the core power level on a zone basis arises from the fact that the core radius is many times larger than the diffusion length, and consequently the control system must be designed to prevent xenon-induced spatial instability.

(3) An additional facility, used both for adjusting the flux shape for optimum peak-to-average ratio and to overcome the xenon transient following a trip from power, consists of 18 adjuster rods which may be inserted vertically among the calandria tubes⁷. Cobalt is used as the absorber in the adjuster rods because of the commercial value of the radioisotope produced by irradiation.

(4) Rapid shutdown is effected by dropping eleven cadmium-loaded shut-off rods operating through the top of the calandria tank, backed up if necessary by dumping of the moderator into a dump tank situated below the calandria. During normal operation, helium gas in the dump tank is maintained at a high pressure. Release of this pressure allows the moderator to fall under gravity into the dump tank. It is interesting to note that, in later CANDU designs, the use of moderator dump has been discontinued in favor of the rapid injection of poison (a salt of gadolinium) into the moderator.

(5) For dealing with reactivity variations beyond the capacity of the other systems, for example, at the start of life, when the reactor is loaded with fresh fuel, boron can be added to the moderator in the form of a soluble oxide. The boron can be removed when necessary by ion exchange.

The void coefficient in CANDU is positive, yielding a reactivity change of about 0.75% for complete voiding of all channels in the equilibrium core. With this magnitude of coefficient, suitable subdivision of the primary circuit limits the rate and extent of the transient associated with any possible void increase to a level which can easily be covered by the shut-down rods.

The Pickering containment consists of a two-stage system. Each of the four reactors in the station is enclosed within its own concrete containment building, which is maintained slightly below atmospheric pressure. In the event of a sudden buildup of pressure, due to a major breach of the primary circuit, pressure relief valves would open and the mixture of steam and air would be discharged to a "vacuum building" which serves all four reactors. This building, which has a volume of $8 \times 10^4 \text{ m}^3$, is constructed from reinforced concrete, and is normally maintained at an absolute pressure of 50mmHg. The vacuum building is capable of containing all the steam produced by the discharge of the entire primary coolant, but the process of containment is aided by a cooling spray which is automatically actuated at a preset pressure to cool the air and condense the steam. The water tank supplying the spray is located in the top of the vacuum building and has a capacity of two million gallons. The combination of reactor building and vacuum building is designed to contain all the energy release in the worst conceivable rupture of the primary circuit, the reactor building being exposed to a pressure transient of not more than 1.4 atm absolute, which is well within its capacity to withstand.

The choice of the CANDU type of reactor for the Canadian nuclear power program was initially motivated by the country's possession of an abundant supply of natural uranium and a large supply of hydroelectric power which could be used for the production of heavy water. Early problems with the prototype at Douglas Point included failure of coolant pumps, difficulties with the refueling machines, and an excessive rate of heavy water leakage, which was of concern not only on economic grounds but also on account of the associated tritium hazard.⁸

The economics of the present CANDU system are based on the natural uranium, once-through, throwaway fuel cycle, that is, the fuel is passed through the reactor once before

being discharged to permanent storage. No-credit is currently allocated for the plutonium content of the irradiated fuel, although this may be worth recovering at a later date. The simple fuel cycle has the advantage of eliminating expensive enrichment costs and reprocessing costs, but in the longer term CANDU offers good development potential through the possibility of combining the use of thorium and plutonium-based fuels with the high thermal efficiency of organic coolant. On a shorter time scale, improvements can be expected from the use of stronger zirconium alloys which are currently under development. These can be employed either to improve the neutron economy by reducing sheathing thickness or to increase the steam conversion efficiency. The latter objective will also be aided by the adoption of indirect cycle boiling, which will be standard in the later reactors of the CANDU type.

A development of the CANDU system which retains the advantages of the heavy water moderator while permitting the use of a direct coolant cycle is the CANDU-BLW(boiling light water)reactor, of which a prototype operated at Gentilly in Quebec from 1971 to 1980.⁹ One of the major concerns at Gentilly was the study of operation with positive void and power coefficients. Measurements during commissioning indicated a strongly positive power coefficient of the order of 10^{-4} per percent increase in power. In practice, no problem was encountered in controlling the reactor, since the long time constant of around 18s associated with the large (19.5mm-diameter) fuel pins in the 18-pin bundle led to an appreciable delay in the formation of voids following an increase in power. Close control of steam drum pressure was necessary, however, during transients, such as a turbine trip, to avoid over rapid variation of core voidage.

19.3 Advanced CANDU Reactor

The Advanced CANDU Reactor (ACR) marks a shift in the CANDU design from using heavy-water coolant to using light water for cooling. Heavy water, used in the original CANDU design for both cooling and moderating the reaction, is now used only as a moderator. That means the ACR-700 design requires 75 percent less heavy water than the previous version, CANDU 6.

Other similarities with PWR and BWR designs include the balance of plant systems (outside the NSSS) and the steam generators, but some differences still remain.

The ACR retains the basic configuration of the CANDU design with reactor pressure tubes in a horizontal configuration passing through a calandria. However, the distance between each fuel channel in the reactor core has been increased to improve access at the reactor face and make the design more amenable to reactor face installation, inspection and maintenance activities.

The ACR, like all previous CANDU designs, includes online fuelling, i.e. fuel is replaced at power and without shutting the plant down. Current light-water reactors shut down on an 18-month or 24-month cycle for refueling, whereas the ACR design aims for a three-year

operational cycle with a 21-day maintenance outage.

The ACR fuel is similar to that of other CANDU designs, in which it is contained in short fuel bundles that are moved progressively through the reactor in the horizontal pressure tubes. The fuel is enriched to only around 2 percent, compared to slightly higher enrichments in PWRs and BWRs, but this represents a change from the conventional CANDU design, which uses natural uranium fuel.

The other difference is that the ACR-700 does not have boron in the reactor coolant system, which ensures long-term plant life.

The ACR design has two independent safety shutdown systems; an inherent passive emergency fuel coolant capability, in which the moderator system can absorb excess heat; and limited challenge to containment barriers, even in the event of a severe accident. The ACR design incorporates further passive safety, such as a highly stable core design, large operating margins and long times available for operator action.

Due to data showing that 65 percent to 75 percent of forced outage items are related to the balance of plant (the area outside the NSSS), AECL is working with Bechtel Hitachi to reduce that rate either by adding redundancies or by re-engineering equipments.

To reduce construction costs, AECL is examining the plant design to standardize pipe sizes, valves, hangers, supports and other equipment across all systems of the plant. This will also help ease maintenance efforts and minimise spare part requirements.

AECL has decided to focus the ACR program on the development of the ACR-1000 product as the lead design, moving on from the lower-power ACR-700. AECL's planned approach to new build in Ontario envisions the first ACR-1000 going into service in 2016.

Self-test

1. During the event of sudden buildup of pressure, the relief valves would open and the steam would be discharge to _____.
 - A. the atmosphere
 - B. storage tank
 - C. containment
 - D. steam generator
2. The adoption of _____ will be latter objective of CANDU type.
 - A. pressure vessel
 - B. indirect cycle boiling
 - C. direct cycle boiling
 - D. boiling light water
3. The main merit to use D_2O as moderator is result from its _____.
 - A. long neutron life
 - B. photo neutrons produced
 - C. low fast leakage
 - D. low neutron cross sections
4. The discharged fuel from CANDU contains more _____ than that from light water reactor.
 - A. fission products
 - B. γ -rays
 - C. ^{239}Pu
 - D. ^{238}U

5. The heavy water reactor is a candidate of ____.
- A. converter B. breeder
C. advanced reactor D. power station
6. The direct cycle operation means ____ the complication of heat exchanger.
- A. reducing B. eliminating C. increasing D. decreasing
7. In order to attain high temperature in a water cooling reactor, ____ is required.
- A. a pump B. higher pressure
C. higher efficiency D. moderator
8. The Pickering reactors are ____.
- A. pressure vessel reactors B. direct steam cycle
C. water cooling reactors D. indirect steam cycle
9. The overall heavy water in a Pickering reactor is about ____.
- A. 280 tons B. 160 tons C. 440 tons D. 200 tons
10. Each pressure tube contains ____.
- A. 12 fuel bundles B. 12 fuel elements
C. one fuel bundle D. one fuel element
11. The CANDU has ____ void coefficient.
- A. negative B. large C. small D. positive
12. The pressure in the Pickering containment is ____ atmosphere.
- A. at B. lower than C. higher than D. above

关键字解释

Calandria: Closed vessel penetrated by pipes so that liquid in each pipe does not mix.

排管容器: 有管群穿过, 但各管内流体不相混合的密封容器。

CANDU: CANadian Deuterium and Uranium reactor. It uses natural uranium fuel canned in zircaloy and heavy water as moderator and coolant.

加拿大重水铀反应堆: 它使用天然铀燃料, 用锆合金做包壳, 重水作冷却剂和慢化剂。

CANDU-BLW: CANadian Deuterium and Uranium-Boiling Light Water reactor.

加拿大氘铀-沸腾轻水反应堆: 重水作慢化剂, 轻水作冷却剂。

CANDU-PHW: CANadian Deuterium and Uranium-Pressurized Heavy Water reactor.

加拿大氘铀-加压重水反应堆

Fast leakage: The leakage of fast neutrons from nuclear reactor.

快中子泄漏: 快中子从反应堆中的泄漏。

Neutron lifetime: The average time between the appearance and disappearance of neutrons in the reactor.

中子寿期: 在反应堆内中子从出现到消失之间的平均时间。

Photoneutron: Neutron resulting from interaction of photon with nucleus.

光激中子: 由光子与核相互作用而产生的中子。

Power coefficient: The change of reactivity of a reactor with the power increasing.

功率系数: 由于反应堆的功率增加而造成反应性的改变。

Power density: Energy released per second per unit volume of a reactor core.

功率密度: 反应堆堆芯中每秒、每单位体积中所释放出的能量。

Pressure tube reactor: Reactor in which the fuel elements are contained in a large number of separate tubes through which the coolant water flows, rather than in a single pressure vessel.

压力管式反应堆: 燃料元件装在很多根相互独立的冷却剂水流过的管子中, 而不是装在一个压力容器中的反应堆。

Refueling machine: A device for inserting or removing fuel in a nuclear reactor.

换料机: 用来插入和取走反应堆内核燃料的设备。

SGHWR: Steam Generating Heavy Water Reactor. The heavy water moderated reactor that light water as coolant and direct cycle is used.

蒸汽发生重水反应堆: 重水作为慢化剂, 轻水作为冷却剂, 采用直接循环的反应堆。

Thermal efficiency: The efficiency of a heat engine. The ratio of the work done by the engine to the mechanical equivalent of the heat supplied in the steam or fuel.

热效率: 热机的效率, 即发动机做的功与蒸汽或燃料所提供的热量的机械当量之比。

Xenon: One of the noble gases, formed as a result of radioactive decay of uranium fission fragments, this isotope forms the most serious reactor poison, and may delay restart of a reactor after a period of shutdown.

氙: 一种惰性气体, 是由铀裂变碎片的放射性衰变而生成的, 是最严重的反应堆毒物, 在停堆一段时间后会延迟反应堆重新启动。

词 汇

actuate ['æktʃueɪt] *v.* 驱动

amenable [ə'mi:nəbl] *a.* 顺从的, 适合于……的

atm. =atmospheric pressure 大气压

auxiliary [ɔ:g'ziljəri] *a.* 辅助的

bidirectional [baɪdi'rekʃənl] *a.* 双向的

booster ['bu:stə] *n.* 增强, 调升

braze [breɪz] *v.* 铜焊

capital ['kæpɪtl] *a.* 基本的, 主要的

cite [saɪt] *v.* 引用, 提到

cobalt ['kəʊbɔ:lt] *n.* 钴

collapsible [ke'læpsəbl] *a.* 可分拆的, 坍塌的

commission [kə'mɪʃən] *n.* 投入运行

conceivable [kən'si:vəbl] *a.* 可能的

consensus [kən'sensəs] *n.* 一致同意

decline [di'klaɪn] *v.* 下降

- deform [di'fɔ:m] *v.* 使变形
densification [densifi'keiʃən] *n.* 密实化
drench [drentʃ] *v.* 湿透
dryout [drai'au] *n.* 干涸
envision [en'viʒən] *vt.* 预见, 展望
fouling *n.* 弄脏, 结垢
fretting *n.* 微振磨损
gross [grəʊs] *a.* 总计的, 重大的
halve [hɑ:v] *v.* 将……减半
hanger ['hæŋə] *n.* 挂钩, 悬架
hardware ['hɑ:dwɛə] *n.* 设备, 硬件
hydrogenated [hai'drɒdʒineitid] *a.* 氢化的
imply [im'plai] *v.* 意味着
incentive [in'sentiv] *n.* 诱因, 动机
interpin 元件之间
inventory ['invəntri] *n.* 装料量, 总装量
lattice ['lætis] *n.* 栅格
mb=milli barn 毫靶恩, 10^{-3} 靶恩
moderate ['mɒdərit] *a.* 中等的, 适度的
motivate ['məutivait] *v.* 推动, 刺激
on-load refueling 不停堆换料
organic [ɔ:'gænik] *a.* 有机的
PHW=Pressurized Heavy Water 加压重水
pipework *n.* 管路
possession [pə'zeʃən] *n.* 占有
precursor [pri'kə:sə] *n.* 产物母体
preset [pri'set] *a.* 预先设定的
relief valve 释放阀
rigorous ['rigərəs] *a.* 严格的
rupture ['rʌptʃə] *n.* 破裂
scaling-up *n.* 按比例放大
sheath [ʃi:θ] *n.* 外壳
shuffling *n.* 重排
silicide ['silisaɪd] *n.* 硅化物
sluggish ['slʌɡɪʃ] *a.* 缓慢的
spatial ['speɪʃəl] *a.* 腔的, 占有空间的
stringent ['strɪndʒənt] *a.* 严格的
subdivision ['sʌbdɪ.vɪʒən] *n.* 一部分, 再分
symmetric [si'metrik] *a.* 对称的

terphenyl *n.* 三联苯

triuranium *n.* 三价铀

tube-in-shell 管壳式

undermoderation *n.* 弱慢化, 欠慢化

uptake ['ʌpteɪk] *n.* 吸入

volatility [vɒlə'tɪlɪtɪ] *n.* 挥发性

withstand [wɪð'stænd] *v.* 经受

due in part to 部分由于

课文注释

1. Since the photo-neutron periods, ...other types.

句中 **determined by** 引导的短语做 **photo neutron periods** 的定语。整句译为: 因为由 γ 辐射母核的 β 衰变周期确定的光激中子周期一般比缓发中子的周期要长, 所以重水反应堆长期的反应动力学特性比其他类型反应堆的慢。

2. The economics of a D_2O ...of the moderator.

句中 **due in part to...** 是介词短语, 做宾语 **capital cost** 的补语。整句译为: 重水反应堆的经济性是较好的燃料利用性与高的系统总投资之间的平衡, 系统总投资高的部分原因是由于慢化剂的费用相对较高。

3. Among the advantages of...uranium reactors.

句中 **readier** 是形容词 **ready** 的比较级, **readier demonstration of safety** 是说安全性易展示; **simpler access for fuel changing** 是说换料时易接近。整句译为: 压力管系统的优点如下: 除 D_2O 之外, 冷却剂有选择的灵活性; 由于压力系统破裂是出现在压力回路的分支上, 因此在压力系统破裂时容易展示安全性; 根据实验可把反应堆的尺寸放大, 因为这一过程涉及的只是附加一些燃料通道; 由于各通道都是分别布置的, 换料时容易接近, 不停堆换料方便, 对于天然铀反应堆有特殊的优点。

4. A further operational...is close to zero.

句中 **positive void coefficient** 是正空泡系数, 即当冷却剂中空泡份额增加时呈正的反应性; **fuel spacing** 指燃料管间距。整句译为: 更进一步运行和安全性方面的考虑是反应性的正空泡系数, 这可能出现在水冷天然铀燃料的压力管反应堆中; 可以使用浓缩燃料和相对紧密的燃料元件间距, 产生一定的弱慢化, 来避免反应性的正空泡系数, 例如蒸汽发生重水反应堆, 其空泡系数接近于零。

5. Whiteshell, Manitoba

Manitoba (马尼托巴) 是加拿大的一个省, 位于加拿大中部, Whiteshell (瓦特森尔) 是该省的一个地名。

6. Ontario Hydro electrical utility

安大略 (加拿大的一个省) 水电公司。

7. An additional facility, ...calandria tubes.

句中 **used** 引导的分词短语做主语的定语, **which**, 引导的定语从句修饰它前面的 **rods**。

整句译为: 一个用来调节中子通量分布形状、优化峰值与平均中子通量比, 并克服甩负荷时氙瞬变的附加设施, 由 18 个调节棒组成。这些棒可以在排管之间立式插入。

8. Early problems with the prototype...tritium hazard.

句中 Douglas point 是地名“道路拉斯角”; which 引导的定语从句修饰它前面的短语 an excessive rate of heavy water leakage。整句译为: 在道路拉斯角的原型反应堆样机上, 早期存在的问题有冷却剂泵的损坏、换料机故障以及重水泄漏的速度过快, 这些问题不仅与经济原因有关还涉及氚的危害。

9. A development of the CANDU...to 1980.

句中第一个 which 引导定语从句修饰主语, 第二个 which 是介词 of 的宾语; Gentilly 是地名珍丁里, Quebec 魁北克是加拿大东部的一个省。整句译为: CANDU 沸腾轻水反应堆是 CANDU 系统的一个改进型, 它保持了重水慢化剂, 但允许使用冷却剂直接循环, 它的原型从 1971 年到 1980 年在魁北克珍丁里运行。

Unit 20 Generation IV Nuclear Power

20.1 Introduction

In the preceding chapters we have considered technologies and policies that would be central to any moves towards a nuclear renaissance in Europe and North America. In doing so we have emphasized near-term technologies with relatively well understood engineering issues and economics and have been considering technologies from the second and third generations of nuclear power. It has become standard to describe the successive developments in nuclear power plant engineering in terms of the following generations of technology.

- Generation I: early prototype reactors of the 1950s and early 1960s (e.g. Shipping port and Magnox).
- Generation II: commercial power reactors of the 1950s and 1980s (e.g. LWR of both PWR and BWR types, CANDU and AGR).
- Generation III: advanced light water reactors (e.g. systems maturing in the late 1990s such as ABWR, AP1000, ACR and EPR).
- Generation III+: these are innovation new reactor designs showing promise for the near term future (e.g. PBMR and GT-MHR high-temperature reactors).

It is expected that Generation III+ deployments will come to an end by approximately 2030, to be superseded by a next generation of economic, safe, low waste and proliferation-resistant plants to be known as “Generation IV”.

For some, Generation IV technologies represent the core of any nuclear renaissance. They would argue that the issues of nuclear renaissance are synonymous with the issues surrounding Generation IV and that the successful deployment of Generation IV technologies will be the defining stage of any nuclear renaissance. This author, however, disagrees. I believe that the issues of near-term deployment of Generation III technologies are the key determinants of the future of commercial nuclear power. Without plans for new build starting in this first decade of the century, it is hard to imagine that nuclear power in the west has any future whatsoever. It is unlikely that North America and Europe would be able to deploy Generation IV technologies in the years after 2020 if nuclear power had not already experienced substantial new build based upon Generation III and Generation III+ technologies¹. In part this is a consequence of the continued erosion of the skills base and the concomitant loss of industrial and regulatory capacity that will occur without near-term new build.

It is important to acknowledge that those working on Generation IV technologies for deployment in 2030 are, of course, extremely conscious of today's prerequisites in terms of

nuclear skills and underpinning research. That is to say, while Generation IV looks to the long term, all acknowledge that it raises technology and policy issues for the present.

Some might argue that the maintenance of nuclear industrial capacity, in the spirit of keeping the nuclear option open, could allow for a late renaissance based upon generation III+ and Generation IV technologies in the 2020s and 2030s without intervening new builds. However, it seems more likely that defining decisions towards or away from a nuclear renaissance are required in the nearer term. Such decisions will concern new builds using existing Generation III technologies.

In the past few years, the term Generation IV has assumed a second, much more specific, meaning than that referred to above. It has come to be associated with a particular international collaborative group led by the United States Department of Energy. This group, known formally as the Generation IV International Forum (GIF), was chartered in the summer of 2001 following an earlier series of preparatory meetings starting in January 2000. Initially led by an inner policy group, the GIF brings together governmental representatives from ten countries (Argentina, Brazil, Canada, France, Japan, South Korea, South Africa, Switzerland (a late entrant), the United Kingdom and the USA).

In December 2002 the GIF published a major report under the title of *A Technology Roadmap for Generation IV Nuclear Energy Systems*. In March 2003 the Roadmap was presented to the US Congress. The Roadmap summarizes the work of the GIF up to the end of 2002 and presents an outline path towards eventual deployment of Generation IV systems based upon international collaboration. The US leadership of the GIF project shaped the process by which competing technologies were assessed for Generation IV development. Furthermore, the US leadership of the GIF process is clear from the scope and emphasis of the GIF documentation.

Prior to the launch of the GIF, the US DOE had set up a Nuclear Energy Research Advisory Committee (NERAC)². A NERAC study published in October 2001 had focused on technologies deployable by 2010. Eight candidate designs were assessed in that process (ABWR, SWR1000, ESBWR, AP600, AP1000, IRIS, PBMR and GT-MHR) against six criteria: regulatory acceptance, industrial infrastructure, the commercialization plan, the cost-sharing plan, economic competitiveness, and fuel cycle industrial structure. In this book we do not attempt to survey all possible nuclear power designs that might contribute to a nuclear renaissance. We merely seek to explore the range of technology and policy issues affecting new build decisions. Consequently this book does not emphasize the small Westinghouse PWR (100~300MWe) known as IRIS, the ABB Westinghouse advanced PWR System 80+, or the GE ESBWR. This latter design has many similarities to the ABWR designs.

The NERAC work informed the early meetings of the GIF, but it was immediately acknowledged that there was a need to broaden the range of designs under consideration to

reflect international interests. The GIF therefore created the International Near Term Deployment Group. This extended the NERAC list of near-term technologies to 16 designs including both the Canadian ACR-700 and the EPR.

The enlarged GIF international near-term deployment list defines a set of reactors from which any nuclear renaissance based upon generation III and generation III+ technologies would need to be emerged.

However, as its name suggests, the primary attention of the GIF is not devoted to near-term Generation III and Generation III+ issues, but rather is devoted to the longer-term task of advancing true Generation IV technologies. Contenders for Generation IV status have been assessed by the GIF against eight equally weighted goals.

The GIF itself acknowledges that the sustainability criteria adopted are narrow and do not span the range of considerations currently accepted internationally as being central to balanced sustainable development. The GIF accepts the Brundtland formulation of sustainable development, namely that our actions should meet the needs of the present while not compromising the ability of future generations to meet their own needs. However, the Brundtland report of 1987 built the fundamental framework for sustainable development based upon three equal criteria—the economic, the environmental and the social. These three pillars of sustainable development are sometimes referred to as “the three Ps”: Prosperity, Planet and People.

Technology goals for Generation IV nuclear energy systems is as follows:

Sustainability—1. Generation IV nuclear energy systems will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.

Sustainability—2. Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long-term stewardship burden in the future, thereby improving protection of the public health and the environment.

Safety and reliability—1. Generation IV nuclear energy systems operations will excel in safety and reliability.

Safety and reliability—2. Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.

Safety and reliability—3. Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

Economics—1. Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.

Economics—2. Generation IV nuclear energy systems will have a level of financial risk comparable with other energy projects.

Proliferation resistance and physical protection—1. Generation IV nuclear energy systems will increase the assurance that they are a very unattractive, and a least desirable route for diversion, or theft, of weapons-usable materials. Furthermore they will provide increased

physical protection against acts of terrorism.

The GIF sustainability criteria focus exclusively on environmental sustainability with considerations for economic sustainability arguably covered under other goals. The economic goals include reference to the full life cycle of the technology including long-term waste management. However, the pillar of sustainability that is completely omitted in the GIF goals is the concept of social sustainability. In author's opinion this omission is the most serious weakness in the GIF approach, which in other respects has been a model of ambition and vision. It seems that the lack of explicit considerations of the social and political needs and concerns of current and future generations have already started to shape the balance of work undertaken by the GIF and the projects favoured by it. The GIF process has neglected to factor in what types of nuclear power plants or fuel cycles the various national publics and their representatives might actually want in the years after 2030. Experiences with consensus conferences and other forms of public engagement have shown that ordinary citizens are willing to participate in, and to contribute to, policy making and prioritization in these intensely technical areas. For instance, any strong public aversions to separated plutonium in the fuel cycle or to nuclear waste "disposal" (as opposed to monitored and retrievable storage) would be important factors to incorporate into Generation IV planning. At present such inputs seem to be lacking. Generation IV priorities, having been developed in the tradition of engineering-led design and shaped by technocratic policy-making, risk falling at the final hurdle³. The risk is real that the publics of liberal western democracies might refuse to allow Generation IV new build for reasons which are deeply held, widespread and visible, but which the technocrats have dismissed as idiosyncratic and naive.

Technocracy is defined as "rule or control by technical experts". The GIF approach seems to be a technocracy tempered by national policy priorities rather than political forces. It is noteworthy that the GIF goals make no explicit reference to what is arguably the greatest factor driving the need for Generation IV technologies—fossil fuel-based CO₂ emissions and their impact on global climate change. Rather, the first GIF sustainability goal emphasizes the role of Generation IV technologies in delivering "cleaner air". It is important at this point to note that this clause, in a US context at least, does not include consideration of CO₂ emissions. The US Environmental Protection Agency regulates emissions of numerous pollutants and harmful gases, but notably CO₂ is not among them. The GIF goals seem to be shaped by current US political considerations that would seem to pull away from best science and engineering practice regarding both global climate and the dangers of uncontrolled greenhouse gas emissions.

Even more amazing than the interest in closed fuel cycles is the fact that three of the six reactors being proposed by the GIF are fast reactors. Some have the possibility of being developed as fast, epithermal or thermal reactors. Only two are clearly designed to operate as thermal reactors similar to all western commercial nuclear power plants in operation today. The

ambition and vision of the GIF does not stop there, because only one of the six proposals is cooled by light water. Two are helium-cooled and the other three have metal or molten salt cooling. The GIF recommendations for Generation IV are indeed radical and ambitious, albeit in ways that refer back to several of the dreams of the nuclear technologists in the years before the slowing of western nuclear power.

The six technologies selected for research and possible development by the GIF as “most promising systems” are as follows.

20.2 The Gas-cooled Fast Reactor (GFR)

This concept builds upon the gas turbine-modular helium reactor. That design, however, uses a thermal neutron spectrum and a once-through fuel cycle. The GIF recommends building upon Generation III+ experience with high-temperature gas-cooled reactors, such as the GT-MHR and the PBMR, and to extend the concept to incorporate the use of a fast neutron spectrum and a closed fuel cycle.

The use of a fast neutron spectrum allows the use of fertile fuels as well as fissile fuels. The successful use of depleted uranium as fuel would greatly increase the nuclear fuel resources available and so enhance the long-term sustainability of nuclear power. This together with a move to a closed reprocessing-based fuel cycle would allow for a very significant reduction in the total quantities of high-level radioactive wastes requiring management or disposal. The GIF estimates that the conversion of mined reserves into useful heat will be two orders of magnitude more efficient than with thermal spectrum gas reactors with once-through fuel cycles. Moving towards a closed fuel cycle in high-temperature gas-cooled reactors also requires some thought as to the form of nuclear fuel to be used. At present both GT-MHR and PBMR intend to make use of Triso granular fuel formed into respectively prismatic blocks or “pebbles”. Triso fuel is, however, poorly suited to reprocessing given the in-built silicon carbide barrier layer in the fuel granules. These concerns have prompted the GIF to adopt a more open-minded approach to possible fuel forms. Currently ceramic fuels and ceramic-clad fuel kernels made from actinide compounds are under consideration complementing developments in Triso fuel. Significant research will be required to establish novel fuels suitable for use at very high temperatures in a fast neutron flux.

As Fig.20.1 makes clear, the GFR concept builds upon the elegant General Atomics proposal employed in the GT-MHR for a single combined turbine and compressor shaft.

The GIF predicts that a conceptual design for an entire GFR prototype system can be developed by 2019 with an international prototype in operation by 2025.

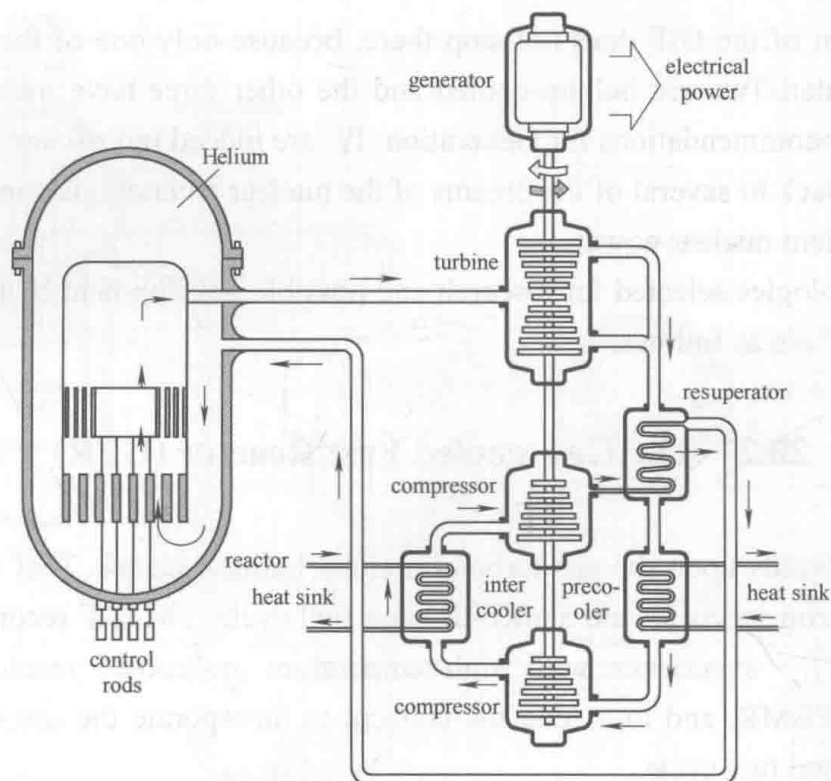


Fig.20.1 schematic of the GIF gas-cooled fast reactor

20.3 Very High Temperature Reactor System (VHTR)

The other reactor system that builds upon Generation III+ experience with high-temperature gas-cooled reactors is the very high temperature reactor system (VHTR) (see Fig.20.2).

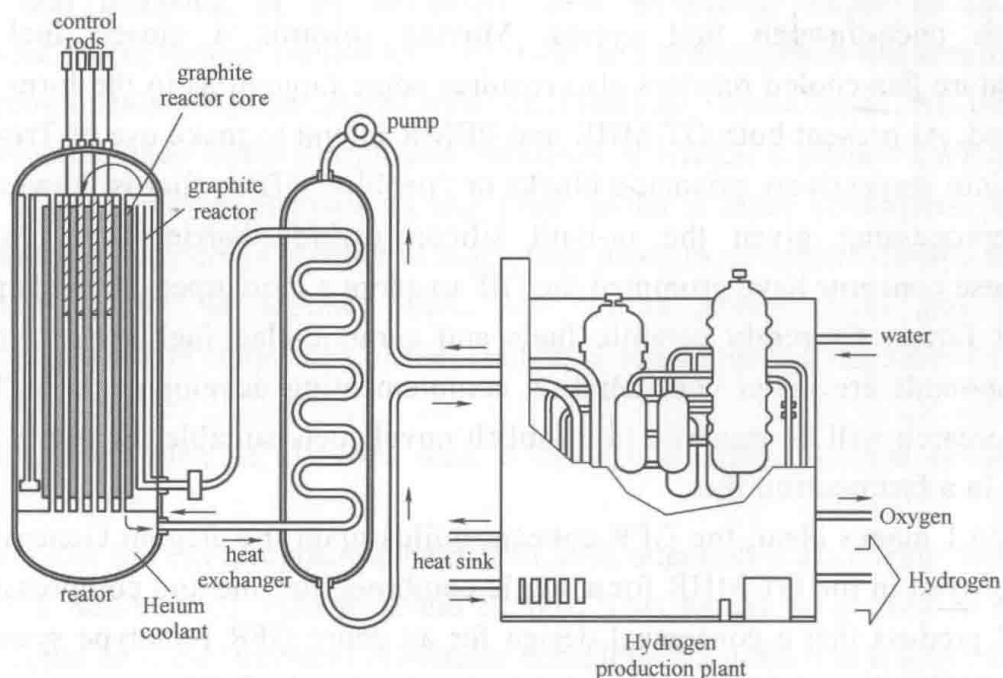


Fig.20.2 schematic of the GIF very high temperature reactor system

This proposal suggests a graphite-moderated, helium-cooled reactor with a once-through

uranium fuel cycle. Attention is also being given to the use of mixed uranium/plutonium fuel cycles. The VHTR is described as a next step in the evolutionary development of high-temperature gas-cooled reactors. The GIF Roadmap acknowledges the technical debt of the VHTR to the Japanese prismatic HTTR and the Chinese pebble bed HTR-10 as well as earlier HTGR experiments⁴. It acknowledges the current development of the PBMR and GT-MHR, but does not emphasize the fact that these two existing projects clearly fit the GIF description for the VHTR. Central to the research needs of the GIF VHTR are the properties of Triso fuel, a topic currently widely studied independent of the existence of the GIF.

The GIF adds one significant new idea to HTGR thinking in the context of the VHTR. The new idea is sufficiently imaginative that it alone takes the VHTR concept from the arena of Generation III+ firmly into the zone of Generation IV. The idea is that the VHTR could separate commercial nuclear energy from its technological bedfellow, the electricity industry.

Thus far this book has focused almost exclusively on the idea that the form and impact of a nuclear renaissance will relate inevitably to the economic and policy issues surrounding electricity generation. The GIF VHTR concept frees nuclear energy from these difficulties and allows nuclear energy directly to address a different societal need that will probably grow to prominence on a timescale similar to the timescale of Generation IV deployment, the resource depletion of fossil fuels needed for transport. The GIF Roadmap discusses the possibility that the VHTR might be used to create hydrogen to fuel a hydrogen economy without the need for wasteful energy conversion steps involving electricity.

20.4 Supercritical Water-cooled Reactor (SCWR)

The GIF acknowledges that the supercritical water-cooled reactor (SCWR) (see Fig.20.3) builds upon much experienced from the Japanese supercritical light water reactor (SCLWR). Key to the SCWR concept is that the water used to cool the reactor is above its thermodynamic critical point of 374°C and 22.1 MPa. At such high temperatures and pressures water no longer exists as two phases (liquid and steam); rather it comprises a single compressible fluid of smoothly varying density in response to changes in pressure or temperature.

The SCWR benefits from much pre-existing technological development. It benefits from experience gained with LWR design and operation (particularly in the development of BWRs). It also benefits from the development of super-critical water energy extraction in fossil-fuelled electricity generation. SCWR systems offer the prospect of significantly higher thermodynamic efficiencies than that available from conventional LWRs. The GIF Roadmap estimates that the SCWR will achieve thermal efficiencies of 44% compared with 33%~35% for conventional LWRs. Supercritical water can absorb more heat than steam per unit mass or volume. That is, its enthalpy is higher. This allows for a reduction in the coolant mass flow required and therefore a reduction in the size of the reactor coolant pumps, the piping and the containment building housing the key equipments. As the process of energy extraction deals

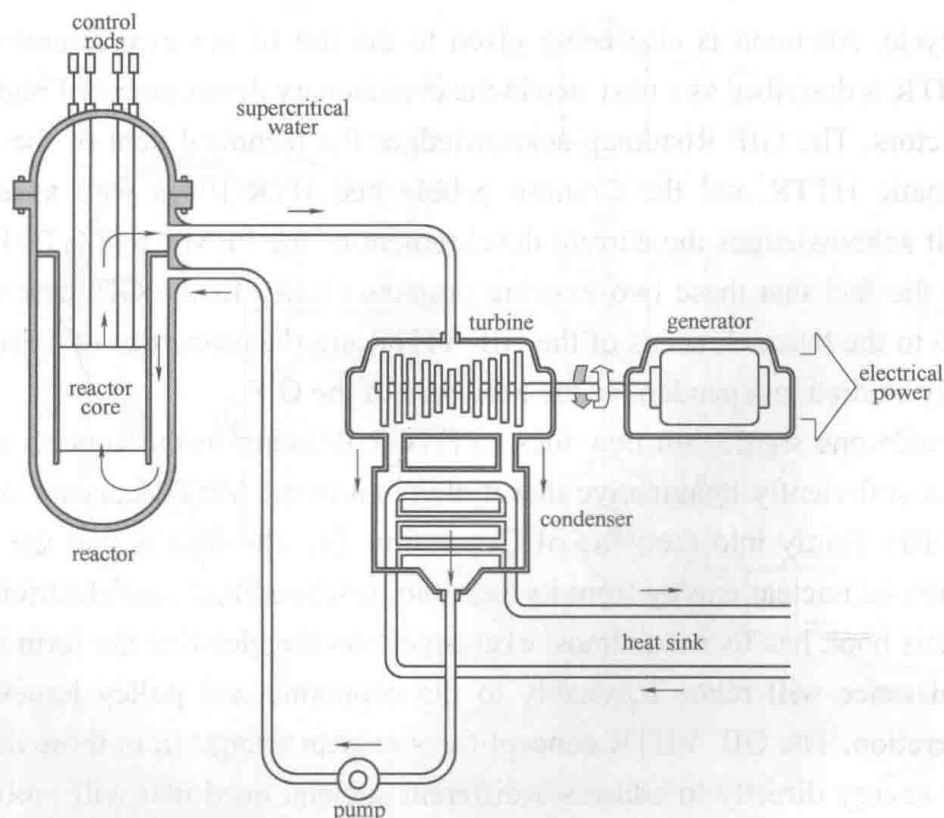


Fig.20.3 schematic of the GIF supercritical water-cooled reactor

with a single fluid phase there is no need for expensive steam dryers, steam separators, recirculation pumps and steam generators. The far greater simplicity of the SCWR results in far lower capital costs. The GIF estimates that an SCWR could be constructed for approximately US\$900 per kWe, that is for roughly half the cost of Generation III LWRs.

Design and licensing of an industrial scale SCWR is likely to be far from straightforward. There remain significant unknowns and challenges in the area of materials for nuclear components in a supercritical water environment. While the SCWR builds upon a large body of experience in the design and operation of LWRs, it pushes LWR technology, placing far larger physical stresses on the plant under normal operations. The SCWR operating pressure of 25MPa, the order of magnitude changes in coolant density and the high operating temperatures at around 500°C all require investigation and engineering development, particularly in the area of fuel claddings and structural materials, to ensure safe and reliable operations⁵.

The GIF reports that, if all goes well, a conceptual design should be available by approximately by 2016.

While initially the GIF proposes a thermal SCWR, attention is also being focused on the possibility of a fast neutron variant of the SCWR. This would be similar to the thermal SCWR, but would have no structural moderator complementing the liquid water flowing into the pressure vessel. Moving to a fast neutron spectrum would bring with it the usual advantages of lower fuel resource depletion, but at the price of far greater radiation damage to the structural elements of the reactor. The Roadmap document estimates that the doses to

the cladding and structural materials could be larger than in the thermal version by a factor of five or more.

20.5 Sodium-cooled Fast Reactor (SFR)

The concept of a sodium-cooled fast reactor (see Fig.20.4) is the most mature of all the Generation IV technologies proposed in the GIF Roadmap. Sodium-cooled fast reactor demonstrators have ranged from the US 1.1 MW_{th} EBR-1 of the 1950s to the 1200 MWe Super Phenix reactor in France in the 1980s. At various times France, Japan, Germany, the UK, the USA and Russia have pursued such programmes. There has been more than 300 reactor years of fast neutron reactor operation experience already in the whole world. At present, however, only Japan and Russia have substantial ongoing plans of their own. Most of the countries listed abandoned their efforts in the face of falling uranium prices and the high levels of technological challenge involved in developing the technology. It seems likely that SFR technologies will not become economic until such time as uranium price rises significantly. In the meantime technology innovations should be dedicated to reducing capital costs of construction, improving intrinsic safety (particularly via a greater use of passive systems), and further works on the fuel reprocessing system essential for the sustainable operation of fast reactor systems. There is much interest within GIF in moving the SFR fuel cycle towards pyrochemical reprocessing and away from aqueous approaches. This has the benefit of improving the proliferation resistance of the technology. A developed SFR would be able to play an important role in the management of current surplus materials arising from Generation II and III technologies. For example, SFR technology would have much to offer the UK in terms of disposal (and use) of the UK's legacy of separated civil plutonium. For countries with legacies arising from the operation to a once-through fuel cycle the SFR would be able to allow for the unlocking of unused energy remaining in the spent fuel. The SFR was originally strongly supported by France and the UK as a means by which to close their reprocessing-based fuel cycles. Indeed the prospect of eventual SFR Dower generation was a major driver behind the development of PU REX-based reprocessing in these countries. The fuels needed by the SFR include the plutonium mentioned above, but also ²³⁸U, which has no beneficial use in conventional thermal neutron fuel cycles.

The proposed SFR would operate at approximately 550°C. Importantly the reactor vessel would be held at a pressure only slightly above atmospheric pressure, the over-pressure required being merely that needed to allow the circulation of the liquid metal coolant. As a power station, the SFR technology could be deployed in either of two sizes: a medium sized machine in the range of 150~500 MWe, and a large-scale machine in the range of 500~1500 MWe.

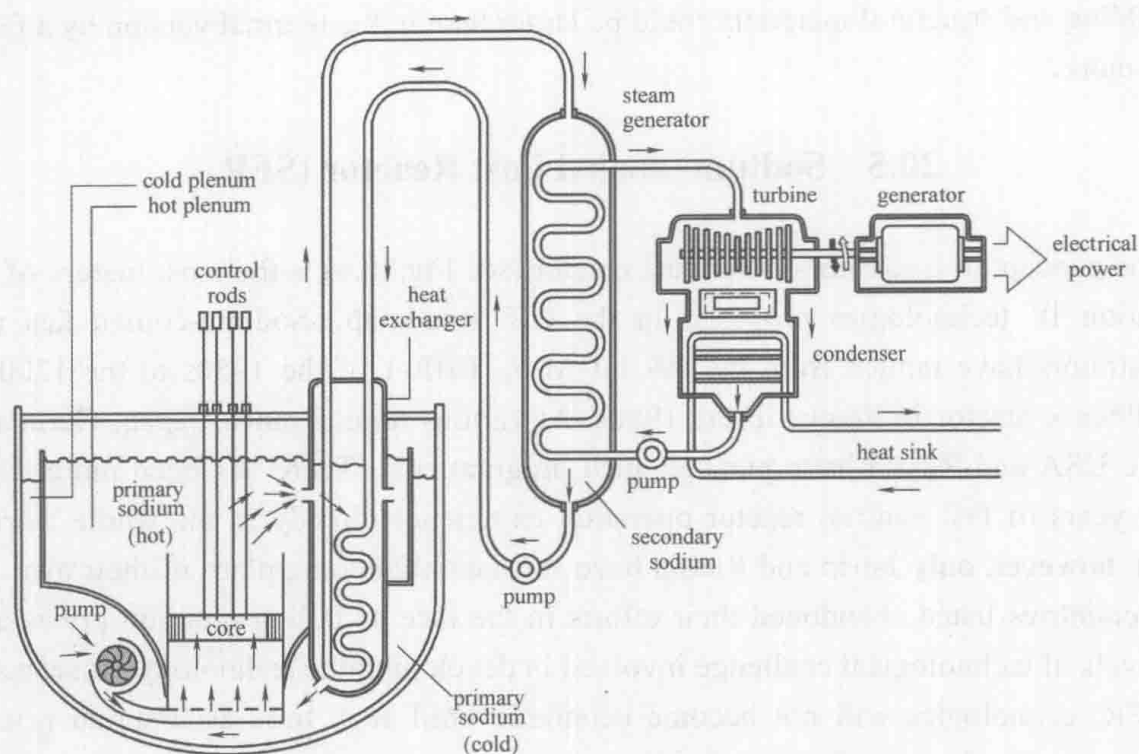


Fig.20.4 schematic of the GIF sodium-cooled fast reactor

20.6 Lead-cooled Fast Reactor (LFR)

The lead-cooled fast reactor (see Fig.20.5) borrows much from the Russian concept of the BREST fast reactor. That technology in turn benefits from much experience gained by the Soviet Navy in the use of lead-bismuth eutectic cooling in naval propulsion reactors in its Alpha class submarines. Russia is not a formal partner in the GIF although links between the Russians and the American-led GIF process do exist. Expertise with lead-cooled and lead-bismuth-cooled reactor systems among GIF member countries is restricted to the US STAR and the Japanese LSPR programmes respectively. In addition, much of the work on materials undertaken in western countries in connection with sodium-cooled fast reactor technology can most probably be applied to the LFR project.

As a fast-spectrum reactor with full actinide fuel recycle, the system has much in common with the sub-critical fast-spectrum nuclear waste transmutation technologies. The GIF restricts itself to consideration of critical fission-based systems, and hence does not include technologies such as the energy amplifier. However, it is through the LFR proposal that the greatest synergies between Generation IV and the energy amplifier will be possible.

The LFR reactor outlet temperature is likely to range between 550°C and 880°C, depending upon the use of advanced materials. If the higher temperatures can be attained (in this version the coolant would not include bismuth), the LFR will be well suited for the thermochemical production of hydrogen. It is envisaged that LFR technology could proceed via step-wise phased development with initial applications being dedicated to electricity

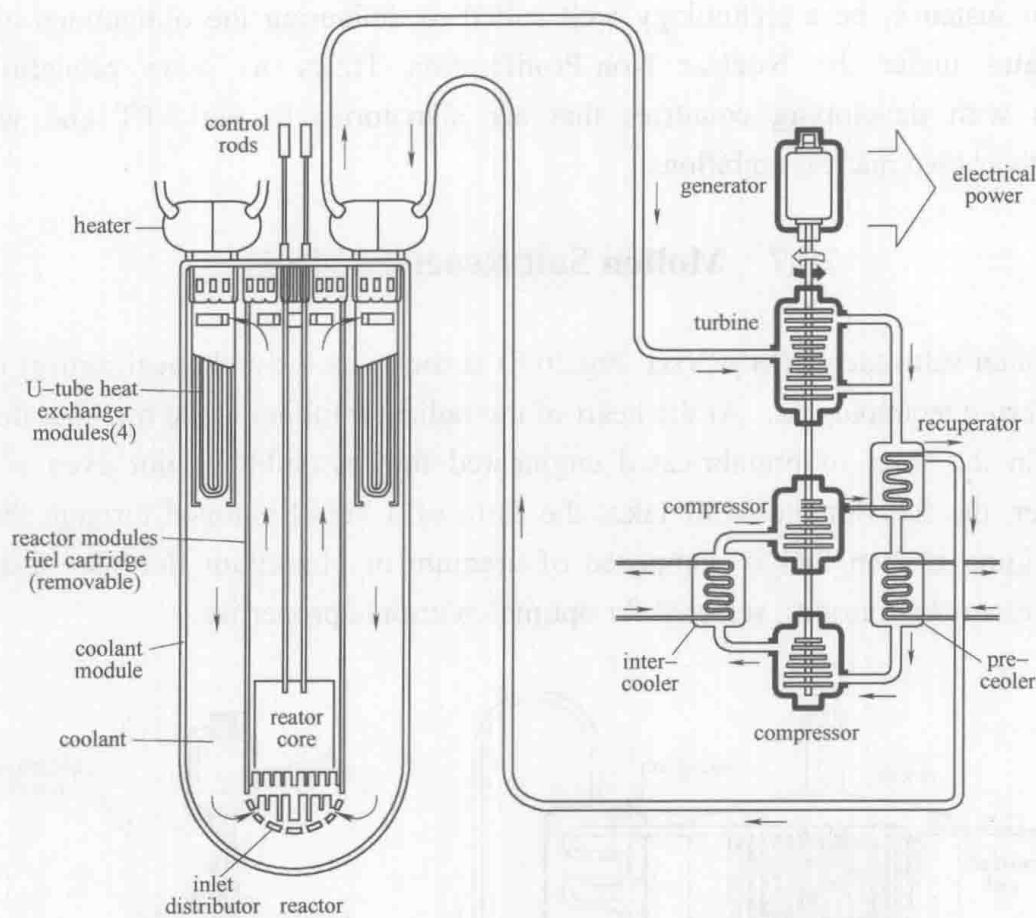


Fig.20.5 schematic of the GIF lead-cooled fast reactor

production and with later development focusing on hydrogen production.

In the short term it is proposed that the LFR would use metal alloy fuel or nitride fuel, if available. Possible metal alloy fuels and the back-end recycling of uranium, zirconium and (using the US definition) TRU waste would also benefit from likely synergies with developments occurring in the parallel SFR programme.

Nitride fuels are required for the high temperature version of the LFR. Moves to full actinide recycle in nitride fuels will require significant new research and development. In this context the GIF Roadmap recommends moves towards non-aqueous pyrochemical reprocessing.

Among the six GIF concepts, the LFR is distinguished by the possibility that it can be developed in three different sizes: a large monolithic plant rated at 1200 MW, a small to medium sized modular system at 300~400 MW, and most interestingly of all, a very small system rated at 50~150 MW with very long refueling intervals. This last system, known as the LFR “battery”, would utilize modular factory-built replaceable reactor cores. The fuel cores needing to be changed only every 15~20 years. The only refueling would be that associated with the changeover of an entire reactor core module or “battery”. In some applications one could even imagine LFR technologies being deployed with no intention to ever refuel the reactor. The small, modular battery concept is extremely well-suited to deployment in developing countries with no interest in developing their own nuclear fuel cycle infrastructure.

It would, for instance, be a technology well suited to delivering the obligations of nuclear weapons states under the Nuclear Non-Proliferation Treaty to share peaceful nuclear technologies with developing countries that are signatories to the NPT and who have constrained their own nuclear ambitions.

20.7 Molten Salt Reactor (MSR)

The molten salt reactor (MSR)(see Fig.20.6) is the most technological radical of all the six GIF preferred technologies. At the heart of the radical thinking is the fact that the nuclear fuel is not in the form of prefabricated engineered fuel assemblies; nor even is the fuel solid. Rather, the fuel for the MSR takes the form of a liquid pumped through the reactor core. The liquid molten fuel is composed of uranium or plutonium fluorides dissolved in sodium or zirconium fluorides, selected for optimal neutronic properties.

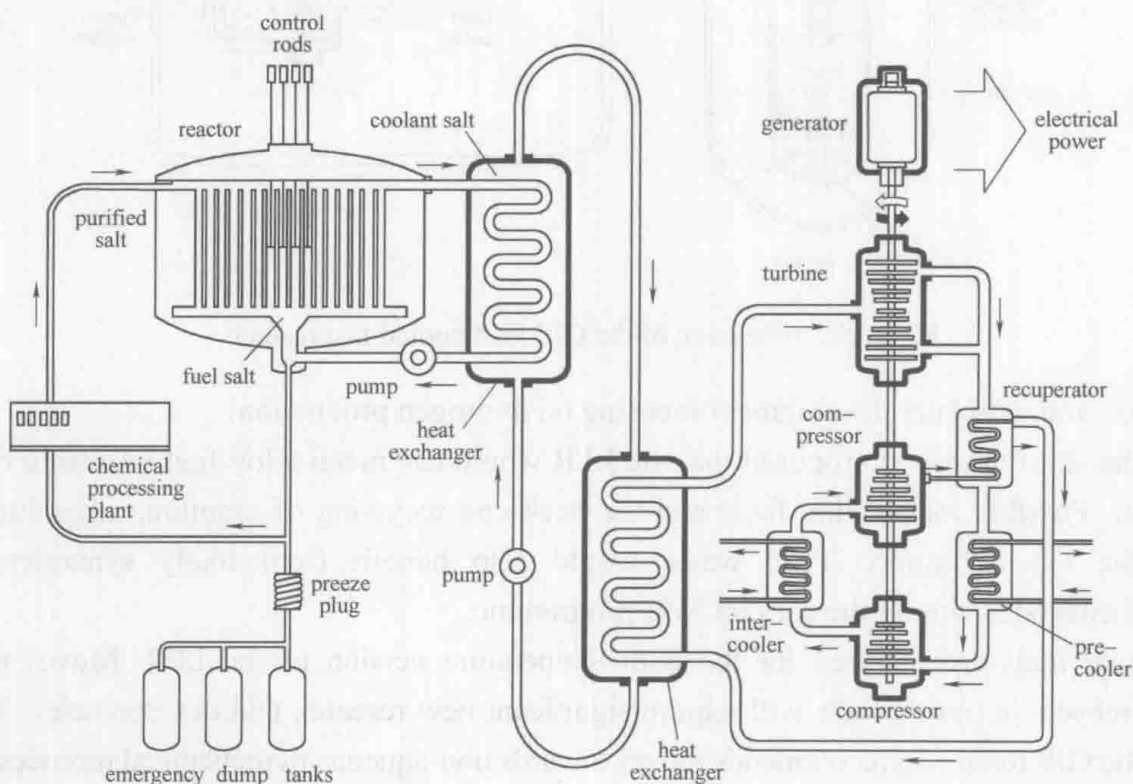


Fig.20.6 schematic of the GIF molten salt reactor

The GIF MSR would be characterized by the following attributes: it would be a large-scale (1000 MWe) base load generator; it would utilize graphite moderated epithermal neutrons; it would operate at high temperature (700~850°C) permitting hydrogen production; it could operate flexibly with a range of fuel compositions. In particular, the reactor could benefit from fertile isotopes in the fuel mix, such as uranium-238 or thorium-232. Also the MSR is well suited to actinide disposal, although this would require offsite fuel processing. Otherwise it would be expected that on-site fuel processing would be sufficient to maintain healthy neutronics of the salt such that it could be used for several years without being

changed. The MSR does not produce significant actinide wastes owing to very high burn-up. This offers good proliferation resistance and excellent primary fuel resource efficiency.

It is believed that, despite its radical and somewhat experimental conceptual status, the MSR concept offers good prospects for reactor safety. The reactor generates almost negligible vapors and so can be regarded as a single liquid phase at all points. Over long periods the build up of radioactive gases could be cause for some concern, but the volume of gas should never be sufficient to jeopardize the fundamental design concept of a uniform liquid fuel. In the event of a serious problem, a “freeze plug” at the base of the reactor would melt passively allowing the reactor fuel to drain safely to a set of emergency dump tanks⁶ (see Fig.20.6).

The MSR is likely to employ two separate molten salt circuits. The primary circuit would contain the nuclear fuel and resulting fission products, while a secondary circuit would exchange heat with the primary circuit and then pass that heat to a second heat exchanger. The heat exchanger could pass the heat to a gas coolant that would pass through a Brayton cycle gas turbine or alternatively pass it to a system designed to catalytically generate hydrogen in a high temperature reaction. The use of primary and secondary molten salt circuits is believed to offer improved safety and easier decommissioning and maintenance.

Although a series of research molten salt reactor did operate successfully in the United States in the 1950s and 1960s. These reactors were originally developed with the intention that they might one day be used to power aircraft, hence the name of the first such reactor: the Aircraft Reactor Experiment. At the end of the 1990s, the second such American reactor, the 8 MW_{th} Molten Salt Reactor Experiment, proved a concern for decommissioning after it had stood idle for almost 30 years. It required attention to remove the radioactive salts that had lain undisturbed in tanks close to the reactor throughout that time. The challenges for safely decommissioning this aged facility were substantial. Despite of the low vapor pressure in molten salt reactors over the long period there had been significant build-up of radio-active uranium hexafluoride gas and hazardous fluorine. In addition, special care was required to avoid criticality incidents in defuelling the facility.

These difficulties relate, in part, to a large set of unknowns regarding the fundamental design and operation of a new MSR concept. Much remains to be understood about the long-term fuel chemistry, the materials behavior of reactor components, risks of metal clustering and precipitation, and the necessary salt processing, both on-site and offsite.

20.8 Generation IV and the Hydrogen Economy

The potential for using nuclear energy for hydrogen production has been discussed. Generation IV International Forum strongly endorses this vision. Of the six preferred technologies in the GIF Roadmap four are deemed suitable for thermo-chemical hydrogen production. Three of these four have expected maximum operating temperatures around 800°C. Maximum operating temperatures of around 800°C are problematically low, given that the

efficiency of thermo-chemical hydrogen production increases rapidly with increasing temperature. These three concepts (GFR, LFR and MSR) are primarily planned to be electricity generating technologies.

The truly exciting technology for hydrogen production is the very-high-temperature reactor. With expected operating temperatures in the range of 1000°C, this technology has the potential to be led by the needs and opportunities of the hydrogen economy. The GIF estimates that the VHTR could reach the stage of a conceptual design by 2015. This timing is well suited to the probable needs of those pursuing hydrogen as an energy vector for vehicular transport. If the links binding nuclear energy and the electricity industry can be weakened, then it seems likely that a wider range of more ambitious new nuclear energy technologies can be developed. Generation IV's emphasis on hydrogen is an important positive step in that direction.

Self-test

1. It has become standard to describe the ____ developments in nuclear power plant engineering.
A. successive B. continue C. large D. fast
2. In this book we do not attempt to survey all possible nuclear power designs ____ contribute to a nuclear renaissance.
A. which can B. which C. that might D. that
3. Generation IV nuclear energy systems ____ for offsite emergency response.
A. don't need B. eliminate the need
C. have a little need D. need
4. Three of the six reactors being proposed by the GIF are ____.
A. gas cooled reactors B. LFR
C. fast reactors D. new type reactors
5. The use of a fast neutron spectrum allows the use of fertile fuels as well as ____.
A. fissile fuels B. UO₂ fuels C. Triso fuels D. plutonium fuels
6. These concerns have prompted the GIF ____ a more open-minded approach to possible fuel forms.
A. play B. introduce C. adopt D. to adopt
7. The VHTR is described as a next step in the ____ of high-temperature gas-cooled reactors.
A. development B. evolutionary development
C. evolutionary D. field
8. Design and licensing of an industrial scale SCWR is likely to be ____.
A. far away B. near
C. no much difficulty D. far from straightforward
9. A developed SFR would be able to ____ in the management of currently surplus

materials arising from Generation II and III technologies.

- A. play an important role B. make it easy
C. production D. be use

10. The small modular battery concept is extremely_____to deployment in developing countries.

- A. well suited B. adopted C. well-suited D. available

11. The MSR does not produce significant actinide wastes _____very high burn-up.

- A. for B. due to C. for it has D. owing to

12. The use of primary and secondary molten salt circuits is_____offer improved safety and easier decommissioning and maintenance.

- A. to B. believed to C. possible D. for

关 键 字 解 释

Actinides: A group name for the series of elements of atomic numbers 89~104 (inclusive), which indicates that they have similar properties to actinium.

锕系: 原子序数从 89 到 104 的整个系列元素的集合名字, 它表示这些元素的性质与锕相似。

AGR: Advanced Gas-cooled Reactor.

先进气冷反应堆。

Fluorine: The lightest halogen and the most electronegative element, symbol of F, which is chemically highly reactive and thus never found as the free element.

氟: 最轻的卤族元素, 也是电负性最强的元素, 用符号 F 表示, 它具有很高的化学活性, 因此不可能以自由元素的状态存在。

GT-MHR: Gas Turbine-Modular Helium Reactor. The reactor is cooled by helium, then the heated helium is introduced to gas turbine, which drives electric generator.

气体透平—模块式氦冷反应堆: 反应堆由氦气冷却, 被加热的氦气送往气体透平, 由透平驱动发电机。

HTTR: High-Temperature Test Reactor.

高温实验堆。

IRIS: International Reactor Innovative and Secure. It is a small integral PWR designed by Westinghouse.

国际革新及安全型反应堆: 是由西屋公司设计的小型一体化压水反应堆。

PBMR: the Pebble Bed Modular Reactor. The fuel for the reactor consists of spherical fuel assemblies a little larger than a billiard ball. These are the “pebbles” of the pebble bed concept. Each pebble contains around 15 000 Triso fuel particles.

球床模块式反应堆: 这种反应堆的燃料由比台球大一点的圆球状燃料组件组成, 这便是球床反应堆的燃料球, 每个这样的燃料球由大约 15 000 个 Triso 型燃料颗粒组成。

Technocrat: Often derogates a highly skilled scientific specialist in charge of the

organization of a country industry, etc.

技术官员: 指担任行政官职的技术专家(一般是贬义)。

VHTR: Very-High-Temperature Reactor.

超高温反应堆。

词 汇

acknowledge [æk'nɒlɪdʒ] *vt.* 承认, 确认

actinides ['æktinaɪdɪz] *n.* 锕系元素

advisory [əd'vaɪzəri] *a.* 顾问的

albeit [ɔ:l'bi:t] *conj.* 虽然, 即使

ambition [æm'biʃən] *n.* 雄心

ambitious [æm'biʃəs] *a.* 有雄心的

aqueous ['eɪkwɪəs] *a.* 含水的, 水成的

arguably ['ɑ:gjuəbli] *ad.* 可论证的

assess [ə'ses] *vt.* 估计

assume [əsju:m] *vt.* 假设

attribute [ə'tribju:t] *v.* 属于

aversion [ə'veɜ:ʃən] *n.* 不喜欢

bedfellow ['bedfeɪləu] *n.* 伙伴

bismuth ['bɪzməθ] *n.* 铋

catalytically *ad.* 催化(作用)

charter ['tʃɑ:tə] *n.* 合同, 许可证

civil ['sɪvɪl] *a.* 民用的

clause [klaʊz] *n.* 条款

cluster ['klʌstə] *n.* 束, 族

complement ['kɒmplɪmənt] *n.* 补充

compromise ['kɒmprəmaɪz] *n.&v.* 折中

concomitant [kən'kɒmɪtənt] *a.* 伴生的

conscious ['kɒnʃəs] *a.* 有意识的, 自觉的

consensus [kən'sensəs] *n.* 共识

context ['kɒntekst] *n.* 前后关系

decommission [ˌdɪkə'mɪʃən] *vt.* 退役

debt [det] *n.* 债务

democracy [dɪ'mɒkrəsi] *n.* 民主

dismiss [dɪs'mɪs] *vt.* 解散

endorse [ɪn'dɔ:s] *vt.* 签署

enthalpy [en'θælpɪ] *n.* 焓

envisage [ɪn'vɪzɪdʒ] *vt.* 设想, 展望

- epithermal [epi'thə:məl] *a.* 超热的
erosion [i'rəʊʒən] *n.* 腐蚀, 削弱
eutectic [ju:'tektik] *n.* 共晶
excel [ik'sel] *v.* 胜过
exclusively *adj.* 独占的, 专门的
expertise [ekspə'ti:z] *n.* 专家
explicit [iks'plisit] *a.* 明白, 显然
formulation [fɔ:mju'leɪʃən] *n.* 公式化
forum [fɔ:rəm] *n.* 论坛
granular ['grænjulə] *a.* 粒状的
granule ['grænju:l] *n.* 颗粒
hurdle ['hə:dl] *n.* 栏杆
idiosyncratic [idiəsin'krætik] *a.* 特别的
idle [aɪdl] *a.* 不工作, 闲置的
infrastructure ['ɪnfre'strʌktʃə] *n.* 基础设施
intervene [ɪntə'vi:n] *v.* 介入
intrinsic [ɪn'trɪnsɪk] *a.* 内在的, 固有的
issue ['ɪʃ(j)u:] *v. & n.* 收益; 结果
jeopardize ['dʒəpədaɪz] *vt.* 危及
kernel ['kə:nəl] *n.* 核
legacy ['leɪɡəsi] *n.* 传统
liberal ['libərəl] *a.* 充足的
mature [mə'tjuə] *a.* 成熟的
monolithic ['mɒnəliθɪk] *a.* 整体的
naive [na:'i:v] *a.* 自然的
near-term *a.* 最近将来的, 短期的
neutronics *n.* 中子物理学
nitride ['naɪtraɪd] *n.* 氮化
novel ['nɒvəl] *n.* 新的
obligation [ɒbli'geɪʃən] *n.* 义务
omission [əu'mɪʃən] *n.* 省略, 疏忽
option ['ɒpʃən] *n.* 选择
pebble ['pebl] *n.* 卵石
pillar ['pɪlə] *n.* 小
precipitation [prɪsɪpi'teɪʃən] *n.* 沉淀
preparatory [pri'pærətəri] *a.* 准备的
prerequisite ['pri:rekwɪzɪt] *a.* 先决条件
prioritization *n.* 优先
prismatic [prɪz'mætɪk] *a.* 棱柱形的

- proliferation [prəʊlɪfə'reɪʃən] *n.* 扩散
- prominence [ˈprɒmɪnəns] *n.* 突起, 显露
- promise [ˈprɒmɪs] *n.* 约定
- prototype [ˈprəʊtəʊtaɪp] *n.* 原型
- pursue [pə'sju:] *vt.* 从事, 采取
- pyrochemical [ˌpaɪrəʊ'kemɪkəl] *a.* 高温化学变化的
- radical [ˈrædɪkəl] *a.* 根本的
- renaissance [rə'neɪsəns] *n.* 复兴
- resistant [rɪ'zɪstənt] *a.* 抵抗
- retrievable [rɪ'tri:vəbl] *a.* 可恢复
- roadmap *n.* 安排, 排定
- signatory [ˈsɪgnətəri] *n.* 缔约国
- silicon [ˈsɪlɪkən] *n.* 硅
- societal [sə'saɪətəl] *a.* 社会的
- span [spæn] *n.* 跨度
- stepwise [ˈstepwaɪz] *a&ad.* 逐步的(地)
- stewardship [ˈstjuədʃɪp] *n.* 管理工作, 代管
- successive [sək'sesɪv] *a.* 连续的
- supersede [sɪjupə'si:d] *vt.* 代替
- surplus [ˈsə:pləs] *a.* 过剩的
- sustainability *n.* 可持续性
- synergy [ˈsɪnə:dʒi] *n.* (最佳) 合作
- synonymous [sɪ'nɒnɪməs] *a.* 同义的
- technocracy [tek'nɒkrəsi] *n.* 专家管理
- temper [ˈtempə] *v. & n.* 回火, 锻炼
- terrorism [ˈterərɪzəm] *n.* 恐怖主义
- treaty [ˈtri:tɪ] *n.* 条约
- underpinning *n.* 基础
- variant [ˈvɛəriənt] *a.* 不同的
- vector [ˈvektə] *n.* 航向, 矢量
- vehicular [vi'hɪkjulə] *a.* 车用的
- whatsoever=whatever 无论什么
- be synonymous with 与……的意义是一样的

课文注释

1. It is unlikely that North America and...Generation III+ technologies.

句中 unlikely 是不大可能的意思, 具有否定的意思, 它否定后面从句的意思; new build 这里指新建造核电站。全句译为: 如果没有根据三代和三代加技术建造的大量核电站, 北美

和欧洲在 2020 年后也不太可能推广四代核电站的技术。

2. Prior to the launch of the GIF, the US DOE had set up a Nuclear Energy Research Advisory Committee.

句中 Prior to 表在……之前, GIF 是第四代反应堆国际论坛, DOE (Department Of Energy) 是能源部。全句译为: 在第四代反应堆国际论坛发起之前, 美国能源部设立了一个核能研究顾问委员会。

3. Generation IV priorities, having been...falling at the final hurdle.

句中 priorities 在这里是指时间上的优先顺序或优先权, 全句译为: 第四代反应堆优先权在传统的工程引导的设计中得以推进, 并由制定政策的专家所定型, 风险就落到了最后的障碍上。

4. The GIF Roadmap acknowledges...earlier HTGR experiments.

句中 acknowledges 是承认和确认的意思, 全句译为: GIF 规划承认日本的棱柱形高温试验堆和中国的球床堆 HTR-10 以及早期的高温气冷堆实验装置在技术上与 VHTR 还有差距。

5. The SCWR operating pressure of 25 MPa, the order...and reliable operations.

句中 the order of magnitude changes in coolant density 是说冷却剂的密度有数量级的改变, engineering development 是工程开发。全句译为: 超临界水堆的运行压力为 25MPa, 冷却剂的密度大小有数量级的改变, 运行温度大约为 500℃, 所有这些都需要研究和工程开发, 特别是在燃料包壳和结构材料方面, 这样才能保证安全和可靠运行。

6. In the event of a serious problem, a “freeze plug” ... of emergency dump tanks.

句中 freeze plug 是指熔盐结块形成的塞子, a set of 是一组的意思。全句译为: 在出现问题时, 在反应堆底部形成的熔盐结块能够非能动地熔化, 使反应堆燃料安全地排到一组应急排放箱中。

附录 I Self-test 习题答案

Unit 1

1. B 2. C 3. A 4. A 5. B 6. C 7. D 8. A 9. D 10. B 11. B 12. C

Unit 2

1. B 2. A 3. D 4. A 5. D 6. C 7. C 8. D 9. B 10. A 11. B 12. B

Unit 3

1. D 2. B 3. C 4. A 5. B 6. C 7. A 8. D 9. B 10. C 11. A 12. C

Unit 4

1. C 2. A 3. B 4. D 5. B 6. A 7. B 8. D 9. B 10. A 11. C 12. A

Unit 5

1. A 2. D 3. B 4. B 5. C 6. A 7. D 8. C 9. A 10. C 11. D 12. C

Unit 6

1. A 2. D 3. B 4. C 5. C 6. A 7. B 8. C 9. D 10. B 11. A 12. C

Unit 7

1. C 2. D 3. B 4. A 5. B 6. D 7. A 8. C 9. C 10. B 11. A 12. D

Unit 8

1. B 2. A 3. C 4. D 5. B 6. A 7. C 8. B 9. D 10. C 11. A 12. D

Unit 9

1. C 2. B 3. A 4. D 5. C 6. B 7. D 8. A 9. A 10. D 11. C 12. B

Unit 10

1. A 2. C 3. B 4. D 5. B 6. D 7. C 8. C 9. A 10. D 11. C 12. D

Unit 11

1. C 2. B 3. B 4. D 5. C 6. B 7. A 8. D 9. C 10. D 11. D 12. A

Unit 12

1. B 2. A 3. D 4. B 5. C 6. C 7. A 8. C 9. B 10. A 11. D 12. C

Unit 13

1. C 2. B 3. A 4. D 5. B 6. C 7. D 8. A 9. B 10. C 11. D 12. B

Unit 14

1. B 2. A 3. D 4. B 5. C 6. D 7. D 8. A 9. B 10. C 11. A 12. B

Unit 15

1. C 2. B 3. A 4. D 5. A 6. D 7. C 8. C 9. B 10. A 11. D 12. B

Unit 16

1. C 2. B 3. A 4. D 5. B 6. C 7. A 8. D 9. B 10. C 11. C 12. A

Unit 17

1. B 2. D 3. D 4. A 5. C 6. C 7. A 8. B 9. D 10. A 11. C 12. B

Unit 18

1. B 2. C 3. A 4. C 5. D 6. B 7. D 8. A 9. B 10. C 11. A 12. D

Unit 19

1. C 2. B 3. D 4. C 5. A 6. B 7. B 8. D 9. C 10. A 11. D 12. B

Unit 20

1. A 2. C 3. B 4. C 5. A 6. D 7. B 8. D 9. A 10. C 11. D 12. B

附录 II 词 汇 表

A

- abandon [ə'bændən] *v.* 放弃
- abbreviate [ə'bri:vieit] *v.* 缩写
- abnormal [æb'nɔ:məl] *a.* 不正常的
- accessible [æk'sesəbl] *a.* 可以接近的
- accumulator [ə'kju:mjuleitə] *n.* 储存器
- acknowledge [æk'nɒlidʒ] *vt.* 承认, 确认
- acoustic [ə'ku:stik] *a.* 声音的
- actinides ['æktinaidz] *n.* 锕系元素
- active ['æktiv] *a.* 主动的, 有源的
- actuate ['æktjueit] *v.* 驱动
- actuation [æktju'eɪʃən] *n.* 动作, 触发
- actuator ['æktjueitə] *n.* 制动器, 执行机构, 传动机构、制动机构
- acute [ə'kju:t] *a.* 急剧的
- address [ə'dres] *vt.* 交, 寻址, 委托
- adequate ['ædikwit] *a.* 足够的
- adequately ['ædikwitli] *ad.* 充分地
- advisory [əd'vaizəri] *a.* 顾问的
- aerodynamic [ɛərəʊdaɪ'næmik] *a.* 空气动力学的
- agent ['eidʒənt] *n.* 介质
- airtight ['ɛətaɪt] *a.* 气密的
- albeit [ɔ:l'bi:it] *conj.* 虽然, 即使
- align [ə'lain] *v.* 与……一起(合作)
- alignment [ə'lainmənt] *n.* 成直线
- allay [ə'lei] *v.* 缓和
- allotropic [æləu'trɒpik] *a.* 同素异形的
- altitude ['æltɪtju:d] *n.* 高度
- aluminium [ə'lju:minjəm] *n.* 铝
- ambient ['æmbiənt] *a.* 周围的, 环境的
- ambition [æm'biʃən] *n.* 雄心
- ambitious [æm'biʃəs] *a.* 有雄心的
- amenable [ə'mi:nəbl] *a.* 顺从的, 适合于……的
- ample ['æmpl] *a.* 足够的
- amplitude ['æmplɪtju:d] *n.* 振幅
- analogous [ə'næləgəs] =similar *a.* 类似的

- analogy [ə'nælədʒi] *n.* 类似
 anisotropic [ə,naisəu'trɒpɪk] *a.* 各向异性的
 annihilate [ə'naɪəleɪt] *v.* 湮灭
 annunciator [ə'nʌnʃieɪtə] *n.* 信号器
 anticipate [æn'tɪsɪpeɪt] *v.* 预先考虑
 antifreeze ['ænti'fri:z] *n.* 防冻液
 antimony ['æntiməni] *n.* 锑
 anxiety [æŋ'zaɪəti] *n.* 担心, 忧虑
 appendix [ə'pendɪks] *n.* 附录
 appreciable [ə'pri:ʃiəbl] *a.* 可观的
 approach [ə'prəʊtʃ] *n.* 方法, 途径
 approval [ə'pru:vəl] *n.* 认可, 批准
 aqueous ['eɪkwɪəs] *a.* 含水的, 水成的
 architect ['ɑ:kitekt] *n.* 建筑师
 arguably ['ɑ:gjuəbli] *ad.* 可论证的
 argument ['ɑ:gju:mənt] *n.* 论证, 讨论
 atm. =atmospheric pressure 大气压
 array [ə'rei] *n.* 组列, 阵列
 assemblage [ə'semblɪdʒ] *n.* 集合, 总体
 assembly [ə'sembli] *n.* 组合, 组合件; 组件, 系统
 assess [ə'ses] *vt.* 估计
 assessment [ə'sesmənt] *n.* 估计, 评定
 assume [ə'sju:m] *vt.* 假设
 attenuation [ə,tenju'eɪʃən] *n.* 衰减
 attributable [ə'trɪbjutəbl] *a.* 基于……的
 attribute [ə'trɪbjʊ:t] *v.* 属于
 augment [ɔ:g'ment] *v.* 增大, 扩增
 auxiliary [ɔ:g'ziljəri] *a.* 辅助的; *n.* 辅机
 availability [ə,veɪlə'bɪlɪti] *n.* 可利用性
 aversion [ə'və:ʃən] *n.* 不喜欢
 axial ['æksɪəl] *a.* 轴向的
 azimuthal ['æzi,mʌθəl] *a.* 方位角的

B

- baffle ['bæfl] *n.* 隔板
 bank [bæŋk] *n.* 棒系, 管束
 barely ['beəli] *ad.* 几乎没有
 barite(s) [bə'reɪt(s)] *n.* 重晶石(混凝土)
 barium ['beəriəm] *n.* 钡

- barrel ['bærəl] *n.* 吊篮, 桶
barrier ['bæriə] *n.* 屏障
baseline *n. & a.* 基线; 基本的
beam [bi:m] *n.* 拉杆, 梁
bedfellow ['bed,feləu] *n.* 伙伴
beryllium [bə'riljəm] *n.* 铍
bid [bid] *v. & n.* 出价, 投标, 表示
bidirectional [ˌbaɪdi'rekʃənəl] *a.* 双向的
billiard ['biljəd] *a.* 台球的
binary ['baɪnəri] *a.* 二元的
biological [baɪə'lɒdʒɪkəl] *a.* 生物(学)的
bismuth ['bɪzməθ] *n.* 铋
blackout ['blækaut] *n.* 停电, 关闭
blind [blaɪnd] *a.* 封闭的
blunt nose *n.* 钝头
board [bɔ:d] *n.* 管理局委员会, 盘, 屏
bombard ['bɒmbɑ:d] *v.* 轰击
bombardment [bɒm'ba:dmənt] *n.* 轰击
bonnet ['bɒnɪt] *n.* 阀盖
booster ['bu:stə] *n.* 增强, 调升
borated ['bɔ:reɪtɪd] *a.* 用硼酸处理过的, 加硼的
bore[bɔ:] *n.* 孔径, 孔
boron ['bɔ:rən] *n.* 硼
boss [bɒs] *n.* 凸台, 浮凸
bounce [baʊns] *v. & n.* 弹, 反冲
boundary ['baʊndəri] *n.* 边界
braze [breɪz] *v.* 铜焊
breeding ['bri:diŋ] *n.* 增殖
bremsstrahlung ['brems'tra:ləŋ] *n.* 韧致辐射
brittle ['brɪtl] *a.* 脆的
broach [brəʊtʃ] *v.* 钻孔
bucket ['bʌkɪt] *n.* 叶片
bulk [bʌlk] *n.* 大块, 大多数
bundle ['bʌndl] *n.* 棒束, 组件
bypass ['baɪpɑ:s] *n.* 旁通, 支路

C

- cadmium ['kædmiəm] *n.* 镉
californium [kæli'fɔ:niəm] *n.* 钚
canal [kə'næl] *n.* 通道

- capital ['kæpitəl] *a.* 基本的
- capsule ['kæpsjʊl] *n.* 腔, 囊, 密封壳
- carbide ['kɑ:baid] *n.* 碳化物
- catalytic [kætə'litik] *a.* 催化的
- catalytically *ad.* 催化 (作用)
- caustic ['kɔ:stik] *a.* 苛性的
- cavitation [kævi'teɪʃən] *n.* 气蚀
- cavity ['kæviti] *n.* 内腔, 坑
- centrifuge ['sentrɪfju:dʒ] *n.* 离心
- ceramic [si'ræmik] *a.* 陶瓷的
- Charpy(impact) [tʃɑ:pi] *n.* 摆锤冲击
- charter ['tʃɑ:tə] *n.* 合同, 许可证
- chiller ['tʃilə] *n.* 冷却装置
- chloride ['klɔ:raid] *v.* 氯化物
- choke [tʃəuk] *n.* 阻塞
- chromium ['krəʊmjəm] *n.* 铬
- circumference [sə'kʌmfərəns] *n.* 圆周
- circumferential [sə'kʌmfə'renʃəl] *a.* 环形的, 周边的, 圆周的
- cite [saɪt] *v.* 引用, 提到
- civil ['sɪvəl] *a.* 土木, 民用的
- claim [kleɪm] *n. & v.* 要求, 申请
- clause [klo:z] *n.* 条款
- clearance ['kliərəns] *n.* 间隙
- clump [klʌmp] *n.* 群
- cluster ['klʌstə] *n.* 束, 族
- clutch [klʌtʃ] *n.* 离合器
- cobalt ['kəʊbɔ:lt] *n.* 钴
- code [kəʊd] *n.* 法规, 规范
- coincidence [kəʊ'ɪnsɪdəns] *n.* 吻合, 同时发生
- cold-worked *a.* 冷加工的
- collaborate [kə'læbəreɪt] *v.* 合作
- collapse [kə'læps] *v. & n.* 消失, 破裂, 坍塌
- collapsible [kə'læpsəbl] *a.* 可分拆的
- collision [kə'lɪʒən] *n.* 碰撞
- coma ['kəʊmə] *n.* 昏迷
- commence [kə'mens] *v.* 开始
- commissariat [kə'mɪ'sɛəriət] *n.* 给养
- commission [kə'mɪʃən] *n.* 投入运行
- commissioning *n.* 试运行
- commit [kə'mɪt] *v.* 委托, 使承担任务

- commitment [kə'mitmənt] *n.* 许诺, 保证
- competition [kəm'pi'tiʃən] *n.* 竞争
- complement ['kɒmplimənt] *n.* 补充
- comply [kəm'plai] *v.* 同意, 遵守
- component [kəm'pəʊnənt] *n.* 元素, 组成部分
- composition [kəm'pə'ziʃən] *n.* 组成, 结构
- compound ['kɒmpaʊnd] *v. & n.* 复合
- compounding *n.* 复合, 综合
- comprise [kəm'praiz] *v.* 由……组成
- compromise ['kɒmprəmaiz] *n. & v.* 折中, 妥善处理, 折中, 妥协
- conceivable [kən'si:vəbl] *a.* 可能的
- concentration [ˌkɒnsən'treɪʃən] *n.* 浓度
- concentric [kən'sentrik] *a.* 同心的
- concomitant [kən'kɒmitənt] *a.* 伴生的
- concrete ['kɒnkri:t] *n.* 混凝土
- concurrence [kən'kʌrəns] *n.* 同意
- condense [kən'dens] *v.* 冷凝
- conditioning [kən'diʃəniŋ] *n.* 处理
- conductivity [ˌkɒndʌk'tiviti] *n.* 传导率
- conduit ['kɒndit] *n.* 导管
- configuration [kən'figjʊ'reiʃən] *n.* 结构, 配置, 形状, 方位
- congestion [kən'dʒestʃən] *n.* 拥挤
- conjecture [kən'dʒektʃə] *n. & v.* 推测
- conscious ['kɒnʃəs] *a.* 有意识的, 自觉的
- consensus [kən'sensəs] *n.* 共识, 一致同意
- consequence ['kɒnsikwəns] *n.* 后果
- conservation [ˌkɒnsə'veiʃən] *n.* 自然资源保护, 保守
- conservative [kən'sə:vətiv] *a.* 保守的
- consistency [kən'sistənsi] *n.* 一致性
- consistent [kən'sistənt] *a.* 一致的
- consistently [kən'sistəntli] *ad.* 一贯地
- consortium [kən'sɔ:tʃəm] *n.* 合作, 联合
- constituent [kən'stitjuənt] *a.* 组成的
- constraint [kən'streint] *n.* 制约
- containment [kən'teinmənt] *n.* 安全壳
- contaminate [kən'tæmineit] *v.* 污染
- contemplate ['kɒntempleit] *v.* 考虑, 预期
- context ['kɒntekst] *n.* 前后关系
- contour ['kɒntʊə] *n.* 形状, 外形

- contrary ['kɒntrəri] *v.* 与……相反
 contrive [kən'traiv] *v.* 竟然弄到……地步, 设法
 conventional [kən'venʃənəl] *a.* 传统的, 常规的
 converging [kən've:dʒɪŋ] *n.* 渐缩的
 corium ['kɔ:riəm] *n.* 堆芯熔化物
 correlation [ˌkɒrɪ'leɪʃən] *n.* 关系式
 corrosion [kə'rəʊʒən] *n.* 腐蚀
 coast down [kəustdaʊn] *n.* 惰走, 下降
 coulomb ['ku:lɒm] *n.* 库(仑)
 counteract [ˌkauntə'rækt] *vt.* 抵抗, 阻碍, 减少, 反作用
 countermeasure ['kauntə'meʒə] *n.* 干扰
 course [kɔ:s] *n.* 一层, 一行, 过程
 crack [kræk] *v. & n.* 破裂
 cramp [kræmp] *n.* 绞痛
 creep [kri:p] *v. & n.* 蠕变
 crescent-shaped *a.* 月牙形的
 crevice ['krevis] *n.* 裂缝, 缝隙
 critical ['kritikəl] *a.* 临界
 criticality [kriti'kælitɪ] *n.* 临界(性, 状态)
 crossover ['krɒs,əʊvə] *n.* 跨越
 crust [krʌst] *n.* 地壳
 crystalline ['kristəlɪn] *a.* 晶状的
 culminate ['kʌlmineɪt] *v.* 达到极点, 结束
 curious ['kjʊəriəs] *a.* 好奇的, 奇妙的
 curved [kə:vd] *a.* 弯曲的

D

- debris ['debri:] *n.* 碎片, 残渣
 debt [det] *n.* 债务
 decay [di'keɪ] *v. & n.* 衰变
 decline [di'klaɪn] *v.* 下降
 decommission [ˌdi:kə'mɪʃən] *vt.* 退役
 deduce [di'dju:s] *v.* 推论, 导出
 deficit ['defɪsɪt] *n.* 亏损
 deflect [di'flekt] *v.* 折射
 deflection [di'flekʃən] *n.* 偏转, 折射
 deform [di:'fɔ:m] *v.* 使变形
 deliberate [di'libəreɪt] *a. & v.* (仔细) 考虑的, 谨慎的; 考虑, 商讨
 democracy [di'mɒkrəsi] *n.* 民主

- densification [densifi'keiʃən] *n.* 密实化
- deplete [di'pli:t] *n.* 用尽
- deployment [di'ploimənt] *n.* 推广
- deposit [di'pɒzɪt] *n. & v.* 储存, 堆焊、覆盖
- deregulate [di:'regjuleɪt] *vt.* 撤销对……的管理规定
- derivation [deri'veiʃən] *n.* 导出, 起源
- derive [di'raɪv] *v.* 导出, 从……得出
- destructive [dis'trʌktɪv] *n.* 破坏性的
- detector [di'tektə] *n.* 探测器
- detonate ['detəuneɪt] *v.* 爆炸、爆燃
- detonation [detəu'neiʃən] *n.* 爆炸
- deuteron [dju'ti:riən] *n.* 氘核, 重氢核
- deuterium [dju'tiəriəm] *n.* 氘, 重氢
- dewatering [di:'wɔ:təriŋ] *n.* 排水, 降低水位
- diagnostic [,daɪəɡ'nɒstɪk] *a.* 诊断的
- diaphragm ['daɪəfræm] *n.* 隔板, 垫圈
- diarrhea [,daɪə'riə] *n.* 腹泻
- dictate [dik'teɪt] *v.* 支配
- diffraction [di'frækʃən] *n.* 衍射
- diffuse [di'fju:z] *n.* 扩散
- deficiency [di'fiʃənsi] *n.* 缺乏
- diluent ['diljuənt] *n.* 稀释剂
- dilution [dai'lju:ʃən] *n.* 稀释, 冲淡
- dimensional [di'menʃənəl] *a.* 尺寸的
- dioxide [dai'ɒksaɪd] *n.* 二氧化物
- disassembly [disə'sembli] *n.* 拆下
- discipline ['disiplɪn] *n.* 规定, 科目
- discrepancy [dis'krepənsi] *n.* 不同, 差异, 亏损
- discrete [dis'kri:t] *a.* 不连续的, 分立的
- dislodge [dis'lɒdʒ] *v.* 移动, 撞出
- dismiss [dis'mɪs] *vt.* 解散
- disrupt [dis'rʌpt] *v.* 使分离
- disseminate [di'semineɪt] *v.* 传播
- dissipate ['disɪpeɪt] *v.* 消散
- distinguish [dis'tɪŋɡwiʃ] *v.* 区别, 分类
- distortion [dis'tɔ:ʃən] *n.* 变形, 扭曲, 畸变
- diverging [dai'və:dʒɪŋ] *a.* 扩张的
- diverse [dai'və:s] *a.* 多样的
- diversity [dai'və:sɪti] *n.* 多样性, 差异

- division [di'viʒən] *n.* 分, 划
 docket ['dɒkit] *v. & n.* 概要, 附加提要
 dome [dəʊm] *n.* 圆拱顶
 domed [dəʊmd] *a.* 圆顶的
 Doppler ['dɒplə] *n.* 多普勒
 dormant ['dɔ:mənt] *a.* 固定的
 dose [dəʊs] *n.* 剂量
 downstream ['daʊnstri:m] *a.* 在下游的
 downtime *n.* 停机, 不工作
 drain [dreɪn] *v. & n.* 排出, 排水管
 drawing *n.* 抽出, 引出, 绘图
 drench [drentʃ] *v.* 湿透
 dryness ['draɪnis] *n.* 干度
 dryout [drai'au] *n.* 干涸
 ductibility [dʌkti'bɪləti] *n.* 延展性, 可锻性
 ductile ['dʌktail] *a.* 可延展的, 易变形
 dumbbell ['dʌmbel] *n.* 哑铃
 dynamic [daɪ'næmɪk] *a.* 动力学的, 动态的

E

- eccentricity [ɛksən'trisiti] *n.* 偏心
 economizer *n.* 节热器, 省煤器
 eductor [i'dʌktə] *n.* 排放装置
 ejection [i'dʒekʃən] *n.* 放射出, 排出
 elastic [i'læstɪk] *a.* 弹性的
 elastomer [i'læstəmə] *n.* 弹性体
 electrostatic [i'lektroʊ'stætɪk] *a.* 静电的
 elevation [əli'veɪʃən] *n.* 正面图, 纵剖图, 标高, 高度
 eliminate [i'limineɪt] *v.* 消除
 embark [ɪm'ba:k] *v.* 开始, 投资
 embrace [ɪm'breɪs] *v. & n.* 包含, 接受
 embrittlement [em'brɪtlmənt] *n.* 脆性, 脆裂
 emission [i'mɪʃən] *n.* 发射
 empirical [em'pɪrɪkəl] *a.* 经验的
 encapsulate [ɪn'kæpsjuleɪt] *v.* 封装
 encounter [ɪn'kaʊntə] *v.* 碰到, 遇到
 endorse [ɪn'dɔ:s] *vt.* 签署
 enrich [ɪn'ritʃ] *v.* 富集, 浓缩
 enthalpy [en'θælpɪ] *n.* 焓

- enthusiasm [in'θju:ziæzəm] *n.* 热心, 积极性
entitle [in'taitl] *v.* 叫作
entrain [in'trein] *v.* 携带
envisage [in'vizidʒ] *vt.* 设想, 展望
envision [in'vʒən] *vt.* 预见, 展望
epidemiological [epi,di:miə'lɒdʒikəl] *a.* 流行病的
epithermal [epi'θɔ:məl] *a.* 超热的
equivalence [i'kwivələns] *n.* 等效, 等价
erection [i'rekʃən] *n.* 设备安装, 装配
erosion [i'rəʊʒən] *n.* 侵蚀, 腐蚀, 削弱
escalation [eskə'leiʃən] *n.* (逐步) 升级
europia [ju'rəupiə] *n.* 氧化铀
eutectic [ju:'tektik] *a.* 共晶的
evacuation [i,vækju'eɪʃən] *n.* 搬空, 抽气
eventuality [iventju'ælitɪ] *n.* 偶然事件
evidence ['evidəns] *n.* 迹象, 证据
evident ['evidənt] *a.* 明显的
evolutionary [i:və'lu:ʃənəri] *a.* 进化
exacerbate [eks'æsə:beɪt] *v.* 加重, 使恶化
excel [ik'sel] *v.* 胜过
exclude [iks'klu:d] *v.* 除去
exclusively 独占的, 专门的
expertise [ekspə'ti:z] *n.* 专家
explicit [iks'plisɪt] *a.* 明白, 显然
explicitly [iks'plisɪtli] *ad.* 明显地, 清楚地
exponential [ekspəu'nenʃəl] *a.* 指数的
exponentially [ekspəu'nenʃəli] *ad.* 按指数规律地
exposure [iks'pəʊʒə] *n.* 辐照
extrapolation [ekstrəpəu'leiʃən] *n.* 外推

F

- fabricate ['fæbrikeit] *v.* 制造, 生产
ring riser *n.* 环形冒口
ferritic [fə'ritik] *a.* 铁素(体)的
fertile ['fə:tail] *a.* 可变成裂变物质的, 增殖的
file [fail] *v.* 提出申请, 把……归档
fin [fin] *n.* 肋片
fissile ['fisail] *n.* 易裂变的
fitting ['fitɪŋ] *n.* 配件, 装配部件

- flange [flændʒ] *v. & n.* 带法兰; 法兰
 flashing ['flæʃɪŋ] *n.* 闪蒸
 flask [flɑːsk] *n.* 储罐
 flexible ['fleksəbl] *a.* 挠性的
 flexural ['flekʃərəl] *a.* 挠性的, 弯曲的
 flick [flik] *n.* 轻按
 fluctuation [ˌflʌktju'eɪʃən] *n.* 起伏, 波动
 fluence [fluəns] *n.* 能量密度, (中子) 注量, 能量密度 (J/cm^2)
 fluorescent [fluə'resənt] *a.* 荧光的
 fluoride ['fluəraɪd] *n.* 氟化物
 flywheel ['flaɪwi:l] *n.* 飞轮
 foil [fɔɪl] *n.* 叶, 箔
 foreseeable [fɔː'siːəbl] *n.* 可预见的
 forging ['fɔːdʒɪŋ] *n.* 锻件
 formality [fɔː'mælɪti] *n.* 手续
 formulation [fɔːmju'leɪʃən] *n.* 公式化
 forum ['fɔːrəm] *n.* 论坛
 fouling [faʊlɪŋ] *n.* 弄脏, 结垢
 fraction ['frækʃən] *n.* 分数, 部分
 fractional ['frækʃənl] *a.* 分部的
 fragment ['frægmənt] *n.* 碎片
 fretting [fretɪŋ] *n.* 微振磨损
 friction ['frikʃən] *n.* 摩擦
 fusion ['fjuːʒən] *n.* 核聚变

G

- gadolinium [ˌɡædə'liniəm] *n.* 钆
 gag [ɡæɡ] *n.* 塞盖, 压板
 gaseous ['geɪzjəs] *a.* 气体的
 generic [dʒi'nerɪk] *a.* 同属的, 通用的
 giga *n.* 十亿, 千兆, 10^9
 GJ=Giga Joule 十亿焦耳
 glance [ɡlɑːns] *vi. & n.* 一瞥
 gradient ['ɡreɪdɪənt] *n.* 梯度
 granite ['ɡrænɪt] *n.* 花岗岩
 grant [ɡrɑːnt] *v. & n.* 给予
 granular ['ɡrænjʊlə] *a.* 粒状的
 granule ['ɡrænju:l] *n.* 颗粒
 graphite ['ɡræfaɪt] *n.* 石墨
 grid [ɡrɪd] *n.* 栅格架

gross [grəʊs] *a.* 总计的, 重大的

H

hafnium ['hæfniəm] *n.* 铪

halve [hɑ:v] *v.* 将……减半

hanger ['hæŋə] *n.* 挂钩, 悬架

hard-facing *n.* 表面硬化

hardware ['hɑ:dwɛə] *n.* 设备

hardwired *a.* 硬连线的

hedge [hedʒ] *n.* 障碍(物)

helical ['helikəl] *a.* 螺旋形的

helicopter ['helikɒptə] *n.* 直升机

helium ['hi:ljəm] *n.* 氦

helical ['helikəl] *a. & n.* 螺旋(的)

hemispherical [hemi'sferikəl] *a.* 半球形的

herald ['herəld] *n.* 预言者 *vt.* 宣布的

hereditary [hi'reditəri] *a.* 遗传性的

hexafluoride *n.* 六氟化物

hitherto [hiðə'tu:] *ad.* 至今

homogeneous [həmə'dʒi:njəs] *a.* 均匀的

hookup ['hukʌp] *n.* 挂钩, 连接器

hostility [hɒs'tiliti] *n.* 敌视, 战争

HP=High Pressure 高压

hurdle ['hə:dl] *n.* 栏杆, 障碍

hydrocarbon ['haɪdrəu'kɑ:bən] *n.* 碳氢化合物

hydroelectric ['haɪdrəi'lektrik] *a.* 水电的

hydrogenated [haɪ'drɒdʒineɪtɪd] *a.* 氢化的

hydroxide [haɪ'drɒksaɪd] *n.* 氢氧化物

hyperbolic [haɪpə:'bɒlik] *a.* 双曲线的

hypothesis [haɪ'pɒθɪsɪs] *n.* 假说

I

idiosyncratic [idiəsɪŋ'krætɪk] *a.* 特别的

idle ['aɪdl] *a.* 不工作, 闲置的

ignitor [ɪg'naitə] *n.* 点火器

immediate [i'mi:diət] *a.* 直接的, 立即的

impair [ɪm'peə] *v.* 削弱

impart [ɪm'pɑ:t] *v.* 给予

impermeability [ɪmpə'miə'biliti] *n.* 不可渗透性

- impermeable [im'pə:mjəbl] *a.* 不渗透的
 impingement [im'pindʒmənt] *n.* 冲击
 imply [im'plai] *v.* 意味着
 impose [im'pəuz] *v.* 施加, 发生影响
 inadvertent [inəd'və:tənt] *a.* 偶然的
 incentive [in'sentiv] *n.* 诱因, 动机
 incident ['insidənt] *a.* 入射的
 incipient [in'sipiənt] *a.* 开始的
 Inconel [in'kɒnəl] *n.* 因科镍合金
 incremental [inkri'mentəl] *a.* 增加的
 indefinite [in'definit] *a.* 不明确的
 index ['indeks] *n.* 指示器, 标志, 指标
 indium ['indiəm] *n.* 铟
 inert [i'nə:t] *a.* 惰性的
 inertia [i'nə:ʃiə] *n.* 惯性, 惰性
 infer [in'fə:] *v.* 推断
 infinite ['infinit] *a.* 无限的
 infinitesimal [infini'tesiməl] *a.* 无限小的
 infrastructure ['infre'straktʃə] *n.* 基础设施
 ingest [in'dʒest] *v.* 吸入
 inhalation [inhə'leiʃən] *n.* 吸入
 inhale [in'heil] *v.* 吸入
 inherent [in'hierənt] *a.* 固有的
 inhibit [in'hibit] *v.* 防止, 防腐蚀
 innovative ['inəuveitiv] *a.* 革新的
 inoperable [in'ɒpərəbl] *a.* 不能操作的
 installation [instə'leiʃən] *n.* 装置
 instantaneous [instən'teinjəs] *a.* 瞬时的
 instruct [in'strakt] *v.* 通知, 指导, 指示
 insulate ['insjuleit] *v.* 使绝热, 隔离
 intact [in'tækt] *a.* 未受损的, 无扰动的
 integral ['intigrəl] *a.* 整体的
 integrity [in'tegriti] *n.* 完整, 一体化
 interaction [intər'ækʃən] *n.* 相互作用
 interactively *ad.* 相互作用地
 interconnect ['intəkə'nekt] *v.* 互连
 interim ['intərim] *n.* 中间
 interim-stored *a.* 临时存储的
 interlocking [intə'lɒkiŋ] *a.* 可联动的, 内锁的

- internal [in'tə:nl] *n.* 堆内构件
interpin 元件之间
intervening [,intə'vi:ŋ] 介入
intriguing [in'triɡiŋ] *vt. & n.* 引起……兴趣; 阴谋
intrinsic [in'trɪnsɪk] *a.* 内在的, 固有的
inventory ['ɪnvəntri] *n.* 装料量, 总装量
investor [in'vestə] *n.* 投资者
involvement [in'vɒlvmənt] *n.* 包含, 牵连
iodine ['aiədi:n] *n.* 碘
ironically [ai'rɒnikəli] *ad.* 令人啼笑皆非
irradiate [i'reɪdiət] *n.* 照射, 辐照
isothermal [aɪsəu'θə:məl] *a. & n.* 等温(的)
issue [ɪsju:] *v.* 颁发

J

- jack [dʒæk] *n.* 传动
jeopardize ['dʒepedaɪz] *vt.* 危及, 使受危害
junction ['dʒʌŋktʃən] *n.* 连接
justifiable ['dʒʌstɪfaɪəbl] *a.* 合理的, 正当的

K

- kelvin ['kelvɪn] *n.* 绝对温度
kernel ['kə:nəl] *n.* 核
key [ki:] *v.* 用键固定
keyway ['ki:,wei] *n.* 键槽, 销座
kilovolt ['kɪləʊvɒlt] *n.* 千伏
krypton ['kriptən] *n.* 氪

L

- lashing ['læʃɪŋ] *n.* 捆索
latch [lætʃ] *n.* 凸轮
lattice ['lætɪs] *n.* 栅格
layout ['leɪaʊt] *n.* 布置
lean [li:n] *a.* 瘦的
ledge [ledʒ] *n.* 突出部分
legacy ['leɪɡəsi] *n.* 传统
letdown ['letdaʊn] *n.* 排出, 下降
liable ['laɪəbl] *a.* 易于……的
liberal ['lɪbəərəl] *a.* 充足的

licensing [laisənsɪŋ] *n.* 批准, 执照, 申请
 lieu [lju:] *n.* 场所
 lithium ['liθiəm] *n.* 锂
 locomotive [ləukə'məutiv] *n.* 机车
 lodge [lɒdʒ] *n.* 射入, 沉淀
 logic [lɒdʒik] *n.* 逻辑
 longitudinal [lɒndʒi'tju:dɪnl] *a. & n.* 纵肋(的); 纵向(的)
 LP=Low Pressure 低压
 lug [lʌg] *n.* 突出部, 凸台

M

macroscopic [ˌmækroʊ'skɒpɪk] *a.* 宏观的
 magnesium [mæg'ni:ziəm] *n.* 镁
 Magnox ['mægnɒks] *n.* 镁诺克斯合金
 maintainability [mentɛɪnə'biliti] *n.* 可保养性
 malfunction [mæl'fʌŋkʃən] *n.* 工作不正常
 mandate ['mændeɪt] *v.* 托管
 maneuvering *n.* 操纵, 机动
 manually ['mænjuəli] *ad.* 手动地
 manufacturer [ˌmænju'fæktʃərə] *n.* 制造商
 marrow ['mærəu] *n.* 骨髓
 massive ['mæsɪv] *a.* 大的, 大量的
 mat [mæt] *n.* 底板
 mating ['meɪtɪŋ] *n.* 配合件, 相连件
 mature [mə'tjuə] *a.* 成熟的
 mb=milli barn 毫靶恩, 10^{-3} 靶恩
 mechanism ['mekənɪzəm] *n.* 机理, 作用原理
 mega ['megə] *n.* 兆, 百万
 membrane ['membrein] *n.* 薄膜
 metallurgical [ˌmetə'lə:dʒɪkəl] *a.* 冶金的
 metallurgy [me'tælədʒi] *n.* 冶金
 meteorology [ˌmi:tjə'rɒlədʒi] *n.* 气象状态, 气象学
 microgram ['maɪkrəgræm] *n.* 微克
 microstructure ['maɪkrə'strʌktʃə] *n.* 微观结构
 midplane ['mɪd,pleɪn] *n.* 中平面
 migrate [maɪ'greɪt] *v.* 迁移
 mild [maɪld] *a.* 轻微的, 温和的
 mill [mɪl] *n.* 密尔, 千分之一美元
 millisievert *n.* 毫希伏

- mineral ['minərəl] *n.* 矿物(质), 无机物
- mineralization [,minərəlai'zeiʃən] *n.* 矿物, 矿化作用
- miniature ['minjətʃə] *a. & n.* 小型(的)
- miscellaneous [misi'leiniəs] *a.* 杂项的, 各种的
- missile ['misail] *n.* 导弹
- mitigate ['mitigeit] *v.* 减轻, 减缓
- moderate ['mɒdərit] *a.* 中等的, 适度的
- moderation [,mɒdə'reiʃən] *n.* 慢化, 减速
- modular ['mɒdjulə] *a.* 模块化的
- modularization *n.* 模块化
- modularity [,mɒdju'lærɪti] *n.* 积木性
- moist [mɔɪst] *a.* 湿的
- mol [məʊl] *n.* 摩尔
- momentum [məu'mentəm] *n.* 动量
- monolithic [,mɒnə'liθɪk] *a.* 整体的
- monopolize [mə'nɒpəlaɪz] *v.* 独占
- mortality [mɔ:'tælɪti] *n.* 致命性
- motivate ['məutɪveɪt] *v.* 推动, 刺激
- motivation [,məuti'veiʃən] *n.* 动力, 刺激
- mount [maʊnt] *v.* 安装
- multiplexer ['mʌltɪpleksə] *n.* 多路调制(转换)器
- multiplication [mʌltɪpli'keɪʃən] *n.* 倍增
- mutation [mju:'teɪʃən] *n.* 变化, 转变
- mutual ['mju:tʃuəl] *a.* 相互的, 共同的

N

- naive [na'iv] *a.* 自然的, 经验不足
- nanosecond ['nænəʊ,sekənd] *n.* 毫微秒, 纳秒, 10^{-9} 秒
- near-term *a.* 目前的
- neptunium [nep'tju:njəm] *n.* 镎
- neutronics *n.* 中子物理学
- nickel ['nikl] *n.* 镍
- nil-ductility *n.* 零延性
- nitride ['naɪtraɪd] *n.* 氮化
- notably ['nəʊtəbli] *ad.* 特别是, 显著地
- novel ['nɒvəl] *n.* 新的
- nozzle ['nɒzl] *n.* 接口
- nut [nʌt] *n.* 螺帽

O

- obligation [ˌɒbli'geɪʃən] *n.* 义务
 obsolescence [ˌɒbsə'lesəns] *n.* 逐渐过时
 off-normal *a.* 超常的, 越界的
 offsite *n.* 场外
 omission [əu'mɪʃən] *n.* 省略, 疏忽
 optimism [ˈɒptɪmɪzəm] *n.* 乐观主义
 optimize [ˈɒptɪmaɪz] *v.* 优选
 option [ˈɒpʃən] *n.* 选择
 ore [ɔ:] *n.* 矿
 organic [ɔ:'gænik] *a.* 有机的
 orifice [ˈɒrɪfɪs] *n.* 孔口, 孔板
 outweigh [aʊt'wei] *v.* 重于, 胜过
 overestimate [ˈəʊvə'estɪmeɪt] *v. & n.* 过高估计
 overflow [əʊvə'fləʊ] *v. & n.* 横流, 溢流
 overhaul [ˌəʊvə'hɔ:l] *n.* 大修
 overlay [ˌəʊvə'lei] *v.* 放上, 盖上
 override [ˌəʊvə'raɪd] *v. & n.* 超越, 克服
 oxide [ˈɒksaɪd] *n. & a.* 氧化物(的)
 oxidize [ˈɒksɪdaɪz] *v.* 使氧化, 生锈

P

- package [ˈpækɪdʒ] *n.* 组件
 parabola [pə'reɪbələ] *n.* 抛物线
 paraffin [ˈpærəfɪn] *n.* 石蜡
 parasitic [ˌpærə'sɪtɪk] *a.* 附加的
 parliamentary [ˌpɑ:lə'mentəri] *a.* 国会
 participant [pɑ:'tɪsɪpənt] *n.* 参与者
 particle [ˈpɑ:tɪkl] *n.* 粒子, 微粒
 passive [ˈpæsɪv] *a.* 被动式的, 无源的, 被动的, 非能动
 pebble ['pebl] *n.* 卵石
 peculiar [pi'kju:ljə] *a.* 特有的
 pellet ['pelɪt] *n.* 燃料芯块, 片
 penetrate ['penɪtreɪt] *v.* 穿透, 透入
 performance [pə'fɔ:məns] *n.* 实施, 运转
 peripheral [pə'rɪfərəl] *a.* 周边的, 外围的(设备)
 periphery [pə'rɪfəri] *n.* 边缘, 周边
 permeation [ˌpə:mi'eɪʃən] *n.* 渗透
 perspex ['pə:speks] *n.* 有机玻璃

- perturbation [ˌpɜ:tə'teɪʃən] *n.* 扰动
- pessimistic [pesi'mistik] *a.* 悲观的
- phase [feɪz] *n.* 相
- phosphorous ['fɒsfərəs] *n.* 磷
- PHW=Pressurized Heavy Water 加压重水
- pillar ['pilə] *n.* 小
- pin [pin] *n.* 柱, 棒
- pipework *n.* 管路
- pit [pit] *n.* 坑
- plugging *n.* 堵塞
- plutonium [plu:'təʊniəm] *n.* 钚
- pneumatically [nju:'mætikli] *ad.* 靠压缩空气
- polonium [pə'ləʊniəm] *n.* 钋
- pond [pɒnd] *n.* 池
- pose [pəʊz] *n. & v.* 姿态; 造成
- possession [pə'zeʃən] *n.* 占有
- post-tensioned *a.* 后张力的
- postulate ['pɒstjuleɪt] *v.* 假设
- potassium [pə'tæsiəm] *n.* 钾
- powder ['paʊdə] *n.* 粉末
- precipitate [pri'sipiteɪt] *n. & a.* 突然(的); *v.* 沉淀, 下落
- precipitation [pri'sipi'teɪʃən] *n.* 沉淀
- preclude [pri'klu:d] *v.* 预防, 阻碍
- precursor [pri'kɜ:sə] *n.* 先驱, 前身, 产物母体
- predate ['pri:'deɪt] *v.* 在……之前
- predominant [pri'dɒmɪnənt] *a.* 主要的
- pre-eminent [pri:'eminənt] *a.* 优秀的
- preheat ['pri:'hi:t] *v. & n.* 预热
- prehistoric ['pri:his'tɔ:rik] *a.* 很久以前的, 史前的
- pre-incident *n.* 事故前
- preparatory [pri'pærətəri] *a.* 准备的
- prerequisite ['pri:rekwɪzɪt] *n.* 先决条件
- preset ['pri:'set] *a.* 预先设定的
- prevailing [pri'veɪlɪŋ] *a.* 主要的, 占优势的
- preventive [pri'ventɪv] *a.* 预防的
- prioritization 优先
- prismatic [prɪz'mætɪk] *a.* 棱柱形的
- probabilistic [ˌprɒbəbi'lɪstɪk] *a.* 概率的
- probability [ˌprɒbə'bɪləti] *n.* 概率

- procure [prə'kjuə] *vt.* 取得, 采购
 procurement [prə'kjuəmənt] *n.* 采购
 project ['prɒdʒekt] *v.* 计划, 设想; *n.* 方案
 projectile [prɒ'dʒɪktaɪl] *n.* 抛射体, 入射粒子
 proliferation [prəu,lɪfə'reɪʃən] *n.* 扩散
 prominence ['prɒmɪnəns] *n.* 突起, 显露
 promise ['prɒmɪs] *n.* 约定
 pronounced [prə'naʊnst] *a.* 显著的
 propagate ['pɒpəgeɪt] *v.* 传播
 proper ['prɒpə] *a.* 本身, 正文, 本体
 proposal [prə'pəʊzəl] *n.* 计划, 提出
 prototype ['prəʊtətaɪp] *n.* 原型
 protrude [prə'tru:d] *v.* 凸出
 proximity [prɒk'sɪmɪti] *n.* 邻近
 psig=pounds per square inch gauge, 1psi=0.07kg/cm²
 punched-out *n.* 切口, 凸起
 pursue [pə'sju:] *vt.* 从事, 采取
 pyrochemical [ˌpaɪrəu'kemɪkəl] *a.* 高温化学变化的
 pyrolytic *a.* 热解的

Q

- quadruple ['kwɒdrʌpl] *a.* 四倍……的
 qualitative ['kwɒlɪtətɪv] *a.* 定性的
 quality ['kwɒlɪti] *n.* 纯度, 干度, 质量
 quantitative ['kwɒntɪtətɪv] *n.* 定量的力
 quantum ['kwɒntəm] *n.* (复数 quanta) 量子
 quench [kwentʃ] *v.* 急冷

R

- radiation [ˌreɪdɪ'eɪʃən] *n.* 放射性, 发射, 射线
 radical ['rædɪkəl] *a.* 根本的
 radiological [ˌreɪdɪəu'lɒdʒɪkəl] *a.* 放射学的
 radon ['reɪdɒn] *n.* 氡
 ramp [ræmp] *n.* 斜面, 上升
 random ['rændəm] *a.* 随机的
 rating ['reɪtɪŋ] *n.* 负荷, 额定值
 reasonable ['ri:znəbl] *a.* 适当的
 recommendation [ˌrekəmen'deɪʃən] *n.* 建议
 redundancy [rɪ'dʌndənsi] *n.* 冗余

- redundant [ri'dʌndənt] *a.* 冗余的
- reed [ri:d] *n.* 弹簧
- refuel [ri:'fjuəl] *v.* 换燃料
- regenerative [ri'dʒenəreitiv] *a.* 再生的
- regulatory ['regjuleitəri] *a.* 管理的, 规章(限制)人的, 调节的
- reiterate [ri:'itəreit] *v.* 重申, 反复地说
- relay ['ri:lei] *n.* 继电器
- reliance [ri'laɪəns] *n.* 信赖, 依赖
- remote [ri'məut] *a.* 远程的
- renaissance [rə'neɪsəns] *n.* 复兴
- render ['rendə] *v.* 提出, 表现
- repetitive [ri'petitiv] *a.* 重复的
- reposition [ri:pə'ziʃən] *n.* 复原位, 改变……位置
- reprocessing [ri:'prəusesɪŋ] *n.* 再加工, 后处理
- repulsion [ri'pʌlʃən] *n.* 排斥
- reservoir ['rezəvwa:] *n.* 储存器; 水池
- resistant [ri'zɪstənt] *a.* 抵抗
- resonance ['rezənəns] *n.* 共振
- restrain [ris'trein] *v.* 约束
- restrict [ris'trikt] *v.* 限制, 约束, 节流
- restrictor [ris'triktə] *n.* 节流阀, 限流器
- retain [ri'tein] *v.* 保持、保留
- retard [ri'tɑ:d] *v.* 延迟
- retention [ri'tenʃən] *n.* 保存
- retrievable [ri'tri:vəbl] *a.* 可恢复
- rev/s=revolutions per second 每秒多少转
- rhodium ['rəudiəm] *n.* 铑
- rigidity [ri'dʒɪdɪti] *n.* 刚性
- rigorous ['rigərəs] *a.* 严格的
- rigorously *ad.* 严格地, 精确地
- ring-forged 锻造出的环
- roadmap *n.* 安排, 排定
- robust [rəu'brʌst] *n.* 坚固的
- rolling ['rəʊlɪŋ] *n.* 滚压
- roughen ['rʌfən] *v.* 使变粗糙
- rpm=revolutions per minute 每分钟多少转
- ruggedness ['rʌɡɪdnɪs] *n.* 强度, 坚固性
- rupture ['rʌptʃə] *n.* 破裂

S

- sacrifice ['sækrɪfaɪs] *v.* 损失
- saturation [ˌsætʃə'reɪʃən] *n.* 饱和
- scale [skeɪl] *n.* 规模, 尺度
- scaling-up *n.* 按比例放大
- scatter ['skætə] *v. & n.* 散射, 扩散
- scattering ['skætərɪŋ] *n.* 散射
- scenario [si'nɑ:riəu] *n.* 情况, 方案, 情景
- schematically ['skimætɪkəli:] *ad.* 用示意图, 大略地
- scheme [ski:m] *n.* 方案, 系统
- scoring *n.* 划痕
- seismic ['saɪzmɪk] *a.* 地震的
- seldom ['seldəm] *ad.* 很少
- self-annealing *n.* 自退火
- self-sustaining *a.* 自持的
- semi-permeable *a.* 半渗透的
- sensing ['sensɪŋ] *a. & n.* 信号传感
- serration [se'reɪʃən] *n.* 锯齿形
- set-on 确定, 安置, 调节
- severe [si'viə] *a.* 严重的
- shale [ʃeɪl] *n.* 页岩
- shallow ['ʃæləu] *a.* 浅的
- shatter ['ʃætə] *v.* 破裂, 松落
- sheath [ʃi:θ] *n.* 外壳, 套管
- shedding ['ʃedɪŋ] *n.* 卸下, 脱开负载
- sheltering *n.* 掩蔽, 保护
- shielding ['ʃi:ldɪŋ] *n.* 屏蔽; *a.* 防护的
- shippable ['ʃɪpəbl] *a.* 可装运的
- shrinkage ['ʃrɪŋkɪdʒ] *n.* 收缩、减少
- shuffling ['ʃʌflɪŋ] *n.* 重排
- shutoff ['ʃʌtɔ:f] *n.* 关闭
- SI=System International 国际单位
- signatory ['sɪgnətəri] *n.* 缔约国
- significance [sɪg'nɪfɪkəns] *n.* 意义, 含义
- signify ['sɪgnɪfaɪ] *v.* 示, 意味
- silicide ['sɪli,said] *n.* 硅化物
- silicon ['sɪlɪkən] *n.* 硅
- simulation [ˌsɪmjʊ'leɪʃən] *n.* 模拟, 仿真
- simultaneously [sɪməl'teɪnjəsli] *ad.* 同时地

- sink [sɪŋk] *n.* 散热器, 热井
sketch [sketʃ] *v.* 画示意图
slab [slæb] *n.* 平板
sleeve [sli:v] *n.* 套管
slot [slɒt] *n. & v.* 槽, 狭缝; 开槽
sluggish ['slʌɡɪʃ] *a.* 缓慢的
slump [slʌmp] *n. & v.* 坍塌
snubber ['snʌbə] *n.* 缓冲器
societal [sə'saiətəl] *a.* 社会的
sodium ['səʊdʒəm] *n.* 钠
solder ['sɒldə] *v.* 焊
solicitation [sɒ.lɪ.sɪ'teɪʃən] *n.* 请求, 要求
somatic [səʊ'mætɪk] *a.* 身体的
spacer ['speɪsə] *n.* 定位架
span [spæn] *n.* 跨度
spatial ['speɪʃəl] *a.* 腔的, 占有空间的
specimen ['spesɪmɪn] *n.* 样品, 试件
spectrum ['spektrəm] *n.* 谱
speculative ['spekjʊlətɪv] *a.* 推测的
spider ['spaɪdə] *n.* 星形接头
spill [spɪl] *v.* 溢出
splice [splaɪs] *v.* 镶接
split [splɪt] *v.* 分开, 分解
spring-back *n.* 回弹, 弹性后效
spun [spʌn] *v.* spin 的过去分词, 旋转
stack [stæk] *n.* 堆积, 许多
staff [stɑ:f] *n.* 工作人员
stage [steɪdʒ] *n.* 级(由一组喷嘴叶片和一组转动叶片组成的)
stamp [stæmp] *v.* 盖章通过
standstill 停滞, 间歇
stationary ['steɪʃənəri] *a.* 静止的
steam line *n.* 蒸汽管路
steepen ['sti:pən] *v.* 变陡
stellite ['stelait] *n.* 斯太立特硬质合金
step-wise ['stepwaɪz] *a. & ad.* 逐步的(地)
stewardship ['stjuədʃɪp] *n.* 管理工作, 代管
stiff [stɪf] *a.* 刚性的
stochastic [stə'kæstɪk] *a.* 随机的
stockpile ['stɒkpail] *n. & v.* 储存, 堆放, 储存

- strategically [strə'ti:dʒikəli] *ad.* 战略上, 策略地
- strategy ['strætɪdʒi] *n.* 战略
- stratification [strætifi'keɪʃən] *n.* 分层
- string [striŋ] *n.* 串排
- stringency ['strɪndʒənsi] *n.* 紧急, 迫切
- stringent ['strɪndʒənt] *a.* 严格的
- strip [stri:p] *n.* 带, 条
- stuck [stʌk] *v.* stick 的过去分词, 卡住
- stud [stʌd] *n.* 双头螺栓
- subcool ['sʌb'ku:l] *v.* 欠热, 过冷
- subcritical [sʌb'kritikəl] *a.* 次临界
- subdivision ['sʌbdɪ,vɪʒən] *n.* 一部分, 再分
- sublime [sə'blaɪm] *v.* 升华
- subscript ['sʌbskript] *n.* 下脚标
- substantial [səb'stænʃəl] *a.* 实质的
- sub-system ['sʌbsistəm] *n.* 辅助系统
- subtle ['sʌtl] *a.* 微妙的, 稀薄的
- subtract [səb'trækt] *v.* 减去
- successive [sək'sesɪv] *a.* 连续的
- sulfur=sulphur ['sʌlfə] *n.* 硫
- sump ['sʌmp] *n.* 储水池, 水坑
- supercritical [ˌsju:pə'kritikəl] *a.* 超临界
- superior [sju'piəriə] *a.* 高级的
- superscript ['sju:pəskript] *n.* 上脚标
- supersede [ˌsju:pə'si:d] *vt.* 代替
- supplement ['sʌplɪmənt] *v.* 补充
- surge [sə:dʒ] *n.* 波动
- surplus ['sə:pləs] *n.* 过剩的
- survivor [sə'vaɪvə] *n.* 幸存者
- susceptible [sə'septəbl] *a.* 敏感的
- suspend [səs'pend] *v.* 悬浮
- sustain [səs'teɪn] *v.* 维持
- sustainability *n.* 可持续性
- sustainer [səs'teɪnə] *n.* 二次中子源, 维持者
- swing [swɪŋ] *n.* 摆动
- symmetric [si'metrik] *a.* 对称的
- symptom ['sɪmptəm] *n.* 征兆
- synergy ['sɪnə:dʒi] *n.* (最佳) 合作
- synonymous [sɪ'nɒnɪməs] *a.* 同义的

T

- tail [teɪl] *n.* 尾渣
- taper ['teɪpə] *v.* 逐渐变细
- tapered ['teɪpəd] *a.* 变截面的
- target ['tɑ:ɡɪt] *n.* 靶, 目标
- teardrop *n.* 泪珠
- technocracy [tek'nɒkrəsi] *n.* 专家管理
- temper ['tempə] *v. & n.* 回火, 锻炼, 调和
- tendon ['tendən] *n.* 筋
- tension ['tenʃən] *n.* 张力
- terminate ['tə:mineɪt] *v.* 终止, 结束
- terminology [tə:'mɪnɒlədʒi] *n.* 专门术语
- terphenyl [tə:'fɪnɪl] *n.* 三联苯
- terrorism ['terərɪzəm] *n.* 恐怖主义
- therapy ['θerəpi] *n.* 治疗
- therein [θeər'in] *ad.* 其中
- thermalize ['θə:məlaɪz] *v.* 使热能化
- thermodynamic [θə:məudai'næmɪk] *a.* 热力学的
- thimble ['θɪmbl] *n.* 套管
- thorium ['θɔ:riəm] *n.* 钍
- threshold ['θreʃhəʊld] *n.* 临界点, 限度, 阈值
- throttle ['θrɒtl] *n.* 节流
- thrust [θrʌst] *v.* 冲入
- thyroid ['θaɪrɔɪd] *n.* 甲状腺
- tilt [tɪlt] *v.* 倾斜, 偏离
- tin [tɪn] *n.* 锡
- tissue ['tɪʃu:] *n.* (细胞)组织
- tornado [tɔ:'neɪdəʊ] *n.* 飓风, 龙卷风
- torsional ['tɔ:ʃənəl] *a.* 扭曲的
- toxic ['tɒksɪk] *a.* 有毒性的
- track [træk] *n.* 轨迹, 记录
- train [treɪn] *n.* 系列, 将不同设备串起来的系统
- transient ['trænzɪənt] *n.* 瞬态
- transition [træn'sɪʃən] *n.* 转变
- transuranic [trænsju'rænik] *a.* 超铀的, 原子序数高于铀的
- transuranium [trænsjuə'reɪnjəm] *n.* 铀后元素
- trap [træp] *n.* 井, 收集器
- treaty ['tri:ti] *n.* 条约
- tremor ['tremə] *n.* 颤抖

triangular [traɪ'æŋɡjələ] *a.* 三角(形)的

trip [trip] *n.* 脱扣, 停堆

triuranium *n.* 三铀

tsunami [tsju'nɑ:mi] *n.* 海啸

tube-in-shell 管壳式

tubular ['tju:bjulə] *a.* 由管组成的, 管形的

turbine ['tə:bin] *n.* 透平, 汽轮机

turbulence ['tə:bjuləns] *n.* 扰动

twist [twist] *v.* 扭转

U

ultrasonic [ʌltrə'sɒnik] *a.* 超声的

unacceptable [ʌnək'septəbl] *a.* 不能接受的, 不合格的

undermoderation *n.* 弱慢化, 欠慢化

underpinning *n.* 基础

undershoot [ʌndəʃu:t] *v. & n.* 下冲

uniform ['ju:nifɔ:m] *a.* 均匀的

unionized [ʌn'aɪənaɪzd] *a.* 未(非)电离的

unmitigated [ʌn'mitigeɪtɪd] *a.* 没缓和的, 绝对的

unwittingly [ʌn'wɪtɪŋli] *ad.* 无意地

upstream [ʌp'stri:m] *a.* 上游的

uptake [ʌpteɪk] *n.* 吸入

utility [ju:'tɪlɪti] *n.* 公用事业公司

V

vacant ['veɪkənt] *a.* 空的

valid ['vælɪd] *a.* 有效的, 正确的

validation [væli'deɪʃən] *n.* 有效, 确认, 证实

variant ['vɛəriənt] *a.* 不同的

vector ['vektə] *n.* 航向, 矢量

vehicular [vi'hɪkjulə] *a.* 车用的

vendor ['vendɔ:] *n.* 卖主, 供货商

verify ['verɪfaɪ] *v.* 验证

version ['vɜ:ʃən] *n.* 型(式), 种类

versus ['vɜ:səs] *prep.* 与……比较, 作为……的函数, 与……关系

viability [vaɪə'bɪlɪti] *n.* 生命力, 生存能

vicinity [vi'sɪnɪti] *n.* 附近

violate ['vaɪəleɪt] *vt.* 违反

viscosity [vis'kɒsɪti] *n.* 黏性

visualize ['vizjuəlaɪz] *v.* 目测, 观察
 vital ['vaɪtəl] *a.* 极重要的
 void [vɔɪd] *n.* 空隙
 volatile ['vɒlətaɪl] *a. & n.* 挥发性的, 挥发物
 volatility [vɒlə'tɪlɪti] *n.* 挥发性
 vomit ['vɒmɪt] *v.* 呕吐

W

warp [wɔ:p] *v.* 扭曲
 waste ['weɪst] *n.* 废物
 wax [wæks] *n.* 蜡
 weld [weld] *v. & n.* 焊
 whatsoever=whatever 无论什么
 whip [hwɪp] *v. & n.* 拍打
 withstand [wɪð'stænd] *v.* 经受

X

xenon ['zenɒn] *n.* 氙

Z

zircaloy ['zə:kəlɔɪ] *n.* 锆合金

词 组

A

a factor of two 二分之一
 as a rule of thumb 根据经验
 as far as 就……
 at all cost 不惜任何代价, 无论如何

B

be synonymous with 与……的意义是一样的
 boric acid 硼酸

C

check valve 止回阀
 chemical shim 化学补偿剂
 conform to 适应

D

dead band 死区, 不需要控制系统工作的区

deal with 处理, 对付
die out 消失, 灭绝
drive mechanism (控制棒)驱动机构
due in part to 部分由于

E

emerge from 出现

F

finned cladding 带肋片的包壳
free from 没有……的

G

a great deal 大量的

H

heat sink 热井
hold...apart 将……隔开

I

in lieu of 代替
in the long run 从长远来看
inherent safety 固有安全性

M

make up 补充
more or less 有点

N

nucleate boiling 核态沸腾

O

on-load refueling 不停堆换料

P

pick up 吸收, 加速
pot type 池式

R

reed switch 弹簧开关
relief valve 释放阀

S

shut down 停堆
so long as 只要, 因为
spacer grid 定位格架
superior to 优于
surge line 波动管

T

taking account of 考虑
tend to 趋于
tube sheet 管板
turbine trip 汽轮机脱扣 (甩负荷)

参 考 文 献

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