NUCLEAR MATERIALS – FISSION, FERTILE AND DUAL-USE STRUCTURAL MATERIALS INVOLVED IN NUCLEAR REACTORS

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The article presents a brief account of nuclear materials, with special emphasis on fissile, fertile and some important dual-use structural materials generally involved in nuclear reactors. The dual-use structural materials utilized in nuclear reactors have got important applications in both nuclear and non-nuclear fields. In the hostile environment, the important phenomena such as interactions between fission products and the surrounding elemental species, radiation-induced effects, corrosion, generation of gases, swelling and so forth, become increasingly complex and the performance of a nuclear reactor system thus becomes very much dependent on the physicochemical stability and nuclear compatibility of the dual-use structural materials used in fuel sub-assembly towards the fuel elements. In this context, the dual-use structural materials like stainless steel, zirconium alloys, etc. as claddings; water, liquid sodium or gases, etc. as coolants and water; boron, etc. as moderators, having good reliability and appropriate nuclear compatibility with the fuels, are of prime importance in reactor technology. In advanced designed reactors, development of novel fuels coupled with efficient dual-use structural materials may mitigate the challenges involved in optimizing the efficiency of the power reactors under specific experimental conditions.

Introduction

Since the discovery of X-rays by Wilhelm C. Roentgen and ‘Becquerel rays’ or ‘Uranic rays’ by Henri Becquerel in the last decade of the nineteenth century, conscientious activities in nuclear science have been pursued in a big way1-4. Pierre and Marie Curie indicated that the emission of these energetic rays is an atomic property characteristic of the element concerned and they introduced the term ‘Radioactivity’ for the phenomenon. Ernest Rutherford in 1919 suggested that in a nuclear reaction, interactions of two nuclear particles give rise to emission of new particles from the interacting system and also result in redistribution of nuclear energy and momentum. He indicated that the nucleus, $^{14}$N, on reaction with alpha particles, undergoes transmutation to form $^{18}$F which disintegrates further to give $^{17}$O and a proton, presented as $^{14}$N($\alpha$, p)$^{17}$O. In 1932, J. Chadwick discovered the fundamental particle, neutron, by bombardment of boron with alpha particle through the nuclear reaction,$^{10}$B($\alpha$, n)$^{13}$N. Subsequently, Irene and Frederic Joliot Curie in 1934 discovered artificial radioactivity and Otto Hahn and F. Strassman along with Lise Meitner in 1939 discovered the phenomenon of nuclear fission suggesting the division of a heavy atomic nucleus into two fragments roughly of equal masses accompanied by the release of a huge amount of nuclear energy. It was shown that when an atom like $^{235}$U absorbs a neutron of suitable energy undergoes fission splitting the nucleus into two parts as illustrated by the fission reaction, $^{235}$U + n $\rightarrow ^{144}$Ba + $^{90}$Kr + 2n + energy.

The nuclear fission can occur with slow or fast neutrons and the release of nuclear energy can be controlled or uncontrolled under specific conditions. The nuclear power has thus gradually been acknowledged as an essential alternative source of sustainable energy. In harnessing this

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nuclear energy, the elemental species, $^{233}$U, $^{235}$U, $^{239}$Pu and to a lesser extent, some heavier transuranic elements are effectively being utilized as the fissionable nuclear source materials in reactor technology.

The term “nuclear materials”\(^1\)-\(^3\), according to the International Atomic Energy Agency (IAEA), in general, refers to uranium, plutonium, and thorium, in any form. Nuclear materials, has been differentiated further into “source materials”, consisting of naturally occurring uranium, thorium and the depleted uranium which is not as such suitable for use as nuclear fuels. Uranium ore concentrates are also sometimes considered as “source material”. Presently, the fissile $^{233}$U, $^{235}$U and uranium enriched in isotopes of $^{233}$U, $^{235}$U or $^{239}$Pu, are defined as “Special Nuclear Materials (SNMs)” or sometimes as “Special Fissile Materials”. Of these SNMs, only $^{235}$U occurs naturally. Advancement in nuclear technology has helped to have the species, $^{233}$U, and $^{239}$Pu, the respective nuclear reaction products of $^{232}$Th and $^{238}$U, as two more fissionable materials and has made it possible to sustain fission chain reactions in nuclear reactors. The naturally occurring heavier $^{238}$U and $^{232}$Th, the respective precursors of the artificial or man-made fissile species $^{239}$Pu and $^{233}$U are designated as fertile nuclear materials. It is interesting that even in 1980 Convention on the Physical Protection of Nuclear Materials, thorium has not been included in the definition of nuclear materials.

In addition to these nuclear fissile and fertile source materials, there are some specific materials like boron, graphite, aluminum, stainless steel, zirconium alloys, etc. termed as ‘dual-use-materials’\(^5\)-\(^7\) which have got important applications in both nuclear and non-nuclear fields. Over the years, with the evolution of different types of advanced designed reactors, varieties of ‘dual-use materials’ having specific nuclear characteristics are “especially designed or prepared for the processing, use or production of special fissionable material” exclusively suitable as fuel and fuel-sub-assembly components in nuclear reactors.

The history of development of nuclear power reactors dates back to the early days of Manhattan Project and the Chicago Pile-1 was the world’s first controlled nuclear fission reactor constructed in 1942 in the University of Chicago. Demonstration of controlled nuclear chain reaction by Enrico Fermi paved the way for the construction of advanced designed reactors. The nuclear reactors are broadly designated as the research reactors primarily used for material research, training and production of medically and industrially important radioisotopes and the power reactors essentially used in harnessing nuclear energy for generation of electricity.

The nuclear materials generally used in reactor operation may be classified as the nuclear fuels\(^8\) and the dual-use structural materials as fuel sub-assembly components mainly in the forms of control rods, fuel-claddings, moderators, coolants and shielding (Figure – 1). In fabrication of nuclear fuels, the fissionable species, $^{233}$U, $^{235}$U, $^{239}$Pu, and the fertile species, $^{238}$U and $^{232}$Th, are of primary importance and to control the fission chain reactions in the fuels, the dual-use materials are essentially utilized both as in-core and out-of core structural materials. The dual-use sub-assembly materials are generally used as (i) control rods containing neutron absorbers like cadmium or boron, (ii) fuel-claddings like zirconium alloys, stainless steel, etc. to ensure retention of the fission products in the fuel matrix and also to prevent coolant from chemically interacting with the fuel, (iii) moderators like light water, heavy water and graphite which usually surround the fuels and the coolants helping in controlling the chain reaction by slowing down or thermalizing neutrons and (iv) coolants like light water, heavy water, liquid sodium or gases like CO\(_2\) and helium to take away the heat generated in the fuels and transfer over directly or indirectly to steam generator as per requirement.

![Figure 1. A Generic Nuclear Reactor\(^5\)](image-url)


**Fissile Materials**

The basic requirement for an atomic energy program is the element uranium as it is the primary fissile fuel in nuclear reactors. Uranium, as found in nature, consists of three varieties of isotopes, namely, $^{234}\text{U}$ (~0.0054%), $^{235}\text{U}$ (~0.72%) and $^{238}\text{U}$ (~99.27%), with specific activity of ~0.67 $\mu$Ci/g. Of these uranium isotopes, only $^{235}\text{U}$ is fissile. The predominant naturally occurring heavier isotope, $^{238}\text{U}$, is not fissile. There are three fissile nuclides, $^{233}\text{U}$, $^{235}\text{U}$ and $^{239}\text{Pu}$, and of these three, nature has thus provided with only one nuclide, $^{235}\text{U}$, and the other two nuclides, $^{233}\text{U}$ and $^{239}\text{Pu}$, are artificially produced or man-made. In producing these two fissile nuclides, $^{239}\text{Pu}$ and $^{233}\text{U}$, in reactors, the fertile nuclides, $^{238}\text{U}$ and $^{232}\text{Th}$, capture neutrons first to form $^{239}\text{U}$ and $^{233}\text{Th}$, and then after two successive $\beta$-decay convert themselves respectively to fissile $^{239}\text{Pu}$ and $^{233}\text{U}$. The nuclear fuels are generally fabricated by judicious combination of these three fissile and two naturally occurring fertile nuclides in the forms of metals, alloys, inter-metallic compounds, ceramics (oxide, carbide, nitride, silicide), or dispersion types, depending on the reactor designed.

**Uranium – 235:** The naturally occurring fissionable $^{235}\text{U}$ although can burn out of all energies in nuclear reactors, but the natural $^{235}\text{U}$-content (~0.72%) is not sufficient enough to sustain a fission chain reaction as such and an increase in specific concentration of $^{235}\text{U}$ is essentially needed to initiate the desired nuclear chain reaction in reactors.

In nuclear technology, uranium recovered from natural ores is primarily concentrated as ‘Yellow Cake’ for fabrication of nuclear fuels. In production of reactor grade nuclear fuel isotopically enriched in $^{235}\text{U}$, the pure uranium product has to be treated further by some sophisticated enrichment procedures in which uranium is fed as UF$_6$ formed by direct fluorination of UF$_4$ as UF$_4$ + F$_2$ → UF$_6$. Most of the reactors operating in the world use enriched uranium. In reactors using ordinary water as moderator, the fuel has to be enriched to about 3 to 4% in $^{235}\text{U}$. If natural uranium is to be used as fuel, the moderator has to be heavy water. Enhancement in isotopic concentration of fissile $^{235}\text{U}$ in uranium more than that occurs in nature is termed as uranium enrichment. Fuel enrichment varies from natural to 90% of $^{235}\text{U}$. Uranium for strategic application needs to be enriched beyond 90% of $^{235}\text{U}$.

**Plutonium-239:** In harnessing nuclear energy, uranium is the most common fuel material, the other being artificial plutonium. Plutonium does not exist in nature. It is an artificial element having a number of isotopes. The most important fissionable plutonium isotope, $^{239}\text{Pu}$, is produced in reactors from uranium $^{238}\text{U}$ by (n,$\beta$) reaction followed by two successive $\beta$ - decays which can be presented as

\[
^{238}\text{U}_{92} (n,) \rightarrow ^{239}\text{U}_{92} \rightarrow ^{239}\text{Np}_{93} \rightarrow ^{239}\text{Pu}_{94}
\]

On neutron irradiation of $^{238}\text{U}$ with neutrons, along with fissile $^{239}\text{Pu}$, some other plutonium isotopes like fissile $^{241}\text{Pu}$ and fertile $^{236}\text{Pu}$, $^{238}\text{Pu}$, $^{240}\text{Pu}$, $^{242}\text{Pu}$ are also produced through different nuclear reactions depending on the type of reactor and burn-up of uranium fuel. The fissionable $^{239}\text{Pu}$ has a half life of 24,110 years and undergoes fission with neutrons of all energies. Since a higher number of neutrons are emitted per fission, it is considered as an important fissile fuel species in fast breeder reactors. The fissile $^{241}\text{Pu}$, with a half life of 14.4 years decays primarily by $\beta$ - emission to alpha -active non-fissile $^{241}\text{Am}$ which further decays to $^{237}\text{Np}$. The nuclides, $^{238}\text{Pu}$ and $^{241}\text{Am}$, are highly alpha-active and generate heat which increases progressively with the aging of plutonium. The daughter products of isotope $^{236}\text{Pu}$ (Tl and Bi) are hard gamma-emitters and may be present in high burn up plutonium.

In the reactor, besides the production of plutonium isotopes, host of other fission products and ‘minor’ actinides like Np, Am, Cm, etc., are also produced in the fuels. Plutonium produced in a reactor during irradiation is separated from spent fuels mainly by PUREX process. In the process, the spent fuel is first dissolved in nitric acid and then uranium along with most of the fission products are separated from plutonium nitrate by solvent extraction/ion exchange exchange processes. Plutonium left in nitrate solution is precipitated as an oxalate and then calcined to get plutonium as PuO$_2$ powder. Conversion of PuO$_2$ to metal plutonium is performed by its fluorination to PuF$_4$ followed by metallo-thermic reduction.

Since late eighties, $^{239}\text{Pu}$, because of its specific nuclear characteristics, is better utilized as mixed oxide (MOX) fuels in light water fast breeder reactors (FBRs) in a significant way. Plutonium is a highly radiotoxic alpha-active material and is utilized with utmost care. Attempt for disposition of plutonium in fast reactors has been made by replacing the depleted UO$_2$ in MOX fuels by some inert materials like SiC and MgO stabilized in ZrO$_2$ and the axial blanket material by some specific inert matrix.

**Uranium-233:** The fissile uranium $^{233}\text{U}$, artificially produced in FBRs, is an important source of nuclear energy for the future. The nuclide, $^{233}\text{U}$, produced by transmutation
of \(^{232}\text{Th}\) through the thorium-uranium breeding cycle provides a large source of fissile materials in advanced power breeder reactors\(^1\). The primary aim of development of the FBR concept is to breed the fissile nuclides, \(^{239}\text{Pu}\) and \(^{233}\text{U}\), through the two respective breeding cycles, \(^{238}\text{U} \rightarrow ^{239}\text{Pu} \rightarrow ^{232}\text{Th} \rightarrow ^{233}\text{U}\). In producing \(^{233}\text{U}\), the natural thorium species, \(^{232}\text{Th}\), like \(^{235}\text{U}\), needs to absorb a neutron to get converted into an unstable nuclide, \(^{233}\text{Th}\), on two successive \(\beta\) - decays leads to the fissionable \(^{233}\text{U}\) with a half life of \(1.6 \times 10^5\) years. The involved nuclear reactions may be presented as

\[ ^{232}\text{Th}_{90} (n, \beta^+) \rightarrow ^{233}\text{Th}_{90} \rightarrow ^{233}\text{Pa}_{91} \rightarrow ^{233}\text{U}_{92} \]

\[ \beta \quad \beta \]

\[ 23.5 \text{ m} \quad 27.4 \text{ d} \]

Thorium based nuclear fuels have some added advantages since the fissile nuclide, \(^{233}\text{U}\), has favorable nuclear characteristics both in thermal and fast neutron spectrum and the \(^{232}\text{Th}–^{233}\text{U}\) fuel cycle can be made self-sustaining using external fissile material, \(^{235}\text{U}\), \(^{239}\text{Pu}\) or an accelerator driven neutron source. On irradiation of thorium in FBRs, along with fissile \(^{233}\text{U}\), small quantities of \(^{232}\text{U}\) (68.9 yr), a hard gamma emitter, is also formed through \((n, 2n)\) nuclear reaction in thorium fuel cycle which may pose some technological problems.

The ability to convert significant quantities of fertile material into useful fissile material is depended on the magnitude of its reproduction factor, \(\eta\) (eta) i.e. the number of neutrons produced per neutron absorbed in the fuel. The value of \(\eta\) in fast neutron spectrum of \(^{239}\text{Pu}\) is comparatively higher than that of \(^{235}\text{U}\) or \(^{233}\text{U}\) implying that more number of neutrons become available for breeding \(^{239}\text{Pu}\) from \(^{238}\text{U}\) in \(^{238}\text{U}–^{239}\text{Pu}\) fuel cycle than that in \(^{232}\text{Th}–^{233}\text{U}\) fuel cycle and hence the breeding ratio. Actually, \(^{239}\text{Pu}\) is the best fissile material in fast neutron spectrum.

**Fertile Materials**

The potentiality of the virtually inexhaustible nuclear energy locked in natural uranium and thorium was realized since mid 1940s\(^2\). The naturally occurring heavier uranium, \(^{238}\text{U}\), and thorium, \(^{232}\text{Th}\), as precursors of fissile \(^{239}\text{Pu}\) and \(^{233}\text{U}\), are designated as the fertile materials and are efficiently utilized in breeding and recycling of the fissile species in FBRs. Presently, in most of the advanced power reactors, the fission chain reactions are maintained at a controlled and steady rate by transformation or breeding of fresh fissile isotopes, \(^{239}\text{Pu}\) and \(^{233}\text{U}\), respectively from \(^{238}\text{U}\) and \(^{232}\text{Th}\). The fissile nuclides have high cross sections for thermal neutron induced fissions but the fertile nuclides, \(^{238}\text{U}\) and \(^{232}\text{Th}\), do not experience significant fission with thermal neutrons, but can undergo fission with fast neutrons.

**Uranium – 238:** The naturally occurring fertile uranium nuclide \(^{238}\text{U}\) through a series of nuclear reactions gets converted into the most important fissionable nuclide, \(^{239}\text{Pu}\). In FBRs, the nuclear chain reactions are sustained by fast neutrons and the reactor cores are surrounded by blankets of natural uranium or thorium to capture the neutrons leaking from the cores producing the respective fissile nuclide, \(^{239}\text{Pu}\) or \(^{233}\text{U}\). The element, in general, should be very low in impurities like boron, cadmium, gadolinium, samarium, etc. as these species are high neutron absorbers.

In nature, uranium occurs in rocks of almost all geological ages and are generally categorized as the Unconformity-related (33%), Sandstone type (18%), Breccia complex (17%), Quartz-pebble conglomerates (13%) and Vein type (10%) mostly located in Australia, USA, Kazakhstan, Uzbekistan, Canada and South Africa. Exploration and evaluation of uranium deposits are generally performed following the estimation of ore reserves and radiometric reconnaissance survey of uranium mineralization. The total world’s reserves of uranium is \(-2.49\) million tons and the annual world production of uranium concentrates as \(\text{U}_3\text{O}_8\) is estimated to be around 40,000 tons.

Uranium, in most of its natural deposits, generally exists in its tetravalent state. The tetravalent uranium, however, gets converted to its water soluble hexavalent form and becomes mobile in oxidizing environment. The process of concentration of uranium from ores involves several stages like (i) dressing of the ores by physical methods of crushing, grinding, gravity separation, froth flotation, etc., (ii) treatment of the dressed materials with dilute acid or alkali carbonate to dissolve uranium depending on the chemical composition of the ores and (iii) recovery of uranium as \(\text{U}_3\text{O}_8\) concentrate (70 – 80%) from the leach solutions. Uranium is further purified either through solvent extraction utilizing organo-phosphoric acid esters or long chain alkyl amines as suitable extractants or by effective ion exchange procedures as anionic complexes. Practically, a better grade of uranium is obtained through solvent extraction, especially when amines are employed as extractants.

**Thorium – 232:** The naturally occurring thorium is mono-isotopic, \(^{232}\text{Th}\) (100%) and as a promising fertile element like uranium\(^{238}\text{U}\), plays a prominent role in world’s nuclear energy programs\(^1\). Thorium-232, because of its nuclear characteristics both in thermal and fast neutron spectrum and the \(^{232}\text{Th}–^{233}\text{U}\) fuel cycle can be made self-sustaining using external fissile material, \(^{235}\text{U}\), \(^{239}\text{Pu}\) or an accelerator driven neutron source. On irradiation of thorium in FBRs, along with fissile \(^{233}\text{U}\), small quantities of \(^{232}\text{U}\) (68.9 yr), a hard gamma emitter, is also formed through \((n, 2n)\) nuclear reaction in thorium fuel cycle which may pose some technological problems.

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The naturally occurring thorium is

\[ {^{232}\text{Th}} \rightarrow {^{233}\text{Th}} \rightarrow {^{233}\text{Pa}} \rightarrow {^{233}\text{U}} \]

\[ \beta \quad \beta \]

\[ 23.5 \text{ m} \quad 27.4 \text{ d} \]

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neutron, is utilized as an efficient fertile fuel material with fertile uranium-238 in breeder reactors. Thoria based nuclear fuels have excellent nuclear characteristics in both thermal and fast neutron spectrum. The thermal neutron absorption cross section of thorium is 7.4 barns as against 2.7 barns for natural uranium and is very effective in neutron capture. Countries which have large deposits of thorium but with limited uranium reserves, the option for using 233U fuel based FBRs for generation of nuclear power is of primary importance.

In FBRs, thorium, in the form of metal or oxide, is used as the reactor core blanket to produce 233U by neutron capture followed by β - decays. The application of thorium as a nuclear fuel comprises several major steps like (i) conversion of natural 238U fed as fuel to a reactor to fissile 239Pu, (ii) transfer of the fissile 239Pu to another reactor using thorium as fuel where thorium is converted to fissile 233U resulting in production of fissile material more than that actually consumed and (iii) to continue the process, all that is required to feed the breeder reactor is addition of thorium. In FBR fuel-assembly, thorium, the precursor of fissionable 233U, acts as an excellent host for 239Pu and enables deeper burning of plutonium. In fuel performance point of view, thorium (ThO2) is suggested to be superior to uranium (UO2) as it has better thermal conductivity, lower thermal expansion, higher radiation resistance and the long lived minor actinides produced by fission of 233U are also much less in quantities.

In reprocessing of thorium fuel, the highly stable ThO2 is dissolved in nitric acid in presence of fluoride ions. The chemical procedures involved in solvent extraction separation of thorium-uranium in the irradiated fuels are almost similar to that of separation of uranium-plutonium system. The extractant, Tributyl phosphate (TBP), in suitable diluent is used to extract 233U, 228Th and 229Th or for selective extraction of 233U alone.

In nature, thorium primarily occurs in orthophosphate rare earth minerals, monazite, as ThO2 (~9%). It is about three times as abundant as uranium. Thorium is recovered from phosphate minerals by dissolving in nitrate solution from which it is precipitated as oxalate and finally converted to oxide by calcination. The oxide is reduced to thorium metal powder and processed to the desired form by powdered metallurgical techniques.

Ceramic Nuclear Fuel Materials: To achieve higher burn up levels than that obtained by using metallic fuels in a reactor, development of ceramic mixed oxide, (U, Pu)O2, and non-oxide carbide (U, Pu)C and nitride (U, Pu)N, nuclear fuels have been initiated. The U-Pu mixed oxide ceramic fuels are the undisputed choice for majority of the nuclear power reactors presently in operation all over the world. The ceramic mixed carbide and nitride fuels, having excellent compatibility with coolant like liquid sodium, are sometimes being used as nuclear fuels. The UO2, either natural or enriched with 235U, and PuO2 are iso-structural and completely solid soluble in the whole range of UO2 - PuO2 composition as mixed oxide. Compared to metal or alloy fuels, oxides of uranium, plutonium and thorium have good chemical stability and compatibility with different dual-use nuclear structural materials.

In addition to these fissile and fertile materials, the radionuclides, 241Am, 239Pu, 210Po, 137Cs, 90Sr, 3H, etc., under specific conditions, sometimes act as radioisotopic nuclear fuels. Similarly, the byproducts that are produced or made radioactive in nuclear reactors as well as the tailings and wastes produced during ore processing of uranium and thorium as source materials, are also considered as nuclear materials.

Dual-use Structural Nuclear Materials

The nuclear fuel primarily comprising of the fissile and fertile species is the heart of a reactor. In successful operation of a reactor, in addition to nuclear fuels, it always needs some ‘dual-use’ special elements or compounds as structural materials with favorable nuclear characteristics highly compatible with the fuels as in-core and out-of-core structural components. In an on-going fission process, a large number of highly radioactive neutron-rich fission fragments of masses of around 100 and 135 with half lives ranging from a fraction of a second to several hundred of years, are produced causing changes/degradations in thermo-chemical properties of the exposed structural materials.

The dual-use structural materials, in general, are effectively utilized in different forms such as (i) fuel dispersing medium like aluminum, aluminum oxide, etc., (ii) neutron moderators like light water, heavy water, graphite, beryllium oxide, etc., (iii) cladding materials like zircalloys, stainless steels, etc. (iv) coolants like light water, heavy water, liquid sodium, carbon dioxide, etc., (v) reflector of neutrons like heavy water and (vi) radioactive shields like lead, steel, water, concrete etc. In designing an advanced reactor, adequate knowledge of physico-chemical properties and nuclear characteristics of the structural materials, in addition to the fissile and fertile elements, are essentially needed in predicting the fuel performance as well as the aspect of nuclear safety. Characteristic properties of some of the important ‘dual-
Coated sensors for neutron counters. Colloidal boron solution is sometimes applied in fabricating neutron absorber/sensor in nuclear instrumentations. Carborundum (boron carbide) enriched in 10B isotope has now become the most suitable control rod material, especially for high temperature reactors. Similarly, titanium diboride with high melting point and good thermal conductivity is also considered as a suitable control rod material, especially for high temperature reactors. Boron of natural isotopic composition as well as enriched with 10B is used in neutron counters as neutron absorber/sensor in nuclear instrumentation. Colloidal boron solution is sometimes applied in fabricating coated sensors for neutron counters.

**Control Rods**

The control rods control the nuclear power level. The substances capable of absorbing neutrons are called controlled materials. To achieve controlled fission in a reactor, the control materials in general, must have very high cross sections for neutron capture so as to absorb excess neutrons to keep the multiplication factor at a state suitable for maintaining desired nuclear reactivity in the reactor core. In this context, boron with a neutron capture cross section of 764 barns is one of the best choices as control rod material.

**Boron:** The element boron and its compounds, boron carbide and titanium boride, having high neutron absorption cross sections, are extensively used as important neutron absorbing materials in control rods in nuclear reactors. In operating condition, the reactivity of a reactor gradually decreases due to decrease in its fissile content and increase in formation of fission products. The steady reactivity, as desired, is however maintained by insertion of control rods containing appropriate control materials to absorb neutrons. The control rods, as part of the fuel assembly, generally contain the element like boron in the form of boron steel or boron carbide as neutron absorber. Boron, besides its essential use as a potential neutron absorber in control rods, finds important applications also in human shielding against neutrons.

Boron has two principal isotopes, 10B and 11B. Of these, 10B with high neutron absorption cross section (thermal neutron cross section, 3835 b) is highly effective as neutron absorber. The (n, α) reaction products of boron, helium and lithium, as shown by the reaction, 

\[ ^{10}\text{B} + n \rightarrow ^{4}\text{He} + ^{7}\text{Li} \]

are stable and non-radioactive and hence the problems of decay heating during reactor shut down or transfer of depleted control rods become minimal. Boron carbide enriched in 10B isotope has now become the most preferred control rod material for controlling neutrons in FBRs. Similarly, titanium diboride with high melting point and good thermal conductivity is also considered as a suitable control rod material, especially for high temperature reactors. Boron of natural isotopic composition as well as enriched with 10B is used in neutron counters as neutron absorber/sensor in nuclear instrumentation. Colloidal boron solution is sometimes applied in fabricating coated sensors for neutron counters.

Boron carbide in the form of powder and as composites like Bocarsil (composites of flexible rubber sheets and boron carbide), Boral, (composites of boron carbide and aluminum), Poly-boron, (composites of boron in polyethylene matrix), etc. have varieties of applications in plugging neutron leaks surrounding beam holes in research reactors, neutron shielding in nuclear instruments and in storage of radioactive materials. The compound, ZrO2, as sintering additive to boron carbide, is found to be highly effective in its densification in reduced temperatures.

In early designed Pressurized Heavy Water Reactors, some specific components containing boron or other neutron absorbers like Cd, Dy, Eu, Gd, Hf, 135Xe, etc. were used along with the fuel rods as nuclear burnable poisons to achieve optimized high burn up of the nuclear fuels. Presently, the neutron poison is added directly to fuel pellets as boron coating and also as gadolinium oxide doping to allow more flexibility in their use. In fuel processing plants, especially in storage tanks and vessels, poison tubes containing materials like gadolinium are incorporated so as to cause substantial reduction in neutron population. The stainless steel vessels clad with cadmium or borated coatings are also sometimes used for the purpose. Addition of dysprosium (ZrO2-Dy2O3) as burnable neutron absorber to the central rod used in connecting the fuel pins loaded with (232Th-233U)O2 has helped in achieving the desired reactivity in Advanced Heavy Water Reactors. Cadmium and hafnium with high capture cross sections but with low melting points are sometimes used in the forms of alloys as neutron absorbers.

In nature, boron generally occurs in Tourmaline, Na2B4O7.10H2O (~10% of boron) and Kernite, Na2B4O7.4H2O minerals. The element is mainly produced through Moissan process by reduction of boric oxide with magnesium, B2O3 + 3Mg → 2B + 3MgO. In the laboratory, it can be produced by thermal decomposition of diborane, B2H6, through thermal-electrolytic reduction techniques. The element is not affected by either HCL or HF acid but gets oxidized by hot oxidizing acids like HNO3, H2SO4, HClO4, etc. It is extremely difficult to get pure boron as it reacts with a majority of metals at high temperatures to form metallic carbides or borides. It is an extremely hard refractory solid material and classified as a metalloid or semiconductor.

Besides boron, pure graphite has also got important use as a moderator in control rods. Graphite is a form of carbon with significant resistance to corrosion, low neutron absorption, high temperature resistance, good resistance to ablation and good machinability. To reduce fuel pellet-clad interaction, the inner surfaces of the fuel tubes are
Sometimes thinly (5-9 microns) coated with graphite. Similarly, the metal hafnium with high thermal neutron cross-section (~600 times more than Zr) make it ideal for use as control rods in fission reactors. The Hf-control rods have long life as many of the hafnium isotopes produced by absorption of neutrons are also strong neutron absorbers. Hf-oxide is often used in the production of shields for nuclear materials.

**Moderators**

L. Szilard and E. Fermi were the first to introduce the concept of moderator in slowing down neutrons. Substances capable of slowing down or thermalizing neutrons are generally termed as moderators to, respectively, two hydrogen isotopes, $^1$H and $^2$H (D), respectively in the forms of light water (H$_2$O) and heavy water (D$_2$O), carbon in the form of graphite and beryllium. In case of graphite, because of its higher mass, the moderating characteristics are not as good as those of water, although a wide variety of coolants can be used. High toxicity of beryllium has restricted its use as a moderator. Amongst these moderators, heavy water is most prominent and perhaps the best (moderating ratio of D$_2$O, H$_2$O and graphite is 21000 : 72 : 188).

**Heavy Water:** The lightest element, hydrogen, with three naturally occurring isotopes, hydrogen $^1$H, deuterium, $^2$H (D) and tritium, $^3$H (T), is the most abundant matter in the universe. Of these three isotopes, $^1$H and $^2$H are stable and $^3$H is radioactive (12.26 yr) which undergoes $\beta$-decay to form $^3$He. Separation of the hydrogen isotopes is of great importance in nuclear science. For example, deuterons (a few MeV) are used as projectiles in studies of nuclear reactions and deuterium and tritium are used in nuclear fusion.

The natural occurrences of H$_2$O, D$_2$O and T$_2$O are in the ratio of about $1 : 1.5 \times 10^{-4} : 2\times10^{-6}$ and the abundance of heavy water in normal water is ~150 ppm. Deuterium discovered by H.C. Urey in 1931, was concentrated in the form of its oxide, D$_2$O, as heavy water, by electrolysis of ordinary water in 1932. The Light Water Reactors (LWRs) which are in maximum use in the world with enriched (~3%) uranium oxide as fuel utilize light water both as moderator and coolant. The requirement of enriched uranium as fuel in LWRs has to some extent been overcome by using natural uranium metal as fuel in heavy water moderated reactors first installed in 1944 at Argonne National Laboratory.

In case of water as moderator, the fast neutrons produced by fission events in reactors are reduced to thermal energies or moderated by collisions with low mass nuclei of $^1$H or $^2$H, usually as light water or heavy water. Comparatively heavy water has been found to be a better moderator as D has a neutron capture cross section of only 0.5 mb and the moderating capacity is only slightly inferior to that of $^1$H, whereas light water has a significant neutron capture cross section (0.3 b for $^1$H). Presence of one additional neutron in the nucleus of D in D$_2$O considerably reduces its neutron absorption capacity compared to that of H$_2$O making D$_2$O as a good moderator for neutrons and thus plays a significant role in sustaining a nuclear chain reaction in reactors with natural uranium. In heavy water reactors, the concentration of T in D$_2$O gradually increases as it is formed by D through neutron capture and also in the fission of uranium. Heavy water is a good reflector of neutrons and is not consumed in reactor operation.

The oxides of H, D and T, light water, heavy water and tritiated water, considerably differ in their physical properties as the mass differences of the hydrogen isotopes differ greatly. It is about 10% more dense than regular water but visibly indistinguishable and non-radioactive. There are some differences in their chemical behaviors also such as the neutral heavy water has a pD of 7.4 as D$_2$O ionizes to a lesser extent than H$_2$O and the solubility of some salts is comparatively lower in D$_2$O than that in H$_2$O. There is also preferential combination of O$_2$ with H$_2$ compared to D$_2$. On radiolysis, the net decomposition for D$_2$O is lower than that for H$_2$O.

Heavy water is generally produced by distillation or electrolysis of water and through different chemical exchange processes. Amongst the important chemical exchange procedures involving H$_2$S-H$_2$O, NH$_3$-H$_2$O and H$_2$O-H$_2$ systems generally employed in production of heavy water, the H$_2$S-H$_2$O exchange process meets over 90% of its global requirement. Cryogenic distillation of hydrogen is an important process for separation of both deuterium and tritium.

In reactor technology, it is an advantage that heavy water, D$_2$O (~100%), as a moderator and coolant, permits the use of natural uranium as fuel in thermal reactor systems eliminating the need of uranium enrichment. Presently, the heavy water reactors fuelled by natural uranium are fruitfully utilized to generate fissile $^{239}$Pu to feed the fast...
breeder reactors to breed fissile $^{233}$U from fertile natural thorium, $^{232}$Th as a nuclear fuel for generation of much needed electricity. Since World War II, due to strategic as well as commercial importance of heavy water, major research on development of heavy water processes have been carried out in the countries like Canada, France, Germany, U. K., Sweden and India.

In Molten Salt Breeder Reactors with a molten salt mixture of UF$_4$, LiF, BeF$_2$ and ThF$_4$ fuels, the graphite rods suspended in the molten salt are used as moderators. In 1954, in Aircraft Nuclear Propulsion Program (ANP), a molten salt, NaF-ZrF$_4$-UF$_4$, was used as fuel and blocks of BeO served as moderator. The Russian ‘RBMK’ reactors using enriched uranium as fuel are graphite moderated reactors.

**Claddings**

The primary function of the cladding materials is to encase the fuel pellets isolating the fuels from the coolants in reactor cores. Claddings act as the barriers or primary containments to fission products entering the coolant stream and minimize the transfer of radioactivity into the reactor shell and beyond. It is the first line of defense for radioactivity. The claddings and the wrapper tube materials in a fast reactor fuel sub-assembly are usually subjected to high operational temperatures, high energy neutron flux, chemical interactions with various fission products and coolants. The cladding materials, in general, should have the preferable properties of (i) lower affinity for neutron absorption, (ii) high corrosion resistance, (iii) good compatibility with fuels, coolants and other structural materials and (iv) resistance to irradiation induced void swelling, creep and embrittlement.

In operating condition, chemical compatibility of the fuel-cladding systems varies from reactor to reactor and even from point to point in a given reactor and thus the phenomenon of the fuel pellet-clad interactions (PCI) demand special attention in reactor technology. The diffusion and deposition of corrosion and fission products, particularly the gases like iodine, which can easily reach the claddings along the fuel cracks and pellet-pellet interfaces, are responsible for fuel failure. For example, in case of stainless steel claddings, deposition of the corrosion products, Fe, Ni, Cr, etc. on fuel surface as oxide or uptake of hydrogen by zircaloy alloy claddings may lead to cladding embrittlement resulting damage to fuel assembly.

In nuclear power reactors, the special dual-use structural materials like zirconium alloys, stainless steel, aluminum, etc. as nuclear claddings play significant roles in maintaining a predetermined geometric integrity of the reactor cores.

**Zirconium:** Zirconium as a congeneric pair of elements always occurs together in nature with hafnium (~2 – 3%). Since hafnium has high thermal neutron absorption cross section, it is essential to have hafnium-free zirconium for its use as cladding materials in reactor technology. It encapsulates fuels in water-cooled thermal reactor components as fuel cladding and other internal reactor components. Zirconium having the property of corrosion resistance in water and stream, U-Zr alloy cladding have been used in early nuclear submarines, first in the nuclear submarine, ‘the Nautilus’, USA, in 1951. In Enrico Fermi Atomic Power Plant, the U-Mo fuel was also bonded with zirconium cladding.

Pure zirconium metal has the unique characteristic properties of (i) low neutron absorption cross section (~0.8 b), (ii) dimensional stability at high temperatures, (iii) high reactivity and alloying behaviors with various elements and (iv) better corrosion resistance, etc. Addition of traces of tin to pure zirconium further enhances its corrosion resistance. The exceptional favorable properties of the element make it an ideal material for use in reactor cores and more than 90% of zirconium production is consumed by commercial nuclear power generation.

Research and development on cladding materials over the years have resulted in the production of different zirconium alloys like zircaloy-2 and zircaloy-4 having excellent compatibility with varieties of nuclear fuels and structural materials, especially in thermal power reactors for generation of electricity. Currently, almost all commercial power reactors, in general, use zircaloy-2 (Sn-1.5%, Fe-0.12%, Cr-0.1%), or zircaloy-4 (slightly different in composition from zircalloy-2) as the cladding material. Addition of some alloying elements like Fe, Cr and Ni to zircalloy-2 provides better mechanical properties for its use in reactors. Zircaloy-4 has higher resistance to hydrogen uptake than zircaloy-2. The problem of hydriding the surface of the clad by hydrogenous impurities resulting in blistering or cracking is eliminated by inclusion of a hydrogen getter inside the fuel rods. Development of a thin liner of pure zirconium ‘barrier’ inside cladding reduces the incidence of PCI failure in water reactors. Similarly, a thin layer of copper or titanium coating on the inside wall of zircaloy clad tubing can provide an effective diffusion barrier against diffusion of iodine. Zirconium alloys meet almost all required criteria for ideal claddings and provide major incentive for use in reactors especially in LWRs using UO$_2$ or (U, Pu)O$_2$ as fuels. The fission product, cesium forms compounds like Cs$_2$ZrO$_3$ and Cs$_2$UO$_4$ near pure
zirconium layer resulting considerable reduction in PCI. For stainless steel or zirconium alloy claddings, sometimes, incorporation of a fuel-clad barrier layer of graphite helps in reducing the PCI in different designed reactors. Development of varieties of zirconium alloying systems like Zr-Nb, Zr-Ti, Zr-Hf, Zr-Ta, Zr-Fe, Zr-Ni, Zr-Cu, Zr-Cr, Zr-H, Zr-Al, Zr-Sn, Zr-O, Zr-N, Zr-Nb-Cu, Zr-Sn-Nb, etc. are under investigations.

Zirconium as a non-fissile dual-use material, in some specific conditions, can act as a thermal bond between the binary U-Pu fuel system in the form of a ternary U-Pu-Zr alloy fuel like ‘Na’. The Zr-bonded fuel, in some cases, has some added advantages over the Na-bonded fuel. In Experimental Breeder Reactors, the U-Zr (~2%) alloy fuel rods encased in Type 347 stainless Steel tubes were individually loaded in the core inside zircaloys tube. The important role played by zirconium includes increasing of solidus temperature, overcoming of the problems of PCI, stabilization of the isotopic phases, etc.

The TRIGA reactors and the CABRI reactor in France used U₃O₈ dispersed in graphite moderator as fuel and zircaloy as claddings. In TRISO fuel particles, the SiC coating layers are sometimes replaced by stable ZrC layers in forming more advantageous ZrC-TRISO fuel particles. The thorium and urania fuels, clad in zircalloys using the blanket concept are being investigated in USA, Canada, Italy and other countries.

The element, zirconium, discovered by Berzelius in 1824, occurs in nature mainly as zircon, ZrSiO₄ and baddeleyite. In separation of the refractory element zirconium, it is first recovered as oxide from zircon and then converted to its metallic form through two major steps of (i) chlorination of the oxide to zirconium tetrachloride and (ii) its reduction with magnesium to metal zirconium. Applications of Van Arkel iodine refinement process and the Kroll’s process led to separation of highly pure zirconium from hafnium and other associated impurities. Countries like France, USA, Russia and India are capable of producing reactor grade zirconium sponge, zirconium alloys and a wide variety of related dual-use structural components effectively used in power reactors.

**Stainless Steel:** In 1940, plutonium alloy clad in mild steel was first used as fuel in mercury cooled fast reactor, Clementine, in USA. Since then various types of both in-core and out-of-core dual-use stainless steel components as nuclear claddings and other major structural materials for advanced reactor systems are being evolved either by addition of suitable stabilizing elements or by modification of the structural compositions.

In course of study, varieties of austenitic stainless steel alloys like 304, 304L (N), 306, 316L (N), 316-Ti, 15-15Ti, D9, etc.; high nickel alloys like PE-16, INC-706, etc. and ferritic/martensitic stainless steel alloys like HT9, T91, EM10, EM2, etc., as effective nuclear cladding/wrapper materials have been developed. These alloy claddings with favorable high temperature mechanical and irradiation resistances exhibit excellent compatibility with different coolants especially with sodium in fast reactors. The CO₂ cooled British AGR reactor has utilized stainless steel alloy (Fe – 20Cr-25Ni-1Nb) with good ductility, corrosion resistance and reasonable strength at high temperatures as cladding. To have better irradiation resistance, in sodium cooled Prototype Fast Breeder Reactor (PFBR), D9 has been used as cladding/wrapper in-core component and the low carbon-nitrogen alloys 304 L(N) and 316 L(N) have been selected as the main out-of-core structural components. In case of the stronger ferritic/martensitic type alloy claddings, HT9 and T91, swelling is comparatively less than that in case of other austenitic stainless steel alloys. The modified ferritic-Cr-Mo steels of the type 2.25Cr-1Mo and 9Cr-1Mo, with microstructure stability and good compatibility with the coolants, sodium and water, are used as stream generator components. In PFBR, nickel-based colmonoy alloys are sometimes selected as suitable replacement for the cobalt-based stellite alloys in steam supply system components. Similarly, austenitic stainless steel of AISI type 304 is the favored structural material for the construction of FBR reprocessing and waste management plants.

Generally, the oxide fuels, UO₂ and (U, Pu)O₂, are very inert in contact with stainless steels and zircalloys. However, in fast reactors with stainless steel alloy cladding of U-Pu fuels, there is a possibility of forming a low melting eutectic liquid phase of uranium and plutonium with the alloying elements, Fe, Cr and Ni, if there is a loss of coolant-flow. Addition of zirconium to U-Pu fuels helps in forming an U-15Pu-10Zr alloy with good compatibility with austenitic stainless steel type 304 cladding and also with sodium coolant which acts as high conductivity bond between the fuel and the clad during reactor operation. The metallic fuel-clad compatibility is further enhanced by the addition of Nb, V and Mo to cladding materials under specific conditions. In case of carbide fuels, there is a possibility of forming inter-metallic compounds like (U, Pu)(Fe, Ni)₂ and (U, Pu)(Ni, Fe)₂ with stainless steel claddings and also carburization of stainless steels due to dissolution of carbon in sodium coolant.
The uranium compound, U-Fs (Fs is an alloy of Mo, Ru, Rh, Pd, Zr and Nb), used as driver fuel in the Experimental Breeder Reactor-II, has been found to be of poor compatibility with stainless steel claddings and it has been replaced by the metal fuel, U-Pu-Zr, with better compatibility with the claddings in ANL. The Russian BR-5, a sodium cooled reactor, was fuelled with PuO2 pellets clad in Type 321 stainless steel and with helium bonding. Development on British DFR (Dounreay Fast Reactor), led to the use of two metal claddings, namely, vanadium for inner cladding and niobium for the outer depending on reactor temperatures.

**Aluminum**: The metal aluminum and its alloys have got important applications both as fuel matrix as well as a fuel cladding materials particularly in research reactors. The first light water-cooled and moderated Material Testing Reactor (MTR) installed in 1952 used an alloy fuel of UAl3 clad in aluminum. In British Magnox type reactor utilizing ‘adjusted uranium’ as fuel, the magnesium-base alloy, Magnox Al80 (Al 0.8%, Be 0.002 –0.05%, Ca 0.008% and Fe 0.006%) applied as cladding material provided good compatibility with CO2 coolant. Application of aluminum as clad in the plate type uranium inter-metallic compound, UAl3 or U3Si4 dispersed in aluminum matrix as fuel, ensured effective fuel-clad contact resulting improvement in corrosion resistance of the reactor fuel.

The advanced High Flux Beam Reactor (HFBR) at ORNL developed mainly to produce transuranic elements for research purposes used cermet fuel of U3O8 dispersed in aluminum matrix and an aluminum alloy 6061 (Mg 1%, Si 0.6%, Ca 0.25% and Cr 0.25%) with good corrosion resistance as cladding material. In water-cooled TRIGA reactor developed by General Atomics Corporation in 1950s used U-Zr hydride fuel clad in aluminum/zirconium.

**Coolants**

In nuclear reactors, the dual-use coolant materials which are circulated through the reactor cores, in general, ensure the fuel assembly systems to maintain structural boundary integrity, minimize the fuel-clad interactions and reduce the out-of-core radiation fields but these themselves should not be easily radioactively. The suitability of a coolant is dictated by its favorable characteristic properties of high thermal conductivity, low neutron absorption cross section, chemical stability, corrosion resistance and higher compatibility with the fuel and other associated structural materials. Both liquids and gases are being used as coolants primarily to transfer the heat released during nuclear fission to a suitable medium to generate electricity or simply dissipated as occurs in a research reactor. The important liquid coolants effectively used in different designed reactors include light water, heavy water, liquid metals like sodium, lead, NaK, Pb-Bi eutectic (LBE) and molten salts of FLiNaK and FLiBe. The most commonly used gaseous coolants are CO2 and helium. The Enrico Fermi Atomic Power Plant at Michigan was a U-Mo fueled and sodium cooled nuclear plant.

**Liquid Sodium**: The un-modерated high neutron flux FBRs with high temperature small reactor core volumes demand very efficient cooling systems. The high neutron flux causes radiation damage and degradation of the claddings and duct materials and may restrict the life of the fuel elements. The alkali metals especially liquid sodium having favorable thermodynamic and nuclear properties, suitable compatibility with different cladding materials and higher heat transfer capabilities has been extensively employed as an effective coolant in FBRs. The nuclear heat from the reactor core is removed by circulating liquid sodium from a cold pool and after transporting its heat to the heat exchangers mixes again with the cold pool. Utilization of liquid sodium metal as coolant makes a fast breeder reactor possible to operate at higher temperatures enhancing its nuclear fuel cycle efficiency as a whole.

Sodium does not moderate neutrons (fast neutron absorption cross section, ~ 0.87 mb) and its possible (n,) activation product, 24Na, is short lived (15 h) and the (n, 2n) product, 22Na, also does not build up much limiting the emission of radiation doses to working environment. The nuclear grade sodium metal has got the properties of low melting point (~186 oC), high boiling point (~1400 oC), high thermal conductivity, good corrosion resistance and high compatibility with the major dual-use structural materials. It is also compatible with mixed carbide fuels as coolant. Liquid sodium having density and viscosity very close to water permits its use in different hydraulic design parameters. The liquid metal cooled fast reactors are under operation in United Kingdom, France, Russia, Japan and India.

Since in case of liquid sodium, there is a possibility of formation of Na2(U, Pu)O4 with mixed U-Pu oxide fuel resulting in swelling and cladding rapture, alternatives like Na-Pb and Pb–Bi alloys have sometimes been considered as coolants. But the characteristic properties of higher melting point, corrosion and especially generation of radioactive 210Po by neutron capture in 209Bi and 208Pb have to some extent restricted their applications.

Sodium metal is generally produced by Down’s process. Sodium being an alkali is highly reactive and forms compounds with all non-metals excepting nitrogen and...
noble gases. It burns in air and reacts extensively with water.

In Molten Salt Reactors (MSRs), the fluid fuels and the blanket circulation systems are separated by an intermediate coolant like liquid sodium, helium or $^7\text{LiF-BeF}_2$. In MSRs, $^7\text{LiF-BeF}_2$ as coolant, displays excellent compatibility with Hastelloy-N (an alloy of Ni with Mo, Cr, Fe, Si and C), a preferred construction material for reactor vessels, pumps and heat exchangers, etc. The electricity generating Gas Cooled Reactors (GCRs), built in Britain and France generally use CO$_2$ as coolant and graphite as moderator. In High Temperature Gas Cooled reactors, the fissile and the fertile materials dispersed in graphite moderator are used as fuels and helium is used as coolant.

The presence of trace and ultra trace elemental constituents in nuclear fuels as well as in fuel sub-assembly components has significant affects on the overall efficiency of a nuclear reactor as these may cause significant changes in neutron economy, fuel integrity, thermal, mechanical and corrosion properties of dual-use structural materials and failure of fuel-clad-coolant interactions. The quality control of the trace constituents to certain specifications in the relevant media needs careful consideration in achieving the desired fuel burn-up in a reactor.

The management of radioactive wastes, in general, is of great environmental concern. In disposal of radioactive wastes, after separation of the most radio-toxic and heat generating fission products such as $^{137}\text{Cs}$ and $^{90}\text{Sr}$, lanthanides and ‘minor’ actinides, Pu, Np, Am, Cm, etc., present in spent fuels, the residual wastes are immobilized in an appropriate solid matrix like borosilicate glass, phosphate glass or ceramics like SYNROC and zirconolite (CaZrTi$_2$O$_7$), the choice of which is governed by its nuclear compatibility with the quality of the specific radioactive wastes. The immobilized solid wastes encased in stainless steel canisters are placed in deep repository of suitable geological formations for long term safety surveillance.

In the last decade of the 19th century, Becquerel in his discovery of radioactivity used photographic films as a suitable medium to detect and measure the intensity of the emitting radiations. During this long period, various sophisticated nuclear techniques and devices have been developed for detection, characterization and precise measurement of intensities of these emitting radiations in different media. The efficiency of radiation detectors depends on the physic-chemical and nuclear characteristics of the specific dual-use materials to be used for nuclear detection and/or the type of measurement to be made. As for example, in gas filled radiation detectors like proportional counters, Geiger Mullar(GM) counters, etc., gases like helium, argon, hydrogen, nitrogen, methane, etc. are a few fill-gases that have got important applications as dual-use nuclear materials. Similarly, BF$_3$, $^3\text{He}$ and $^4\text{He}$-gas filled detectors are exclusively used for neutron detection.

In gamma-ray spectroscopy, the detectors are also based on some specific dual-use semiconductor materials like germanium, silicon, etc. The inorganic scintillators, Bismuth germinate, Thallium activated NaI and CsI, High Purity Germanium (HPGe) and Lithium drifted Germanium detectors Ge(Li) are effectively employed in gamma-spectroscopic measurements. Silicon surface barrier detectors and Si(Li) detectors are respectively used in Alpha-spectroscopic studies and X-ray measurements. Beryllium metal has an useful application as a metal window for X-ray machine. Organic materials like anthracene, stilbene and some specific plastics are also utilized as scintillation detector materials. In early developments, paraffin was used in fabrication of moderator assemblies in neutron counters. In Thermo-Luminescence Dosimeters (TLD), generally used as TLD badges, small capsules of LiF and CaSO$_4$ (50–100 mg) doped with manganese impurity constitute the thermo-luminescent materials to measure the cumulative radiation doses received by the workers in radiation environment. Thus, varieties of organic and inorganic materials found their effective applications in nuclear instrumentations (detection systems/nuclear devices) as essential dual-use nuclear materials.

**Conclusions**

Extensive research on the development of advanced nuclear fuels and efficient fuel sub-assembly structural materials having favorable elemental properties especially nuclear characteristics highly compatible with the fuels has been continued since the installation of Fermi’s first atomic pile. Over the decades, development of varieties of improved fuel-assembly designs has resulted in significant increase in higher thermal margins in the reactor cores for a longer life-span (40–60 yr). Since the life and efficacy of nuclear fuels and the loss of integrity of the dual-use structural material used in a reactor are primarily dictated by the effective changes/degradations caused by the fuel-clad-coolant interactions in its hostile environment, it is essential to know the out-of-pile properties of the fuels and the major and minor structural components particularly their nuclear compatibilities towards the fuels so that corrective measures can be taken at the design stage of a reactor.
Knowledge on the evolution of novel nuclear fuels coupled with more efficient new alloys, ceramics, composites, etc. as dual-use structural materials may mitigate the challenges in achieving optimum reactor system performance under demanding conditions. Understanding of the characteristic properties and specifications of the fuel as well as fuel sub-assembly components can help to keep pace with the increasing world’s demand of energy with minimum environmental impact but with maximum safety as par regulations stipulated by the authorized Bodies like IAEA, ICRP and others.

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