Fundamentals of Nuclear Power Generation
Dr. Dipankar N. Basu
Department of Mechanical Engineering
Indian Institute of Technology, Guwahati

Module - 08
Breeder Reactors
Lecture – 02
Fuel cycles & FBR

Hello friends, welcome back to the second lecture of our module 8 where you are discussing about the breeder reactors. We already had one lecture and I personally feel I use quite awful in the second part of that lecture; as I was feeling a bit exhausted and drowsy and just went through the motion. So, probably I would like to clarify some of the topics which I feel I have not explained clearly at least in the later part of that second lecture.

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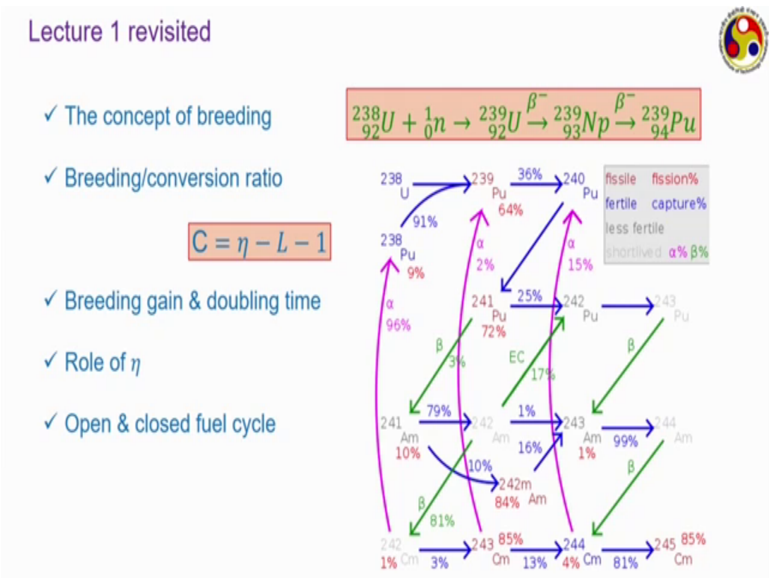

But what we have done; so, far is we have discussed about the concept of breeding. Breeding are refers to an example which is shown here this kind of reactions we term as breeding, where a fertile isotope or a fertile nucleus consumes one neutron in a non fission capture reaction that is like an n gamma kind of reaction where its absorbs a neutron.

So, thereby produces another isotope of the same one like uranium 238 in this example is capturing in neutron and producing uranium 239 plus gamma radiation. But this uranium 239 as we have seen has an extremely small half life and therefore, almost instantaneously it gets converted to neptunium 239 that also has a quiets small half life something in the order of 2; 2 days therefore, it also undergoes another beta decay to finally, produce plutonium 239 and plutonium 239 is a fissile isotope.

So, we started with the fertile isotope of uranium 238 and after one step of neutron capture and two steps of beta decay; it gets converted to a fissile isotope. This particular kind of reactions are called breeding and breeding is an extremely important concept from nuclear reaction or nuclear reactor operation point of view; because if you think that uranium 235 is a only fuel that we can use in nuclear reactor then our option is quite limited. Of course, among naturally occurring or among all the fissile isotopes that we can visualize, uranium 235 is probable the only on which is naturally occurring.

And isotopes like plutonium 239 or uranium 233 or a few others can artificially be produced by virtue of such kind of breeding reactions. But suppose if the power generation is dependent only on uranium then we know that the uranium ore that we get from the mines contains only about 0.7 percent uranium 235 and rest is this uranium 238; which is fertile in nature. And therefore, the if we have to depend only on uranium 235 for power production, we are going to utilize only 0.7 percent of what we are going to get from the ore. Or I should say maximum of 0.7 percent because there will always be some kind of practical loss, some portions of that U 235 itself also will remain unutilized etcetera.

But because of this option of breeding only that 99.3 percent of uranium 235 which till this chapter was looking almost an useless kind of thing; that actually can be very very useful because while it goes through those neutron capture reactions maybe in the resonant absorption zone or maybe in the thermal neutron zone that can get converted to plutonium 239; there by producing or adding some more fuel into the reactor; that means, we are loading the reactor with just uranium 238 and 235.

And uranium 235 is the only fuel if it is just loaded with natural uranium we are having only 0.7 percent of the total mass acting as the real fuel, but else time goes on of course, there will be depletion of uranium 235 because of the fission reaction, but they will also be depletion of uranium 238 and that will lead to the formation of this new fuel that is plutonium 239.

So, breeding can lead to this conversion of fertile isotope to fissile isotope; thereby adding some new fuels or developing some new fuels into the reactor. In fact, in modern day thermal reactors which are fuel purely with uranium they are as much as 30 to 40 percent of the total power production may come from plutonium 239.

Because Pu 239 is an extremely fissile isotope with significantly high fission absorption cross section and also the power that we generally harness from one fission reaction of Pu 239 is slightly higher than what we can get from U 238; U 235. I should say there can be quite a few other kinds of fission other kinds of breeding reactions also like we have shown through this diagram in the last lecture and just repeating once. Uranium 238 generally can have only one kind of nuclear reaction which is this neutron capture leading to uranium 239.

Now, U 239 and neptunium 239 both have extremely small half life hence therefore, we can just neglect them and just we can consider that U 238 can directly get converted to Pu 239; which is a fissile material and it also has as shown here it has a 64 percent probability of participating in fission reaction therefore, fission cross section is reasonably high, But it can also participate there is 36 percent probability of having a capture reaction that is U 239 after capturing a neutron in 36 percent of the total cases can go through a n gamma reaction producing Pu 240.

Now, Pu 240 is also a fertile material and it again has just a single probability of neutron capture and conversion of Pu 240 on which again is a highly fissile isotope. It has 72 percent possibility of fission, but it can also participate in couple of other reactions like there is 25 percent probability of a neutron capture reaction to produce Pu 242 there is also about 3 percent probability of having a beta reaction to produce am 241. But these are generally less fertile kind of isotopes and their half lives are quite small as well, but there are possibilities that this am 241 can and go through another neutron capture reaction; there will producing m 242 am which is a highly fertile sorry highly fissile isotope and it is 84 percent probability of participating in a fission reaction.

This way we can see almost a chain kind of reaction like another a very important part of this chain is that this A m 242 which let me erase this A m 242 which is appearing here is a very very short lived isotope with extremely small half life can go through a beta decay and to produce this curium 242 which can participate through a series of reactions to finally, get converted to Cm 240 first Cm 243 and then Cm 245 both them who are extremely fissile isotope extremely active. And generally these are the two isotopes they which are primarily considered for nuclear weapon fabrication.

So, both of them are extremely important another possibility the Cm 242 is to go through a alpha decay they were producing this Pu 238, which again being a fertile one in nature can participate in neutron capture to produce Pu 239. So, we are back where we started it is a therefore, you can clearly see it is a whole net your starting with U 238. And then there are different kind of possibilities of getting different kinds of fissile nucleus formation through breeding; how many fissile nucleus we can see in this net? Like Pu 239 as number 1, then we can find Pu 241 as the other one these are two different isotopes of plutonium, we can also see this curium 243 and 245 and in some cases we can also have this American 242 m.

So, there were five different fissile isotopes which we can get just by this U 238 participation in breeding reaction. And therefore, though we are having all uranium 235 initially as the only fissile nucleus; we can have five more in a nucleus with plutonium 239 being the most common on for the obvious reason because that is the deduct product of this uranium 238 breeding.

And as the fraction of uranium 238 is; so, high 99.3 percent in case of natural uranium natural uranium fuel reactors. So, there is a very high percentage of plutonium 239 also can be found in the reactors, but the most important thing that comes out from all these concepts of breeding is that by virtue of this breeding reaction a nuclear reactor can produce its own fuel. Or the potion of initial supply that is uranium 238 which is not a fuel a reactor can convert that to a fuel; thereby adding some more fuel to the originally supplied one.

And also something I shall be talking about later on after the reactions are over whatever is left with say once we have operated the reactor for a long period of 1 or 2 years then the spent fuel that is left with that also can contain significant amount of plutonium. And also small fraction of uranium 235 which can again through some kind of reprocessing can you head back to can be fed back to the reactors nuclear reactors. Therefore, even the waste material can also used as fuel the so, called waste material can act as fuel for nuclear reactors.

And I hope this terms or this whatever I am saying that sounds familiar to you. Because this here if we can make this breeding reaction or make proper use of this bleeding reaction, the total duration over which the same fuel can be used is enhanced by a large amount or significantly larged amount. For getting suppose we have a reactor which is giving you 1000 megawatt of power output.

Now, if we want to get that same 1000 megawatt only from uranium 235; then we have to supply a significant amount of fuel. But only because of this production of plutonium in course of the reaction we can supply much less amount of initial fuel and still attain the critical condition that is the total critical mass requirement that definitely reduces because of the breeding. And also in the very first lecture, I have mentioned under while I was talking about different advantages of nuclear power I have mentioned the term renewable.

And this is where nuclear energy can be renewable quite similar to the conventional renewable energy like solar or wind, it can produce its own fuel and thereby the supply of fuel can be almost infinite just to put that in context with the plane uranium base reactions. If we are considering only uranium 235 was the fuel uranium being in metal which is found in normal ores its total life is not very high.

Like if we consider the present if we take the present day electricity consumption or power requirement over the entire world as the benchmark then presents stock of oils and natural gaseous may lost only about 30 years whereas, present stock of coal is expected to last something like 50 to 70 years. In for uranium 235 or just natural uranium I should say if we are depending only on that then this nuclear energy also is expected to last only something like 70, 80 years because after that probably the stock of uranium will get exhausted.

But we can use this breeding technology go to this more fuel thereby reducing the consumption of natural uranium. And proper use of this breeding technology a projection shows that the total life of the same uranium which we presently have can be something like 30000 years. So, that definitely is the remarkable figure to use and also another fuel that another kind of breeding reaction that we have seen in the previous lecture thorium

232; thorium 232 can absorb neutron and produce uranium 233 which is again an extremely fissile isotope.

And thorium stock at the world at the present is at least 3 times more than uranium. So, thorium will definitely be lasting much more compare to uranium and we can produce this uranium 233 from thorium; there by getting a very very long supply of fuel almost over an infinite period of time. So, renewable energy from that point of you can also be treated as the or I should say nuclear energy from that point of view you can also be treated as the renewable source of energy. Because it is near infinite or 30000 years for the moment we can also definitely consider to be an infinite period of time.

Now, moving further we have discussed about the breeding or conversion ratio which is defined by this C and it is defined as the ratio of number of fissile isotopes available through breeding divided by initial number of fissile isotopes which are consumed during the reaction. And that can be related to the thermal fission factor and the total leakage by relation like this.

Here one thing I would like to add with our previous discussion; C is generally called the conversion ratio and as we have already seen depending upon the value of C we can call a nuclear reactor to be a breeder a converter or a burner. When thus value of C is greater than 1 then we call that a breeder reactor because then it is able to produce is one fuel.

And then instead of using the convert term conversion ratio generally the term breeding ratio I used; as long as C is less than 1 like in case of burners and converters who is take with the conversion ratio, but when C is greater than 1 that is when we are talking about breeder we call it breeding ratio. We have discussed our breeding gain and doubling time doubling time is the time required to double; the number of fissile nucleus that is there that for common thermal reactor the doubling time is something approximately of 10 years.

So, doubling time is an in a way is an indicative of the reactor producing the fuel for another reactor means we have supplied some amount of fuel to get the reaction started. And after this doubling time something like the 10 years or over then there will be enough fuel store inside which is sufficient to run two reactors. So, half of that can be taken out of the reactor and we can start another reactor for this. So, the doubling time is also very important to know at least for from the operator point of view.

The role of eta we have discussed and we briefly discussed about the open and closed fuel cycles; this is where I went through the motion a bit and I would like to come back here again.

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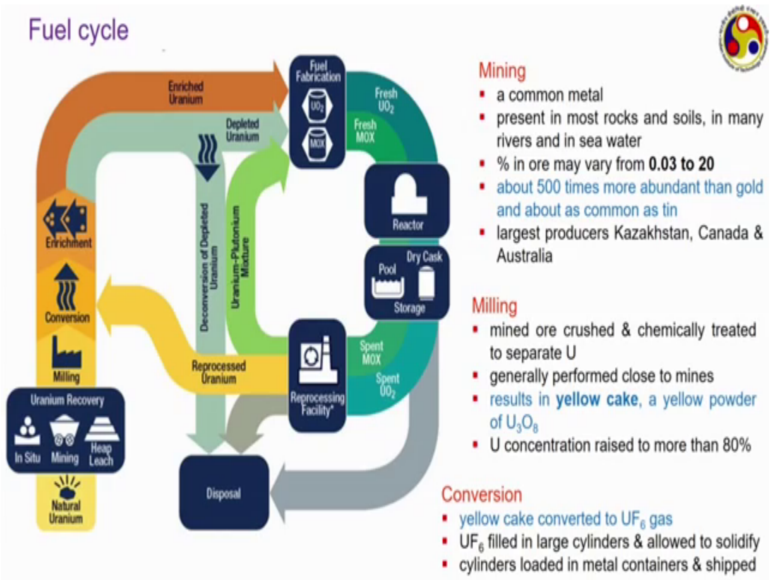

So, to start with we have this which is the fuel cycle or the nuclear fuel cycle; there you can see there are several steps here the first step of course, is the mining the mining from here there refers to the harnessing the uranium from the core of the earth.

Now, my uranium is a very common metal and it is present in most of the rocks and soils also in many rivers and even in the sea water also. And the ore that we get generally from the mines contains 0.3 to 20 percent of uranium. So, that is quite large range, but of course, here we are talking about the natural uranium that is whatever we are getting from the ore 0.03 percent or 20 percent whatever only 0.3 percent of that is uranium 235. There are principally 3 kinds of methods of mining which are shown here; it can be the in situ mining or in situ harnessing the mining from the depth of the mines or also the heap leach technology.

Now, uranium actually is a quite abundant metal about 500 times more abundant than gold and something very common metal like tin has nearly the same kind of awareness as uranium. So, getting uranium at least at the moment is not very difficult there are several sources or several countries where uranium is available the largest producers being Kazakhstan, Canada and also Australia. These are the countries which are the largest contributor in total uranium stock of the world, but there are several other countries where uranium is available.

Once we have got this raw ores from the mines then we need to go through the milling process. Milling refers to the ores that we have harness from the mines they are crushed and they are also chemically treated to separate out the uranium from whatever else was there in that ore. Generally the milling sites are located very close to the mines because then uranium ore that we get from a mines that is a very mildly radioactive, but still that should be considered very very carefully. And operator should be cautious that is why general it is not allowed to travel allowed or that ore is not allowed to travel long distance before the milling process is run we do it quite close to the mines itself.

And then the result of that milling process is something known as the yellow cake; it is actually an yellow powder of U 3 0 8; uranium can have several chemical balances accordingly it can have several kinds of oxides U 3 0 8 is just one of them. And, but the from the milling process the principal product is this yellow powder of U 3 0 8 which conventionally is called the yellow cake. And the uranium concentration in original where it was just 0.03 to 20 percent through this milling process it can be raised something more than 70 percent quite often in the common mills; it can be generally in the range of 90 percent or more.

So; the yellow cake that we have found which is U 3 0 8 that contains more than 80 percent uranium; next is the conversion strip. Conversion refers to the conversion of this U 3 0 8 to UF 6 uranium hexafluoride which is actually a gaseous material. So, UF 6 after this conversion UF 6 is filled in large cylinders and allowed to solidify and then the cylinders are loaded into metal containers and shipped to the plants or generally if some other enrichment or some other kind of chemical processes are required it is supplied here.

So, what we are getting at the enrichment location is this UF 6; I repeat UF 6 is the gaseous material whereas, the U 3 0 8 that we are getting from the milling plant that is that is an yellow powder.

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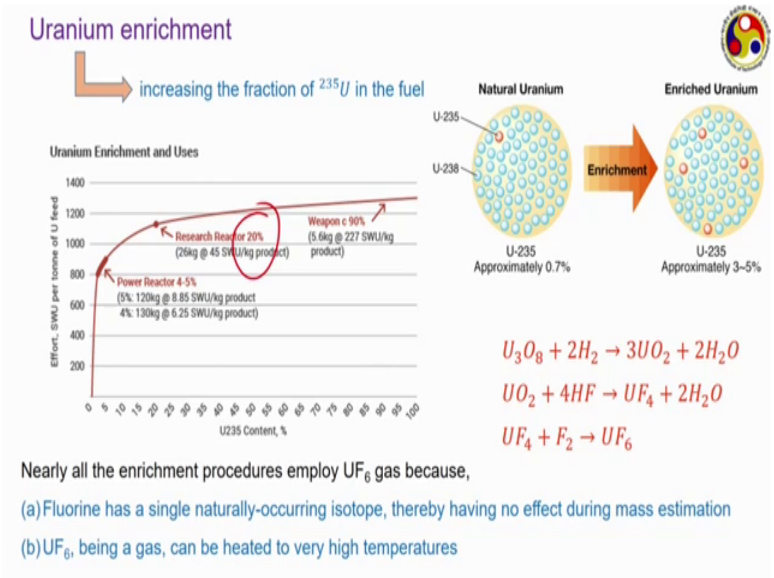

Next comes the enrichment; enrichment is an extremely important topic in nuclear power generation because we have already discussed in an earlier module about the thermal fission factors etcetera. And from there you know that as the fraction or percentage of U 235 in the uranium increases thermal fission factor also increases when in natural uranium were where we have only about 0.3 percent of U 235; the thermal fission factor is something like 1.36 and with increase enrichment that increases rapidly and around 18 to 20 percent of enrichment it more or less saturates to something like 2.05 or 2.06.

So, it is and as the thermal fission factor increases overall multiplication factor reactivity of the plant also increases and hence the power output also keeps on increasing. And enrichment is also a quite controversial technology because this is generally while the major process of enrichments are quite standard, but exactly how that is done or exactly how those processes are controlled those are quite complicated technologies and every country has their own process and generally likes to keep that as a secret.

Very few countries properly can enrich the uranium fuel; whereas, other countries who do not have the technology does not have the technology, but still are running nuclear plant has to go for either natural uranium or export that from some other countries. So, the term use the enrichment definitely refers to increasing the fraction of uranium 235 in the fuel something like this in natural uranium it is only about 0.7 percent.

So, the natural uranium ore that we get or even after the milling and conversions to shut down; whatever uranium we are getting that is natural uranium only which contains only 0.7 percent of U 235. And because of this enrichment process we can increase the fraction of U 235 to whatever level we would like to have.

Common thermal reactors use enrichment in the level of 3 to 5 percent. In fact, it can be quite low like 1.5 to 2 percent as well in PWRs, but pressure is heavy water reactors I hope you remember does not use enrich fuel because it uses deuterium as the moderator which has near 0 absorption cross section. And therefore, it can go for a natural uranium, but PWRs or BWRs uses ordinary water and water has a small absorption cross section.

So, some portions of the neutrons will get absorbed; to increase the probability of fission occurrence on neutron nucleus interaction we need to go for some kind of enrichment which is generally limited to 4 to 5 percent, but that is in thermal reactors. When you are talking about fast reactors we need much higher level of enrichment because I hope you remember the fast fission cross section or the cross section fission cross section subjected to fast neutrons are extremely low for all common fissile nucleus U 233 or U 235 compared to their thermal fission cross section fast fission cross sections are quite small. And therefore, we need a high neutron density high neutron flux density to achieve the fast reactions or fast fission.

And that is possible only when we can increase this number of U 235 isotopes in the total fuel. Generally the power reactors or industrial reactors uses an enrichment of the level of 4 to 5 percent whereas, research reactor can go up to 20 percent, but in case of fast reactors just what I was talking about it is it is not just 4 to 5 percent it can be quite high something in the range of 10 to 15 percent or even 20 percent in some rare cases. And for nuclear weapons we need even larger enrichment which can be while above 80 percent can be in the range of 90 to 95 percent as well.

So, depending upon for what purpose we are trying to go for enrichment they have to decide which level you are which level or which kind of applications we are focusing on. There are several ways we can handle or we can have this enrichment, but almost all of them utilizes UF 6. The advantage of fluorine is that it has a signal naturally occur in isotope; therefore, whenever we are dealing with UF 6 generally all those enrichment procedures focuses on separating uranium 235 and 238. And their chemical properties are virtually the same because the number of protons in both the isotopes are same. And it is the number of protons that is the atomic number which determines the chemical properties.

But as the number of neutrons are different that is has U 238 nucleus has 3 more neutrons; so, it is slightly heavier. And that small difference in mass is the one on which this enrichment technology or separation technology is based on. Now if fluorine has multiple kinds of isotopes then it will become very difficult to identify the difference between 2 UF 6 molecules are coming because of the difference in fluorine or difference in uranium, but thankfully fluorine has single naturally occurring isotope and therefore, whenever 2 molecules are having different weight we can clearly say that the heavier one condense U 238.

And another advantage is UF 6 is gaseous and so, we can go to very high temperature levels which are generally required for the process enrichment process. This is the way UF 6 is produced during that conversion process for that U 3 0 8 that we are getting in the form of that yellow cake that is treated with the hydrogen to form uranium oxide UO 2 uranium dioxide. I should say and that UO 2 is then reduced using hydrogen fluoride to get UF 4 which further reacts with fluorine to form UF 6 on that UF 6 gas is then taken to that enrichment plant.

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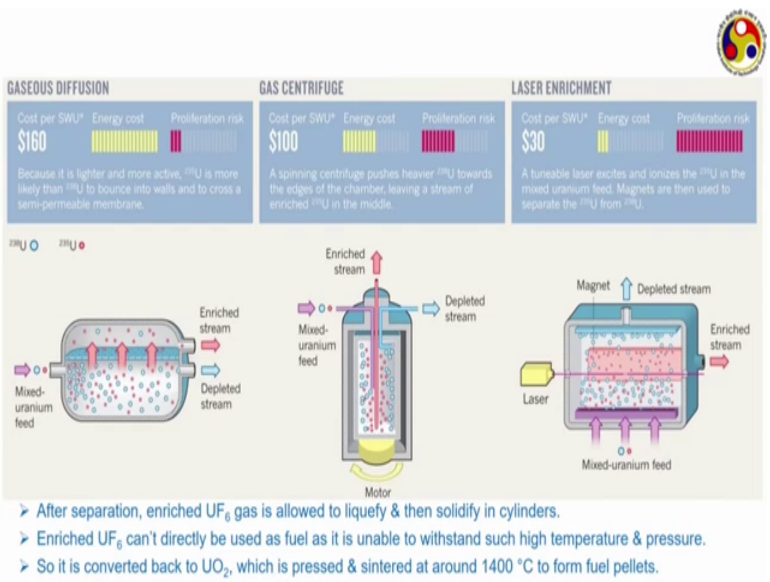

These are the 3 methods of enrichment which are the most popular one; gaseous diffusion is the older technology initial enrichment procedures where based upon the gaseous diffusion only, but nowadays it is becoming almost absolute in the fever of the gas centrifuge. The biggest problem in gaseous diffusion is it is extremely costlier it requires extremely high level of energy which is significantly smaller in this centrifuge gas centrifuge process.

In fact, an estimation shows that the amount of energy consumed during gaseous diffusion can be as much as 45 to 50 times larger then what requires in a gas centrifuge with equal amount of output. Now, coming to the technology in case of gaseous diffusion, we generally use a membrane a porous membrane. Now uranium 235 atoms being lighter and also more active they are more likely to move around in the container and strike with the walls and whenever this strives that porous membrane it can cause that permeable membrane. So, there is a larger probability of uranium 235 to cross this membrane compared to uranium 238.

But of course, just one step of membrane is not sufficient rather a diffusion process can involved a few hundreds maybe more than 1000 number of such membranes; all having same dimensions sometimes small difference in the nature, but ultimately what we get is on one side of this membrane or group of membranes, we have the original fuel or I should not say original fuel rather on one side of the membrane we get enriched fuel on the other side you get the depleted fuel. Here enriched means the one which contains more uranium 235 than the natural one whereas, the depleted one refers to the one which contains less amount of uranium 235 compared to the original one.

So, the diffusion process is associated with or it is based upon the lighter uranium 235 showing larger activity and thereby crossing the membrane. In case of gas centrifuges we use a centrifugal action that is as the centrifugal force is put into the action on the UF 6 meter gas through a spinning centrifuge, the uranium uranium 238 or the molecules continue uranium 238 being heavier they will move closer to the walls. Whereas, uranium 235 containing molecules are more likely to stay close to the center of the centrifuge or centrifugal force and therefore, if we collect the gas close to the centerline there will be a much higher fraction of uranium 235 compared to uranium 238.

So, the enriched stream will come out collected somewhere very close to the close to the center of this centrifuge; whereas, the depleted steam is collected from depleted stream is collected from the ages of the container towards the or near to the wall of the container. The third one laser enrichment gas centrifuge by the way is the technology which is used in or used almost everywhere. Now laser enrichment is a newer technology which is even cheaper energy cost is extremely low here a tunable laser excise and ionize the uranium 235 in the initial supplied mixture and then used to separate the uranium 235 from uranium 233 using a magnetic field.

So, I do not want to discuss much about this laser enrichment because it is something still under research and here to go for any kind of commercial application. After this separation process is done following any one of them gas diffuse centrifuge being gas centrifuge being the most common one enriched UF 6 gas is allowed to liquefy.

And then solidify in the cylinders and then we of course, cannot supply the UF 6 directly as a fuel in a nuclear reactor because the temperature UF 6 can survive up to is not sufficient for that; at such high pressure and temperature conditions inside a nuclear reactor UF 6 will not be able to sustain. And so, we have to convert this UF 6 back to UO 2, but remember that you the we are talking about is now an enriched one that is more number of UF 6 molecules will comprise U 235 then U 238 or then originally we had.

And then this conversion to UO 2 is done which is generally a solid material or a powdery kind of material it is for pressed and sintered at around 1400 degree Celsius to form fuel pellets. Pellets can be of rectangular block shaped or cylindrical shaped like you have talked about earlier. This 1400 degree Celsius up to which we need to center this UO 2 is too is too high for UF 6 and that is why we do not use UF 6 directly as the fuel.

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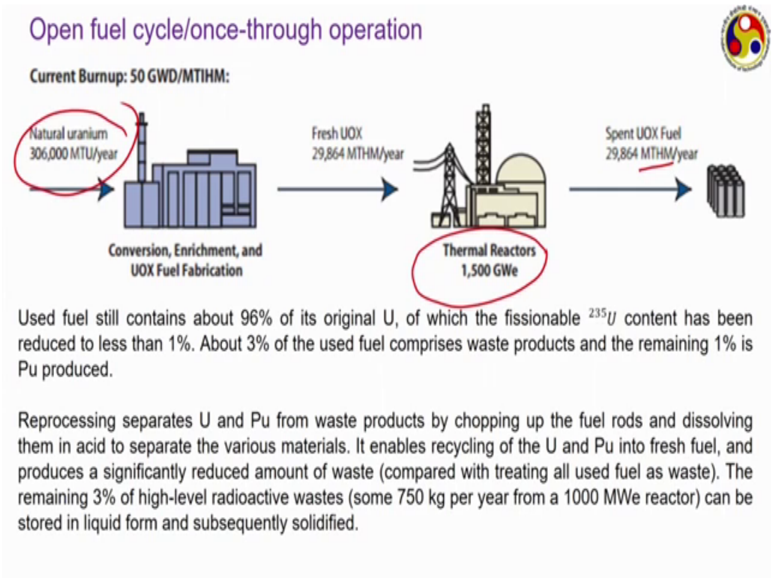

Now, first in the fuel cycle in the first one we can have first kind of operation is the open fuel cycle where this natural uranium after going to conversion enrichment and uranium oxide fabrication is supplied to the thermal reactor and whatever comes out of the thermal reactor that is just thrown away as the spent fuel. Therefore, to produce a thermal power of 1500 Gigawatt using this thermal reactor we need something like this amount 306000 megaton or metric ton of uranium over a period of 1 year.

And 29.864000 of MTHM per year becomes goes for waste. Here this unit MTHM refers to metric ton of heavy metal that is we are not talking about uranium oxide mass rather; we are talking about the mass of this metal part that is uranium part only. Initially the supplier 306000 natural uranium, but we are getting nearly 10 percent of that as the final spent fuel which is quite large.

And the answer is spent fuel that comes out there still contains about 96 percent of the original uranium; out of which the U 235 content can be less than 1 percent. And also there can be 1 percent of plutonium and rest about 3 percent used for are the use fuel comprising waste products which can further go for processing.

The step after this conversion or often after the power production is the reprocessing to; in order to harness the fuel whatever fuel lived in the spinned portion we go through the reprocessing process. Here the uranium and plutonium are separated from the waste product very first chopping up the fuel rods and dissolving them into an acid to separate.

The various products the recycling of uranium plutonium it is the fresh fuel produce a significant reduction in the total amount of waste.

So, whatever total amount of waste that we are getting if there is no recycling then that will also contain all unburned uranium or plutonium or I should not say unburned, it is that portion of uranium plutonium which is not participate in the reaction. And if we can somehow separate them out then it can be mixed with the fresh fuel and can again go back to the reactor.

So, about a significant amount of this waste can be saved of course, about 3 percent of the total mass contains high level radioactive wastes which need to be stored in liquid form and subsequently solidify.

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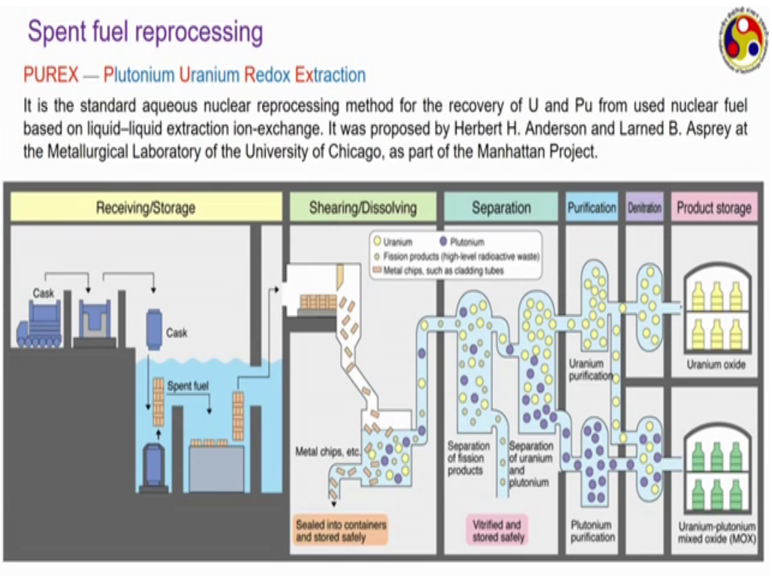

For the standard procedure that is followed for spent fuel reprocessing and it is called PUREX Plutonium Uranium Redox Extraction. It is a standard aqueous nuclear processing method for the recovery of uranium and plutonium from the used nuclear fuel. It was proposed by Anderson and Asprey at the Metallurgical Laboratory of the University of Chicago as a part of the Manhattan project it was done around 1947. This is the standard chart of reprocessing the un use fuel or the spent fuel from here that is taken to the shearing or dissolving unit where they are like a mentioned in the previous slide they are chopped into smaller units.

The cladding is also removed from this metal chips refer to the cladding material. And then which is going out here and then whatever is left those uranium and plutonium they go for the separation stage where the fission products are separated and only uranium and plutonium that remains. So, this is the step where plutonium the fission products all those long living fission products are separated out from the uranium and plutonium that is the remaining fuel.

The fission products which are separated in this block; they are then stored in suitable containers and goes for long term storage; we shall be discussing about the process of this fuel storage or I should say the storage of the waste product into in our last module. But the uranium and plutonium that we have that goes for purification where they are also separated from each other generally using some kind of chemical procedure. Then we can have two ways generally nitric acid plays a big role in this all chemical reactions that we are having.

So, denitration is the process of removing the nitrate part from the fuel compound that we are having. And before we go for the final product storage we can generally go for two kinds of option one is restoring pure uranium oxide, other is storing the uranium plutonium mixed oxide or the; so, called MOX. The role of MOX is somewhat similar to the enrichment because the objective of enrichment is to increase the U 239 percentages total fuel or I should say the increase in the fraction of fissile nucleus in the total fuel.

And if we go for a MOX that effect is also similar because plutonium itself is radioactive for I should say it has a high fission cross section. And therefore, adding plutonium into uranium that will also lead to a increase in the fission fraction of fissile nucleus in the total mixture. So, these are very rough diagram instead of going to the detail here, we can just can identify what are the possible steps during this reprocessing.

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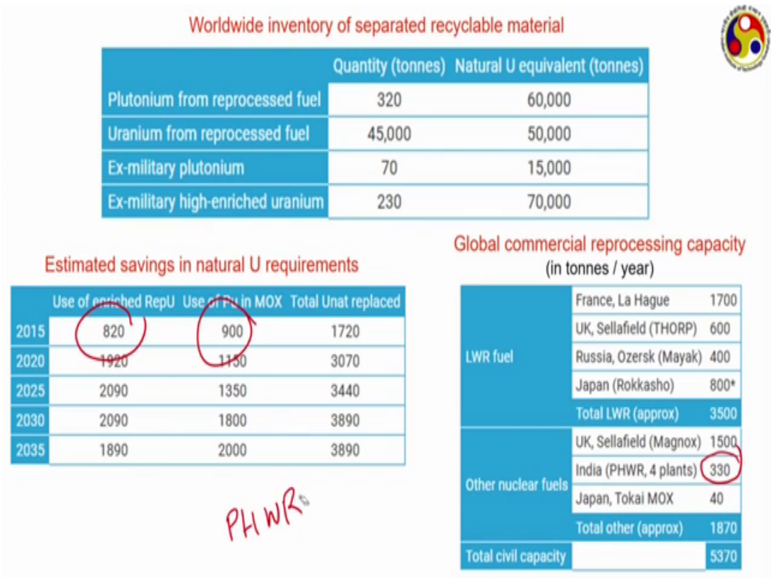

Worldwide inventory of separated recyclable material

	Quantity (tonnes)	Natural U equivalent (tonnes)
Plutonium from reprocessed fuel	320	60,000
Uranium from reprocessed fuel	45,000	50,000
Ex-military plutonium	70	15,000
Ex-military high-enriched uranium	230	70,000

Estimated savings in natural U requirements

	Use of enriched RepU	Use of Pu in MOX	Total Unat replaced
2015	820	900	1720
2020	1920	1150	3070
2025	2090	1350	3440
2030	2090	1800	3890
2035	1890	2000	3890

Global commercial reprocessing capacity (in tonnes / year)

LWR fuel	France, La Hague	1700
	UK, Sellafield (THORP)	600
	Russia, Ozersk (Mayak)	400
	Japan (Rokkasho)	800*
	Total LWR (approx)	3500
Other nuclear fuels	UK, Sellafield (Magnox)	1500
	India (PHWR, 4 plants)	330
	Japan, Tokai MOX	40
	Total other (approx)	1870
Total civil capacity		5370

Reprocessing very important role in nuclear industry because only through this reprocessing step; we can recover a significant portion of uranium and plutonium plutonium from the spent fuel; if this stay for not done then though all those uranium and plutonium plutonium would have been lost to the waste product.

If we see the global inventory of separated recyclable material the values for uranium and plutonium this are given here, but natural uranium equivalent that is the 45000 of uranium that we can reprocess or we can get from the reprocess fuel is equivalent to 50000 tonnes of natural uranium.

And very interestingly, if we can get 320 tonnes of plutonium from the reprocess fuels; it is equivalent to 60000 tonnes of natural uranium because of its much higher burn up. And the corresponding saving in natural uranium requirement is shown in this particular table in 2015; it is only a projection starting from 2015; total amount of enriched and reprocessed uranium comes to be something like this and use of plutonium or plutonium used in MOX is something in this.

So, the total uranium natural uranium replaced can be 1720 tonnes that is in 2015; we have received about 820 tonnes of enrich uranium and 900 tonnes of plutonium has been gone to fabricate this MOX. So, a total 1720 tonnes of; so, 1720 tonnes of uranium natural uranium has been replaced by this. There are several other countries or several countries where this process is also going on this reprocessing it is where for LWR; that is LWR type of fuel where like PWR BWR etcetera.

And these are the common facilities that we have whereas, other nuclear fuels are being strike are being tried in here the role of India is also important. In India PHWR is a primary kind of reactor which use a heavy water has about moderator and coolant. And we can see from this about 330 tonnes of material or uranium can be recovered per year from this 4 PHWR plants. So, a total of 5370 tonnes per year of fuel can be saved.

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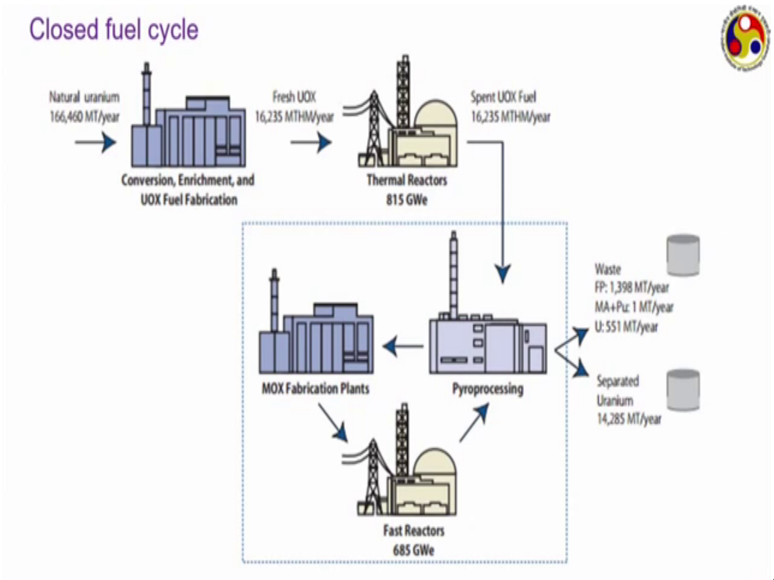

So, the closed fuel cycle is the one which makes use of this MOX; you can see the natural uranium is supplied here which gets converted to the uranium oxide during the enrichment process enrichment after or after enrichment I should say which goes to a thermal reactors producing this amount of energy. But the spent fuel from a thermal reactor goes through this reprocessing process like by the process and also the MOX fabrication this MOX is supplied to the fast reactor to get 685 Gigawatt of energy.

So, if we add this 850 or sorry if we add this 815 Gigawatt with 685 Gigawatt; we get the total 1500 megawatt of power, but this will lead to a significant reduction in the total fuel consumption because we are using MOXs. And finally, the waste product that we are getting I would urged to compare the values with the what you got in the previous one.

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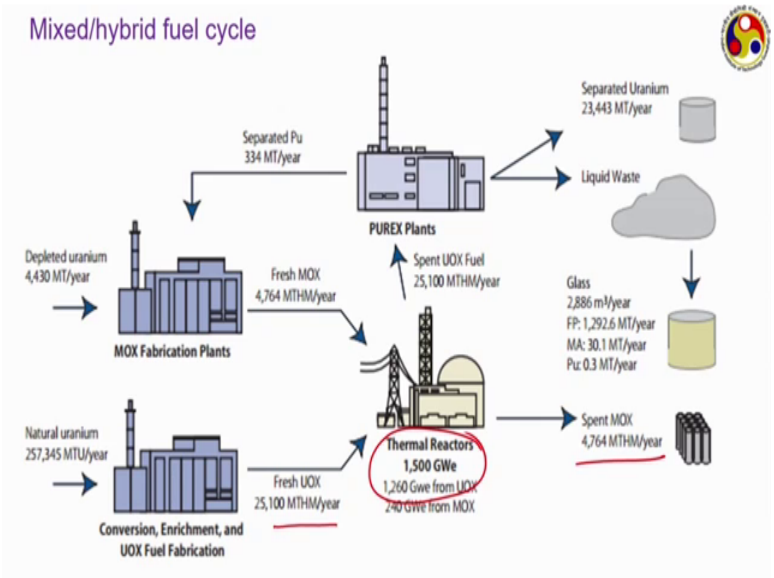

This is the mixture hybrid fuel cycle where the thermal reactor itself is the capacity 1500 Gigawatt, but it is receiving both a fresh stock of uranium oxide from and also a fresh stock of MOX from the MOX salutation plant. Both are supply to the thermal reactor and finally, this amount of thermal this amount of MOX will not be spent; spent is this comes out after from the reactor. All these 3 models have its one advantages and disadvantages, but generally the closed fuel cycle or hybrid fuel cycle are more prepared in modern day generation to a generation 4 reactors.

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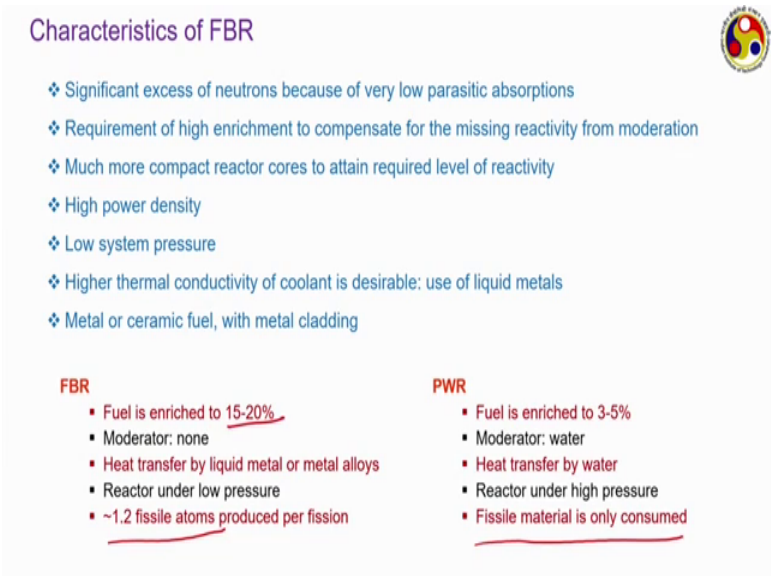

Now, let us come back to the fast breeder reactors; our discussion starting fast breeder reactors, but we have discussed about that concept of breeding in the previous lecture and here so, for we have talked only a mostly about fuel and fuel processing. Let us quickly check what can be the characteristic of fast breeder reactor which I am sure all of you can guess by now. Significant excess of neutrons because of very low parasitic absorption, there is no moderator inside this and coolant can be an liquid metal coolant whose parasitic absorption generally is quite low.

So, high enrichment is generally required as there is no moderation and so, the neutrons will not be able to come to the thermal neutron level and for fast neutron level their absorption cross sections are quite low. So, we need a large enrichment in order to provide more sites for interaction within neutron a neutron and nucleus. Then much more compact reactor core to attain required level of reactivity the total size are the critical mass total critical mass requirement for a given power output generally smaller for this breeder fast breeder reactors. And therefore, core size needs to be much smaller, power density reasonably high, system pressure is low they or can work models is a that much pressure itself.

And the thermal conductivity showed by the coolant thermal conductivity for the coolant has to be very high as because of this high energy density earlier. That is the reason to go for liquid metals metal or ceramic fuels are used with metal cladding. Now we are in a position to compare the fast breeder reactors and PWRs, if we go point by point in case of PWRs we use only lightly slightly enriched fuel 3 to 5 percent whereas, in FBR we go for very high level of enrichment 15 to 20 percent. The moderator imperial versus water no moderator required in case of fast breeder reactors.

Because you want to operate with the fast reactor itself; heat transfer happens to the water because water is the coolant in PWRs, but for breeder reactors we have liquid metal or metal alloys acting as the coolant. Breeder reactor also works at low pressure, but the pressure is high for the PWRs something in the something in the range of 70 bar. And also breeder reactor by deficient they are able to breed new or nucleus and their total number of nucleus total number of nucleus that has been developed through the breeding is replaced with this 1.2 number of fissile isotopes.

But there is no such kind of provision in PWRs the fissile materials are consumed and corresponding replacement with plutonium is extremely small.

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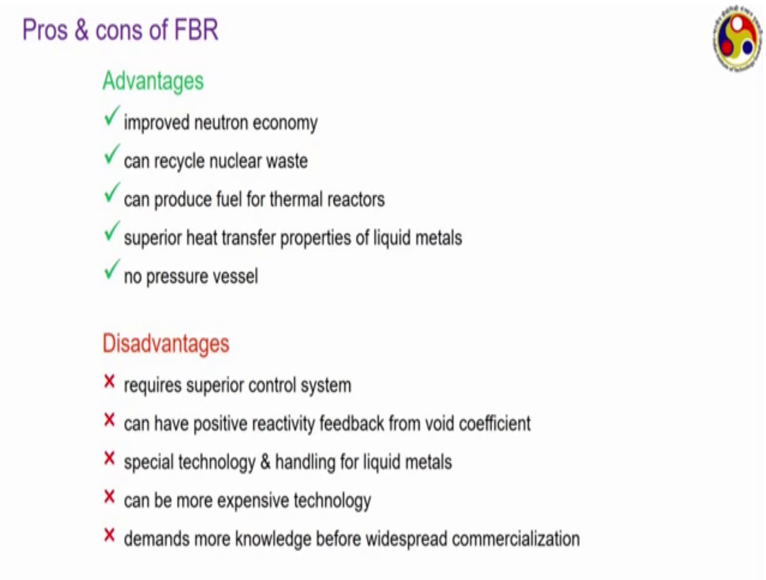

What are the pros and cons of fast breeder reactors? Of course, improved neutron economy because it can give much higher power output corresponding to a given neutron flask, it can recycle the nuclear waste which on which we have had lots of discussion so, far. It can produce fuel for thermal reactors in like the doubling time is something that refers to doubling in the number of total fission level nucleus.

Now when doubling time has been attained that reactor or excess waste fuels can be taken out; can be strip process slightly those waste product can be taken out and can be process slightly. And in course of doubling time; it will produce the same number of new fuels as it was or really had. So, that new fuel can be used to start a new thermal realtor. Then liquid metal has much better heat transfer characteristics compared to water which is advantageous, no pressure vessel that is another big advantage also, but there are also disadvantages like superior control systems are required the fuel being highly enrich.

So, control should be optimum in certain situation a fast breed reactor can have positive reactivity effect from void coefficient. The void reactivity effect or temperature reactivity has to be negative as for definition, the handling of liquid metal is also not easy they are heavy, they are corrosive and they are generally electrically conducting as well. Therefore, special technology and handling is required FBR can be expensive compared to PWRs. And one final point the knowledge of FBR is still not as FBR as like PWRs or BWRs; only very few countries just 4 or 5 countries are presently having active fast breeder reactor plants in there in their city and the technology that they apply they are or employ they are that those are also not publicly available.

So, we need to know more need to understand more about before going for common; you need to know more, we need to discuss more about possible kinds of ways we can control the reactor. And at the design stage itself everything should be proper and that can come only through rigorous research and experimentation at laboratories.

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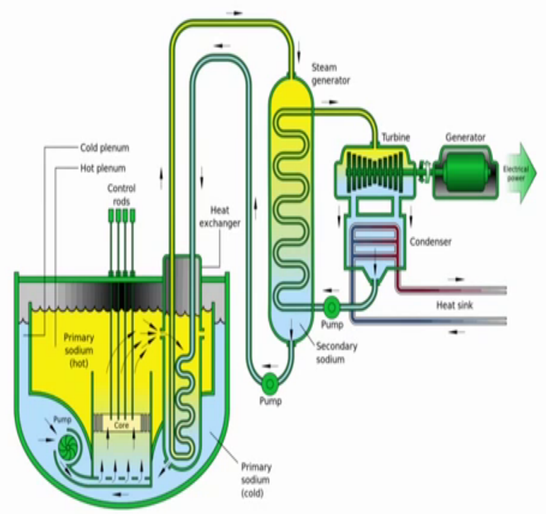

Sodium-cooled fast reactor (SFR)

This is one sodium cooled fast breeder reactor, you can clearly see here you have a big pool of sodium. The core is immersed into this and this intermediate heat exchanger is also immersed in this big pool of sodium.

So, the energy released by fission is observed by this coolant, which passes through this line going through the steam generator. In the steam generator water the water is allowed to receive the heat from this hot sodium; there by getting converted to steam and that steam is can be utilize in turbines and generators.

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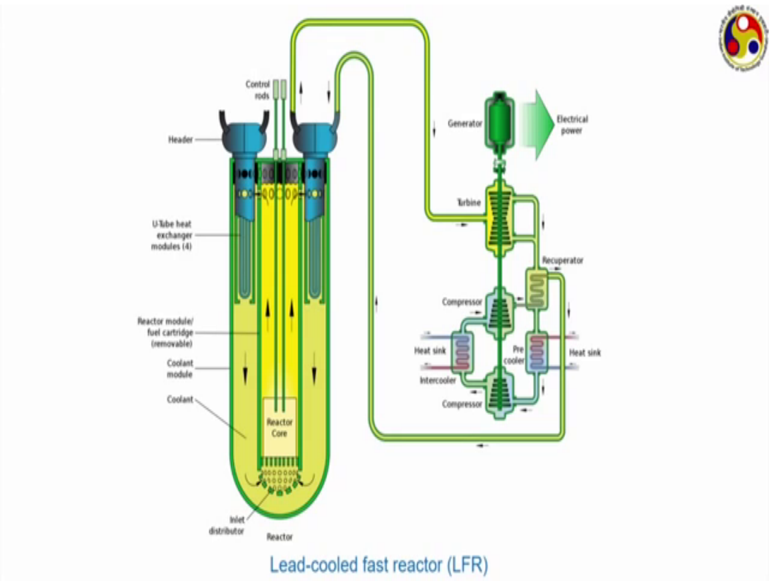

This is another type of design which is the lead cooled fast reactor, where the working medium can be a generally is a gaseous one. And the working medium passes through a gas turbine then you can go through fresh steps of ostriches of recuperator and compressor and heat sink.

So, both this lead cooled fast reactor and the sodium cooled fast reactor; they are under the generation 4 initiative and. So, none of them have any active plant at the moment.

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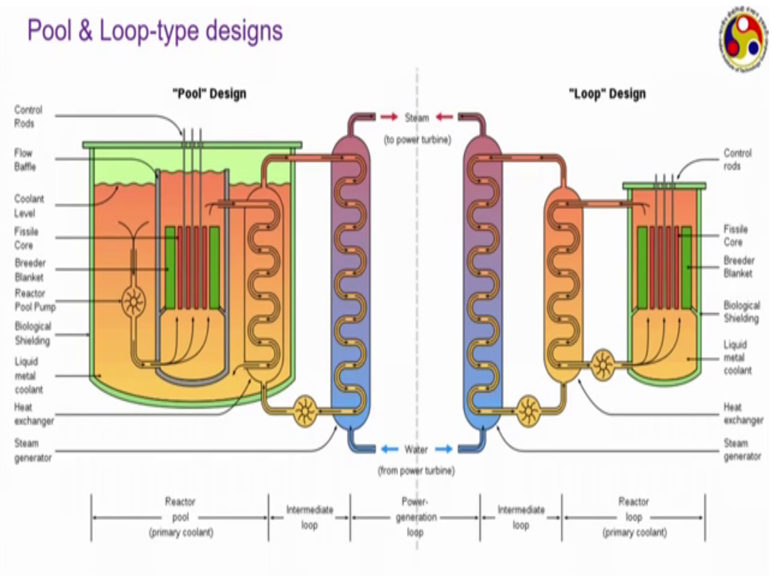

Finally, is the pool and loop type design; nuclear reactor as the two diagrams that we have seen, if you look carefully their figures are looking quite different and accordingly fast breed reactors can have two kinds of design. So, one is the pool type design another is a loop type design; the sodium cooled fast reactor that you have seen that actually adults the (Refer Time: 49:21) that is the pool type of design. In case of pool type design we have a large uranium pool sorry large pool of liquid sodium.

And the reactor core, the pump which is this one and also the intermediate heat exchanger like this one; all are immersed into the pool itself. Liquid sodium is pumped by this pump to pass through the reactor core from here it gets the energy. And then it enters this intermediate heat exchanger, through the intermediate heat exchange while passing through the intermediate heat exchanger; it is also in indirect contact with the, while passing through this intermediate heat exchanger liquid sodium transports it energy to another line; where the secondary side we have steam or water this water boils to produce the steam.

Sorry I am wrong, actually I would like to correct here this intermediate a loop is also having sodium as its working fluid. So, thus I am drawing again the sodium which is coming from the core here with high temperature; while passing through this intermediate loop that passes is energy to another stream of sodium which in turn goes through this second heat exchanger and transfer this energy to the water which is entering this from here. So, steam comes out which can be supply to the turbines.

But in case of loop type design; here this pump and intermediate heat exchangers are outside the liquid fuel. The core is still immersed into the pool, but the pump is outside which forces the liquid sodium to pass through the core and then enter to this line to the intermediate heat exchanger. Intermediate heat exchanger in the sodium in the intermediate heat exchanger transfers heat to another stream of sodium and goes to a steam generator like in the previous case.

So, in case of cool type design; we have core intermediate heat exchanger and the sodium pump all immersed into a pool of liquid sodium. Whereas in case of loop time design it is only the core which is immersed into liquid sodium, the intermediate heat exchanger and the pump are outside. Both designs have their own structural advantages and disadvantages; while the pool type design is mostly preferred in USA loop type design is more popular in European countries.

And also the designs that you have seen in the previous slide if we just go back is a sodium cooled fast reactor, you can clearly see this is the core this is the pump and this is the intermediate heat exchanger. And they all are immersed in this pool of liquid; so, this is a pool type design. And then this sodium is going to this sodium is transferring energy to this second stream of sodium, liquid sodium and that in turn is supplying its energy in this steam generator to this stream of water which is coming via this line.

So, this is a pool type design, but if you see this lead cooled fast reactor; here the reactor core is the most, but this heat exchangers are kept separately and it is revealed not a loop type design. But it is also not a proper classical pool type design as well or loop because here the reactor core and the heat exchangers are kept in separate blocks.

So, modern generation 4 designs are sometimes trying some kind of combination between pool and loop type in order to take advantage from both of them. So, today's lecture is up to this we I have one more lecture left on this fast breeder reactor, where I shall be talking about a few other relevant aspects of this fast breeder reactor. So, whatever queries you have please keep on sending the mails and I am I shall try my best to give you a give you or respond to your queries immediately so.

Thanks for the day bye.