İSTANBUL TECHNICAL UNIVERSITY*INSTITUTE FOR NUCLEAR ENERGY

INTEGRAL FAST REACTORS

M.S. THESIS

NECDET KÜLÇE, B.S.ME

DEPARTMENT : NUCLEAR SCIENCES

PROGRAM : NUCLEAR ENERGY

JANUARY 1995

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Supervisor : **Prof.Dr. Şarman GENCAY**

Members : **Prof.Dr.Melih GEÇKİNLİ**

Member : **Prof.Dr.Ahmet BAYÜLKEN**

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FOREWORD

The Integral Fast Reactor (IFR) concept tries to take advantage of the inherent properties of liquid-metal cooling metallic fuel in a way that leads to important improvements in the characteristics of the complete reactor system. The key technical properties and potential advantages of the IFR concept, its technology status, and its future research and development requirements are described and discussed in this work.

This study contains the introductory information and recent developments in IFR concept. I hope, this work would be very useful to the researchers of ITU Institute for Nuclear Energy.

Every chapter carries a few (hopefully) helpful comments meant to emphasis the significance of the material presented. I am certain that the literature contains many other references that some readers may fault me for not including. I am attracted to add to the list myself but the constraints of keeping the work to manageable size made this impossible, so I have no claim that my coverage of the subject is complete.

In preparing this thesis, I had a great deal of help from many sources, without whose assistance and cooperation this work would have been impossible. I should like to express my thanks and appreciation to all of them. Especially, I would like to thank Prof. Dr. Şarman GENCAY and lovely Ayşen SOMUNKIRAN for helping and supporting me with their best in every stage of work.

Necdet KÜLÇE

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ÖZET

Dünyada enerji arz talep dengesinin sağlanmasında oldukça fazla çaba harcanmış ve bu çabaların sonucunda geliştirilmiş nükleer reaktör tasarımlarının önemi ortaya çıkmıştır.

Dünya enerji talebinin gelecekte günümüzden daha aşağı bir seviyede olmayacağı bilinmektedir. Bu durumun da ciddi çevre sorunları veya nükleer enerji olmadan çözülemeyeceği ortadadır.

1979'da Three Mile Island (TMI) kazasından sonra durgunlaşan nükleer enerji endüstrisi 1986'da Chernobyl kazasından sonra daha büyük bir bunalım içine girdi. Ancak bu bunalımın tek nedeni kazalar değil aynı zamanda alışılagelmiş reaktör tasarımlarının getirdiği bazı sorunlardır. Özellikle de son yıllarda etkisini gösteren ve fosil yakıtların kullanımı ile oluşan çevre kirlenmesi, asit yağmurları, ve sera etkisi nedeniyle nükleer enerjinin yeniden gündeme gelme fırsatı ortaya çıkmıştır. O halde geliştirilmiş nükleer reaktör tasarımları üzerinde çalışmanın tam zamanıdır.

Fakat ileri nükleer reaktör tasarımları için teknolojik yetersizlikler yenilmelidir. Böylece, konuyla ilgili çalışmalar bilimsel ve teknolojik ilerlemelere yol açacaktır. Ayrıca hafif ve ağır sulu güç reaktörleri üzerinde yapılan denemeler de ileri reaktör tasarımı konusuna ışık tutacaktır.

Bütün bu olaylar TMI kazasından sonra oluşan yeni ve daha güvenli reaktör tipleri arayışının önemini göstermektedir. Bu yeni tip reaktörlerin en önemli özellikleri pasif güvenlik sistemleri ile insandan kaynaklanan hataların giderilebilmesidir. Ayrıca düşük güç yoğunluğu ve düşük güçte modüler sistemlerin tasarımı yeni güvenlik anlayışının ürünleridir. Elektrik üretimi dışında bölgesel ısıtma sistemleri ve proses ısısı üretimi içinde gerçekleştirilen reaktör tasarımları vardır.

Nükleer teknolojiye sahip ülkelerde oluşan ileri reaktör tasarımı arayışı pek çok yeni tasarımların ortaya çıkmasına yol açmıştır. ABD nükleer teknolojisini hafif sulu sistemlere bağladığı için yeni tasarımları da eski tip reaktörlerde bazı değişiklikler yaparak gerçekleştirmiştir. Bunlara örnek Westinghouse tasarımı basınçlı su reaktörü AP-600 ve General Electrics tasarımı kaynar su reaktörü SBWR verilebilir. Bunlar dışında General Atomics tasarımı modüler tipte gaz soğutmalı MHTGR dikkati çeken bir tasarımdır. Ayrıca sıvı metal soğutmalı hızlı reaktör tasarımları da bulunmaktadır. Japonya' da da benzer çalışmalar vardır. Avrupa'da ise su havuzuna yerleştirilen kora sahip hafif su soğutmalı PIUS olağandışı yapısal güvenlikli bir tasarımdır. Japon tasarımı ISER de PIUS'a benzer bir öneridir. Alman HTR firması ise yüksek sıcaklıklı gaz soğutmalı bir sistem geliştirilmiştir. Bütün bu tasarımlarda yeni güvenlik anlayışı etkin olarak kullanılmaktadır. Tasarım olarak hafif su soğutmalı ileri reaktörler eldeki

teknolojiyi değişik biçimlerde kullanmayı önerirken yüksek sıcaklık gaz soğutmalı reaktörler ile sıvı metal soğutmalı hızlı reaktörler yeni teknoloji sunmaktadır.

Genel olarak ileri reaktör tasarımları incelendiğinde görülen önemli değişiklikler vardır. Bu yeni kuşak reaktörler pasif güvenlik sistemleri ile donatılmışlardır. Bunu gerçekleştirmek için doğal taşınım sağlanmaktadır. Böylelikle korun soğutulması basit fiziksel ilkeler ile reaktörde dolaşan soğutucu ile gerçekleştirilmektedir. Operatör müdehalesi, elektrikli ve mekanik sistemlere gerek kalmadan reaktör çalışmasını sürdürebilmektedir. Bu da pompa ve benzeri sistemleri gereksiz kılmaktadır. Sonuç olarak reaktör kendinden güvenli olarak çalışmaktadır. Doğal taşınımla soğutma, reaktör çalışırken olduğu kadar kapanma sonrası bozunma ısısını giderecek etkinliğe sahip olmaktadır. Reaktördeki toplam su miktarı arttığından herhangi bir kaza durumunda müdehale süresi artmaktadır. Bu tip reaktörlerin pasif koruma ve güvenlik sistemleri özetlenecek olursa:

- Yüksek ısıl kapasite : Nükleer reaktörlerin yüksek güç yoğunluğu ve düşük ısıl kapasiteleri bu reaktörlerdeki normal çalışma koşullarından bir sapma durumunda sıcaklık dolayısıyla basınçta büyük değişimlere neden olur. Reaktörün birincil sisteminin ısıl kapasitesinin arttırılması ile bu değişimlerin daha yavaş gerçekleşmesi sağlanabilir. Yani, hem reaktör operatörlerinin hem de reaktör kontrol sistemlerinin müdehalesi için gereken zaman daha da uzatılmış olur. Ayrıca birincil devredeki yüksek ısıl kapasite birincil devreyi nükleer güç santralinin diğer kısımlarındaki değişimlerinden korur; böylece reaktörün güvenliğine olumsuz yönde etki yapabilecek sistemlerin sayısı azalmış olur.

- Doğal taşınım: Reaktörler kapandıktan sonra güçleri ile orantılı olarak bir bozunma ısısına sahiptirler. Bu ısıyı almak için bazı aktif sistemler kullanılır. Reaktör gücü ne kadar düşük ve boyutu ne kadar küçük olursa doğal taşınım ile soğutulması o kadar kolay olur.

- Güçlü negatif reaktivite katsayısı: Reaktörler için tanımlanan reaktivite katsayısı, sıcaklık değişimlerinde reaktivitenin nasıl değiştiğini belirten bir sayıdır. Reaktivite katsayısının güçlü negatif olması, sıcaklık arttığında reaktivitenin hızla düştüğünü gösterir. Böyle bir durumda reaktör kendini güvenli bir şekilde kapatır.

- Düşük güç yoğunluğu: Reaktörün güç yoğunluğu, birim hacimden alınan güç demektir. Bir reaktörde bu yoğunluk ne kadar düşük olursa, olası bir kaza durumunda reaktörün yaratabileceği zarar o ölçüde azalır. Buna ek olarak, böyle bir kaza durumunda aktif sistemler devre dışı kalsa bile, reaktörün soğutulması daha kolay olur.

Herhangi bir kaza durumunda radyoaktif maddelerin çevreye yayılması olasılığını daha aza indirmek için önlemler alınmaktadır. Reaktör koruma kabı kor erimesi kazası sonucu basıncın artması durumunda da bozulmadan kalabilecek şekilde tasarımlanmakta ve böylece radyoaktif malzemelerin dışarıya sızmaları engellenmektedir.

Yeni tasarımlarda kordaki güç yoğunluğu düşürülmektedir. Bu da reaktör kontrolunu kolaylaştırmakta ve reaktörün güvenlik sınırlarının genişlemesine yol açmaktadır. Reaktör kontrolü de bilgisayarlar yardımı ile yapılıp insan etkeni en alt düzeye indirilmektedir.

Ancak böyle amaçlanana ulaşılabilir; **"Nükleer güç santralleri sadece güvenli değil, çok çok fazla güvenli olmalıdır."**

Günümüz teknolojisiyle üretilen nükleer santraller bu amaca ulaşamayacağına göre, yeni tasarımlar üzerinde çalışma zorunluluğu yeniden ortaya çıkmaktadır.

İleri reaktör tasarımları genellikle düşük güçlü reaktörler için yapılmıştır. Dünyanın çeşitli ülkelerinde yaygın olarak kullanılan 3000 MWt güçteki reaktörler yerine daha düşük güçte (500-600 MWe düzeylerinde) orta boy nükleer reaktör sistemleri önerilmektedir. Bunun nedeni işletim ve bakım kolaylığıdır. Ayrıca daha kısa sürede bitirilip devreye sokulabilmektedir. Herhangi bir nedenle devre dışı kaldığında etkisinin daha az olması da olumlu bir yön olarak gösterilebilir . Düşük güçteki reaktörlerin en büyük olumsuz yönü birim güç başına ilk yatırım maliyetlerinin fazla olmasıdır.

Bütün bu tasarım değişiklikleri hem yapım sürecini kısaltacak ham de lisanslama sırasında çıkacak sorunları azaltacaktır. Böylelikle bazı ülkelerde toplam yapım sürecinin 4-5 yıl düzeylerine inmesi beklenmektedir.

İleri reaktör tasarımları toplam enerji üretim maliyetini düşürmek için daha etken yakıt çevrimleri önermektedirler. Normal olarak termal reaktörlerde üretilen plutonyumun en etkin olarak kullanımı hızlı üretken reaktörlerde olacaktır.

Bu reaktörlerin hızlı üretken reaktör olarak tanımlanmasının nedeni nötron enerji spektrumunun çok gruplu olması yüzündendir. Eşik enerjilerinin üstündeki değerlerde üretken nüklidler fizyona uğrar böylece hızları yavaşlamadan önce fizyon yapmış olurlar.

Korda üretken malzemelerin bulunması çoğaltma katsayısını azaltmasına rağmen fizyon yapabilen malzemelerin oluşmasına yol açar. Böylece nötron yakalanması ve radyoaktif bozunumla üretken nüklitten fizyon yapabilen nüklit oluşur. Bu da yakılan Pu-239 dan daha fazla Pu-239 üretilmesi demek olur, diğer bir deyişle reaktör "üretken" sıfatını kazanır.

İleri sıvı metal-soğutmalı hızlı üretken reaktörler yapısal güvenli olarak adlandırılır. Modüler yapıya sahip reaktörlerin bir kaçı bir araya gelerek bir güç bloğu oluşturuyor. Bu blokların her birinin ayrı bir türbine bağlı olarak çalışması isteniyor. Böylece, her bir blok birbirinden bağımsız olarak çalışabilecek hale geliyor. Blok içerisinde her bir reaktör diğerlerinden ayrı olarak işletilebilecek ve kapatılabilecek şekilde yer alıyor.

Hızlı üretken reaktörlerde soğutucu olarak sodyum kullanılıyor. Soğutucunun ısı aktarım özelliklerinin iyi olması pompalar devre dışı kalsa bile doğal taşınım ile reaktör korunun yeterince soğutulmasını sağlıyor. Sodyumun normal şartlarda tek fazda çalışması diğer önemli bir özelliğidir.

Bu reaktörlerde metalik yakıtlar kullanılmaktadır. Bu yakıtlar sayesinde kor içerisindeki maksimum sıcaklık düşürülüyor. Reaktörün herhangi bir kaza anındaki davranışı oksit yakıt elemanı kullanımına göre daha güvenli olmaktadır. Metalik yakıt elemanları ve paslanmaz çelik zarf malzemelerinin kullanımı sonucunda oldukça yüksek yanma oranları elde etmek olanağı olmuştur. Yakıtın basitliği sebebiyle, reaktör yakınında kurulabilecek bir "yeniden işleme" tesisinin maliyeti diğerlerine oranla daha düşük olacağı tahmin edilmektedir.

Yeniden işleme tesisinde kullanılan işlem pyrometallurjik metottur. Bu metotta fizyon ürünü ayrılması diğer sulu yöntemlere göre 10 kat fazladır. Yakıt yeniden işleme tesisi kapalı döngü şeklinde çalışıp yeniden işleme kayıpları da reaktörün üretken olma özelliğinden dolayı dengelenmektedir.

Metalik yakıt elemanlı havuz tipli sodyum soğutmalı bir deneysel reaktör olan EBR-II' de bu tip reaktörlerin yapısal güvenliği deneysel olarak kanıtlanmıştır. Bu reaktörlerde Nisan 1986'da yapılan soğutma kaybı (Loss of Heat Sink) ve akış kaybı (Loss of Flow) kazalarından reaktörün hiç bir kontrol sistemine gerek duymadan güvenli bir duruma geldiği görülmüştür. Normal olarak en şiddetli reaktör kazalarından sayılan bu iki kazanın bu tip bir reaktörde reaktörün güvenliğini tehdit etmemesinin deneysel olarak kanıtlanması oldukça önemli bir gelişmedir.

Üstünde çalışılması gereken ve bilinen reaktör kazaları senaryoları da özetlenecek olursa;

- ATWS, anticipated transient without scram, scram olmadan istenmeyen geçiş.
- LOFWS, loss of flow without scram, scram olmadan akış kaybı.
- LOHSWS, loss of heat sink without scram, scram olmadan akış kaybı.
- TOWS, rod run out transient overpower without scram, scram olmadan fazla yüklenme.
- Fazla soğutma kazaları: hızlı pompa ve giriş sıcaklığı düşüklüğü.

Deneysel üretken reaktörde (EBR-II) yapılan bazı test sonuçlarında:

- Doğal taşınım, herhangi bir sebeple oluşabilecek bir kaza durumunda doğal taşınım ile güvenli bir şekilde soğutmak mümküm olacağı.

- Soğutucu kaybı kazası, soğutucu pompaların durmasıyla oluşabilecek böyle bir kaza durumunda kor içindeki sıcaklıklar belirlenen limitlerin altında kalacağı.

- İkincil devrenin işletim dışı kalması,

A) Sodyum çıkış sıcaklıkları çok çabuk normal işletim sıcaklıklarının altına inebileceği.

B) Geçiş sıcaklıklarının ihmal edilebilir düzeyde olduğu.

C) Önceden hesaplanan sıcaklıklar ile ölçülen sıcaklıklar arasında tam bir uyum olduğu görülmüştür.

Hızlı üretken reaktörler ile yakıt sorunu daha da azaltılabileceği belirtilmektedir.Bir örnek te kullanılmış yakıtların yeniden işlenerek aktinitlerin elde edilmesidir. Bu da uzun ömürlü transuranik izotopların transmutasyon ve fizyon yolu ile kısa ömürlü ürünlere dönüştürülmesidir. Transuranik izotoplarının transmutasyon sebebi, bunların kullanılmış yakıtlardan elde edilip sıvı metal reaktörü yakıtı olarak yeniden işlenmesidir.

Sulu yöntem ve pyroproses prosesleri yeniden işleme ana metodlarıdır. Pyroprosesin önemli bir avantajı da Pu'nun diğer sulu yöntemlerde olduğu gibi transuranik izotoplardan ayrılmamasıdır. Bu da bomba yapılmasını önleme açısından önemli bir noktadır.

Ayrıca atıkların saklanması açısından çok büyük yatırımlar yapılmasını engellemektedir.

Günümüzde hızlı reaktör programı bu aktinid geri kazanım sisteminin en önemli araçlarından biri olmak üzere gelişimin tamamlamak üzeredir.

Hızlı reaktör sistemini ve bileşenlerinin güvenilirliği, işletme kolaylığı ile çok önemli olanaklar sağlamaktadır. Tesis genel kapasitesi de bakım gereklerinin azaltılmasıyla arttırılmıştır. Çünkü soğutucu olarak kullanılan sodyumun korrozif özelliği yoktur.

Hızlı reaktör teknolojisindeki gelişmeler akademik çalışmaların katılımı açısından oldukça önemli firsatlar sağlamaktadır. Her şeyden önce, hızlı reaktör teknolojisi ileri reaktör teknolojisidir. Bu yüzden de hızlı reaktör teknolojisi için çalışmanın tam zamanıdır. Hızlı reaktör teknolojisi yeni ve gelişmekte olan teknikleri gerektirmektedir; metallik yakıtlar, elektrorefining' e dayalı yeniden işleme teknikleri, yeni atık çeşitleridir. Bu yüzden de hızlı reaktör teknolojisinin önemli noktalarından biri de akademik çalışmalar için çekici bir konu olmasıdır.

SUMMARY

The world energy demand in the future will not be under the present level, because much longer population with energy consumption less than world average. It seems impossible to solve the lack of energy problem without nuclear energy and an environmental side effect.

In the 21st century, an opportunity will be for nuclear power to solve the world electric energy source problem. The rate of improvement in power reactor technology is rising which indicates that rapid technology advances will continue to be made in existing reactors designed for near future. Most of the reactors will be based on the large amount of performance experience from existing reactors. The researchers declare that new technologies should be adopted to new plants. In addition, this is the way of improving the safety of the future nuclear power plants. Because, the current nuclear technologies are not able to meet this challenge. This is the result of shortcomings in the technology that the next generation nuclear power technology must overcome.

The presence of fertile materials in the core causes to create new fissionable materials. So, new fuel is created converting the fertile nuclei to fissile nuclei through neutron capture and radioactive decay. This means that it is possible to produce more Pu-239 than is burned, which is called to "breed" new fuel. The idea behind breeder idea is this fact. Metallic fuel is the key to the IFR concept. Metallic fuel has an excellent steady state performance and high burn-up potential characteristics. Metallic fuel has a much higher breeding ratio than oxide or carbide fuels. The easy fabrication of metallic fuels combined with simple, compact pyrometallurgical reprocessing promises a major cost reduction in the LMR reactor fuel cycle.

The IFR idea has a number of specific technical advantages satisfying the design goals of advanced LMRs. These areas are fuel performance, inherent passive safety, economics, waste management, operability and reliability, as a system for actinide recycles program. In the actinide recycle system, the longer lived transuranic (TRU) isotopes are converted to shorter lived products by means of transmutation and fission. The basic system of this process is pyroprocess. Recently, the IFR research program is getting nearer to completion and is in the early stages of developing LWR actinide recovery for recycle in the ALMR.

The core reactivity shutdown and decay heat removal without relying on devices requiring switches and outside sources of power are the main purpose of the inherent safety feature.

The IFR technology development effort provides opportunities for active participation by the academic studies. First, the IFR is a next generation advanced reactor concept. Now is the right time to pursue fundamental research and development to advance its technology base. The IFR incorporates many new technologies that are different from conventional nuclear technologies; metallic fuel, reprocessing based on electrorefining, new waste forms etc.,. New emerging technologies and further innovations in the reactor design and the fuel cycle closure can be used.

1. INTRODUCTION

Although there are lots of developments in worldwide energy supply and demand situation in recent years, it has been proven that research efforts towards the next generation nuclear development are definitely required and should be encouraged.

The world energy demand in the future will not be under the present level, because much larger population with energy consumption less than the world average, and it is difficult to provide such a demand without nuclear energy or a serious environmental problem.

It is well known that the safety is particularly most important in nuclear energy field. There is scientific indication with the great number of reactor years, the accumulated operation records of at least 4,453 power-reactor years at the end of 1987, that the peaceful uses of nuclear energy including the Chernobyl accident have a much higher safety level than the safety levels of other systems such as traffic and housing systems which have been accepted for a long time by the public.

In the medium- and long-range view, there will again be a high possibility of a supply and demand situation for oil. [1]

The 20th century has been a period of technological revolution; aviation, electronics, automobiles, computers, and space flight are but a few of the technologies that have emerged and blossomed into integral components of everyday life. Nuclear power emerged midway through the century, with great promise for service to society. But unlike the other technologies, nuclear power has encountered serious difficulties, at least in the United States.

1- No new orders for nuclear power plants have been placed since 1977.
2- Many orders have been cancelled since the Three Mile Island Unit 2 (TMI-2) accident in March 1979.
3- Plants under construction have experienced major delays and escalation of construction cost.
4- New plants are experiencing difficulty in getting operating licence.
5- The Chernobyl accident erased hopes for a near-term rebirth of new plant construction and caused serious concern over nuclear power plant safety worldwide.
6- Utility leaders appear to have more or less given up on further expansion of nuclear power capacity.

Yet, it is clear that nuclear power will continue to be a major energy source both in the USA and worldwide. There are now 423 operating reactors worldwide, with 100 in the USA. Sixteen percent of USA electric power is generated by nuclear energy, and USA light water reactor (LWR) technology leads the world. Indeed, USA nuclear companies and maintenance of nuclear power plants are doing well. A shortage of electric power in the northeast and southeast USA is expected by the end of the century. Nuclear power could play a major role in meeting this demand. In any case, the use of nuclear power will continue to expand in countries such as France, Japan, Canada Germany, Korea, Czechoslovakia and previous UDSSR.

With increasing emphasis on nuclear power in many foreign countries and decreasing emphasis in the United States, technical leadership in nuclear power is shifting away from the United States:

1- The pioneering scientists and engineers are slowly retiring out of the industry.
2- Competent engineers are moving out of the field.
3- The influx of young engineers is inadequate.
4- Graduate programs in nuclear engineering have been in a state of general decline.
5- There no longer exists an atmosphere of technical excitement.
6- "Tiredness" in the field in the United States is obvious.
7- Japan is exerting its energy to take on a new leadership role in the technology: the advanced pressurized water reactor (APWR) and advanced boiling water reactor (ABWR) programs are under way, Japan is undertaking international initiatives this year, and the Japan Atomic Industrial Forum's annual meeting is becoming a focal point for the new technology.
8- France is taking a bigger role in the international market with Korea Nuclear Projects 9 and 10 being completed, the China Guandong Projects initiated, and fuel services extended to include reprocessing and enrichment.

The Chernobyl accident resulted in the international community taking a serious look at the next generation of reactor technology. Kremlin previous leader Gorbachev has urged the development of next-generation nuclear power plant technology. Secretary - General Hans Blix has recommended that International Atomic Energy Agency undertake an initiative to begin an international cooperative effort to design a next-generation nuclear power plant. Japan is emphasizing research on inherently safe reactor systems. Sweden and Korea once considered construction of a process inherent ultimate safety (PIUS) plant in Korea. The U.S. Department of Energy appears to have increased its emphasis on the integral fast reactor (IFR) system.

The nuclear power environment, in the time interval which the darkness decreases and some plant installations are in planning stage, the long term expectation for nuclear power shows cause for optimism:

1- Nuclear power remains an important alternative energy source.
2- The acid rain and CO2 issues have a clear dampening affect on the future use of coal.
3- Oil is still a volatile commodity and an expensive source of energy for electricity.
4- There remain opportunities for improving technologies.
5- Closing the back end of the fuel cycle appears feasible, but is yet unattempted.
6- Some nations have successful nuclear power programs, in France, the cost of nuclear-generated electricity is two- thirds the cost of coal-generated electricity, Canada exports nuclear-generated electricity to United States, and nuclear energy provides Japan with an edge in economic competitiveness.

In the environment of the 21st century, an opportunity will once again exist for nuclear power to emerge as an important electric energy source. But the current nuclear technologies are unlikely to be enable to meet this challenge.

In contrast to many other emering technologies of this century, current nuclear power technology has failed to blossom successfully. This is the result of shortcomings in the technology that the next generation of technology must overcome.

In addition to the above situation, the nuclear energy is the key energy source to overcome the scarcity of energy in industrialized countries where less fossil energy resources exists.

The hard efforts to develop the nuclear technology in the next generation will give rise to a further advancement in science and technology in the future. Nuclear energy technology is fairly experienced one concerning the typical light and heavy water power reactors, but there is big area for advanced technological development in the nuclear energy field.[2]

2. IMPROVEMENTS ON THE CURRENT REACTOR TECHNOLOGY

The rate of improvement in power reactor technology is raising. This indicates that rapid technology advances will continue to be made in existing reactors. Those reactors are designed for the near future, and may be introduced in the long term. Most of the efforts will be based on the large amount of performance data gained from existing reactors.

The status and progress on advanced technology development and trends in water reactor designs are reviewed at meetings of international working groups. A particular highlight in the field of new design concepts are the further progress in the joint project for a 1,450-1,500 MWe PWR licensable in France and Germany.

In other meetings, the working groups on advanced water cooled reactors has been declaring their targets including the need for public acceptance as a precondition for the construction of new plants; promising new technologies should be adopted, after complete testing; and improved elements are a possible way to increase safety of future nuclear power plants.

The status and trends of fast reactors are reported that France has initiated a program on creative studies aimed at proposing new concepts for fast breeder reactors (FBRs). In Japan, construction work on a prototype reactor continued; and in the previous Soviet Union, four fast experimental and prototype reactors were in operation and considerable operating experience has been obtained. In the USA, emphasis was on the comprehensive development of technology for the integral fast reactor to be followed by a period of demonstration to verify economic feasibility. This was done, while the conceptual design of a fast reactor has been completed in China.

Design activities for the European fast reactor (EFR) continued. Major emphasis was put on possible action for cost reduction and further safety. In March 1991, the construction of the high temperature test reactor (HTTR) at the Oarai Research Establishment in Japan started. It was noted that an important milestone in gas-cooled reactor development had been reached. Technology development activities for modular high-temperature reactors continued in the USA, the previous USSR, and Germany. [3]

2.1. Shortcomings Of Current Nuclear Power Technology

Nuclear power plant incidents and accidents raised severe doubts, particularly among amateur people, regarding the ultimate safety of nuclear power. Amateur people see credible scientists and engineers on both sides of the safety issue, but those who have vested interests in nuclear power claim nuclear power to be safe and those with no vested interests tend to take the other side. Seeing this split, amateur people have significant fears regarding nuclear power safety. With little economic advantage for nuclear power, safety issues have become critical factors in plant licensing. On the other hand, airplane accidents do result in investigations and substantial debate, but no one seriously suggests closing down the nation's air transportation system. In Japan and France, where nuclear power is critical to national economy well-being, its risks are far more accepted.

In the USA, crucial to the failure of nuclear power technology was the process by which the present technology was chosen. This process was insensitive to the utility market and to customer's requirements. The LWR technology dominates nuclear power; but LWR technology was chosen for commercial use by the military following chosen for the nuclear submarine. Utilities did not participate in this choice. More importantly, this early closure of technical choice shut the door on much of the innovation process that could have followed. Rather than being responsive to market needs, subsequent R&D efforts were directed at solving the problems of adapting the technology to power generation, and economic factors were given lower priority. Yet, to the consumers, the cost per kilowatt-hour remained paramount.

After the imposed selection of LWR technology, the technology was transferred to major vendors - Westinghouse Electric Corporation, General Electric Company, Combustion Engineering , and Babcock & Wilcox Company. These nuclear steam supply system (NSSS) vendors initially engaged in "turnkey" projects and lost money. Westinghouse and General Electric soon learned to make the turnkey projects profitable but, rather than pursuing them further, turned over the overall plant engineering job to architect-engineer (A/E) firms. These firms did not have the expertise in high technology needed to deal with the nuclear components of the plant, and they continued to rely on the NSSS suppliers for the "nuclear island". This separation of the NSSS and the balance of plant led to an environment in which the total system was never adequately examined nor optimized as an integrated system. Frankly, the A/E firms, which are paid on an hourly basis for their work, had no incentive to optimize the system. After all, the systems would be paid for by consumers who were out of the loop altogether.

The A/E firms filled the gap left by the NSSS vendors for utilities, and plants have since been constructed and operated on a "component approach". To make this approach successful requires a strong technical leader as the project manager, preferably the utility. But there are hundreds of utilities in the U.S.A., few of which have more than one nuclear power plant, and more than a dozen A/E firms have overseen the construction of one or more nuclear power plants. With such a diffuse market for the technology and such disorganized leadership, the A/E firms have proven incapable of mastering the nuclear power plant system as a whole, and the utilities have failed to provide the needed technical leadership. They are unlikely to do better in the future, as they lack the culture for the technological innovation as well as the incentive to improve the overall product.

From the very beginning, it was recognized that nuclear power carried with it hazards that needed to be controlled. As concerns for plant safety increased, regulators indirectly suppressed technological innovation, largely out of conservatism. Utilities became very protective of their plants for fear that any acknowledgment of imperfection would result in their shutdown. The argument was that if a system has no imperfections it is perfect and cannot be improved. Technological innovation was therefore viewed as an admission that a plant is imperfect by some conservative utilities.

TABLE 2.1
NUCLEAR STEAM GENERATION CAPACITY ORDERED

Year	Nuclear Units Ordered	Nuclear Capacity MW(electric)	Nuclear Power as a percent of total
1951 to 61	14	1170	1
1962	2	628	7
1963	4	2495	15
1964	0	0	0
1965	7	4423	15
1966	20	16345	43
1967	30	24427	47
1968	14	12872	36
1969	7	7203	22
1970	14	14305	30
1971	20	19921	50
1972	36	39705	63
1973	38	43068	59

Through the late 1960s and early 1970s, the USA nuclear power plant market experienced rapid growth (see Table 2.1). almost any A/E firm could get a contract. Competition based on the quality of the A/E firm was minimal, and the industry lacked technical leadership. At the same time, as shown in Figure 2.1 and Table 2.2, the government provided strong incentives for plant size increases that further prevented experiential feedback and learning.

In contrast, the nuclear programs of France, Canada, and Japan have developed under the strong leadership of a single entity. In France, the single utility, Electricité, de France, took the leadership role and was able to standardize nuclear power plants. From these standardized power plants, they accumulated expertise and gained strong protection by the state, and long term planning has been feasible. The french technology was transferred from Westinghouse to Commisariat ... l' Energie Atomique/Framatome. In Canada, a single NSSS vendor, Atomic Energy of Canada Ltd. (AECL), took leadership. A strong utility, Ontaria Hydro provided a strong management team and was involved in the decisions leading to the selection of the Canada deuterium uranium reactor configuration. AECL was also strongly protected by its government. In Japan, the heavy industries - Mitsubishi, Toshiba, and Hitachi - provided leadership. They undertook turnkey projects for nine Japanese utilities. These giant firms could integrate the technology vertically and even provide for construction. They have also been able to take profits from their business and invest them to provide for technological innovation, making possible projects such as the APWR and ABWR.

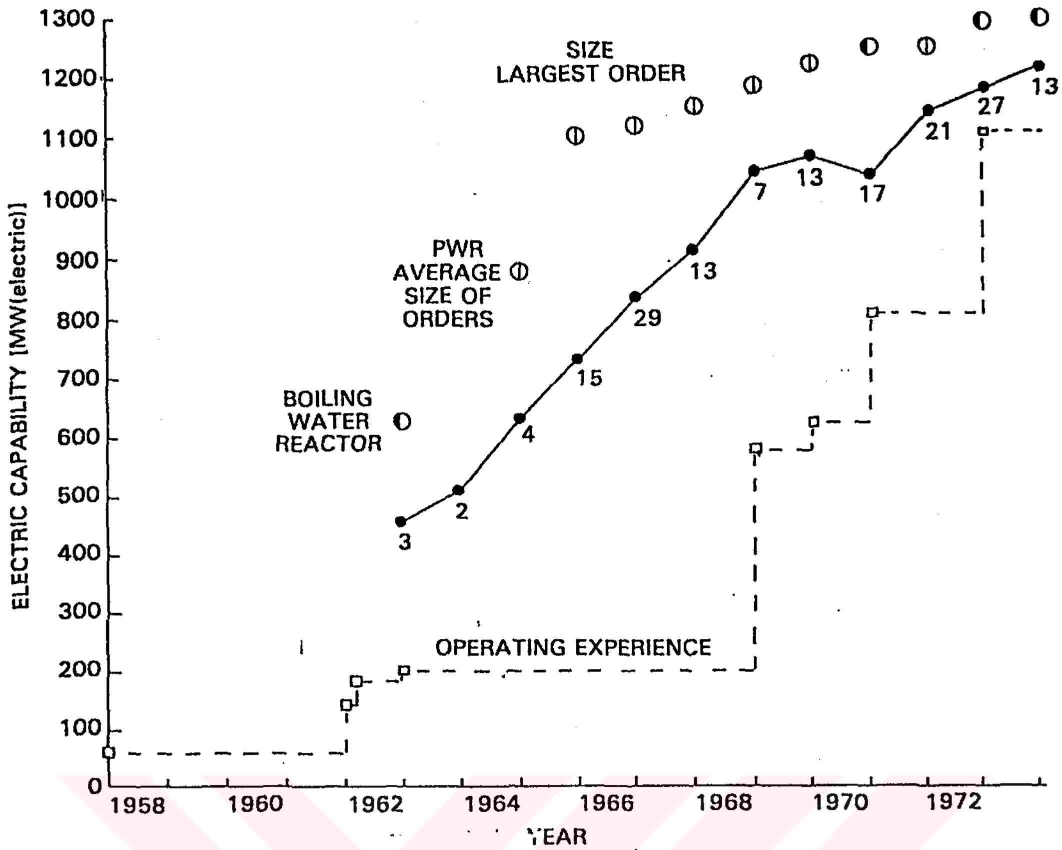

FIGURE 2.1 THE MEGAWATT RACE

TABLE 2.2

LARGEST CAPACITY OF LWRS ORDERED DURING 1962 TO 1977

Year	Largest BWR MW(electric)	Largest PWR MW(electric)	Leader by Percent Capacity
1962to 1963	620	575	BWR-7.3
1964	No orders	No Orders	———
1965	800	873	PWR-8.4
1966	1067	1090	PWR-2.1
1967	1067	1115	PWR-4.3
1968	1100	1148	PWR-4.2
1969	1067	1180	PWR-9.8
1970	1078	1213	PWR-11.1
1971	1250	1150	BWR-8.0
1972	1233	1250	PWR-1.4
1973	1288	1280	BWR-0.6
1974	1288	1285	BWR-0.2
1975	No Domestic Orders	No Domestic Orders	———
1976	No Domestic Orders	1270	PWR----
1977	No orders	No Orders	———

In the 1960s and 1970s, during the great expansion of nuclear power, computational tools - hardware and software - for support of nuclear power plant design, manufacture and construction, operation, and maintenance were not available. These tools could be available for the next generation of nuclear technology. Integration of the NSSS with the balance of plant and system optimization could be feasible. The entire plant description - design, specifications, bill of materials, construction schedule, operating procedures, and maintenance schedules and requirements - could be contained and updated in a single computer data base. But the long duration of nuclear power plant projects requires new approaches for feedback and optimization. French pressurized water reactors (PWRs) are basically Westinghouse PWRs; however, through standardization, engineering has improved performance, A/E work is much better, and maintenance is easier. Their power plants cost less and work better.

Before the TMI-2 accident, nuclear power plant operators were basically undereducated. Their training did not provide them with an understanding of the physical processes by which the plant operated, and the TMI-2 accident was largely the result of the poor quality of the operator's training. Efforts have been made to improve training, including the establishment of the Institute for Nuclear Power Operations. But these efforts are made more difficult by the lack of power plant standardization in the U.S.A. and the resulting difficulty and cost of providing adequate training simulators. There is also a need for nuclear power plants to be designed with more consideration for the ergonomic aspects of design.

The following problems seem to have beset the nuclear industry can be given as follows:

1. failure to produce a competitive technology
2. failure of continued technological innovation
3. failure to obtain strong leadership either in industry or government
4. failure to demonstrate adequate levels of safety
5. failure to initially provide for high-quality plant operation
6. failure to design the system to be responsive to its customers.

The next generation of nuclear technologies must correct these failures if nuclear power technology is to be blossom in the next century. In meeting this challenge, the industry should increase institutes on nuclear technology such as Electric Power research Institute.[2]

2.2. The Need For The Change For Nuclear Technology: Next Generation Of Nuclear Technology

It typically takes 10 years or longer to bring a large technical system such as a nuclear power plant from the laboratory to the marketplace. Thus, in selecting research topics for the next generation of nuclear technology, the time of interest is that of power plants that will begin construction after the year 2000. Even so, the conceptualization of these plant designs is already well under way. But for any next generation nuclear technology to be viable, it must meet certain basic requirements: safety, economics, utility system compatibility, and continuing technological change.

Nuclear power plants of the next generation must not only be safe, they must demonstrably safe. That is, reasonable scientists and engineers must agree that poses no significant threat to human health. In fact, an even more stringent requirement may be imposed, namely, that no credible accident can occur that poses an economic threat to utility.

These safety levels might not be achievable with LWR technology, and even if achievable, they would not be demonstrable. This consideration alone may force the choice of "inherently safe" reactor configurations for the next generation of the technology. These are reactor designs for which no credible accident scenario can be developed.

The next generation of nuclear technology must achieve a complete turnaround in the cost of nuclear power plants. Only if the generation cost can be brought down to a level that is 30 to 50% below that of competing alternatives will nuclear power see widespread implementation. Marginal competitiveness is not a sufficient attraction to utility planners who have suffered through delays, uncertainties, and cost escalations. Today's utility industry is run on a "penalty" system. Utility decision makers are not rewarded for incremental improvements. They are penalized for mistakes. Thus ultraconservatism prevails.[2]

Three second-generation designs have been developed to an advanced stage in France (SPX-2), in Germany (SNR-2), and the United Kingdom (CDFR). These designs have all adopted the pool arrangement of the primary circuit and will now evolve within the framework of the European cooperation in a sequence of demonstration reactors that will produce a common model that incorporates all of improvements generated by the earlier cooperation and is ready for full commercial extrapolation. From these design requirements, the operational experience gained from Phenix and PFR, and from the construction and first operation of SPX-1, some general guidelines have been established for the R&D program. These general objectives, having the potential for substantial cost reductions, are

1. longer station life: 40 years more
2. increased availability
3. increased output
4. high fuel burnup of the order of 20%
5. improved safety technology.

As an example of capital cost reduction, we can mention the studies carried out in France and in the Germany. They show a 20 to 30% weight loss in the nuclear steam supply system (NSSS) materials and the simplifications at the balance-of- plant level related to extended use of passive features in the NSSS. The excess weight of fast reactors is one of main causes of high capital costs; therefore, a reduction in weight results directly in cost reduction.

Relative to the fuel cycle, the need for a single demonstration reprocessing plant has been recognized in Europe to serve the demonstration reactor program as a joint project sharing design, R&D, and funding. In the same context, it is recognized that the existing fuel fabrication capacity is adequate to cover the needs of the proposed reactors.[4]

2.3 The Worldwide Experience And Efforts For The Next Generation Reactors

2.3.1 European Experience:

The FBRs (Fast Breeder Reactors) have been under development in Europe for >30 years and have progressed through several stages. Following theoretical and technological studies, some experimental reactors have been built in various European countries with the aim of acquiring technological experience on the construction and exploitation of a small but significant sodium-cooled fast reactor. We should mention the Dounreay fast reactor (DFR) in the United Kingdom, which first reached criticality in 1959, Rapsodie in France, and KompakteNatrium-gekühlte Kernreaktoranlage Unit 2 (KNK-2) in the Germany.

The next step was the construction of demonstration power stations with outputs of c.a 250 to 300 MW (electric) using components on a scale that required only a modest extrapolation to commercial size. We should consider that

1. in the United Kingdom, the prototype fast reactor (PFR), operating since 1974, has provided useful information on station behavior

2. in France, Phoenix, operating since 1973 with a good load factor, reached a 16.4-TW.h gross electrical production by the end of 1986.

3. in the FRG, construction of the Schneller Natriumgekühlter Reaktor (SNR)-300 was completed in 1985 and fuel loading is expected in the near future. [4]

Based on this important experience, the utilities of France, Italy, and FRG signed an agreement in 1973 for the construction of two commercial-sized FBRs. The first step of this agreement has been Superphenix (SPX-1) construction at Creys Malville by the NERSA company (51% Electricite de France, 33% Ente Nazionale per l'Energia, 16% Schnell-Bruter- Kernwerksgesellschaft. The reactor, without major difficulties, reached full power in December 1986 but then had a sodium leakage problem in the fuel storage vessel. At present, a full understanding of the event has been reached, restart authorization was given in January 1989, and the reactor is now operating at full power.

More generally, in the period between 1973 and 1983, the European situation has been characterized by significant progress in bilateral cooperation both on the industrial and R&D sides. In 1974, France and Italy signed some agreements between research organizations [Commissariat a l'Energie Atomique (CEA) and the Comitato Nazionale per l'Energia Nucleare a Alternativa (ENEA)], design companies

(Novatoma and Ansaldo-NIRA), and manufacturers. In 1977 an important R&D agreement was established between France (CEA) and the previous FRG [Interatom and Kernforschungszentrum (KfK) in association with Belgium (Belgonucleare) and the Netherlands (Neratom).

Some power station reference designs, adopting the common pool-type concept, with power in the 1300 to 1500-MW (electric) range, have also been developed during this period:

1. SNR-2 in the FRG, following the above mentioned 1973 utility agreement
2. SPX-2 in France, a reactor concept in which the experience gained during SPX-1 design and construction has been taken into account.
3. Commercial Demonstration Fast Reactor (CDFR), where the substantial experience gained in the United Kingdom is fully transferred to a comprehensive conceptual design.

2.3.1.1. European Cooperation:

The common European effort to be continued over a long period of time needs to be shared in order to minimize the economic and financial risks of the different partners.

The current phase of FBR collaboration was initiated by an intergovernmental memorandum of understanding dated January 10, 1984. The governments of the United Kingdom, France, the previous FRG, Italy and Belgium engaged in

1. harmonization of R&D programs
2. industrial cooperation
3. cooperation among electric utilities
4. fuel cycle cooperation.

Under this memorandum, the various national organizations will conclude agreements in separate agreements in separate areas. The relevant agreements are

1. Reactor memorandum of understanding signed on March 2, 1984, by the following R&D and design organizations:
   a. CEA
   b. Novatome, French FBR manufacturing company
   c. KfK
   d. Interatom, FBR manufacturing company in the FRG
   e. United Kingdom Atomic Energy Authority (UKAEA)
   f. National Nuclear Cooperation (NNC) FBR manufacturing company in the United Kingdom
   g. ENEA
   h. Nucleare Italiana Reattori Avanzati (NIRA) FBR manufacturing company Italy (now Ansaldo)
   i. Nuclear Research Center Mol, Belgium, SCK-CEN
   j. Belgonucleaire, industrial consortium for the development of nuclear energy in Belgium.

2. Utilities arrangements

3. fuel cycle plant design and R&D.

The United Kingdom and France signed general agreements in the last two areas in February and March 1984, and negotiations to include other countries are continuing.[4]

2.3.2 The Japanese Effort

The mainstream of R&D on fast breeder reactors in Japan has been carried out by Power Reactor and Nuclear Fuel Development Cooperation PNC and the utilities, and JAERI has made R&D on carbide and nitride fuels, some specific safety related items, related basic nuclear data, and a tentative conceptual design of a high breeding fast reactor with a new concept of fission-product-gas-purge/tube-in-shell type metallic fuel assemblies which may have a breeding ratio higher than 1.8.

Conceptual study on actinide burner reactor has also been made where two types of reactor cores, a metallic-pin- fuel/sodium-cooled and coated-particle nitride-fuel/helium- cooled core types, are assessed to be promising. [1]

2.3.3 Liquid Metal Fast Breeder Reactor (LMFBR) Demonstration Plants In USA [5]

The history of the civilian power water cooled reactor program has shown that in addition to program oriented test facilities, a number of demonstration plants are essential to validate the results of the overall R&D program, to provide an industrial base, and to obtain acceptance of the plant type by the utilities. Similarly, demonstration plants will be used for the LMFBR program.

The first demonstration utility plants included in the overall LMFBR plant will have a capacity somewhere in the 300-500 MW(e) range, will be an integral part of a utility power grid, and will prototype and demonstrate as many features as possible of a 1000 MW(e) or larger plant.

The first demonstration plants will have to prototype and validate technical understanding and engineering control of most of these features. Many other important aspects of the LMFBR program can only be confirmed in a demonstration plant in the utility environment. These include demonstration of overall fuel cycle, long-term reliability and safety of operation, and high plant factor and maintainability, along with related potential economics of a system designed for and operated by utilities. Such plants represent the focal point of immediate R&D program, and these will be the ultimate tests of the initial phase of the overall program.

In looking forward to the commitment for the first demonstration plants, the status of certain key program elements must be considered. These include:

a) Proof test of fuel, fuel clad, fuel structure, and fuel assembly to over 50,000 MWd/t and over 1023 nvt.
b) Successful development and demonstration of key components.
c) Resolution of key sodium technology problems.
d) Successful operation of EBR-II or Fermi at adequate plant factors.
e) Resolution of key physics problems requiring use of crossection accelerators, critical assemblies, and SEFOR.
f) Resolution of key safety problems requiring use of critical assemblies, TREAT, and SEFOR.
g) Development of the industrial capability.
h) Meaningful industrial and utility participation.

TABLE 2.3 "Fast Breeding Reactors (Operating in May'80 and in Construction)" [6]

Item No:	Year Of Criticality	Country	Reactor Name	Place	Power MWt	Power MWe
Low Power Experimental Reactors						
1	1958	UDSSR	SBR-10	Obninsk	10	
2	1963	USA	EBR-II	Idaho	20	
3	1967	FRANCE	RAPSODIE	Cadarache	40	
4	1969	UDSSR	BOR-60	Melekess	60	
5	1977	GERMANY	KNK-2	Karlsruhe	60	
6	1977	JAPAN	JOYO	Oarai	100	
7	1980	USA	FFTF	Hanford	400	
8	1980	ITALY	PEC	Brasimone	130	
9		INDIA	FBTR	Kalpakkam	40	
Not In Operation: EBR-I (USA) 1955, E.Fermi (USA) 1966, SEFOR (USA) 1972, Dounrey (UK) 1977						
Medium Power Demonstration Reactors						
10	1972	UDSSR	BN-350	Shevchenko		350
11	1972	UK	PFR	Dounrey		230
12	1973	FRANCE	Phoenix	Marcoule		250
13	1980	UDSSR	BN-600	Sverdlosk		600
14	1987	JAPAN	Monju			250
15		GERMANY	SNR-300	Kalkar		300
16		USA	C.River	Tennessee		350
Note: Construction of the last two reactors were stopped due to political and social reasons.						
High Power Commercial Reactors						
17	1983	FRANCE	S.Phoenix	Creys-Melville		1200

3. FAST REACTORS

3.1. Fast Reactor Physics, an Overwiev [7], [8], [9]

3.1.1 Neutron Spectrum and Breeder Reactors

As it is known, the energy of neutrons of interest in reactor physics ranges from about 0.01 eV to 10 MeV. In thermal reactors average energy is 0.025 ev and most probable energy is 0.01 ev. Hardest neutron spectrum in reactor physics is fission spectrum. The neutron spectrum of fast reactors is between thermal neutron and fission spectrum.

Reactors are classified as thermal or fast according to their neutron energy spectrum. To understand the neutronic meaning of energy spectrum, the cross sections of fissile and fertile materials should be examined. Analysis of neutron spectrum is out of the scope of this work. However, it should be noted that fission is possible in fertile nuclide if the neutron energy is above a threshold value Figure 3.1. gives fission cross sections of fissile and fertile materials, and also gives the neutrons produced per fission in fast and thermal fission.

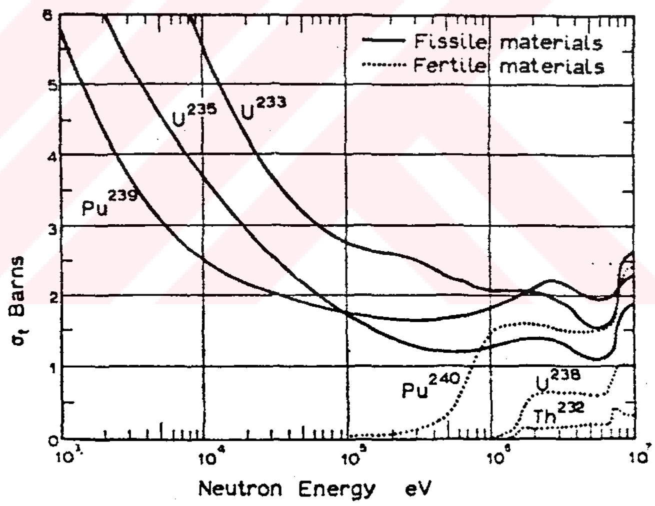

FIGURE 3.1.FISSION CROSS SECTIONS OF FISSILE AND FERTILE MATERIALS AT HIGH ENERGIES

In fast reactors light weight materials are eliminated and 10-20% enriched fuel is used. Thus, neutrons cause sufficient fission in the high energy range before they slowed down into the high capture resonance ranges. Those ranges exist for U-235 and U-238 about the energy interval of 1 ev - 1 kev.

The presence of fertile materials in the core has an effect of decreasing multiplication factor. However, they also cause to create new fissionable materials. Thus, new fuel is created converting the fertile nuclei to fissile nuclei through neutron capture and radioactive decay that is shown in the Figure 3.2.

$n + {}_{92}U^{238} \longrightarrow {}_{92}U^{239} + \gamma$

${}_{93}Np^{239} + \beta$

${}_{94}Pu^{239} + \beta$

$n + {}_{90}Th^{232} \longrightarrow {}_{90}Th^{233} + \gamma$

${}_{91}Pa^{233} + \beta$

${}_{92}U^{233} + \beta$

FIGURE 3.2.CONVERSION OF FERTILE NUCLEI TO FISSILE NUCLEI

η is an important quantity, if the production of fissile material is of concern. The examination of the Figure 3.3. shows that η is higher than 2 in some regions that implies the possibility of converting at least one fertile nucleus to a fissile one. In thermal region U-233 (converted from Th-232) and in fast region Pu-239 (converted from U-238) seems attractive to maintain breeding cycle. The latter one is the motivation behind the development of "Fast Breeder Reactors."

3.1.2. Breeding And Conversion Ratio

It might even be possible to produce more Pu-239 than is burned - that is to "breed" new fuel. This is the idea behind the breeder concept. [10]

The number of fissionable nuclei produced for each fissile Pu-239 nuclei destroyed in plutonium loaded fast reactor is called "Conversion Ratio" which its maximum value is

$$CRmax = \eta - 1$$

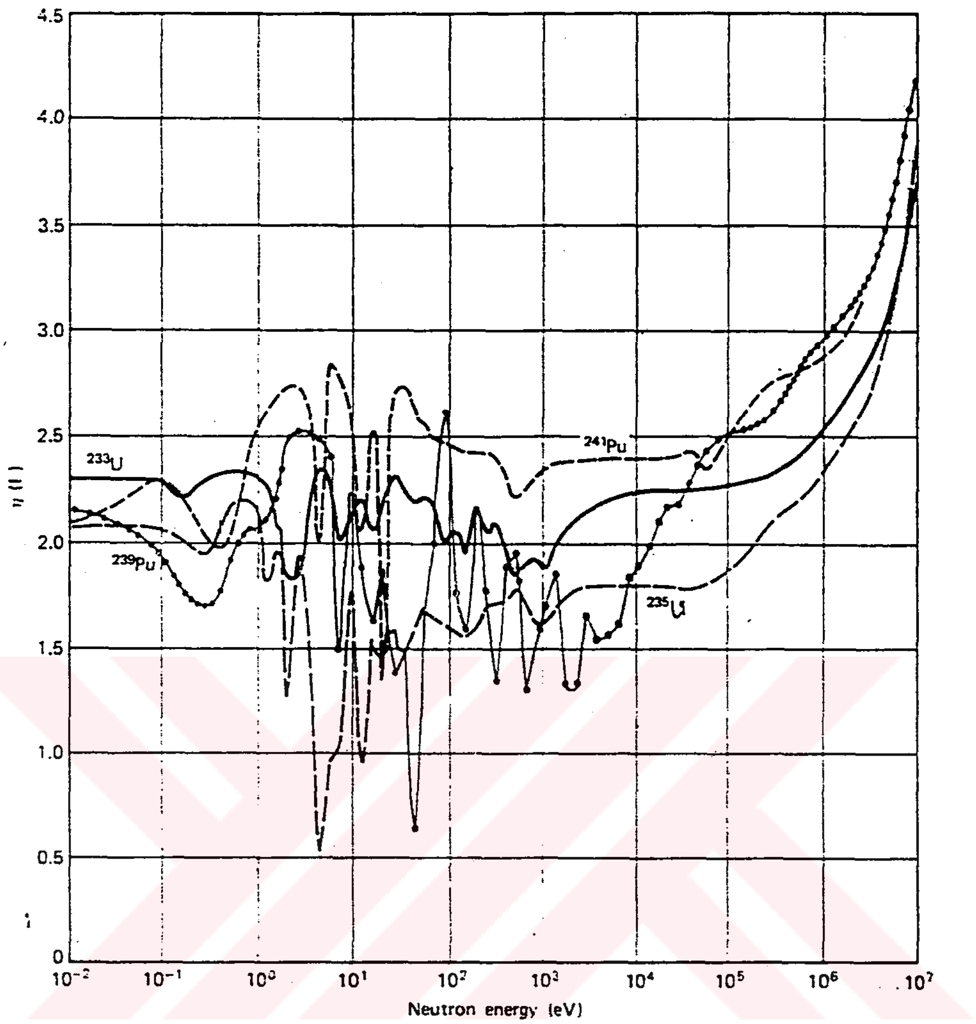

FIGURE 3.3. VARIATION OF η WITH ENERGY FOR U-233, U-235, PU-239, PU-241

η is the average number of fast neutron which are released per neutron absorbed in the fuel and one neutron is needed to maintain chain reaction.

Since, there is always some leakage from the core and some loss of neutrons through poison absorption,

$$CR = \eta - 1 - A$$

where A denotes the additional neutron loses. The "Breeding Gain," G, is defined as

$$G = CR - 1$$

G gives the number of extra neutrons to produce more than one fissile nuclei per fissile nuclei destroyed. Pu-239 has the highest η in the hardest spectrum. Table 3.1 shows the η values. As it is understood breeding is very good in Pu-239 fuelled fast reactors. Conversion ratio, C, for LWR and HTGR's is about 0.6 and 0.8 respectively but for fast breeder Pu- 239 loaded reactors C may have a value about 1.2 -1.5.

TABLE 3.1 [8]

NEUTRONS PRODUCED PER FISSION IN FAST AND THERMAL FISSION*

ISOTOPE	Fast Fission (LMFBR)	Thermal Fission (PWR)
U-233	2.64	2.28
U-235	2.46	2.05
Pu-239	3.03	1.96

3.1.3 Fast Reactor Core

Fast reactors are fuelled with Pu and/or U-235 and reflector material is fertile U-238 which is also called "Blanket". The requirements of high neutron flux levels and the need to minimize the amount of moderating material in such fast reactor systems result in cores consisting of tightly packed hexagonal lattices. The fuel elements are uranium-plutonium dioxide pellets that are enriched to 15 to 20% in stainless steel cladding of about 6.35 mm. in diameter. Around the core, there is a blanket of depleted uranium oxide pins, that serve as reflectors. They also deflect neutrons back into the core, but they are constructed of depleted uranium or other fertile material, in order to increase the production of fissile material and to avoid degrading the neutron energy spectrum. In other words, that serve to enhance the breeding properties.

Both sodium and helium have been proposed as fast reactor coolants. In both cases, the core designs have power densities that are much higher than those in thermal reactors, and therefore the cores are much smaller, as indicated in Table 3.2. Fast reactors are refuelled off-line at relatively infrequent intervals. They do not, however, require large excess reactivities to operate for sustained periods. There are two reasons for this.

The capture cross sections of the fission products that build up in the core very small for the fast neutron spectrum. And the depletion of fissile material is partially or completely countered by the build-up of plutonium-239. As a consequence and because of the fact that fast reactors are not neutronically loosely coupled, relatively few control rods suffice to control the reactivity.

TABLE 3.2. PROPERTIES OF 3000 MW(T) POWER REACTORS (1973)

Type	Description	Core Average Power Density (W/m^3)	Core Volume V(m^3)	Migration Length M (cm)	Diameter in Migration Length D/M	Nonescape Probablity
HTGR	Graphite Mod. He cooled	7.0	12.0	428.6	62.8	0.9917
BWR	Boiling Water	50.0	60.0	178.0	0.9990	
PWR	Pressurized Water	75.0	1.8	40.0	190.0	0.9991
GCFR	Gas Cooled Fast Reactor	280.0	6.6	10.7	33.4	0.9713
LMFBR	Na -Cooled Fast Breeder	530.0	5.0	5.7	35.6	0.9749

In fast reactor core calculation, the accuracy of a single group equation is not adequate. Therefore multigroup diffusion equation has to be used. There are some codes that were prepared for the fast reactor core calculations. In a thermal reactor the strong elastic scattering is present in the core. However, in a fast reactor, fission is induced by the neutrons whose energies in the ranges in which inelastic scattering and resonance absorption are most important. The importance of thermal neutrons and thermal fission in thermal reactors lower the importance of these details that should be considered in the calculation of the fast cores.

As it was noted that in thermal reactors the multigroup constants are sensitive functions of core composition and core operating conditions. Because of this fact they should be recalculated many times in core design. In fast reactors the expense of a direct calculation of the multigroup constants coupled with a somewhat lower sensitivity to core environment has motivated the development of sets of universal microscopic multigroup constants which can be used for any fast reactor core design. [10]

3.1.4 Reactivity Feedback Effects in Fast Reactors

Fast reactors differ from thermal reactors in several aspects as discussed previously. The enrichment of the fuel is higher than that in a thermal system, with typical enrichments ranging from 10 to 20 %. There is also an absence of low atomic weight moderator material, resulting in a fast neutron spectrum shown in Figure 3.4. As a rule, crossections tend to decrease with increasing neutron energy, causing the neutron spectrum averaged crossections in a fast reactor to be smaller than those in a thermal system.

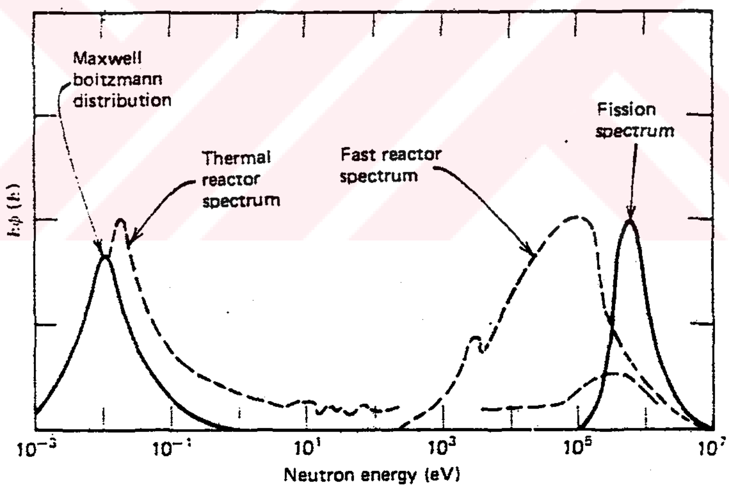

FIGURE 3.4. TYPICAL NEUTRON FLUX SPECTRA

3.1.4.1 Fuel Temperature Coefficient

Both the Doppler effect and thermal expansion contribute to the fuel temperature coefficient. As fast reactor designs have evolved toward enrichment oxide fuels and softer neutron spectra, however, the Doppler coefficient has come to be the dominant contribution. The Doppler effect has practically no effect on leakage properties, but affects the fuel temperature coefficient primarily by causing changes in k_{∞}.

In determining the temperature dependence of k_{∞} resulting from the Doppler effect, several simplifications may be made. Because of the threshold in the fertile material above which fission takes place is well above the energy range where resonance absent.

The Doppler effect in fast systems also would be expected to be smaller than that in the thermal reactor because the enrichment is larger and therefore the atom density of fertile material in the fuel is smaller.

A more important phenomenon in fast reactors is that harder neutron spectrum causes a smaller fraction of the neutrons to be in the energy range where the resonance crossections occur.

3.1.4.2 Coolant Reactivity Coefficient

Transients that overheat fast reactor coolants lead to decreased coolant density. This density decrease is through one of two mechanisms. First, the coolant remains as a single phase and the temperature rises at approximately constant pressure. The second mechanism is through the boiling of liquid coolant such as sodium.

As noted in the treatment of Doppler coefficients, a decrease in coolant density tends to shift the neutron spectrum upward in energy because of the decreased density of lighter-weight atoms that slow down neutrons through elastic collisions. [7]

3.2. FUELS IN FAST REACTORS [11]

3.2.1. Differences Of LMFBR Fuels From LWR Fuels

Table 3.3 lists the principal differences between irradiated fuel to be reprocessed from an LMFBR and LWR.

Because some of the sodium coolant used in the LMFBR fuel that may have adhered to the cladding or penetrated leaks in it would react vigorously with water or nitric acid, it is necessary to oxidize all sodium by exposing the fuel to an inert gas containing a controlled amount of water vapor before the dissolution step. LMFBR fuel may not be stored with water cooling till after all sodium has been removed.

Use of stainless steel cladding in the LMFBR instead of zircaloy has little effect on mechanical decladding or on dissolution. Stainless steel, like zircaloy, is not rapidly dissolved by nitric acid of the concentrations used in the Purex process.

Fuel in the core of the LMFBR is operated at a specific power over three times that of the LWR. During the cooling period, the specific power of LMFBR core fuel from radioactive decay remains about three times that of LWR fuel cooled for the same length of time. This makes shipping, handling, and storage of LMFBR fuel prior to reprocessing much more difficult than LWR fuel.

To reduce the specific power somewhat in reprocessing, it is planned to combine irradiated fuel from the LMFBR core with irradiated fuel from the LMFBR blankets in proportion to the rates at which they are discharged from the reactor. Even so, the specific power of LMFBR fuel cooled 150 days is 1.4 times that of LWR fuel cooled the same length of time.

The burnup of fuel in the LMFBR core is two more times that of LWR fuel, leading to higher concentrations of fission products, gaseous and solid, and greater radiation effects on cladding and fuel. The average burnup of combined LMFBR cores and blanket material is about 15 percent higher than that of LWR fuel.

The concentration of plutonium in combined core and blanket fuel from the LMFBR is more than 10 times that of LWR fuel. This is most significant difference between the two fuels with respect to reprocessing. Other important differences are the greater amounts of tritium and I-131, the 140 percent greater ruthenium activity, and the 60 percent greater overall specific activity of 150-day cooled LMFBR fuel.

Because of the high plutonium content of spent fuel from the LMFBR, there is strong economic incentive to return this plutonium to reactor with minimum delay for cooling, reprocessing, and refabrication. Consequently, the foregoing comparison of LMFBR and LWR reprocessing conditions for equal cooling periods of 150 days does not tell the whole story. For example, if LMFBR fuel were reprocessed 90 days after discharge from the reactor instead of 150 days, the activity of 8.05-day which is half life of I-131 would be 175 times greater, and the specific power of fuel from the core would be 1.5 times greater. As a result, this discussion of reprocessing LMFBR fuel is thus less complete and less up to date than would be desired, due to the fact that international dissemination of reprocessing information has been restricted since 1973.

3.2.2. Purex Process[11]

The principle of the Purex process, now commonly used for processing irradiated uranium by solvent extraction , is illustrated in Figure 3.5. It depends on the principle of the difusion which provides the transfer of a solid from a aqueous phase to an another aqueous phase. The partial process is realized by means of columns working with the principle of counterflow. The solvent used in this process is a solution of tributyl phosphate (TBP) in a high-boiling hydrocarbon, frequently n-dodecane or a mixture of similar hydrocarbons. In Purex process, TBP and hydrocarbons are stable compounds. TBP forms complexes with uranyl nitrate [$UO_2(NO_3)_2$] and tetravalent plutonium nitrate [$Pu(NO_3)_4$] very slowly, whose concentration in the hydrocarbon phase is higher than in an aqueous solution of nitric acid in equilibrium with the hydrocarbon phase. On the other hand, TBP complexes of most fission products and trivalent plutonium nitrate have lower concentrations in the hydrocarbon phase than in the aqueous phase in equilibrium.

TABLE 3.3 PRINCIPAL DIFFERENCES BETWEEN IRRADIATED FUEL FROM LMFBR AND LWR

REACTOR		
	LMFBR	LWR
Coolant	Sodium	Water
Cladding Material	Stainless Steel	Zircaloy
Fuel Rod Material, cm	0.6-0.8	1.0-1.2
Reactor Specific Power, MW/Mg HM*		
Core	98	
Average, Core And Blankets	49.3	30
Burnup, MWd/MT		
Core	67600	
Axial Blanket	4700	
Radial Blanket	8000	
Mixed Core And Blankets	37000	33000
Composition Of Mixed Core And Blanket Cooled 150 days w/o		
Uranium	85.6	95.4
Neptunium	0.025	0.075
Plutonium	10.3	0.90
Americium	0.035	0.014
Curium	0.0011	0.0047
Fission Products	3.9	3.1
Specific Activity Of Mixed Core And Blanket Cooled 150 days, Ci/Mg Hm		
Tritium	1050	690
Kr-85	8430	11000
I-131	3.55	2.22
Strontium	162500	174000
Cesium	152000	321000
Ruthenium	1210000	500000
Total	6980000	4310000

* Mg Hm, megagrams (metric ton) heavy metal (uranium+plutonium) charged to reactor.

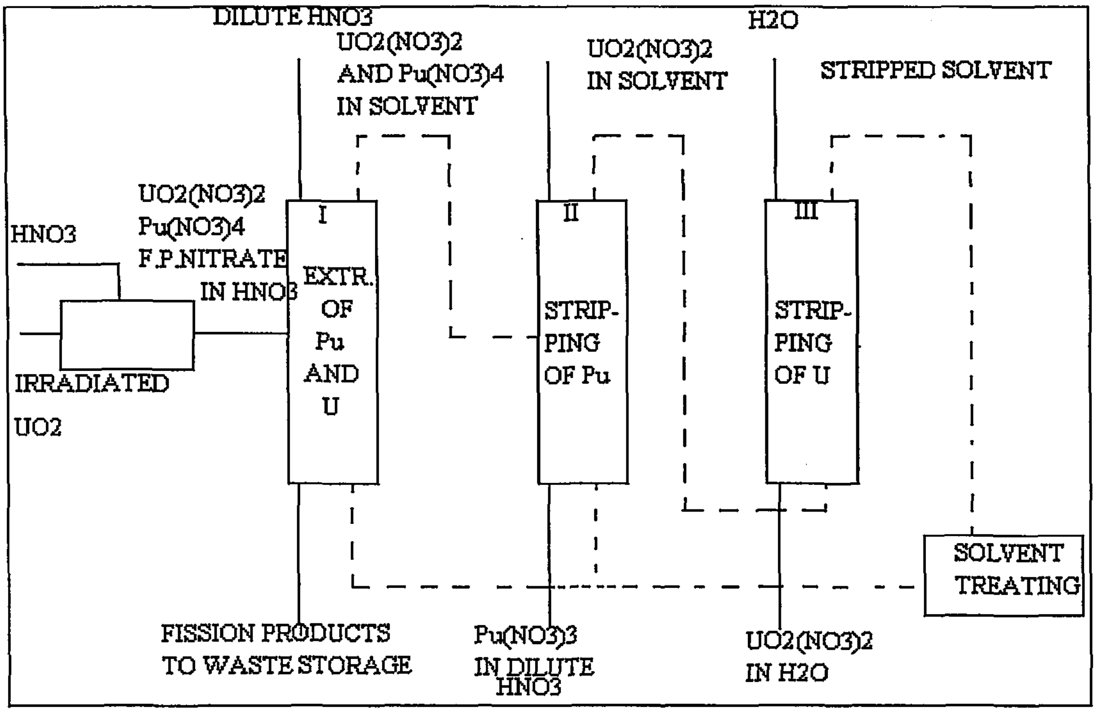

FIGURE 3.5 PRINCIPLE OF PUREX

In the Purex process, irradiated UO_2 is dissolved in nitric acid under such conditions that uranium is oxidized to uranyl nitrate and plutonium to $Pu(NO_3)_4$. The resulting aqueous solution of uranyl, plutonium, and fission- product nitrates is fed to the center of countercurrent solvent extraction contactor I, which may be either a pulse column or a battery of mixer-settlers.

Filled Columns: They contain some kind of filtering buffer (glass, metal, porcelain) in the form of coil and spiral. The function of the filled column is to force the organic liquid to flow in the aqueous phase through the resisting buffer, so that it causes to the dropwise flow. As a result, contact surface between two surfaces increases.

Pulse Column: It contains thin drilled plates along the column. The liquid drops contact these through these holes and so direct effective contact for two liquids are provided. Mixer-Settler: They are made up of two parts. Mixer, it leads the liquid to the contact of phase mixer and then to the settling part. Its small dimensions are the advantage of it.

This contactor is refluxed at one end by clean solvent and at the other by a dilute nitric acid scrub solution. The solvent extracts all the uranium and plutonium from the aqueous phase and some of the fission products. The fission products are removed from the solvent by the nitric acid scrub solution. Fission products leave contactor I in solut ion in aqueous nitric acid.

Solvent from contactor I containing uranyl nitrate and $Pu(NO_3)_4$ is fed to the center of contactor II. This is refluxed at one end by clean solvent and at the other by a dilute nitric acid solution of a reducing agent strong enough to reduce plutonium to the trivalent form, but not so strong as to reduce uranium from the hexavalent form. Ferrous sulfamateid frequently used. In contactor II plutonium is transferred to the aqueous phase, while uranium remains in the solvent. Solvent from the contactor II is fed to one end of contactor III, which is stripped at the other end by water, which transfers the uranium to the aqueous phase leaving the contactor.

After chemical treatment to remove degradation products, the solvent leaving III is reused in contactors I and II.

3.2.2.1 Some Characteristics Of The Purex Method

- Purex has a lower operating cost than other conventional processes.

- Product purity is around %99 and decontamination factor is 10^7.

- It has a good safety factor for the criticality and corrosion.

- Reaction is slow.

- U and Pu are extracted in water solution.

- To prevent chain reactions, a special geometry should be used.

- The equipments used in reprocessing of irradiated fuels should be simple and small.

4. INTEGRAL FAST REACTORS

The IFR concept has a number of specific technical advantages that collectively address the design goals of the advanced LMR. These advantages are in the areas of fuel performance, inherent passive safety, economics, waste management, and operability and reliability.

4.1 Introduction and Background

The Integral Fast reactor (IFR) concept consists of four technical features: (1) liquid sodium cooling, (2) pool-type reactor configuration, (3) metallic fuel, and (4) an integral fuel cycle, based on pyrometallurgical processing and injection-cast fuel fabrication, with the fuel cycle facility collocated with the reactor, if so desired.

Much of the technology for the IFR is based on Experimental Breeder Reactor II (EBR-II) experience which will be also discussed in next section. The EBR-II was the first pool- type liquid-metal reactor (LMR). Metallic fuel was developed as the driver fuel in EBR-II. From 1964 through 1969, 35.000 fuel pins were reprocessed and refabricated in the EBR-II fuel cycle facility, which was based on an early pyroprocess with some characteristics similar to that now proposed for the IFR.

Only in recent years have developments in metallic fuel taken place that now make the metallic fuel-based IFR a promising development choice. Even with its potential fuel cycle advantages, metallic fuel was thought to be unacceptable for many years because of its poor irradiation behavior in the 1950s and early 1960s. Discoveries at EBR-II in the late 1960s and design developments and irradiation experience in the 1970s have totally changed this picture. Generic metallic fuel can now be designed for very superior irradiation performance. Over 20.000 older design EBR-II fuel pins achieved their 80 MWd/kg design burnup without any failures. With simple design changes, the new EBR-II fuel has a design burnup of 140 MWd/kg.

Furthermore, very recent metallurgical processing discoveries and developments have radically altered both the pyroprocess itself and the outlook for the major breakthroughs in both fuel and blanket processing. Pyroprocessing was inadequate for scale-up in the early EBR- II melt refining pyroprocess. Losses were several percent, the product fuel still contained all the noble metal fission products, and blanket material was not processed. The new IFR process replaces melt refining with a new electrorefining process. Electrorefining using a liquid cadmium anode and a fused chloride salt extracts the fuel uranium-plutonium mixture from the dissolved mixture of fuel and fission products.

In the development of the next generation of advanced reactors, particular attention must be given to inherent passive safety, fuel cycle closure, including waste management, and economics.

The basic thrust of IFR development is to exploit the inherent properties of the reactor materials to achieve the desired characteristics of next-generation nuclear power, which is also capable of extending uranium resources to meet large energy demands expected in the future.

4.2. Fuels In Integral Fast Reactors

4.2.1. Fuel Performance [12]:

Metallic fuel is the key to the IFR concept. Metallic fuel is easy to fabricate. Metallic fuel slugs, typically of 0.5 cm in diameter and 30 to 45 cm in length, are produced by injection casting. One to three fuel slugs are then loaded into an individual cladding jacket and cladding is filled with sodium bond. The cross-sectional area of the fuel slug is 75% of the available area inside the cladding. As the fuel is irradiated, the fission gas bubbles grow, inducting fuel swelling. As the fuel swells out to the cladding wall, the fission gas bubbles interconnect and provide a gas release path to plenum located on top of the fuel column. This interconnected porosity is the key to stopping further fuel swelling and mitigating any significant stresses due to fuel/cladding contact.

Metallic fuel has excellent steady-state and off-normal performance characteristics. High burnup potential has been demonstrated with EBR-II uranium-fissium driver fuel. The new IFR U-ZR and U-Pu-Zr fuel irradiation experience has been better than expected. The lead tests have achieved 185 MWd/kg. [13]

The metallic fuels have a higher metal density, thermal conductivity, temperature induced coefficient of expansion and breeding potential when compared to the ceramics.

Metallic fuel has excellent transient capabilities. The metallic fuel itself does not impose any restrictions on transient operations or load-following capabilities. The robustness of metallic fuel is illustrated by the following sample history of a typical EBR-II driver fuel:

40 start-ups and shutdowns
5 15% overpower transients
3 60% overpower transients
45 loss-of-flow (LOF) and loss-of-heat-sink tests including a LOF test from 100% power without scram.

The metallic- fuelled cores also have a higher breeding potential due to a superior neutron economy associated with a hardened spectrum. As illustrated in Figure 4.1, metallic fuel has a much higher breeding ratio than oxide or carbide fuels. The superior neutron economy further allows reactor core design improvements, such as minimizing the reactivity swing over the operating cycle, increased cycle length, etc.

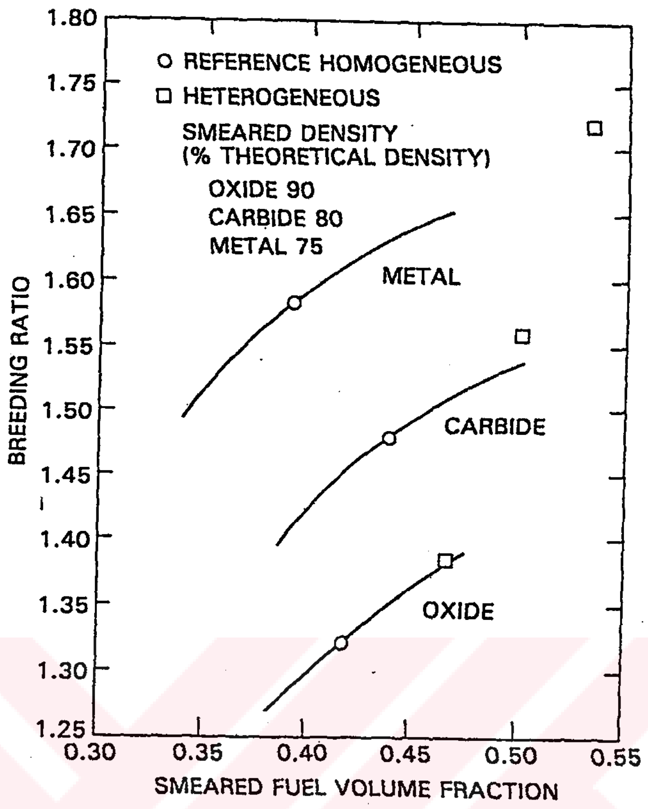

FIGURE 4.1.COMPARISON OF BREEDING RATIO POTENTIALS OF OXIDE, CARBIDE, AND METAL FUELS [12]

Three recent developments in uranium-plutonium alloy fuels fast breeder power reactors should be emphasized: (1) additions of zirconium and titanium have been found to increase the melting temperature of uranium-plutonium alloys from 820-910 °C for U-Pu-Fs alloys to 1100 °C for the U-Pu-Zr and U-Pu-Ti alloys, (2) uranium-plutonium-zirconium alloys have been found to be compatible with type 304-SS up to temperatures ranging between 800 °C and 850 °C, and (3) a burnup of 8.7 at.% (33x1020 fiss/cm3) has been demonstrated successfully in fuel pins of a uranium-plutonium-fissium alloy.

4.2.2.Fabrication Of Fuel Rods [13]

The metallic fuel is easily fabricated, requires no finishing, and allows a relaxed specification that further simplifies the fabrication process. The easy fabrication combined with simple, compact pyrometallurgical reprocessing promises a major cost reduction in the liquid metal reactor complete fuel cycle.

A procedure known as injection casting was effectively used for the fuel rods for EBR-II and remotely over 35.000 fuel elements have been produced for EBR-II using the integral fuel cycle between 1964 and 1969. More than 100.000 fuel elements with fresh uranium have been fabricated.

4.2.2.1 IFR Fuel Cycle Based On Pyrometallurgical Reprocessing [14]

The metallic fuel form facilitates the use of a pyrometallurgical process for fuel reprocessing wherein the fuel remains in a metallic form throughout the process, the uranium and plutonium remain intimately mixed, and the fission product removal fractions are 10 to 100 times less than in aqueous processes. Figure 4.2.summarizes the process; blanket elements and driver elements undergo separate electrorefining operations wherein the cladding and fission products are separated in the salt phase and the uranium and plutonium are deposited in metallic form on cathodes. The cathode from the blanket process is subjected to a halide slagging step to separate a plutonium-rich feed for fissile makeup in driver refabrication from a uranium-rich feed for the blanket refabrication. These metallic feed streams supply a fuel and blanket remote refabrication process housed in the same hot cell and based on injection casting of fuel slugs, followed by insertion and seal welding into sodium-bonded clad. The fuel cycle can be operated in a closed, fissile self-sufficient mode wherein the reactor breeding just compensates for reprocessing/refabrication losses.

4.2.3. Fuel Element Lifetime [13]

Based upon achieving 185,000 MWd/T in experimental elements, a lifetime of 150,000 to 200,000 MWd/T can be projected for the design considered for the innovative reactors. Once the end-of-life failure mechanism is learned, the design could be optimized along with fuel and cladding compositions to maximize lifetime.

The character and behaviour of metalic fuel allows excellent irradiation performance and inherent safety features. Both laboratory and reactor test results continue to substantiate the expected excellent fuel performance under both steady- state and off-normal conditions, and the inherent safety potential is being demonstrated through the EBR-II tests and TREAT experiments.

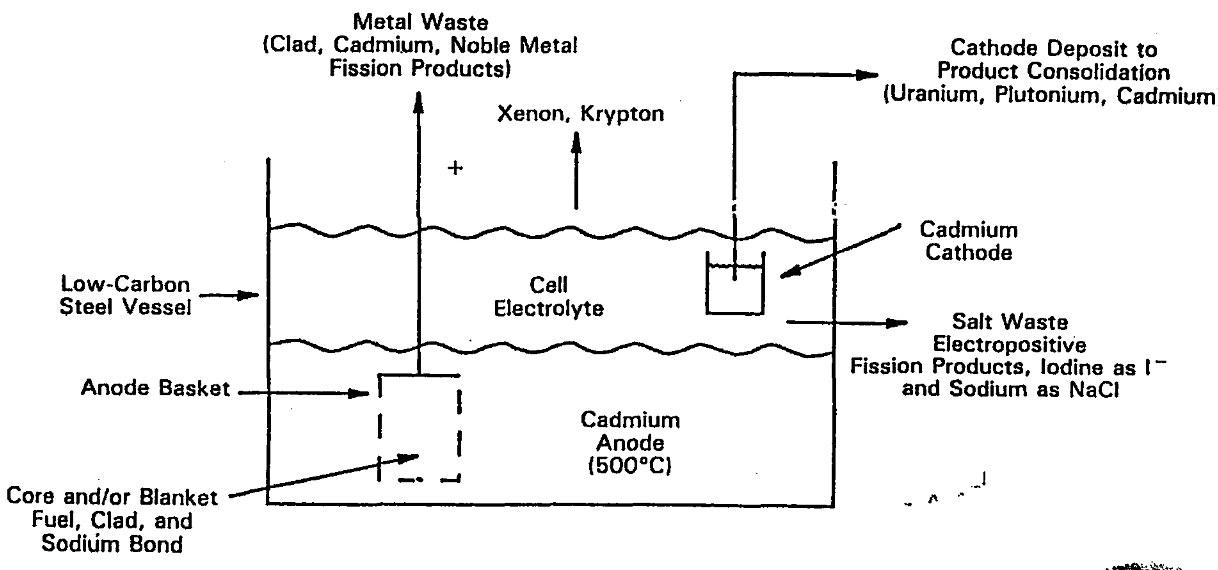

Figure 4.2.Schematic Representation Of Electrorefining

4.3. Operation And Safety Physics Of IFR's

4.3.1. Physics Of Inherent Safety [14]

There are two desirable goals of inherent safety; the core reactivity shutdown and decay heat removal, neither relying on devices requiring switching and/or outside sources of power.

The requirement for the heat removal of decay heat inherently include, a) arranging the reactor vessel internals and primary coolant circuit to achieve natural circulation cooling of the core and a smooth transition to natural circulation upon scrammed and unscrammed loss of primary and secondary pumping action, b) providing an ultimate heat sink with a cooling capacity for a long term decay heat power level, c) providing a thermal mass and internals sufficient to absorb, that initial transient decay heat exceeds the inherent decay heat removal capacity.

Inherent decay heat removal has been approached in U.S.A modular sized plants by use of pool layout.

Inherent control of reactivity in the case of ab-normal events without reliance on control rod scram can be approached by core design for favorable ratios of the power, power to flow, and inlet temperature coefficients of reactivity and by design for a high internal conversion ratio.

It is possible to inherently bring the power to a safe shutdown level with a core average temperature rise that is well below that rise causing sodium boiling or structural damage.

4.3.2 Review Of Core Physics Of Inherent Shutdown

A liquid-metal-cooled reactor (LMR) core can be influenced by means of the changes in the coolant inlet temperature and flow rate, through reactivity changes due to control rod motion or seismeicly induced core geometry changes.

The influence of these three paths to the reactor core can be investigated by studying anticipated -transient-without-scram (ATWS) events, which are loss-of flow without scram (LOHSWS), loss-of-heat-sink without scram (LOHWS), and rod runout transient overpower without scram (TOP), plus two overcooling accidents; pump overspeed and chilled T_{inlet}.

The quasi-static reactivity balance for the core to express the results of events can be written:

$$0 = \delta\rho = (P-1)A + (P/F-1)B + \delta\rho TinC + \delta\rho ext \qquad (4.1)$$

P,F = normalized power and flow, respectively

δTin = change from normal coolant inlet temperature
$\delta\rho ext$ = externally imposed reactivity
A, B, C = integral reactivity parameters that are measurable at the operating plant via perturbations introduced through the communication paths.
C = inlet temperature coefficient of reactivity (cent/°C)
(A+B) = reactivity decrement experienced in going to full power and flow from zero power isothermal at coolant inlet temperature (cents).
B = power flow coefficient (cent/100% P/F)
A = net (power-flow) reactivity decrement (cents)

In other words, A, B,C can be expressed as following:
$A = (\alpha_D + \alpha_e).\ \delta\ T_f$ (4.2.)
$B = (\alpha_D + \alpha_e + \alpha_{Na} + 2(\alpha_R + 2/3\ \alpha_B))\ .\ \delta\ T_o$ (4.3.)
$C = \alpha_D + \alpha e + \alpha_{Na} + \alpha_B$ (4.4)

where α_D, αe, α_{Na}, α_B α_R are Doppler, fuel expansion, sodium density, rod drive line, core radial expansion reactivity feedback coefficients respectively. And δT_f fuel, δT_o coolant temperature changes Core design goals are investigated [14] through three anticipated transients without scram (ATWS) plus two overcooling accidents and chilled T_{inlet}.

Core design goals are given in Table 4.1. And Table 4.2. gives the result of investigations to be prepared Table 4.1.

In transients, equation (4.1.) can be solved for the new set of condition; coolant flow, inlet temperature, externally induced reactivity.

The linearity of the $\delta\rho$ can not be said for the large changes in power, power/flow and inlet temperatures. This equation is useful for the trend analysis, otherwise for detailed studies computer codes are used. The typical coefficient values for oxide fuelled LMR are; A=-1.50, B= -0.50 and C= -0.005 $/°C.

To reduce power from 100 to 5 % by inherent means with no rod motion at constant flow equation (4.1.) requires the inlet temperature to rise by 380 °C

$$0 = -(0.05-1)(1.5) - (0.05/1-1)(0.5) - \delta Tin(0.005) + 0$$

$$\delta Tin = 380\ °C$$

From this simple example can be understood that reactor inherently is brought to safe condition in 5 % power with a large increase in coolant temperature.

If the temperature rise is required to be bring the power to zero is too large, it may induce sodium boiling, fuel element failure or structural damage to reactor components, which should be considered. This simple methodology is used to discern relationship between the reactivity parameters A, B, C given in Table 4.1. which are obtained using Eqn.4.1. and shown on Table 4.2. Those terms can be used for the several events which can externally affect the reactor core. Once the desired trends in A,B,C are known, core design choices to achieve these trends can bee selected. The summary of quasi- static reactivity balance results for unprotected accidents are also tabulated in Table 4.1. Detailed calculations are given in Reference14.

TABLE 4.1.
CORE DESIGN GOALS TO PROMOTE INHERENT SHUTDOWN [14]

Goal	Rationale
1. A, B, and C should be negative	Negative A implies a negative prompt power coefficient. Negative B and C imply a negative temperature coefficient
2. A/B should be small	A/B must be small to inherently control the asymptotic temperature rise in LOFWS.
3. $C\delta T_0/B$ should lie 1 and 2.	This range, with goals 1 and 2, should provide a proper balance between the LOHSWS and chilled inlet temperature inherent responses.
4. $\delta\rho_{TOP}$/abs(B) should be small,e.g.	$\delta\rho_{TOP}$/abs(B) must be small to inherently control the TOP.
5. $\Gamma.\lambda.(1+A/B)^2$.abs(B) should be large relative to 1 $ of reactivity.	Pump coast down time Γ should be suitablyadjusted relative to delayed neutron decay time λ.

Table 4.3. give the design strategy considering the numerical values of the reactivity feedback coefficients and the changes which are possible to make. Some of the values have to be taken as they are, but some of them can be adjusted to increase the inherent shutdown trend.

As it is observed in the last column of the table 4.3.low ΔT_f of the metal fuel increase the inherent safety of the IFR. The radial expansion coefficient also is better for metal fuel.

TABLE 4.2.
QUASI-STATIC REACTIVITY BALANCE RESULTS FOR UNPROTECTED ACCIDENTS

	Asymptotic State			
	P	F	δTin	Indicated Trend for Inherent Shutdown**
LOHSWS	0	1	(A+B)/C	A+B, small C,large
TOP	1 (after rise in Tin due to BOP heat removal limit)	1	$\Delta\rho_{TOP}/(-C)$	A+B,large C,large $\Delta\rho_{TOP}$, small
LOFWS	0	Natural circulation	0	A, small B, large τ, long
Chilled Inlet*	(1-C.δTin)/(A+B)	1	abs(1- δTin)≤ 1,5.ΔTc	C,small A+B,large
Pump Overspeed	(1+A/B)/ (1/F+A/B)	F>1	0	A, negative B, negative

* Condition is obtained in the last column considering the conflict existing between desirable trends for different ATWS events.

** Conflicts exist between desirable trends for different ATWS events. Resolution of these conflicts is discussed in the text.

TABLE 4.3.IFR CORE DESIGN STRATEGY FOR INHERENT SHUTDOWN

DESIRE	DOPPLER	FUEL AXIAL EXPANS.	SODIUM DENSITY	ROD DRIVE LINE	RADIAL EXPANS.	COOL-ANT ΔT	FUEL ΔT
SMALL A=	(α_D	+α_e					)* ΔT_f
LARGE B=	(α_D	+α_e	+α_{Na}	+2(α_R	+2/3α_B))	* ΔT_C/2	
LARGE C=	(α_D	+α_e	+α_{Na}		+α_G)		
METAL	-0.10	-0.12	+0.18		-0.16 -0.33	150	150
OXIDE	-0.16	-0.10	+0.11		-0.12 -0.27	150	750
DESIGN STRATG.	TAKE AS IS	TAKE AS IS	MAKE LESS +	TAKE AS IS	MAKE LARGE	TAKE AS IS	MAKE SMALL

TABLE 4.4.FUEL CYCLE COST COMPARISON FOR
1350 MW (ELECTRIC) PLANTS

	Metal (mill/kWh)	Oxide (mill/kWh)
Fuel Cycle Facility Capital Fixed Charges	0.68	3.39
Fuel Cycle Facility O&M Cost	1.28	2.31
Hardware Components	0.80	0.80
Fissile Inventory Carrying Changes		
Plutonium Start-up (25$/g)	0.73	0.88
Uranium Start-up (35$/g)	1.66	1.99
Waste Disposal Fee	1.00	1.00
Total		
Plutonium Start-up	4.99	8.38
Uranium start-up	5.42	9.49

The reactor system mass per unit energy production is a strong function of the reactor size. Because, modular construction approaches based on small size may well make sense under conditions where initial capital costs are the dominant concern.

4.5. Waste Management

The liquid metal reactor (LMR) systems has a potential for extending uranium resource by a factor of 50-100 compared to other commercial systems. Recently, all experimental and demonstration works are based on the fact that the LMR's are power reactors, whereas, with time being the importance of resource arguments will be understood. Fuel cycle closure is important in two aspects; a) The recovery of spent LMR fuel is important for the competitiveness of the fuel cycle costs, b) Waste disposal is a problem for public and a closed cycle provides more acceptance.

The pyroprocess system leads to much simpler waste treatments and lesser waste. Long-lived fission products in purex are no more problem in the pyroprocess by means of immobilize. Tritium is collected as HTO by cell purification system. Removal of krypton by cyrogenic distillation is straight forward. Carbon-14 remains in the salt or metal waste as carbides, and iodine is contained in salt waste as CsI or NaI. Ruthenium is contained as metal in the cadmium waste. The chloride salt waste can be obtained in the form of glass waste, but the extraction of actinides and convert the salt waste into an concrete waste. The high level wastes are capsulated as a metal matrix or lead containers.

In pyroprocess, the actinide elements behave usually together, so that they are recycled along with plutonium. Besides, there is a big separation between actinides and rare earth fission products causing easy extraction of the remaining actinides removed, the fission products in the high level waste will decay sufficiently in 300 years so that their long term radiological risk drops below the cancer level of natural uranium ore.

The entire system, including the waste process, must be considered, and feasibility of each element should be demonstrated right from the start.

4.6. Operability And Reliability

A mandate for change about reliability is; the system must be reliable - it should have low outage rate to meet the prescribed loss of load probability (LOLP). Desired values of 21 st century is lower than present values. This will naturally cause internal reductions in a power plant and provision for extensive maintenance during operation [2].

Although there are lots of desires for the future, the recent properties of the IFR lead to ease of operation, reliability in many components and systems that are much less so in water-cooled systems. The plant capacity can also be improved by means of easing the maintenance requirement.

For example, this maintenance requirement easiness can be achieved with the fact that sodium is non corrosive to the components in the reactor structure. So that there is no significant radioactive deposits causing some maintenance difficulties. However, this an important problem for LWR's compared to the LMR's. As a result, sodium usage causes easiness in the maintenance and low radiation levels for the personnel. And during the maintenance and inspection of the equipments like steam generators, turbine generator, steam and feed water pumps, there are no problems expected. Even after >20 years of operation, EBR II and other LMR's are experiencing <0.2 person-Sv/year personnel exposure, as compared to some LWRs that now approach 10 person-Sv/year.

In addition, non corrosive coolants provide sodium coolant components with high reliability and higher availability.The EBR II steam generators have operated without leaks over 23 years of continuous service. In contrary to this situation, in LWRs the water chemistry has caused frequent failures in generators.

5. A CASE STUDY: EBR-II INITIAL OPERATION -- SIGNIFICANCE

The Experimental Breeder Reactor-II has progressed successfully from the preoperational testing phase to operational phase. Some of the highlights of that transition and of the operating experience gained will be discussed in this chapter. Included are descriptions of the approach to power, selected operational aspects of special interest, fuel element performance, irradiation experiments, and future plans.

5.1. Description Of Facility

The EBR-II facility is an integral nuclear power plant; i.e. it includes a complete fuel processing and fabrication facility in addition to the reactor, heat transfer systems, and steam-electric plant. The reactor is a fast power breeder, sodium cooled. Maximum power rating is 62500 kW. Gross electrical output is 20000 kW. Steam conditions are 86 atm and 447 °C.

The reactor and the entire primary system are submerged in sodium contained in a tank of 8 m diameter and 8 m depth. This primary tank has a cover fitted with seventy nozzles through which operating mechanisms, instrument leads, etc., extend. These mechanisms include the primary pumps, intermediate heat exchanger, storage basket for spent fuel, fuel transfer arm, and immersion heaters. Two large rotating plugs, one eccentrically located in the other, are employed in the centre of the cover. These carry the fuel gripper, control rod drives, and reactor cover lifting mechanism. The entire tank weighs about 900 tons (including 320 tons of sodium) and is suspended from an overhead support structure by six hangers equipped with rollers at their top ends. The general fuel handling scheme is depicted in Fig.5.1.

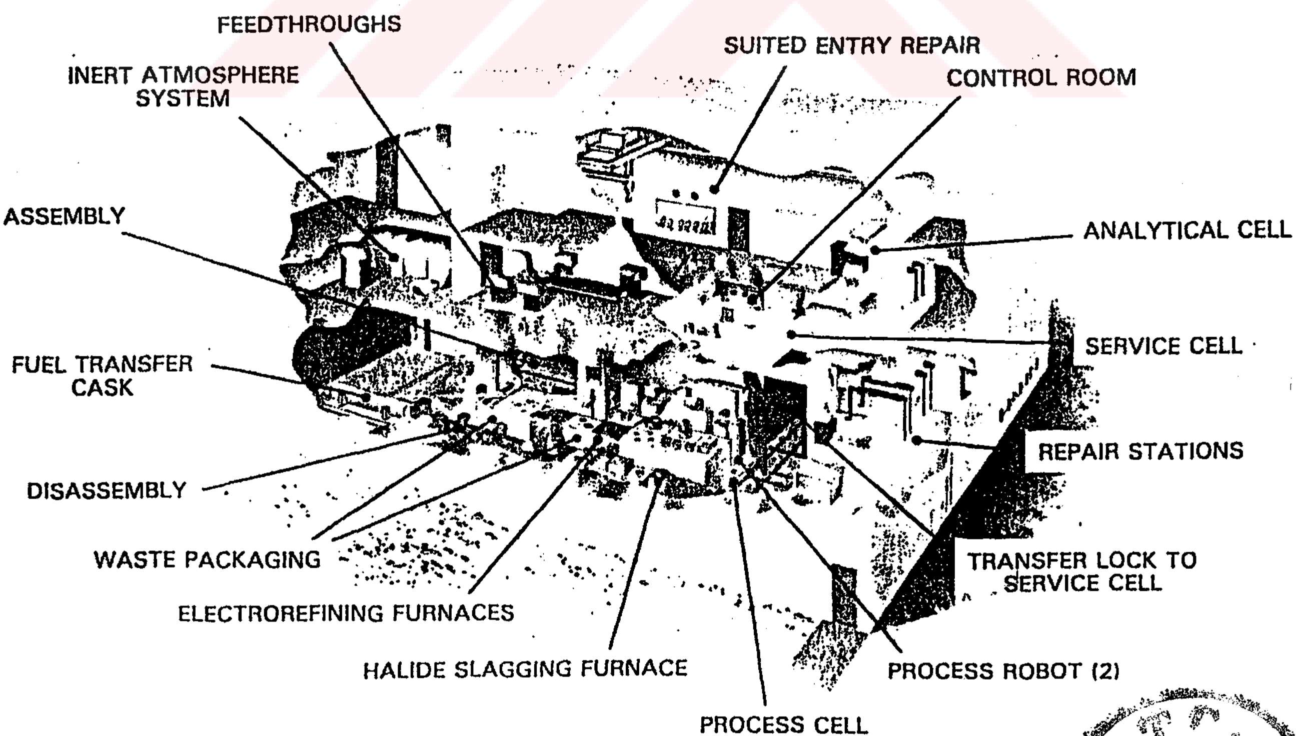

FIG 5.1.COMMERCIAL IFR FUEL CYCLE FACILITY

The reactor core is approximately hexagonal in cross-section, with length to diameter ratio of about 0.8. The fuel is U-5% Fissium. At full power, maximum power density is 1230 kW/litre of core volume. Reactor inlet coolant temperature is 371 °C; outlet temperature is 473 °C.

Breeding blankets of depleted uranium surround the core. Reactor control is effected by movement of fuel into and out of the core. A total of twelve peripherally located control rods are employed, of which any one may be used for regulation.

5.2. Selected Aspects Of Operation

5.2.1. Reactor And Primary System

Performance of the reactor and primary system has been excellent. The reactor is highly stable and readily controllable. All core power densities, coolant flow rates, and subassembly outlet coolant temperatures are as predicted (one out of the total of twenty core subassembly outlet thermocouples reads higher than calculated, but this couple is believed to be incorrectly positioned). The reactor inlet coolant temperature is easily maintained at the desired level with precision (in the submerged concept used, mixing of the large mass of primary bulk sodium effectively minimizes reflection of steam or secondary system load changes). Primary and secondary system thermal power measurements agree within a few percentages.

Mechanical stability of core components is good. Examination of the entire first core loading after removal from the reactor has shown no evidence of mechanical weakness of any kind. No wear or fretting marks or other indication of vibration, either of fuel elements or of subassemblies, has appeared. Similarly, no suggestion of coolant flow blockage, surface deposition, or any sort of harmful effect from particulate or other coolant contamination has arisen. No gas entrainment in the coolant has been discerned. No sticking of subassemblies in the reactor vessel grid has occurred. Because no fuel element failures have occurred, virtually no fission product build-up outside the reactor has occurred.

Neutron and gamma shielding provisions have proved more than adequate. Even in the complex primary tank cover area with its numerous nozzles and mechanisms, no shielding additional to that provided in the original design has become necessary. Effectiveness of the primary neutron shielding immediately surrounding the reactor vessel is illustrated by the fact that after more than 5000 MEd of reactor operation, stainless steel heater assemblies removed from the primary tank generally exhibited surface activity levels of less than 30 m/hr.

Maintenance of the reactor and primary system has proved entirely feasible and practical. (The oscillator failure caused substantial delay; however, this temporarily installed experimental unit was not considered a part of the regular primary system). Repair or replacement of numerous components was accomplished without difficulty. Included among those partially or completely removed from the system and than reinsert were several control rod drive mechanisms, nuclear instrument assemblies, primary tank heaters, auxiliary gripper plug, and reactor vessel cover lock mechanisms.

A number of minor difficulties, and one accident that delayed reactor operation a month, occurred with fuel handling mechanisms and were surmounted.

5.2.2. Secondary And Steam Systems

Operating experience with the secondary sodium system has been very good. No significant troubles have developed. Intermediate heat exchanger performance is as predicted, both thermally and hydraulically. Sodium activity induced by passage through the heat exchanger (located in the primary tank) is negligible, only a few mr/h can be observed at system pipe surfaces. The 251 kW induction heating employed on the entire secondary has worked well, although minor improvements in its control system are now planned. Draining and refilling of the system, usually to facilitate repair or maintenance of steam system components, has been accomplished repeatedly and easily.

Steam system performance type date has been fair. As seems normal with nuclear plants, most troubles have developed with conventional, non-nuclear components. Only indicative difficulties will be noted here. The steam turbine driving the boiler feedwater pump failed early in operation, requiring replacement of the turbine wheel. The positive displacement start-up feedwater pump malfunctioned repeatedly and is to be replaced. Steam drum and main steam header safety valves frequently relieved at unacceptability low pressures or failed to reseat properly, requiring numerous checks and adjustments. Steam bypass valves often stuck full open or controlled erratically; several new sets of trim and a new actuator were required. Bellows failures occurred in conventional expansion joints in the steam piping system, requiring redesign.

Experience with the sodium-to-water steam generator has been better than expected. No sodium-to-water leaks have occurred. Pressure drops are lower than anticipated, and dissipation of full 62.5 MW reactor power is possible with only the two superheaters presently installed (four were called for in design). Although detailed analyses have not been made, heat transfer coefficients appear to be essentially as predicted. The only difficulty encountered to date has been the single occurrence of a pinhole leak of water to atmosphere, and this was repaired with little effort. At the 45 MW level employed to date, steam conditions are 86 atm and 424 °C.

5.2.3. Pump Performance

During preoperational testing, both of the primary pumps (centrifugal type, 18.9 m^3/min each) failed soon after star-up. The cause was rubbing of the pump shaft at the lower labyrinth seal. This seal had been provided with small clearance to minimize migration of sodium vapour. Before start of the approach to power, the pumps were reworked to incorporate shafts and seals or improved design. Since then, both pumps have accumulated about 7000 hours' operating time, most of this being at full reactor flow. One pump started sticking as a result of sodium or sodium oxide accumulation above the seal. The sticking was easily overcome by vertical exercising of the shaft. An increase in argon backflow along the shaft has prevent reappearance. By overcoming this minor difficulty, operation has been completely satisfactory.

The secondary system pump (electromagnetic induction type, 24.6 m^3/min) also was repaired during preoperational testing. It earlier had developed a leak owing to fatigue cracks caused by duct wall vibration at high flow. As insurance against further cracking, operation of the repaired pump presently is limited to 85% of full flow. Since start of the approach to power, the pump has accumulated approximately 10,000 hours' operating time and has performed completely satisfactorily.

The secondary pump control system enables stable flow at rates down to a small fraction of 1%. It also enables reverse thrust. These characteristics are needed during plant standby when extremely low secondary flow (frequently lower than that effected by natural convection alone) is required. Despite a small control dead-band in the zero thrust region, operation of this system has been excellent. [17]

6. ACTINIDE BURNERS

6.1. General Concept

The main purpose of the Advanced Liquid Metal Reactors (ALMR) Actinide Recycle Program is to achieve a standardised design licensed and certified. This system consists of a compact liquid metal (sodium) cooled reactor with inherently passive safety characteristics and pyrometallurgical fuel cycle (IFR) present under development.

The idea of this program is closely connected to transmutation or fissioning of the longer lived transuranic (TRU) isotopes to shorter-lived products. The primary requirements for the transmutation of these TRU isotopes are to eliminate them from ultimate waste stream through processing spent fuel and recycle the TRU as an ALMR fuel source. In this ultimate waste stream, the transuranic material is 1% in heavy metals and the minor actinides (MA) such as neptunium, americium and curium are 6 % in it.

There are two basic processes for recycling to achieve the ALMR; aqueous and pyroprocess.

Unlike the aqueous process, the pyroprocess does not separate Pu from other TRU. This is also something very important guaranty against production of Pu used for bomb.

The conceptual limit for transuranics burning is a "pure" burner core where only transuranics are used as a fuel material. Therefore, as 840 MWt ALMR pure burner design operating at an 85 % capacity factor would consume 280 kg/yr of transuranics.

6.2. Reactor And Fuel Cycle Description

6.2.1.Reactor System

As the reactor system, the ALMR is designed to achieve a safe reliable liquid metal cooled, fast neutron plant. Meanwhile, it would also be economically competitive with other sources of electrical energy generation such as coal.

The commercial ALMR plant is planned to be consisting of three power blocks having six reactor modules and the total electrical production rating is 1866 MWe.

The superheated steam is supplied to a single turbine generator from these reactor modules. The steam generator system is separated from the nuclear safety-related portion of the plant. This system and the intermediate heat transport system (IHTS) are also designed to achieve industrial standards. The reactor module is 9.6 m in diameter and about 18 m high. The reactor module, the IHTS and major portion of the steam generator are underground.

Primary sodium is circulated in the reactor by four submerged self cooled electromagnetic pumps during normal operation. See Figure 6.1 and Figure 6.2.

TABLE 6.1 ALMR DESIGN DATA

Overall Plant	
No.of Reactors/Power Block	Two
No. of Power Blocks	One/Two/Three
Net Electric Output	622/1244/1866 MWe
Net Station Efficiency	37 %
Turbine Throttle Conditions	14.8 MPa at 452 °C
Reactor Module	
Thermal Power	840 MWt
Primary Sodium Inlet/ Outlet Temperature	360 °C/500°C
Secondary Sodium Inlet/ Outlet Temperature	327°C/477°C
Reactor Core	
Fuel	Metal
Refuelling	23 months
Breeding Ratio	1.05 Reference 1.23-0.6 Capability (0.02 possible with a Pu-Zr fueled burner)

6.2.2. Core Design

As the core design for light water reactor (the minimal core height 66 cm, 358 diameter and 192 fuel assemblies) actinide recycle system calculation showed us that the small burner destroyed approximately 315 kg of MA and 2400 kg TRU per core over 60 years.

TABLE 6.2. CORE NEUTRONIC CHARACTERICTICS

	Small Burner
Core Height (cm)	66
Core Diameter	358
No.of Fuel Assemblies	192
No.of Blanket Assemblies	-
Conversion Ratio	0.72
Cycle Length (months)	12
Burnup Reactivity Swing ($)	8.99
Peak Linear Power	10.4
Sodium Void Worth ($)	2.54
TRU Enrichment (wt % in U-TRU-Zr)	19/23
TRU Inventory (kg/core)	2554
TRU consumption Rate	
kg/year/core	83.2
%inventory/year	3.3

6.2.3. Metal Fuel Cycle

The ALMR Actinide Recycle System consists of pyrometallurgical fuel cycle. Fig 6.1 shows the ALMR reference fuel cycle.

The pyrometallurgical processes produce two primary waste streams. One of them is a mineral form with no TRU and second is metallic waste consisting only TRU. After about 60 % of core and blanket fuels from the ALMR were processed in an electrorefining campaign, the actinides are completely removed from the cadmium pool and overall Pu/U ratio in a electrorefiner is raised to above 10 by means of electrotransport of uranium to a solid cathode.

In the ALMR's hard neutron spectrum, the actinides from both the LWR and ALMR spent fuels are reduced to shorter lived fission products. So, the fission products are removed from the fuel cycle as waste products whose radioactive level, from standpoint of toxicity will be less than their source, natural uranium in a few hundred years. [15]

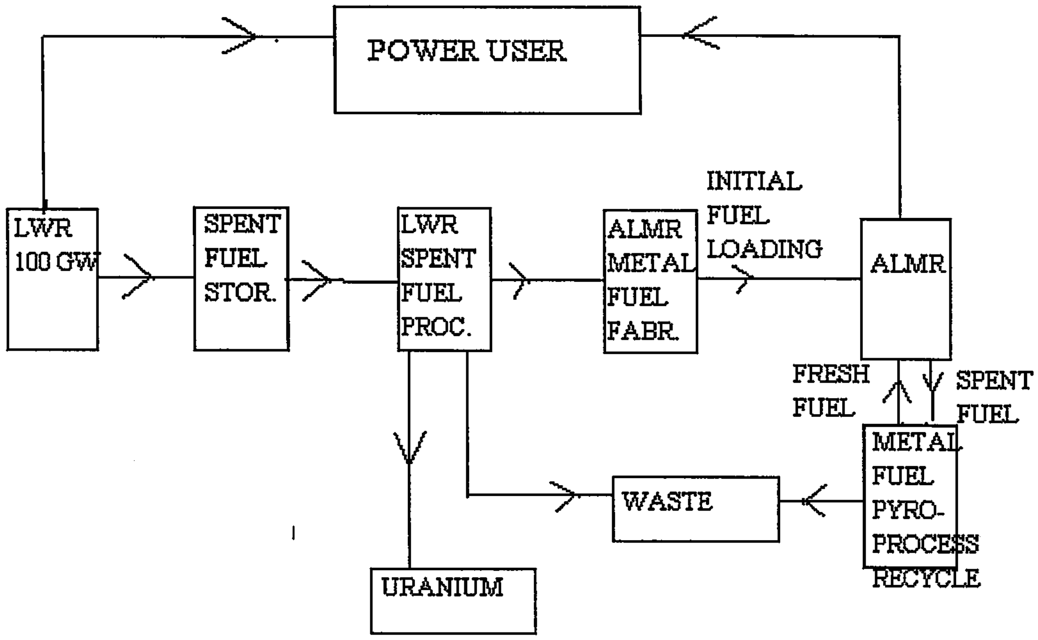

FIGURE 6.3. ALMR REFERENCE FUEL CYCLE

6.3. DEGREE OF USEFULNESS FOR ALMR

Up to end of the year 2015, it expected to have spent fuels with totally more than 600 tonnes actinides. Those 600 tonnes of actinides to be graved are a valuable energy source. These fuels can fuel new reactors almost all types of new reactors. Unless the recycle process is used, there will be a big demand for new depositories having building difficulties.

Recently, the IFR research program is getting nearer to completion and is in the early stages of developing LWR actinide recovery for recycle in the ALMR. From this point of view, the IFR research program has a very big importance for the future. [16]

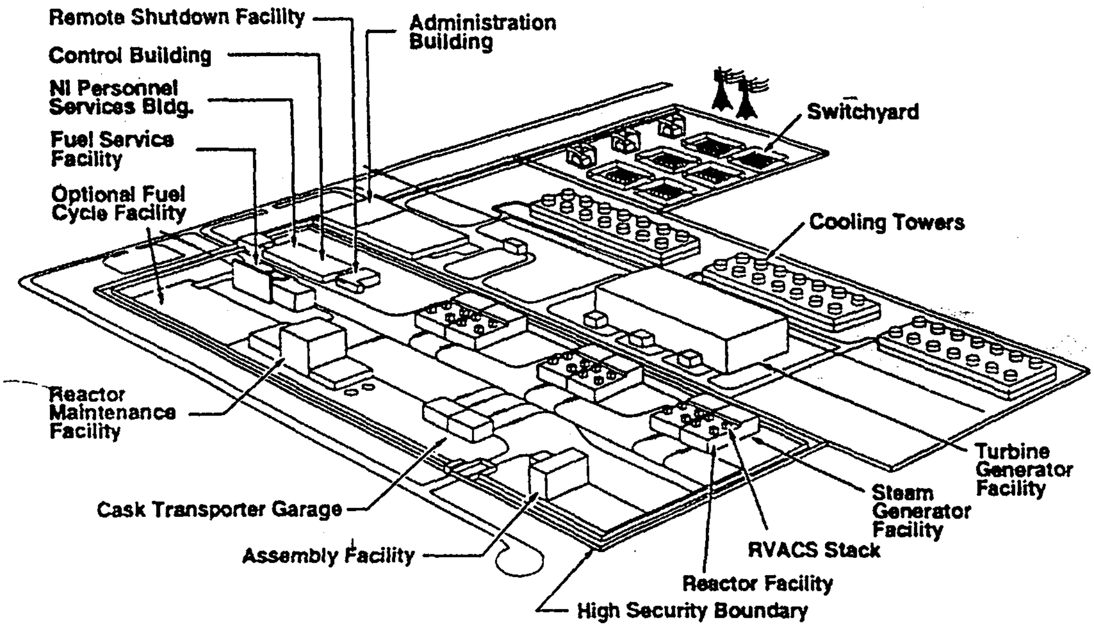

FIGURE 6.1.ALMR POWER PLANT

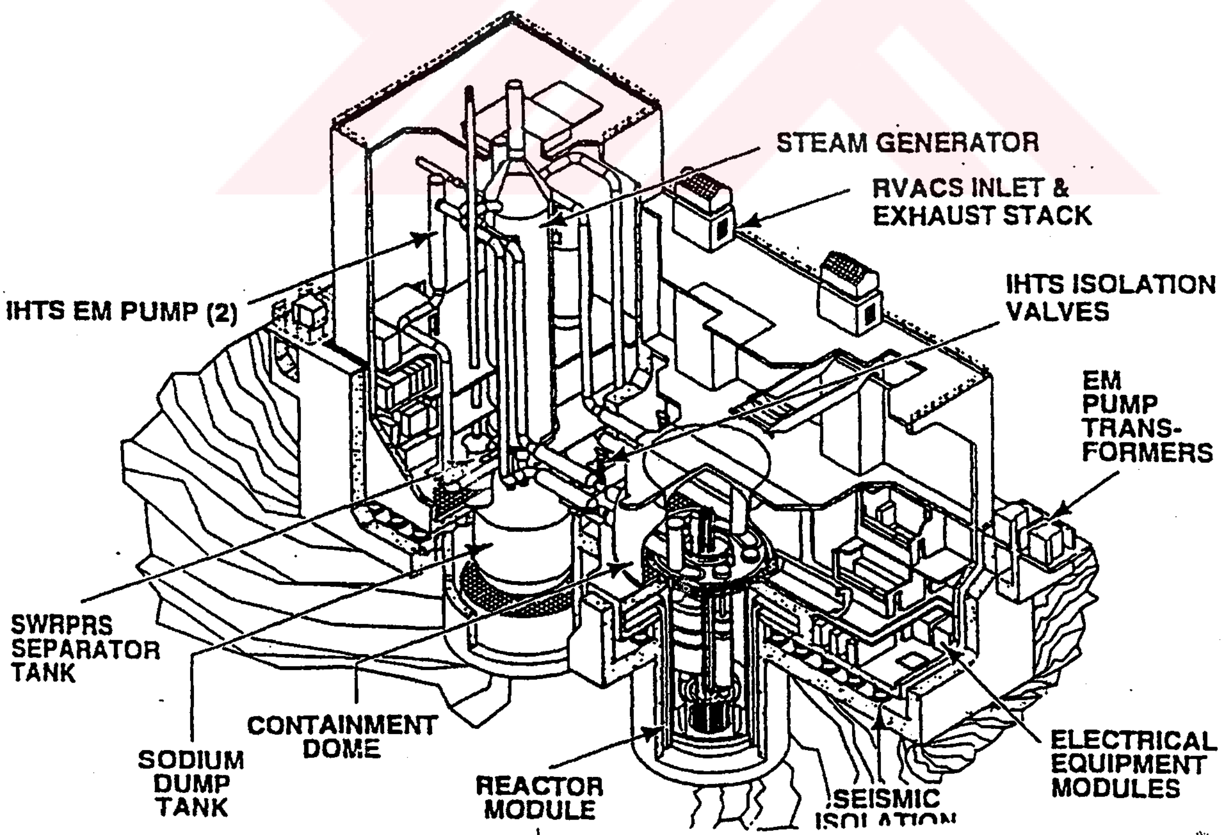

FIGURE 6.2. REACTOR AND STEAM GENERATOR FACILITY GENERAL ARRANGEMENT

7. CONCLUSION

The world energy demand in the future will not be under the present level, because much longer population with energy consumption less than world average. In recent years, there are many developments in worldwide energy supply researches, and the results have shown us that the research efforts for next generation nuclear power plants should be encouraged. It seems impossible to solve the lack of energy problem without nuclear energy and an environmental side effect. Moreover, there will be a high possibility of a supply and demand situation for oil in the medium- and long range view. To increase the energy supply, there is an increasing emphasis on nuclear power in many countries such as France, Japan, Canada, Germany, Czechoslovakia, former USSR. However, there is a decreasing trend for emphasis about nuclear power in USA. This situation has also caused to the fact that technical leadership in nuclear power is shifting from USA.

In 21st century, an opportunity will be for nuclear power to solve the world electric energy source problem. The rate of improvement in power reactor technology is rising which indicates that rapid technology advances will continue to be made in existing reactors designed for near future. Most of the reactors will be based on the large amount of performance experience from existing reactors. The researchers declare that new technologies should be adopted to new plants. In addition, this is the way of improving the safety of the future nuclear power plants. Because, the current nuclear technologies are not able to meet this challenge. This is the result of shortcomings in the technology that the next generation nuclear power technology must overcome.

The efforts in the next generation nuclear technology has showed that there is a big area for advanced technological developments in the nuclear energy field.

For any next generation nuclear technology to be viable, it must meet certain basic requirements: safety, economics, utility system compatibility and continuing technological change.

Nuclear power plants of the next generation must not only be safe, they must be demonstrably safe.

To reach this safety level for LWR is almost impossible. If it were possible, it would not be demonstrable. This consideration forces the choice of "inherently safe" reactor configurations for next generation technology. These are such designs developing no credible accident. Therefore, the next generation nuclear technology is necessary. As the next generation nuclear power plants, fast breeder reactors have been under development in Europe for more than 30 years and have progressed through several stages. Following theoretical and technological studies, some experimental reactors have been built in various European countries.

The Japanese effort is also about fast breeder reactors and specially JAERI has made R&D for metallic fuels. In addition, they also focus on the conceptual study on actinide burner reactors.

In USA, there are also studies for next generation nuclear technology under the title of LMFBR program. And some demonstration plants already exist. In that sense, the importance of the fast reactor idea is increasing more and more.

Reactors are classified as thermal or fast according to their neutron energy spectrum. The fission is possible in fertile nuclide if the neutron energy is above a threshold value. In fast reactors, light weight materials are eliminated as 10-20% enriched fuel is used. Thus neutrons cause sufficient fissions in the high energy range before they slow down.

The presence of fertile materials in the core has an effect of decreasing multiplication factor. However, they also cause to create new fissionable materials. So, new fuel is created converting the fertile nuclei to fissile nuclei through neutron capture and radioactive decay. This means that it is possible to produce more Pu-239 than is burned, which is called to "breed" new fuel. The idea behind breeder idea is this fact.

The number of fissionable nuclei produced for each fissile Pu-239 nuclei destroyed in plutonium loaded fast reactor is called “conversion ratio” ($CR_{max} = \eta - 1$), where η is the average number of fast neutron which are released per neutron absorbed in the fuel.

Fast reactors are fuelled with Pu and/or U-235 and reflector material is fertile U-238 which is called “blanket”. Around the core, there is a blanket of depleted uranium oxide pins serving as reflectors. These reflectors serve to enhance the breeding properties.

Both sodium and helium have been proposed as fast reactor coolants. The core designs have higher power densities for both coolants. The cores of fast reactor are smaller than thermal reactor’s cores.

Multigroup diffusion equations should be used in fast reactor calculations due to fast neutron spectrum.

The principal differences of fast reactors and thermal reactors are listed in Table 3.3. Fuel in the core of LMFBR is operated at a specific power over three times that of the LWR. During the cooling period, the specific power of LMFBR core fuel from radioactive decay remains about three times that LWR fuel cooled for the same time. This makes shipping, handling, and storage of LMFBR fuel prior to reprocessing much more difficult than LWR fuel. Because of the high plutonium content of spent fuel from the LMFBR, there is a strong economic incentive to return this plutonium to reactor with minimum delay for cooling, reprocessing and fabrication.

One of the common used fuel refabrication and reprocessing method is the purex process. The irradiated fuels are reprocessed by means of solvent extraction. It depends on the principle of the diffusion that provides the transfer of a solid from an aqueous phase to an another aqueous phase.

The IFR idea has a number of specific technical advantages satisfying the design goals of advanced LMRs. These areas are fuel performance, inherent passive safety, economics, waste management, operability and reliability, as a system for actinide recycles program.

Metallic fuel is the key to the IFR concept. Metallic fuel has an excellent steady state performance and high burn-up potential characteristics. The metallic fuelled cores also have a higher breeding ratio associated with a hardened spectrum. Metallic fuel has a much higher breeding ratio than oxide or carbide fuels.

The metallic fuel is easily fabricated, requires no finishing, and allows a relaxed specification that further simplifies the fabrication process. The easy fabrication combined with simple, compact pyrometallurgical reprocessing promises a major cost reduction in the LMR reactor fuel cycle. Fission product removal in this process is 10 to 100 times less than in aqueous processes. Blanket and driver elements undergo separate electrorefining operations where the cladding and fission products are separated in the salt phase and the uranium and plutonium are deposited in the metallic form on cathodes. The fuel cycle can be operated in a closed, fissile self sufficient mode wherein the reactor breeding just compensates for reprocessing refabrication losses. The pyroprocessing system leads to much simpler waste treatments and lesser waste. Long lived fission products in purex are no more problem in the pyroprocess by means of immobilize.

The core reactivity shutdown and decay heat removal without relying on devices requiring switches and outside sourses of power is the main purpose of the inherent safety feature. Inherently control of the reactivity in the case of ab-normal events without reliance on control rod scram can be approached by core design for favourable ratios of the power, power to flow, and the inlet temperature coefficients of reactivity and by design for a high internal conversion ratio.

The IFR concept provides a large reduction in fuel cycle cost compared to the conventional purex-based oxide fuel cycle. For the IFR fuel cycle, capital fixed charges, operating maintenance costs, drive and blanket supplies, fissile carrying charges and waste disposal fees are most important contributions. In addition to these contributions, IFR concept has a very important connection to actinide recycle program for ALMRs regarding waste management and cost.

In the actinide recycle system, the longer lived transuranic (TRU) isotopes are converted to shorter lived products by means of transmutation and fission. The basic system of this process is pyroprocess. Recently, the IFR research program is getting nearer to completion and is in the early stages of developing LWR actinide recovery for recycle in the ALMR.

The entire system including the waste process must be considered and feasibility of each element should be demonstrated right from the start.

The recent properties of the IFR lead to ease of operation, reliability in many components and the systems. The plant capacity can also be improved by means of lowering the maintenance requirement. This maintenance requirement easiness can be achieved with the fact that sodium is noncorrosive to the components in the reactor.

The IFR technology development effort provides opportunities for active participation by the academic studies. First of all, the IFR is a next generation advanced reactor concept. Now is the right time to pursue fundamental research and development to advance its technology base. The IFR incorporates many new technologies that are different from conventional nuclear technologies; metallic fuel, reprocessing based on electrorefining, new waste forms etc,. New emerging technologies and further innovations in the reactor design and the fuel cycle closure can be used.

This study contains the introductory information and recent developments in IFR concept. I hope, this work would be very useful to the researchers of ITU Institute For Nuclear Energy.

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