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Nuclear Reactor Hazards

Ongoing Dangers of Operating Nuclear Technology in the 21st Century

Report Prepared for GREENPEACE International

April 2005

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Ongoing Dangers of Operating Nuclear Technology in the 21st Century

Report Prepared for GREENPEACE International

by Helmut Hirsch, Oda Becker,
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April 2005

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Executive summary

This report gives a comprehensive assessment of the hazards of operational reactors, new 'evolutionary' designs and future reactor concepts. It also addresses the risks associated with the management of spent nuclear fuel. The first part of the report describes the characteristics and inherent flaws of the main reactor designs in operation today; the second part assesses the risks associated to new designs; the third part the 'ageing' of operational reactors; the fourth part the terrorist threat to nuclear power and the fifth and final part the risks associated with climate change impacts – such as flooding – on nuclear power.

The main conclusions are:

- All operational reactors have very serious inherent safety flaws which cannot be eliminated by safety upgrading;
- A major accident in a light-water reactor – the large majority of the reactors – can lead to radioactive releases equivalent to several times the release at Chernobyl and about 1000 times that released by a fission weapon. Relocation of the population can become necessary for large areas (up to 100.000 km^2). The number of cancer deaths could exceed 1 million;
- New reactor lines are envisaged which are heralded as fundamentally safe. However, apart from having their own specific safety problems, those new reactors would require enormous sums for their development, with uncertain outcome;
- The average age of the world's reactors is 21 years and many countries are planning to extend the lifetime of their reactors beyond the original design lifetime. This leads to the degradation of critical components and the increase of severe incidents. The age-related degradation mechanisms are not well understood and difficult to predict;
- De-regulation (liberalisation) of electricity markets has pushed nuclear utilities to decrease safety-related investments and limit staff. Utilities are also upgrading their reactors by increasing reactor pressure and operational temperature and the burn-up of the fuel. This accelerates ageing and decreases safety margins. Nuclear regulators are not always able to fully cope with this new regime;
- Highly radioactive spent fuel mostly is stored employing active cooling. If this fails, this could lead to a major release of radioactivity, far more important than the 1986 Chernobyl accident;
- Reactors cannot be sufficiently protected against a terrorist threat. There are several scenario's – aside from a crash of an airliner on the reactor building – which could lead to a major accident;
- Climate change impacts, such as flooding, sea level rises and extreme droughts, seriously increase nuclear risks.

Commercial Reactor Types and Their Shortcomings

At the start of 2005 there were 441 nuclear power reactors, operating in 31 countries. The age, size and design type of all of these reactors vary considerably.

The most prevalent design in operation is the **Pressurised Water Reactors** (PWR), with 215 in operation around the world. The PWR design was originally conceived to propel military submarines. Therefore, the reactors are, relative to other designs, small, but with a high-energy

output. Consequently, the cooling water in the reactor's primary circuit is at a higher temperature and pressure than other comparable reactor designs. These factors can accelerate the corrosion of components; in particular, the steam generators now frequently have to be replaced.

Similarly, extensive documentation now exists on the problems of cracking in the vessel head penetrations. This cap at the top of the reactor pressure vessel contains the pipes that allow the control rods to be inserted into the reactor core, to control the chain reaction. In the early 1990s cracks began to appear in the reactor vessel heads of some reactors in France.

Worldwide investigations were carried out and similar problems were found in reactors in France, Sweden, Switzerland and the U.S. The most serious example discovered to date occurred at the Davis Besse reactor in Ohio, USA. In this case the cracking had been allowed to continue unchecked for around a decade, despite routine checks and when discovered the crack had penetrated through the 160 mm thick pressure vessel with only the 5 mm steel lining of the vessel -which was bulging from the pressure- stopping a breach of the primary cooling system, the most important safety barrier.

Of all commercial reactor types, the PWR has accumulated the largest number of reactor-years in operating experience. It is remarkable that despite this, this reactor type can still experience new and completely unexpected problems. A striking example is the risk of sump strainer clogging that was not recognized until 2000.

Of similar design and history to the PWR is the Russian **VVER** reactor. There are currently 53 of these reactors deployed in seven countries in Eastern Europe in three main reactor designs. The oldest, VVER 440-230, has significant and serious design flaws and consequently, the G8 and EU believe that they cannot economically be brought up to an acceptable safety standard. The lack of a secondary containment system and adequate emergency core cooling system are of particular concern.

The second generation of VVERs, the 440-213s, has introduced a more effective emergency core cooling system but does not deploy a full secondary containment system

A third design of VVER, the 1000-320s, introduced further design changes but despite this, the reactors are not considered as safe as contemporary PWRs. In fact, following unification of Germany, VVERs of all generation were closed or construction abandoned. Both safety and economic considerations were given for these decisions, with safety concerns more heavily weighted.

The second most prevalent reactor design is the **Boiling Water Reactor** (BWR) (there are 90 in operation around the world), which was developed from the PWR. The modifications were undertaken to increase the simplicity of the design and create higher thermal efficiency by using a single circuit and generating steam within the reactor core. However, this modification has failed to improve safety. The result is a reactor that still exhibits most of the hazardous features of the PWR, while introducing a number of new problems.

BWRs have high power density in the core as well as high pressure and temperature in their cooling circuit, although all of these parameters are somewhat lower than in a PWR. Furthermore, the plumbing of the emergency core cooling system is much more complex in a BWR and the control rod injection comes from underneath the pressure vessel. Thus, emergency shutdown cannot depend on gravity, as is the case in PWRs, necessitating additional active safety systems.

Significant corrosion problems have been observed in many BWRs. In the early nineties, a vast amount of cracking has been detected in a number of German BWRs, in piping of a material that was regarded as resistant to so-called stress corrosion cracking.

There is another persisting problem in BWRs occurred in 2001. Pipes ruptured at Hamaoka-1 (Japan) and at Brunsbüttel (Germany). The cause in both cases was an explosion of a mixture of hydrogen and oxygen, which was produced by hydrolysis in the coolant water. If an oxyhydrogen explosion damages crucial components of the reactor's control and protection system and/or the containment envelope, a severe accident with catastrophic radioactive releases (comparable to those at the Chernobyl accident) will develop.

The next most prevalent reactor currently deployed is the **Pressurised Heavy Water Reactor**, of which there are 39 currently in operation in seven countries. The main design is the Canadian CANDU reactor, which is fuelled by natural uranium and is heavy water cooled and moderated. The reactor's primary containment surrounds the 390 individual pressure tubes. The reactor design has some inherent design flaws, most notably that it suffers from positive void coefficient, whereby should the reactor loose coolant the level of reactivity increases. Secondly, the use of natural uranium significantly increases the volume of uranium in the core, which can lead to instabilities. The pressure tubes that contain the uranium tubes are subject to significant neutron bombardment. Experience in Canada has shown that they subsequently degrade and that expensive repair programmes have had to be undertaken, in some cases after only twenty years of operation.

These and other operational problems have caused huge safety and economic problems for the CANDU fleet. In June 1990, six reactors of the top ten in world lifetime performance were CANDU reactors, four of those from Ontario Hydro. Within six years, load factors dropped drastically due to what a technical journal called a "maintenance meltdown". Furthermore, the operation of eight of Ontario Hydro's CANDU reactors was suspended or indefinitely deferred in the late 90s – although some have now restarted.

The other design serialised in Russia was the **RBMK** reactor, which is a graphite moderated boiling water reactor and used at the Chernobyl station in Ukraine, which was the site of the world's worst civilian nuclear power accident in 1986. The reactor exhibits some of the same design problems of the CANDU, namely positive void coefficient and core instability but has a series of additional problems that exacerbate these problems - in particular, the large number of pressure tubes (1693 in the RBMK 1000s).

Some of the design problems of the RBMK have been rectified as a result of the experiences learned from Chernobyl and this has led to an increase in uranium enrichment and a change in the control rods. However, for technical or economic reasons other problems remain. For example, only two of the remaining twelve reactors have installed fully independent and diverse second shut-down systems, and hence the remaining ten do not conform to IAEA safety requirements.

RBMK reactors also contain more zirconium alloy in the core than any other reactor type (about 50 % more than a conventional BWR). They also contain a large amount of graphite (about 1,700 tonnes). A graphite fire can seriously aggravate an accident situation - it can also react violently with water at higher temperatures, producing explosive hydrogen.

Failure of a single pressure tube in an RBMK does not necessarily lead to catastrophic consequences. However, the large number of tubes and pipes necessitates a similarly large number of welds, and constitutes a system that is difficult to inspect and to maintain. The pressure suppression capacity of the containment system of RBMKs has been improved so that simultaneous rupture of up to nine pressure tubes can be controlled. However, in the case of flow blockage after a loss-of-coolant accident, sufficiently high temperatures could be reached that would lead to ruptures in up to 40 channels. A catastrophic destruction of the whole reactor core could follow.

The fundamental design flaws of these reactors have lead to the international community classifying these reactors as 'non-upgradable' and seeking their closure. Closure has occurred or will occur in Lithuania and Ukraine, but despite this, in Russia, efforts are underway to extend the lives of these reactors rather than retire them early.

The United Kingdom has developed from the plutonium production reactors two designs - the **Magnox** - air-cooled, graphite-moderated natural uranium reactor – and subsequently - the **Advanced Gas Reactor** (AGR). Magnox reactors have very low power density and consequently large cores. In an attempt to overcome this perceived weakness, power density was increased by a factor of two in the AGR, but it is still low compared to light water reactors. Carbon dioxide gas circulates in the primary circuit. Gas circulation is more complex in AGRs as the higher temperature requires a special gas flow through the graphite moderator.

In both designs, the reactor core is located inside a large pressure vessel. The Magnox reactors with older steel pressure vessels have suffered from corrosion. These problems are aggravated by thermal ageing and material degradation caused by neutron-induced embrittlement.

Brittle failure of the pressure vessel could lead to total loss of the primary coolant, and possibly large radioactive releases. For this and other reasons, a number of Magnox stations have already been shut down.

Neither Magnox nor AGRs reactors have a secondary containment. Both reactor types have a high potential for large radioactive releases. The old Magnox reactor fleet must be regarded as particularly hazardous due to these many safety deficiencies.

In addition to the different inherent problems of the various reactor designs, operational internal and external factors may conspire to further reduce the safety margins. These factors include:

Ageing:

There is general consensus that the extension of the life of reactors is of the foremost importance today for the nuclear industry. The International Energy Agency pointedly sums it up as follows: *"If there are no changes in policy towards nuclear power, plant lifetime is the single most important determinant of nuclear electricity production in the coming decade."*

Across the world over the last two decades there has been a general trend against ordering new reactors. As a consequence, the average age of nuclear reactors around the world has increased year on year and is now 21.

At the time of their construction it was assumed that these reactors would not operate beyond 40 years. However, in order to maximise profits, lifetime extension offers an attractive proposition for the nuclear operators.

Ageing processes are difficult to detect because they usually occur on the microscopic level of the inner structure of materials. They frequently become apparent only after a component failure, for example when pipe breakages have occurred.

The consequences of ageing can roughly be described as two-fold. Firstly, the number of incidents and reportable events at an NPP will increase – small leakages, cracks, short-circuits due to cable failure etc. Secondly, the aging process is leading to the gradual weakening of materials that could lead to catastrophic failures of components with subsequent severe radioactive releases. Most notable among these is the embrittlement of the reactor pressure vessel, which increases the risk of the vessel bursting. Failure of the pressure vessel of a PWR or a BWR constitutes an accident beyond the design basis for which there is no safety system - inevitably leading to a catastrophic release of radioactive material to the environment. As the world's nuclear power plants get older, there are efforts to play down the role of ageing. Those efforts include conveniently narrowing the definition of ageing. Furthermore, the most basic and

severe shortcoming of international regulatory norms resides in the fact that no country has a comprehensive set of technical criteria for deciding when further operation of a nuclear power plant is no longer permitted. As a consequence reactors are being allowed to operate longer.

It is clear that the risk of a nuclear accident grows significantly each year, once a nuclear power plant has been in operation for about two decades.

Terrorist Threats to:

Nuclear Power Plants:

Even before the attacks in New York and Washington in 2001, concerns had been raised over the risk of nuclear facilities from terrorist attacks. Nuclear facilities have been targeted in the past leading to their destruction – such as the attack by Israel on the Osirak reactor in Iraq. The threats to nuclear power plants from terrorist attacks and acts of war can be summarized as follows:

- Because of their importance for the electricity supply system, the severe consequences of radioactive releases as well as because of their symbolic character, nuclear power plants are "attractive" targets for terrorist as well as for military attacks.
- An attack on a nuclear power plant can lead to radioactive releases equivalent to several times the release at Chernobyl. Relocation of the population can become necessary for large areas (up to 100.000 km^2). The number of cancer deaths could exceed 1 million.
- Nuclear power plants could be targets in case of war if a military use is suspected.
- The spectrum of possible modes of attack is very diverse. Attacks could be performed by air, on the ground and from the water. Different means/weapons can be used.
- Protective measures against terror attacks are of very limited use. Furthermore, a number of conceivable measures cannot be implemented in a democratic society.

Reprocessing Plants and Spent Fuel Storage Areas.

The amount of plutonium in storage is steadily increasing. While the US and Russia agreed to dispose each of 34t of "excess" weapons grade plutonium, the world's "civil" plutonium stockpile exceeds 230t. As of the end of 2002, the largest holder of plutonium is the UK with over 90t followed by France with 80t and Russia with over 37t. Plutonium has two particular characteristics; it is of high strategic value as primary weapon ingredient and it is highly radiotoxic. A few kilograms are sufficient in order to manufacture a simple nuclear weapon with only a few micrograms inhaled sufficient to develop cancer.

Unfortunately, none of the buildings at Sellafield or La Hague have been designed to withstand extreme impacts, for example by fully-fuelled large-capacity aircraft or ballistic missiles. The probability that they would resist such impact is limited. The worst release mechanism for plutonium, usually stored in oxide form, is a large fire that would render plutonium particles airborne in micron sizes that are easily inhalable.

Spent nuclear fuel and central radioactive waste storage facilities contain by far the largest inventories of radioactive substances of any facility throughout the nuclear fuel chain. Spent nuclear fuel in cooling pools as well as unconditioned high level radioactive wastes in liquid and sludge form, are particularly vulnerable to attack. The main reason for this is that they are present in readily dispersible form in storage facilities that are not designed to withstand a large aircraft crash or an attack with heavy weapons. Storage facilities at reprocessing plants contain

several hundred of times more than the radioactive inventory that was released as a consequence of the Chernobyl disaster.

Cask Storage Facilities:

Like other forms of storage, spent fuel in casks are vulnerable to terrorist attacks. Resultant radioactive releases are likely to be smaller than those that would result from attacks on storage pools. On the other hand, accessibility to casks appears to be greater than of spent fuel pools located in massive buildings. Improvements to the storage concept are conceivable. However, they are only likely to have a chance of being implemented if the inventories are not too large.

Nuclear Transports:

Terrorist attacks against the transportation of radioactive material can occur almost anywhere in any industrialised country. Since each shipment is unlikely to exceed several tonnes, the expected release will be smaller by orders of magnitude than those which would result from an attack on a storage facility – even if the transport containers are severely damaged. On the other hand, the place where the release occurs cannot be foreseen, as attacks can occur, in principle, anywhere along urban transportation routes such as rail or in ports.

***Climate Change and Nuclear Technology**:*

Global climate change is a reality. There is a broad consensus among scientists regarding this issue. The global average surface temperature has increased by 0.6 ±0.2°C since the late 19th century. The results of research performed by climate scientists indicate that even slight changes in temperature have a tremendous impact on the corresponding number of extreme weather events. More intense precipitations as well as storms will occur more frequently, which can and have impacted upon the operation of nuclear facilities and in particular nuclear power plants. About 700 natural hazardous events were registered globally in 2003. 300 of these events were storms and severe weather events, and about 200 were major flood events.

These unusual severe weather events impact upon nuclear power operation by causing flooding or draughts affecting the cooling or other safety systems. In addition, storms can directly impact upon nuclear operation or indirectly, by damaging the electricity grids. Heavy storms can lead to multiple damage of the transmission lines, and hence to loss of off-site power. Every nuclear power plant has emergency power supplies, which are often diesel-driven. However, emergency power systems with diesel generator are notoriously trouble-prone. If the emergency diesel generators fail, the situation at the plant becomes critical ("station blackout"). A station blackout at a nuclear power station is a major contributor to severe core damage frequency. Without electricity the operator looses instrumentation and control power leading to an inability to cool the reactor core. A natural disaster that disables the incoming power lines to a nuclear power station coupled with the failure of on-site emergency generators can result in severe accident.

Regulations and practices governing these precautions still reflect the conditions of the 1980s and are not appropriate for the present situation of increasing hazards to the electricity grid due to climate change as well as due to the liberalization of the electricity markets and the increased threat of terrorist attacks.

New Reactor Designs

While there are only about 25 reactors under construction around the world – some of which may never be completed, further development of the technology continues, which is distinguished in two categories.

Generation III

Throughout the world there are around 20 different concepts for the next generation of reactor design, known as Generation III. Most of them are "evolutionary" designs that have been developed from Generation II (i.e. current) reactor types with some modifications, but without introducing drastic changes. Some of them represent more innovative approaches. However, only in Japan are there any commercial scale reactors of Generation III in operation - the Advanced Boiling Water Reactors (ABWR). The next most advanced design is the European Pressurised Water Reactor (EPR), which is being built in Finland and may be also sited in France.

These reactors tend to be modified version of existing reactors, in the case of the EPR, it is simply a later version of current reactor designs – the French N4 reactor and Germany Konvoi- with some improvements, but also with reductions in safety margins and fewer redundancies for some safety systems.

Generation IV

Under the leadership of the United States the "Generation IV International Forum" (GIF) has been established in 2000. Currently, there are six reactor designs being considered, including; Gas-Cooled Fast Reactor System; Lead-Cooled Fast Reactor System; Molten Salt Reactor System; Supercritical-Water-Cooled Reactor System; Sodium-Cooled Fast Reactor System; Very-High-Temperature Reactor System. However, it is unclear what design of reactor will be promoted, what is the most appropriate size, should there be an open or closed fuel cycle or what is the target date for commercialisation. The basic concepts of the "new generation" have been around as long as nuclear power, but they were forced out of the market in the early years by the light water reactors – not without reason, considering the experiences so far, which are dominated by technical and economic problems, and safety deficits. In order to overcome these problems, materials, processes and operating regimes that are significantly different from those of currently operating systems or previous systems have to be developed. Research and development are needed to confirm the viability and safety of new design approaches.

Each of these reactors has large variations and relative advantages and disadvantages over one another. However, currently, they are only paper designs and the expected best case for commercialisation is 2045.

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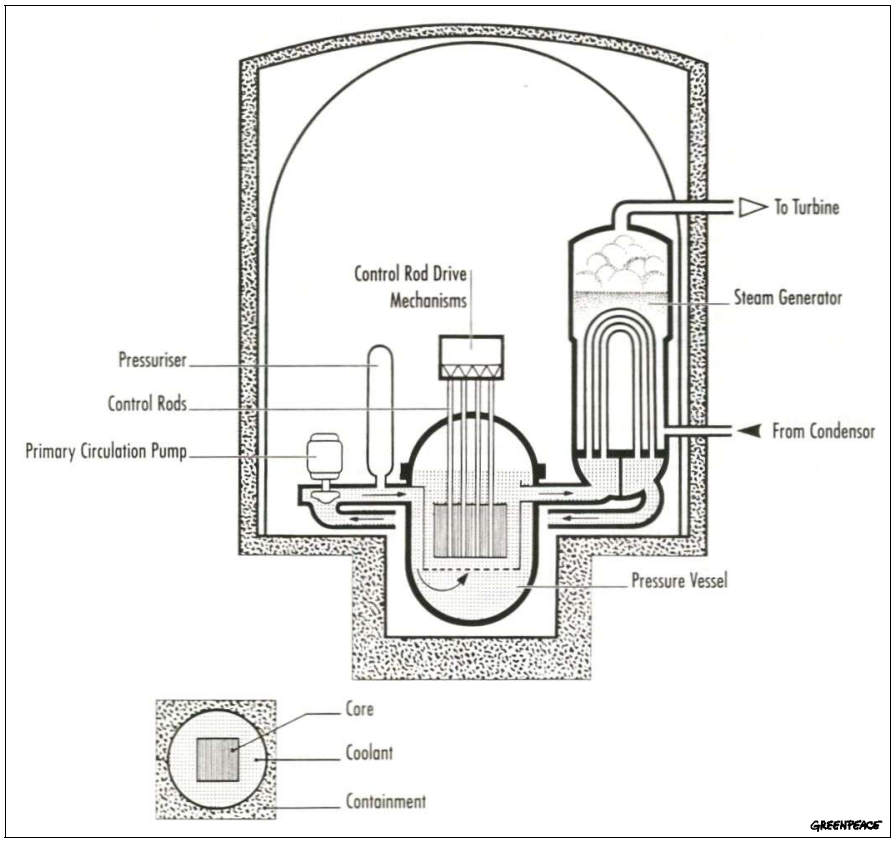
A. Commercial Reactor Types and Their Shortcomings

Introduction

At the start of 2005 there were 441 nuclear power reactors, operating in 31 countries. The age, size and design type of all of these reactors vary considerably. Some of these are still the 'first' generation of nuclear reactors designs, built in the 1950s and 60's directly from military uses – plutonium production or submarine propulsion. However, most of the reactors are the second generation of design, developed in the 1970s and beyond. The majority of these are Light Water Reactors (LWR). They fall in three main categories, the Pressurized Water Reactor (PWR) of which there are 215 in operation, the Boiling Water Reactors (BWR), 90 in operation and the Soviet-designed Pressurized Water Reactors (VVER), 53 in operation.

Pressurized Water Reactors (PWR)



The pressurized water reactor (PWR) was developed from the reactors used to propel submarines. However, PWRs use low enriched uranium whereas submarine reactors use high-enriched uranium fuel. Despite this significant difference, the PWR still exhibits the basic properties of a military reactor, optimized to give high power output while taking up as little volume as possible.

Pressurized water reactors are the most common commercial reactor type, by far: About 50 % of the world's operating nuclear power plants are PWRs.

PWRs have the highest core power density of all reactor types currently in general use. Their primary circuit is characterized by high pressure and high temperature. Chemically reactive zirconium alloy is used as fuel cladding in the core, although in smaller quantities than, for example, in BWRs or RBMKs. When zirconium reacts with steam, hydrogen is produced,

leading to the danger of a hydrogen explosion in the course of an accident. The core is located inside a steel reactor pressure vessel, the integrity of which is crucial to safety. Pressure vessel steel embrittlement under neutron bombardment is a known phenomenon (especially in older plants with a high copper or nickel content in the welds). Its effect on the behaviour of the vessel under high stress, however, is still neither entirely known nor predictable.

Flawless vessel walls and, in particular, flawless welds are rarely achieved in practice. Therefore, manufacturers and utilities have to apply stringent control procedures. In-service inspection for internal flaws is undertaken using ultrasonic and eddy current techniques. Experience shows, however, that these techniques are limited in accuracy and reliability. In cases of high embrittlement, critical crack sizes can be close or even below the resolution limit of these techniques. (This can also constitute a problem for other parts of the reactor like the main coolant pipes, where embrittlement plays no role but corrosion and erosion mechanisms endanger the integrity of the materials.)

In spite of the limitations of inspection techniques, official safety philosophy assumes that the reactor vessel will and cannot burst, or that the probability for vessel burst is so low it can be safely neglected (vessel embrittlement is also discussed in section C).

The steam generators, the link between the primary and secondary circuits, are a notoriously weak point in many PWRs. Damage occurs frequently, up to rupture of generator tubes. Leakages provide a pathway for radioactive releases outside the containment; they also require action to prevent a severe accident.

Steam generator failures like the rupture occurring at Mihama-2 power station in Japan in February 1991 have been traced back to a variety of causes, such as manufacturing defects or corrosion, as well as installation faults leading to vibrations and fretting. In-service inspection of the often more then 10,000 individual tubes in a single steam generator is very difficult and potentially hazardous flaws may remain undetected.

Due to the high power density and the correspondingly high density of decay heat generation after shutdown, PWRs depend heavily on a large array of complicated, active safety systems. These systems have to function fast and reliably. Active systems depend on a continuous electricity supply. Emergency power supplies of nuclear power plants therefore must be considerably more reliable than in other industrial plants. Experience indicates that this has not been achieved. This issue is very important in case of natural hazards (see section D.2).

Safety systems are usually redundant (i.e. more components are provided for a task than needed). Failures of individual components therefore do not necessarily lead to a catastrophic accident. However, redundancy becomes useless if a so-called ‘common cause-failure’ disables all parallel trains of a safety system. For example, the emergency core cooling system of modern German PWRs consists of four parallel trains with four storage tanks for borated water. At the Philippsburg-2 plant, it was discovered in 2001 that all four tanks were not filled to the required level during start-up of the plant – hence, the emergency core cooling system had not been fully operational. Investigations showed that this irregularity had persisted despite 16 yearly revisions, with one exception. Additionally, in three of the four tanks boron concentration was below the required value. In the same year, similar problems were reported at two other German PWRs [BMU 2001]. In the case of the storage tank problem at German PWRs, the reason was poor safety culture – in other words, carelessness of the plant personnel, who apparently had been taking shortcuts wherever possible.

There is a continuous trend worldwide towards increasing automation in nuclear power plants, which potentially can reduce the hazards of human error. On the other hand, increasing reliance on software creates its own problems. The interface between machine and human is notoriously

prone to errors. Furthermore, automation can be seriously disadvantageous in cases of accidents with unforeseen developments, if it impedes improvised measures, which experienced personnel might attempt. In addition, it actually prevents the acquisition of practical knowledge and experience by the personnel, which would be needed in such critical situations.

Due to its negative void coefficient –whereby when reactor power increases and the water moderator starts boiling, power is reduced again -, a PWR will become sub-critical when heating up and left to itself. However, a considerable amount of energy would be released before such an "inherent" shutdown. Therefore, rapid and reliable control rod injection (reactor scram) is required to achieve sub-criticality fast. The scram system, however, is also sensitive to common cause-failures.

The primary circuit of the reactor, plus some auxiliary components, are usually located within a steel or concrete containment designed to withstand internal pressure building up during anticipated accidents (design basis accidents- DBAs). However, this containment is penetrated in many places. If isolation fails, radioactivity will be released even in the case of DBAs. To prevent containment failure during more severe accidents, filtered venting systems have been installed in many PWR plants. Nevertheless, accidents with early destruction of the containment are possible which render the venting system useless – for example, reactor pressure vessel burst, steam explosion, or ejection of molten core material out of the reactor vessel at high pressure.

Hydrogen explosions constitute another mechanism for early containment failure. In the last decade, passive autocatalytic recombiners have been installed in many PWRs world-wide, which recombine hydrogen with oxygen at concentrations below the flammability limit, without requiring external power supply. As a result of the heat they produce, they also increase convection in the containment, which leads to better mixing of the atmosphere and can prevent high local hydrogen concentrations. Thus, the hydrogen hazard has been somewhat reduced, at least for large, dry containments (i.e. the type of containment of most PWRs).

A considerable hazard is presented by the spent fuel storage pool, which is located in or near the reactor building, in some cases inside the containment. Reactor accidents can be the trigger of fuel pool accidents and vice versa, leading to increased radioactive releases. Furthermore, in two-unit plants, sometimes with interlocking systems, an accident in one reactor can affect the safety of the other.

The radioactive releases associated with severe accidents in a PWR can be very high, comparable to or even higher than the releases from the Chernobyl accident. Up to 90 % of the caesium inventory of the reactor core may be released [HAHN 1999]. For a reactor with a power of 1300 MWe and high burn-up fuel, this corresponds to about 350.000 Tera-Becquerel of caesium-137. The release of the same nuclide in Chernobyl was about 85.000 Tera-Becquerel. The difference stems mainly from the fact that in Chernobyl, "only" 20 – 40 % of the core inventory was released; furthermore, due to lower burn-up compared to a PWR, the inventory was somewhat lower.

The design of PWRs varies considerably in different countries. However, there is no straightforward way to judge which design is "better" or "worse". For instance, the individual loops of the emergency core cooling system are less interlocked in the German PWRs than in the US, French, Soviet or Japanese types. This reduces the risk of failure propagation through the system. On the other hand, it gives less flexibility in switching over components from one loop to another.

The levels of diversity (provision of different systems to perform the same task and redundancy -multiple provision of systems for one task-) also vary between countries. US reactors seem to have more diversity in their emergency core cooling systems. Regarding redundancy, the

German principle of 4x50% capacity compares favourably to the 2x100% found in some other PWRs (for example, in France), since it allows for failure of one loop with simultaneous repair of another one. Design pressures of the emergency core cooling systems also vary considerably.

Regarding the containment, many US plants have a significantly lower design pressure than the average plant worldwide. A small number of US and Japanese PWRs have a particularly problematic ice condenser containment. This type of containment is equipped with a pressure suppression system consisting of large baskets of ice that serve to condense steam in the event of coolant loss from the primary circuit. It is notable for a particularly low design pressure and small containment volume.

Safety problems of PWRs are exacerbated by increasing fuel burn-up, i.e. by increasing the energy gained per ton of fuel. The trend to raise burn-up has persisted for many years, for economic reasons; it has intensified in the last decade. Whereas originally, 30 or 35 MWd/kg were typical fuel burn-ups, values above 50 MWd/kg are increasingly achieved in many countries. This is accompanied with increases in the enrichment of the fresh fuel.

Higher burn-up leads to higher loads to fuel element hulls. It is to be expected that the failure rate of fuel rods will increase with increasing burn-up. Also, the handling of the spent fuel elements becomes more difficult –see section D.1.ii-.

Of all commercial reactor types, the PWR has accumulated the largest number of reactor-years in operating experience. It is remarkable that nevertheless, this reactor type can still experience new and completely unexpected problems. A striking example is the risk of sump strainer clogging which was not recognized until after the year 2000.

The emergency core cooling system, when activated because of coolant loss from the primary circuit, draws water from the borated water tanks already mentioned. When those tanks have been emptied, intake is switched over to the so-called sump, i.e. the lowest part of the containment where the water leaking during a loss-of-coolant accident collects by virtue of gravity. Thus, a kind of cooling circuit is established.

However, in the course of a loss-of-coolant accident, insulation from pipes might be dislodged and trapped to the containment sump, together with other debris, which might be present. This can lead to clogging of the sump intake of the emergency core cooling system, seriously impairing this system's operation – unless sump strainers are adequately designed.

This problem has been known since 1992, after an incident in a Swedish BWR. Backfits took place in many BWRs worldwide. PWR operators, however, argued that their risk was much lower because of larger sump screens. Only recently, however, experiments in France and the U.S. showed that strainer size was not the only relevant factor and the clogging potential represented a substantial increase in core damage risk.

In early 2005, the problem is still not completely resolved as in many plants; backfits are still under way or in the planning stages [NUCWEEK 03_04].

Because of this problem, the Biblis-A PWR (Germany) was not permitted to resume operation after a brief planned shutdown in mid-April 2003. After long and controversial debates between licensing authorities and plant owner RWE Power, strainers were upgraded and the plant started up again on December 30 of the same year [DATF 2003].

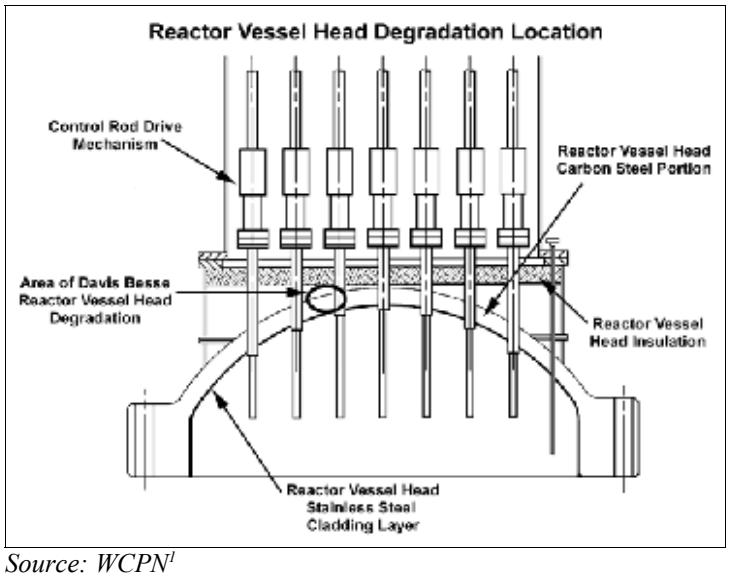
In December 2003, Electricité de France (EdF) became the first operator of a large PWR fleet to plan backfits to all of its reactors on a rapid schedule. Backfits are to begin 2005. Backfits were also being pushed in Belgium.

The U.S.NRC initially was slow to follow up the issue, holding up progress in other countries like Spain or Japan that traditionally follow the NRC's lead. However, efforts gained momentum

during 2004, following a joint workshop of the OECD's Nuclear Energy Agency and the NRC in February. Representatives attended this workshop from most countries operating PWRs worldwide, including VVERs [NEA 2004].

In December 2004, the NRC finally approved an evaluation method for analyzing the sump performance in U.S. PWRs and laid down a timetable for operators to resolve this issue. However, it appears that the issue, in spite of its high significance for risk, is not regarded as particularly urgent. The beginning of corrective actions can be postponed until April 1, 2006, and the conclusions of these actions only have to be completed by December 31st, 2007 [NRC 2004].

Another problem that only emerged after decades of PWR operation and has persisted for over a decade is reactor vessel head penetration cracking. Vessel head penetrations allow the control rods to manoeuvre into the pressure vessel. The rupture of one or several such penetrations could therefore lead to loss-of-coolant combined with a severe reduction of reactor control.



Source: WCPN[1]

Vessel head penetration cracking (VHPC) was first discovered at several French reactors in 1991. Following that discovery, cracks were also found at PWRs in Sweden, Switzerland, the USA and other countries [SCHNEIDER 1993]. This ageing-related problem is still not completely resolved; in the USA, in particular, extensive vessel head replacing is under way and scheduled to be completed by 2007 [NUCWEEK 23_03].

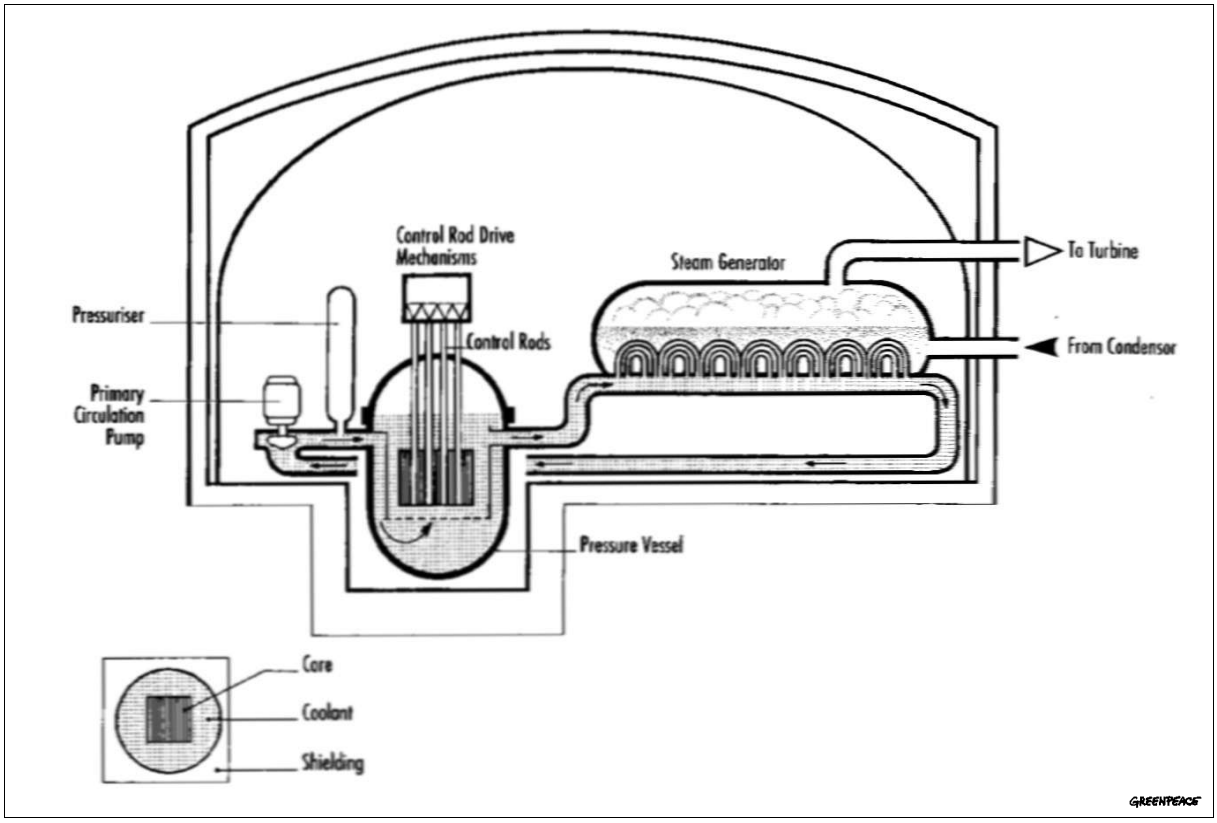
The degradation of the vessel head did not in every case stop at "mere" cracks: During the repair of a cracked nozzle at Davis Besse PWR (USA), extensive corrosive damage to the vessel head was discovered. The case of this "record" damage – only a thin stainless steel liner, which had already started bulging, prevented a very severe loss-of-coolant accident – is treated further in section C.

Soviet-designed Pressurized Water Reactors (VVER)

The VVER (water-water-energy-reactor) reactor line was developed in the Soviet Union. VVERs are pressurized water reactors, and basically the section on PWRs also applies to them. However, VVERs not only have certain design features in common, they are also beset with specific and particularly severe safety deficiencies which deserve separate treatment. In particular, the first generation (VVER-440/230) – along with the RBMKs – has come into focus

[1] http://www.wcpn.org/news/2002/07-09/images/rock_radiation/vessel-head-deg-large.gif

because of the effect on plant safety of their substandard design features. These reactors have been declared as “non upgradeable” or high risk reactors by the European Union and the G7. [G7] They must be closed in all new EU member states.



Originally, those reactors were not equipped with an emergency core cooling system deserving this name, but with a make-up water system of low capacity, designed only for rupture of a pipe with 100 mm inside diameter. As the main coolant pipes’ diameter is 500 mm, far worse leakages are possible.

The reactor does not have a secondary containment system instead a so-called confinement system comprising several sealed and interconnected compartments is present as a barrier against radioactive releases. A high leak rate, low design overpressure and unreliable vent valves venting directly into the atmosphere characterize this confinement system.

The safety of first-generation VVERs is further reduced by deficiencies in redundancy and diversity of equipment, as well as by problems with reactor materials. The pressure vessel is particularly prone to embrittlement.

Significant backfitting has been implemented at VVER 440/230-plants in the 1990s. Most notably, the water system used for emergency core cooling was upgraded and the leak tightness of the confinement system was improved by up to a factor of 100. However, design basis accidents still do not include the break of a main coolant pipe; only pipe breaks up to 200 mm diameter can be controlled. Furthermore, it does not appear feasible to backfit the plants with a containment system that could provide a similar degree of protection as the containments of modern western PWRs [WENRA 2000]. Nevertheless, eight first-generation VVERs are still operating by early 2005: Two units at Bohunice in Slovakia, two at Kozloduy (Bulgaria), two at Kola (Russia) as well as the two prototypes at Novo-Voronezh (Russia).

Second-generation VVERs (440/213) are fitted with an emergency core cooling system designed to cope with the break of a 500 mm main coolant pipe. They have a containment with pressure suppression via a bubble tower system, in principle resembling western BWR containments. Containment leak rates are high compared to western reactors, and the containment system generally is inferior to the full-pressure containments of most western PWRs. The complex behaviour of the water-filled pressure-suppression trays has been tested for design basis accidents and appears to be adequate for those; however, no tests have been performed for severe, beyond-design-basis accidents [NEA 2003]. It is to be feared that safety margins are very small in case of such event.

Another problem of second-generation VVERs is the poor quality of materials (for example, regarding reactor pressure vessel and piping). Also, problems with quality of equipment and the hazards of common-cause failures due to lack of spatial separation of pipes, cables and trains for instrumentation & control remain to some extent in spite of recent backfitting activities. Furthermore, there is an unfavourable arrangement of the turbines relative to the reactor building, leading to the possibility of consequential damage following turbine failure [GRS 1991].

The newer VVER-1000 model is the first Soviet reactor to be fitted with a full-pressure single containment, and there are further improvements regarding the redundancy of safety and control systems. The containment, however, has a basic shortcoming not encountered in western PWRs. The lower containment boundary (containment basemat) is not in contact with the ground, but is located at a higher level inside the reactor building. In case of a severe accident, melt-through can occur within about 48 hours. The containment atmosphere will then blow down into parts of the reactor building that are not leak-tight. High radioactive releases result. Furthermore, the reactor building – including the main and emergency control rooms – will have to be abandoned [FEA 2004; ATPP 2001].

Other safety concerns lie with the quality and reliability of individual equipment, especially with the instrumentation and control systems. The plant layout has weaknesses that make the redundant safety systems vulnerable to hazardous systems interactions and common-cause failures due to fires, internal floods or external hazards [WENRA 2000].

The embrittlement of the reactor pressure vessel also constitutes a potential problem and the data base for the prediction of embrittlement progress over the years is generally inadequate with formulas used for prediction are not necessarily conservative. The problem is particularly severe since welds in VVER-1000 pressure vessels frequently have a rather high content of impurities that accelerate embrittlement, like nickel and manganese. The processes occurring are still not completely understood.

Steam generator integrity is a further issue of concern, in particular regarding the steam generator collector. Up to 1999, cracks have developed on 25 steam generators in operating VVER-1000s. In three cases, damage was found because a leakage of radioactive water into the secondary circuit had already taken place [IAEA 1999]. Improved materials are being used now in the steam generators of some VVER-1000 plants.

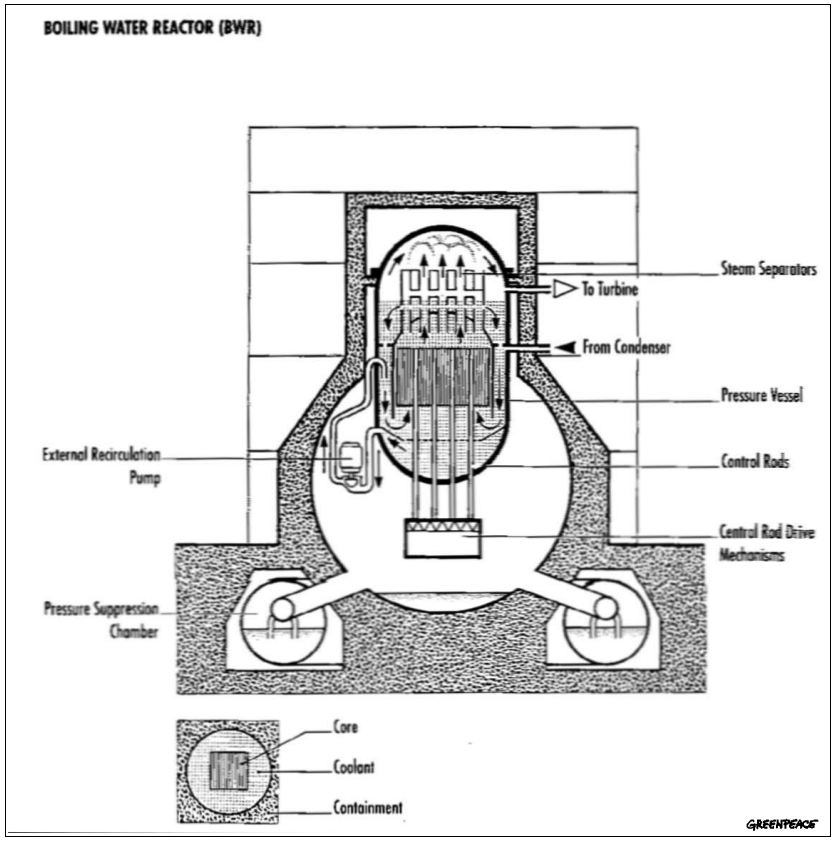
Regarding protection against external events, the level found at VVERs is lower than that corresponding to the best western practice but roughly comparable with that of western plants of the same vintage.

There are extensive debates on the hazards of all generations of VVER plants in comparison to western PWRs. The issue of whether they can reach a western standard by backfitting has been subject of many studies and discussions. However, the simple fact of what happened to the only genuine VVERs that came under the supervision of the authorities and technical support organisations of a Western European state should not be forgotten.

The units of Greifswald-1 to -4 (first generation VVERs) were shut down immediately when Germany was reunified. Greifswald-5, a second-generation VVER that had reached first criticality in early 1989, was decommissioned while still in the start-up phase. And the units Greifswald-6 to -8 as well as Stendal-1 and -2, second- and third-generation VVERs in varying stages of construction were never completed. Both safety and economic considerations were given for these decisions, with safety concerns, however, predominant.

Design deficits, embrittlement and other materials' problems made the need for immediate shutdown obvious for the first generation VVERs. For the other plants, uncertainties as to the success of back fitting measures lead to the conclusion that further investments were not worth the effort. The situation was exacerbated by the less than well-ordered state of plant documentation; a great number of modifications had been implemented during construction and it proved extremely difficult to obtain a clear picture on the up-to-date plant status.

Boiling Water Reactors (BWR)



The boiling water reactor (BWR) was developed from the pressurized water reactor, in an attempt to modify the PWR towards greater simplicity of design and higher thermal efficiency by using a single circuit and by generating steam within the reactor core. However, this modification has failed to improve safety. The result is a reactor that still exhibits most of the hazardous features of the PWR, while introducing a large number of new problems.

BWRs have high power density in the core as well as high pressure and high temperature in their cooling circuit, although all of these parameters are somewhat lower than in a PWR. The uranium inventory in the core is higher than in PWRs. (The water inventory in the cooling circuit is also higher than in PWRs, this can be advantageous in some accident situations). The

amount of chemically reactive zirconium alloy fuel cladding is two to three times that of a PWR. The "primary circuit" of a BWR passes outside the reactor containment. Thus, a leak in this circuit, coupled with failure of isolation valves, offers a direct pathway for uncontrolled releases into the atmosphere.

As in a PWR, the reactor core of a BWR is located in a pressure vessel. The basic problems of the PWR vessel apply here too, but with modifications. Neutron fluxes are considerably lower than in a PWR vessel (by a factor of 10), leading to significantly less embrittlement. On the other hand, the vessel is much larger; longitudinal welds may be required, whereas there are only circumferential welds in a PWR vessel. There is also a much more complicated inner structure, as well as many penetrations at the bottom. Flawless manufacture of such vessels is particularly difficult, their inspection hindered by the penetrations.

Like a PWR, a BWR depends heavily on fast and reliable active safety systems, but the plumbing of the emergency core cooling system is much more complex in a BWR. Control rod injection is from underneath the pressure vessel. Thus, it cannot depend on gravity, as in PWRs, necessitating additional active systems. Regulating the operation of a BWR is generally more complex than in a PWR. Under certain circumstances, the collapse of so-called steam voids in the core can lead to increasing reactivity and thus increasing power during an accident. (BWRs, like PWRs, have a negative void coefficient. Thus, when the reactor heats up and more bubbles form, the chain reaction will become weaker, creating less power. This feature can become hazardous, however, when steam bubbles collapse.)

Old BWRs have an external water recirculation circuit with a pipe inlet below the top of the reactor core. A break in this pipe would lead to a particularly hazardous situation, since the core would rapidly be exposed, as water was lost. Modern BWRs have internal recirculation pumps, avoiding the external circuit but necessitating additional penetrations of the reactor vessel from below.

As opposed to PWRs, the coolant of BWRs generally has comparatively high oxygen content, and significant corrosion problems have been observed in many BWRs. In the early nineties, a vast amount of cracking has been detected in a number of German BWRs, in piping of a material (stabilized austenitic steel) that was regarded as resistant to so-called stress corrosion cracking.

BWR containments exhibit one crucial difference from most PWRs: even for design basis accidents (DBAs), they depend on a pressure suppression system to retain containment integrity. During an accident, the pressure suppression pool would be subject to heavy stresses. As in PWRs, beyond design basis accidents (BDBAs) are possible that could lead to containment destruction, even with a functioning pressure suppression. In older BWR designs, such as the German "Series 69" and the US Mark 1 containment, core melt will almost inevitably lead to a rapid breach of containment, resulting in very high releases of radioactivity.

Containment isolation in the BWR is generally poor and susceptible to failure. Most BWRs now employ containment inertisation (i. e. limiting the amount of free oxygen by introducing inert gases into the containment) with nitrogen to prevent hydrogen explosions that could lead to containment failure in case of accidents. Access to the containment during operation, however, is seriously impeded by this measure. This can be very problematic, as illustrated by the Brunsbuettel event of December 2001 described below.

The basic BWR design varies in different countries. All US BWRs have an external water circuit. Older Swedish BWRs also have external recirculation, but the four more recent plants do not. In Germany, none of the BWRs still operating has external recirculation.

Redundancy in the emergency core cooling system is somewhat higher in German BWRs. On the other hand, US BWRs have higher, though still limited, diversity. Some Swedish power plants have more diversity in their options for emergency power supply.

Most of the European BWRs have systems for filtered containment venting as a precaution against over-pressurization. As in the case of PWRs, there are accident sequences for which venting is useless. In the USA and Japan, venting systems are not considered necessary to prevent containment over-pressurization.

Even after decades of operation of BWRs, safety problems persist which have been known and studied for a long time, and can even get more serious with new fuel types. A typical example is neutron flux oscillations. Such oscillations can occur during (otherwise comparatively harmless) transients and permissible loads to fuel rod cladding may be exceeded if they are not rapidly suppressed, leading to cladding failure.

In the 1980s and early 90s, several such events have been observed in BWRs in Sweden, the USA, Germany and other countries. After a pause, flux oscillations then occurred again at Oskarshamn-2 BWR (Sweden) and Philippsburg-1 BWR (Germany). A new core design aiming at low neutron leakage, new fuel assemblies for higher burn-up and other changes had reduced the margin between the normal operational parameters and the instability region.

At present, so-called in-phase oscillations (when the entire core oscillates in phase) seem to be sufficiently understood to be avoided. However, the measures to control out-of-phase oscillations still need to be developed further. (In this case, parts of the core oscillate in counter-phase to each other.) [GRS 2003]

There is another persisting problem in BWRs that became more prominent in 2001, with firstly pipes ruptured on November 7 at Hamaoka-1 BWR (Japan) and then on December 14 at Brunsbüttel BWR (Germany). The cause in both cases: An explosion of a mixture of hydrogen and oxygen, which was produced by hydrolysis in the coolant water.

Oxyhydrogen is generated continuously during BWR operation. It is present in the cooling circuit and pipes. Normally, the gas is mixed with steam, and the explosive potential is therefore suppressed. However, slight changes of temperature can lead to steam condensation, leading to the formation of oxyhydrogen bubbles.

In the late 1980s, there had already been occasional problems with explosive gases collecting in German BWRs, leading to valve damage. Counter measures had been taken, such as installation of recombiners and temperature monitors for the timely recognition of the cooler zones where oxyhydrogen bubbles might collect. Clearly, however, those measures were not sufficient, or had not been implemented to a sufficient degree at all plants.

After the Brunsbüttel accident, lengthy investigations of the issue took place in Germany. Additional counter measures were implemented at BWR plants. It became clear that there are basic problems in connection with the explosion hazard in BWRs which are not yet fully understood – in particular, concerning the strength of detonations if complex gas mixtures, containing other components in addition to oxyhydrogen, are involved [AMNT 2004].

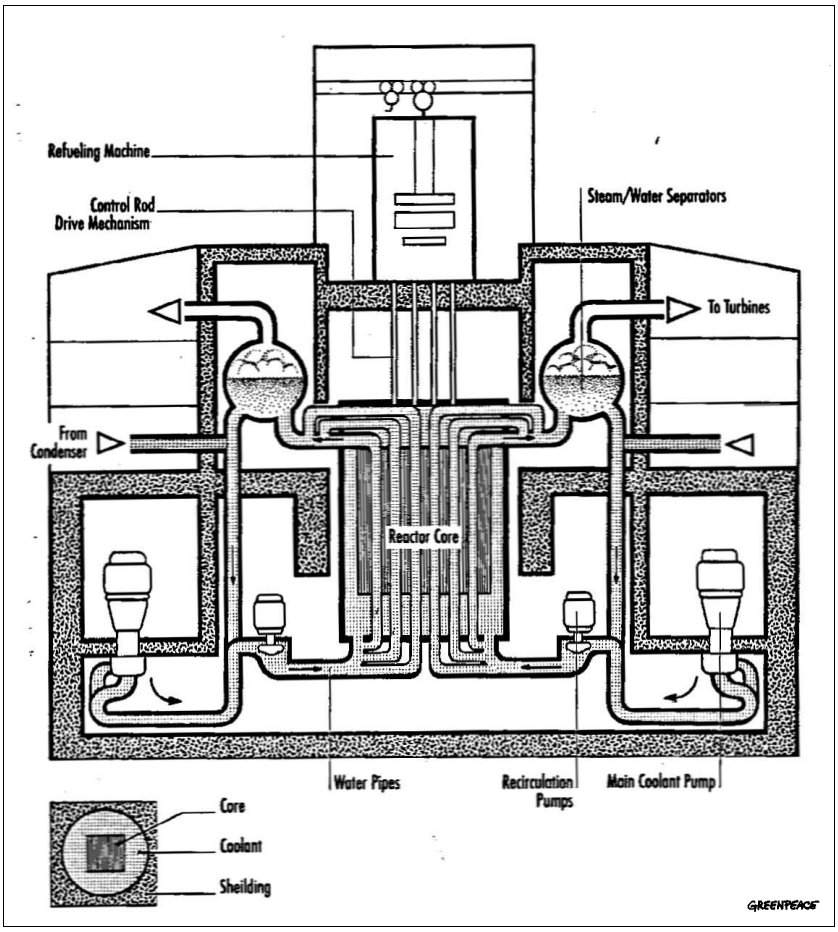
This issue is crucial for the safety of BWRs: A pipe rupture as had occurred at Brunsbüttel can lead to a loss-of-coolant accident, which in principle can be controlled if the safety systems function according to design. However, if an oxyhydrogen explosion also damages crucial components of the reactor's control and protection system and/or the containment envelope, a severe accident with catastrophic radioactive releases (comparable to those at the Chernobyl accident) will develop.

Another problem hit Japanese BWRs in 2002/2003. All 17 BWRs operated by TEPCO had to be shut down when it was revealed that there were defects in core internals. This was linked with very serious deficits in "safety culture": TEPCO staff had routinely covered up inspection findings that should have been reported to regulators immediately. During the investigations, additional problems surfaced – for example, debris items were found in the suppression pools of several units. Replacement of core shrouds was required in some cases [NUCWEEK 02_04; NUCWEEK 48_04].

Several BWRs operated by other utilities suffered from similar problems Chubu Electric's Hamaoka-1 and -2 have already been shut down for several years; core shroud replacement is to performed and expected to be completed by 2007 and mid-2005, respectively [NUCWEEK 48_04].

With TEPCO, the problems also led to long outages and the BWR units were only gradually taken back on line. By April 2004, 8 of the 17 reactors were operating again [NEI 2004]; even by December 2004, over two years after the problem was first discovered, not all units concerned were back on line, with two BWRs still not operating [NUCWEEK 06_05].

Graphite Moderated Boiling Water Reactors (RBMK)



The Soviet RBMK reactor was originally designed for dual-purpose civilian-military operation. It is built for on-load refuelling, and can therefore be used for the production of high-grade plutonium in fuel elements that remain in the core for a short period of time, while at the same time being operated for electricity production. On-load refuelling also has considerable economic advantages. In water-cooled reactors, it necessitates the use of pressure tubes

surrounding individual fuel elements (1693 in an RBMK-1000) rather than a pressure vessel containing the whole core.

Like first-generation VVERs, RBMK reactors are regarded as “non upgradeable” by EU and G7 [G7]

Because of their combination of graphite moderator and light water cooling, RMBK reactors have a positive void coefficient. This means that if the water coolant evaporates, the number of neutrons increases and the fission process can run away with itself. This is a particularly dangerous feature; it requires extremely rapid control rod insertion in the event of an accident and can lead to a large energy release within the core. (The void coefficient is negative only for low fuel burn-up.) In the original RBMK design, this hazard was exacerbated by the so-called “positive scram effect” – control rods, being rapidly introduced into the core, were actually increasing reactivity at first because of a design shortcoming. The result of those deficiencies became apparent at the Chernobyl accident in 1986.

Since 1986, measures have been taken to reduce the positive void coefficient and to correct the deficiencies of the control rods to eliminate the positive scram effect [DONDERER 1996; BUTCHER 2001]. Fuel enrichment has been increased, additional absorber rods have been installed to provide a larger reactivity margin, and the time needed for complete rod insertion has been reduced. It is no longer possible to switch off the scram system while the reactor is in operation. To make human error less probable, control rooms have been fitted with new computerised facilities to provide a more comprehensive reactor control system.

However, a major safety deficit still persists in early 2005, in all RBMK plants but two: The reactors do not have a fully independent and diverse second shutdown system, and hence do not conform to IAEA safety requirements. This issue has long been identified as having high priority [IAEA 1999]. However, there are only two RBMKs where such a system has been inserted. This system consists of a separate set of fast-acting control rods that can provide rapid shutdown of the reactor. One of the RBMKs supplied with such a system is Ignalina-2 in Lithuania, where this improvement was implemented, after four and a half years of planning and installing work, in autumn of 2004, more than 18 years after the Chernobyl accident [BUTCHER 2001; NUCWEEK 39_04]. Even now, however, implementation at Ignalina-2 is not fully concluded since further improvements are required in the diversity of control rod drives, which should be completed by the end of 2006.

The other RBMK unit where an independent and diverse shutdown system has been installed is Kursk-1, a Russian first-generation RBMK. However, various shortcomings have been identified in connection with this system at Kursk-1. There is no complete fault schedule for the system, sufficient reliability of the system software has not been demonstrated so far, and system maintenance during operation is permitted, greatly reducing diversity. Apart from the shut-down system, the safety analyses performed for Kursk-1 by the operators has many other deficits, regarding missing analyses of severe accidents, lack of an analysis of systems’ reliability, lack of a discussion of safety culture etc. [CHOUHA 2004]. Kursk-2 is to be the next plant where an independent and diverse shutdown system is to be installed.

The other RBMKs so far only have a very limited secondary shutdown capability through boron injection via the emergency core cooling system.

There is another issue connected to reactivity problems which is still not completely resolved: If coolant is lost in the channels of the reactor’s control and protection system (CPS), an increase in reactivity results which the CPS cannot cope with. By introduction of neutron absorber material around the tip of the control rod, this effect can be halved, but not entirely eliminated [IAEA 1999].

RBMK reactors contain more zirconium alloy in the core than any other reactor type (about 50 % more than a conventional BWR). They also contain a large amount of graphite (about 1,700 tonnes). A graphite fire can seriously aggravate an accident situation; graphite can also react violently with water at higher temperatures, producing hydrogen. On the other hand, without air intrusion, the large graphite mass will slow down considerably the heating up of the reactor core after cooling failure.

Failure of a single pressure tube in an RBMK does not necessarily lead to catastrophic consequences, as would major failure of the reactor vessel in PWRs and BWRs. However, a large number of tubes and pipes necessitates a similarly large number of welds, and constitutes a system that is difficult to inspect and maintain. Multiple pressure tube failure constitutes an important safety issue.

The pressure suppression capacity of the containment system of RBMKs has been improved so that simultaneous rupture of up to nine pressure tubes can be controlled. However, in case of flow blockage after a loss-of-coolant accident, high temperatures could be reached, leading to ruptures in up to 40 channels of the total of about 1700 [BUTCHER 2001]. A catastrophic destruction of the whole reactor core can follow.

Since 1997, cracking occurred in the stabilized stainless steel piping of RBMK reactors. It has the characteristics of stress corrosion cracking as experienced by western BWRs. Rupture of the pipes concerned has the potential to damage the reactor core; releases of radioactivity into the atmosphere can result. The causes of the cracking phenomena are very complex. So far, no through wall cracks have been detected. There is no consensus, however, between western and Russian experts whether crack growth through pipe walls can be safely excluded [IAEA 2002].

The reactor core of an RBMK is very large, with low power density. This can lead to reactivity instabilities due to the heterogeneous distribution of xenon, a fission product exhibiting high neutron absorption. In addition, fuel elements are frequently changed, which means the configuration of fuel elements of different burn-ups is varying. These factors, as well as the positive void coefficient (which has only been reduced by upgrading, not eliminated), make monitoring and regulation of the reactor complicated and cumbersome.

In RBMKs, scram rods enter the reactor from above as well as from below (not only from below as in ordinary BWRs), which can be seen as a safety advantage. Likewise, the RBMK's emergency core cooling system is equipped with a pressure accumulator for fast core flooding, which is not found in ordinary BWRs.

The containment of an RBMK consists of several cells (for the main components) designed to withstand increased pressure, which for some cells are considerably higher than the design pressure of an ordinary BWR containment. This "cell-type" containment, however, is not total. Between the reactor and the refuelling hall directly above it, there is no high-pressure barrier, despite the upper cap of the working channels being a critical point in the whole design. The functioning of the containment depends on a pressure suppression system. (As has been pointed out above, this system cannot cope with the rupture of more than 9 pressure tubes out of about 1700).

As in ordinary BWR designs, there is a broad spectrum of event chains in RBMK reactors that can lead to large radioactive releases. The complexity of the reactor's control system makes it particularly vulnerable to human error and sabotage as accident initiating events. On-load refuelling opens up additional possibilities for loss-of-coolant accidents.

In RBMK reactors, spent fuel is stored in ponds inside the reactor building. The reactors are built exclusively as twin installations, with the two units sharing common systems, among them the spent fuel pool.

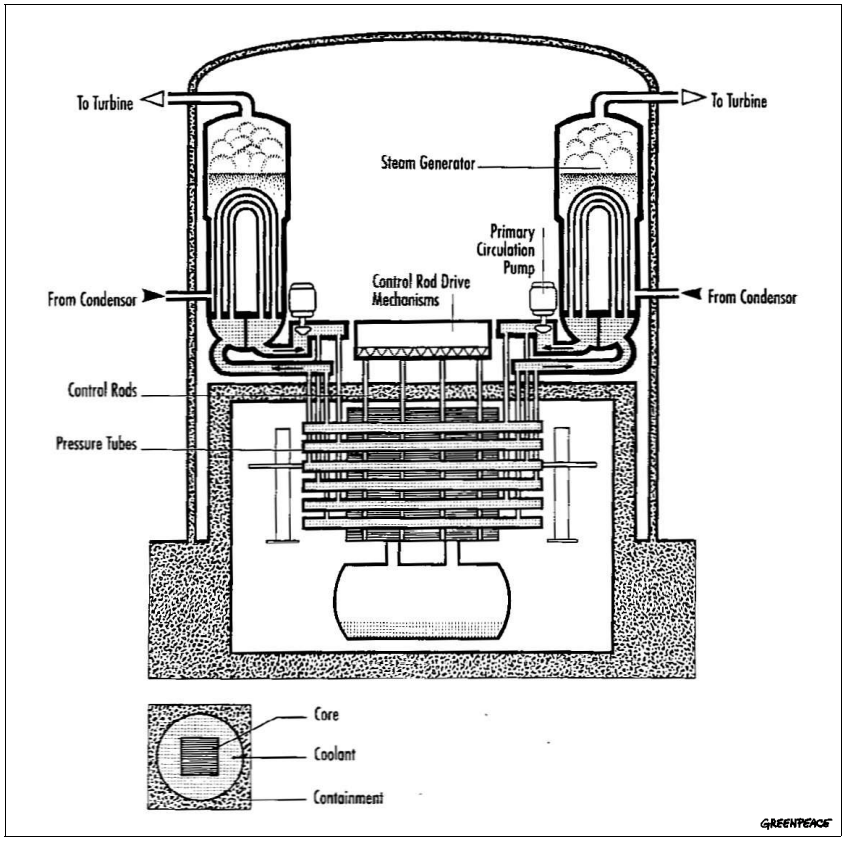
Radioactive releases from an RBMK can be very high, as was illustrated by the 1986 catastrophe at the Chernobyl power plant. It is startling to note that more than 18 years after the accident, the precise sequence of events is still not known, and expert opinions differ widely.

The pressure tubes are subjected to massive neutron-induced embrittlement and other ageing effects. Tube replacement programmes have been initiated in many RBMKs in the late 1980s. Furthermore, the gas gap between the pressure tube and the surrounding graphite closes after approximately 17 years of plant operation, leading to conditions in which the plant cannot be operated further. Re-tubing strategies have been developed by Russian designers [IAEA 1999].

All in all, there have been significant improvements at RBMK power plants since the Chernobyl accident, but none of the potentially hazardous features discussed here has been completely eliminated. Another catastrophic accident, with comparable or even more severe releases than the accident of April 26, 1986, cannot be excluded.

In spite of clear and present hazards of the RBMK type, all sixteen RBMKs in operation at the time of the Chernobyl accident (apart from the stricken reactor) continued to produce electricity for more than five years, and only four have been shut down since, the last one at December 31, 2004. By early 2005, eleven RBMKs are still operating in Russia, and one in Lithuania, in the European Union. For some plants, life extension has even been licensed (see section C).

Pressurized Heavy Water Reactors (PHWR)



Different lines of pressurized heavy water reactors have been developed in various countries. However, almost all PHWRs operating worldwide today belong to the Canadian CANDU line or are based to some extent on CANDU technology. This technology has been exported worldwide

to several Asian countries (China, India, South Korea and Pakistan) as well as to Romania and Argentina. Its natural uranium, heavy water cooled and moderated, pressure tube design has both advantages and disadvantages from a safety perspective, but the possibility of uncontained accidents involving melting fuel has not been eliminated.

Due to the predominance of CANDUs among PHWRs, this section will concentrate on the CANDU type.

As for the RBMK, the pressure tube design precludes the possibility of massive pressure vessel failure, but the accompanying greater length, surface area and complexity of the primary system piping results in a greater risk of loss-of-coolant accidents. The capability for on-load refuelling also introduces additional means by which loss-of-coolant can be initiated. The refuelling machine is also the major pathway for releases of radioactive "hot particles" – particles that have broken off the fuel or other activated metal particles, in particular the long-lived cobalt-60.

The main pressure bearing components, in this case the pressure tubes, are exposed to the full neutron flux, with consequent weakening effects. There have been problems with delayed hydride cracking as a result of deuterium-zirconium alloy reactions. The amount of blistering and cracking was found to be so serious at the Pickering and Bruce reactors that complete tube replacement schemes had to be devised for all 16 reactors at those sites in the late eighties and early nineties. (One reactor contains 390 pressure tubes.) Also, pressure tube fretting corrosion appears to be a generic flaw of the CANDU design. This degradation mechanism has been traced back to vibrations of the pressure tubes and could lead to a loss-of-coolant accident. Advanced tube fretting has been discovered in the early nineties at the Bruce and Darlington reactors.

Problems with CANDU pressure tubes persist. At Point Lepreau power station, premature degradation of the tubes resulting from ageing was recently reported after only about 20 years of commercial operation, requiring a costly refurbishment program [ENBP 2002]. At the Korean CANDU Wolsong-1, pressure tube replacement also might become necessary before the design lifetime (30 years) is reached [KYOUNG-SOO 2003]. Hydride cracking and fretting were observed in the last years at the Cernavoda-1 plant in Romania, which only started operating in 1996 [RADU 2003].

Although the large pool of relatively cool heavy water moderator provides an additional heat sink for decay heat removal, and a comparatively benign environment for control and safety instrumentation, there have been problems with unreliable neutron flux monitoring. The combination of natural uranium and heavy water has serious negative safety implications. The void coefficient of reactivity is positive, so that any loss-of-coolant accident could lead to a power excursion. A loss-of-coolant with scram failure in a CANDU will result in rapid melting of the fuel and possibly common mode breach of the containment. The extensive use of zirconium in the core (about the same amount as in a BWR) leads to a large zirconium-steam reaction potential in case of accidents.

Due to its relatively large size, the core is "decoupled", which means that neutron flux may significantly vary in different parts of the core, leading to flux oscillations. This design-inherent characteristic makes the CANDU reactor particularly vulnerable to loss-of-regulation (LORA) accidents, with subsequent power excursion. In September 1990, a "severe flux tilt" with large power shifts in the reactor core occurred at the Pickering-2 reactor. Plant personnel spent two days trying to stabilize the reactor before finally shutting it down [NAP 1997].

The use of heavy water as coolant and moderator results in the production of large and hazardous quantities of tritium. When coolant leaks in the environment occur, tritium releases cannot be avoided. In April 1996, for example, 50 trillion Bq of tritium were released into Lake

Ontario following a heavy water leak from a heat exchanger at Pickering-4. Several other cases of leaks occurred in the 1990s [NAP 1997].

CANDU designers have attempted to respond to the inherent safety problems by employing two separate scram systems and generally resorting to high levels of diversity and redundancy in the control and safety systems. A probabilistic approach to safety has been taken, however, at the expense of attention to common-mode and common-cause failures. The record of the industry indicates persistent failure in achieving safety system reliabilities and over-reliance on the containment as the final barrier against a large release.

Two different containment designs have been applied to CANDU reactors. The standard 600 MWe reactor (CANDU 600) which has been marketed internationally has a stand-alone containment consisting of a concrete dome that encloses the entire steam generating plant. The containment relies on an active spray system for pressure suppression in combination with an active system for filtered air discharge.

Ontario Hydro's multi-unit stations have a common containment envelope in which several reactors are connected to a single large "vacuum building" by pressure relief ducts. These systems depend on the operation of valves that normally keep the reactor buildings isolated from the vacuum building but are designed to open when the pressure increases beyond a certain point inside the reactor building. Of particular concern are the possibilities for common-mode accidents that damage the containment and/or the vacuum building, which is designed to withstand a release into the containment of a single reactor, but not accident conditions in several reactors at once.

In some stations there is the possibility for loss of primary coolant outside the containment, since components like boilers or maintenance cooling circuits penetrate the massive containment structure.

The basic safety features of the CANDU 600 have not developed very much over the years [WENRA 1999]. The design has not changed fundamentally; the safety deficiencies have persisted into the new century. In particular, the containments of the two units of this type at Cernavoda in Romania (one reactor in operation, one under construction) have the same shortcomings as older CANDU 600 containments [WENISCH 2002].

As with other reactors, CANDU containments are not designed to withstand worst case accidents, for example hydrogen detonations.

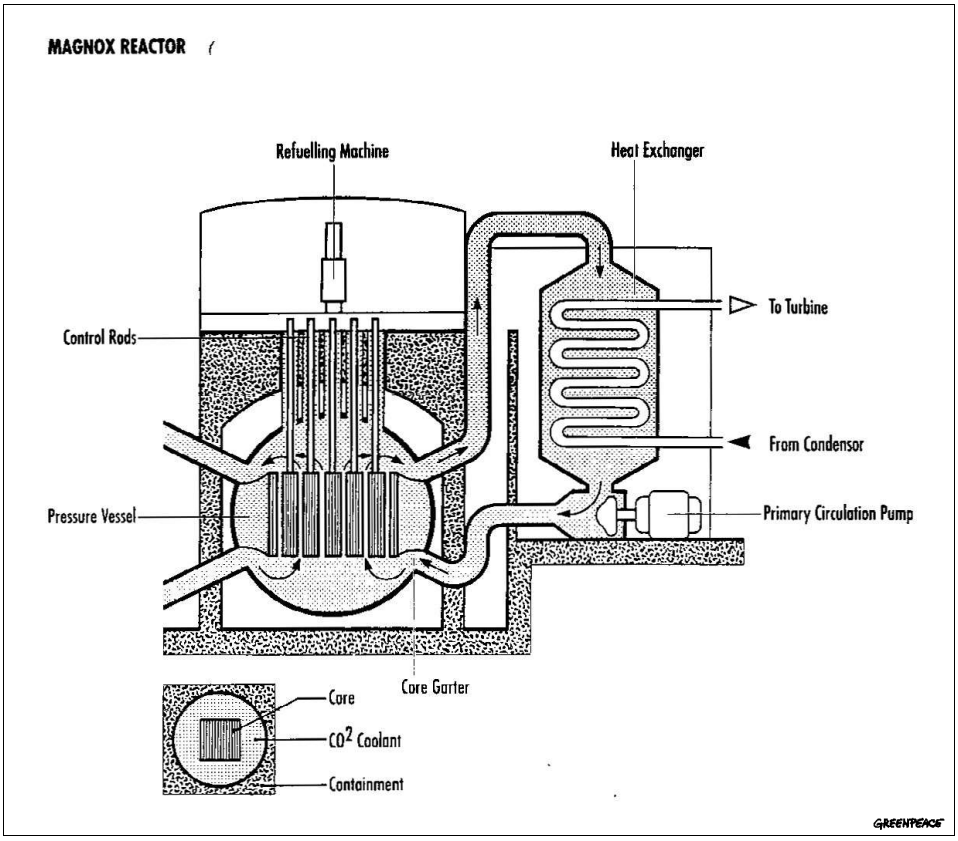
The general trend shows operational problems to become more and more severe at the existing CANDU plants. At older plants, which started up in the early 1970s, the effects of ageing are beginning to take their toll. Economic pressure and the resulting neglect of maintenance programmes are other important factors contributing to the decline of the CANDUs' performance.

This decline was dramatic indeed. As of June 1990, six reactors out of the top ten in worldwide lifetime performance were CANDU reactors, four of those from Ontario Hydro. Within six years, load factors dropped drastically due to what a technical journal called a "maintenance meltdown" [PEM 1998]. The operation of eight of Ontario Hydro's CANDU reactors was suspended or indefinitely deferred in the late 90s (three of those were reported "operable" by the end of 2003 [WNIH 2003, 2004]). As of June 2004, the only CANDUs among the top ten worldwide lifetime performers are the three South Korean Reactors Wolsong-2 to -4, which started commercial operation in the late 90s [KNOX 2004].

Magnox and Advanced Gas-Cooled Reactors (AGR)

Magnox reactors were developed in Great Britain from early air-cooled, graphite-moderated natural uranium reactors that produced plutonium for the British weapons programme. Magnox plants can be adapted to operate for the dual purpose of plutonium production and electricity generation. The Advanced Gas-Cooled Reactor (AGR) represents a further development of the Magnox design, with a significantly higher operating temperature and many technical modifications. Both reactor types are designed for on-load refuelling.

Magnox reactors have very low power density in the core, leading to large cores and, hence, large facilities. In an attempt to overcome this perceived weakness, power density was increased by a factor of two in the AGR, but it is still low compared to light water reactors. Carbon dioxide gas circulates in the primary circuit. Gas circulation is more complex in AGRs as the higher temperature necessitates a special re-entrant gas flow through the graphite moderator.

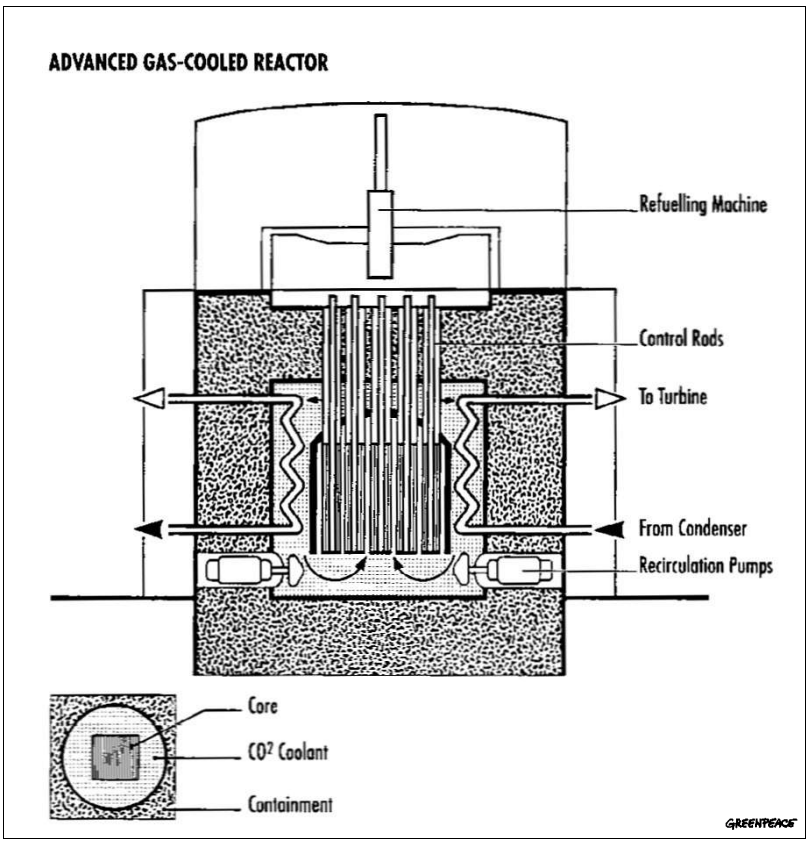


There is no zirconium in either reactor design. However, a fire risk exists, since a significant mass of graphite is located in the core, which can ignite after an air intrusion. This risk is enhanced by the so-called “Wigner effect”, which leads to an increase in the graphite temperature during the operational life of the reactor. Furthermore, small carbonaceous particles generated by radiolysis reduce the ignition threshold. Magnox reactors contain inflammable magnesium and uranium alloys in the core.

AGRs have experienced problems with the dynamics of the gas flow through the core, which as led to vibrations in the fuel assembly stringers (clusters of fuel elements are joined together end-to-end in stringers which are placed in vertical holes in the graphite moderator), especially when refuelling at full reactor power. As a consequence, on-load refuelling at full power was suspended in 1988. Because of economic pressure (achieving high load factors), it was attempted again in the mid-90s. However, in January 1996, an incident occurred at Heysham-2

AGR when a fuel rod became stuck in the reactor core during on-load refuelling. The reactor was automatically shut down. Another attempt also failed, leading to a second shutdown. A serious accident can result if a fuel rod gets stuck and the core overheats [WISE 1996]. Only refuelling at low power is now undertaken at AGRs.

In both designs, the reactor core is located inside a large pressure vessel. The older Magnox reactors, with a steel pressure vessel, have suffered from steel corrosion in the vessel and its internals. Thermal ageing and material degradation aggravate these problems by neutron-induced embrittlement.



Brittle failure of the pressure vessel could lead to total loss of the primary coolant, and possibly large radioactive releases. For this and other reasons, a number of Magnox stations have already been shut down. Operators claim that there would be a leak giving adequate warning of an impending problem before any rupture in the pressure vessel. However, it is unlikely that the leak-before-break (LBB) concept is generally applicable to the entire primary circuit. The existing leakage detection systems must therefore be considered insufficient to rule out major primary circuit ruptures. Furthermore, volumetric in-service inspection does not cover all critical components of the primary gas circuit – pressure vessel, standpipes, ductwork and boilers. Their structural integrity therefore cannot be guaranteed. This particularly applies to the refuelling machine, inspection of which is regularly done only for surface defects.

Emergency shutdown is accomplished in these gas-cooled reactors by rapid control rod insertion. If the inner core restraint system fails, causing loss of core, integrity, and the control rods cannot enter, the chain reaction will continue. There are, however, several secondary shut-down systems. Some of these are terminal in the sense that the reactor cannot be operated any longer afterwards; an example of this is the water flooding provision in the AGR. In Magnox reactors, a back-up boron ball shutdown system has been provided, activation of which is not fail-safe.

Four, maybe six out of fourteen AGRs have a reduced safety margin in their pressure vessel, due to incorrect pre-stressing of the tendons of the concrete pressure vessel.

For Magnox as well as AGRs, post-trip cooling depends on active systems. The design basis accident scenarios assume continued functioning of boilers and/or gas circulators. In case of complete loss of power, graphite temperature will increase dramatically in AGRs.

The support and safety systems of both reactor types are very simple compared to the complex systems of light water reactors. They generally fall short of modern standards due to their lack of diversity and segregation, particularly the electrical systems. Steam pressure in the secondary circuit is considerably higher than gas pressure in the primary loops. The possibility of a multiple boiler tube leakage is a major weakness in AGRs. In this event, water intrusion could be followed by a violent graphite-steam reaction.

Neither Magnox nor AGRs have secondary containment. Both reactor types have a high potential for large radioactive releases. A possible event chain in an AGR, for example, is steam intrusion in the reactor core due to multiple boiler tube failure, followed by failure of the pressure vessel. In a Magnox reactor, air intrusion after pressure vessel failure, and subsequent graphite ignition, could lead to a large release.

All in all, the rather ancient Magnox reactor line must be regarded as particularly hazardous due to many safety deficiencies. This reactor type is being phased out. But only half of the 24 Magnox units in Great Britain have been shut down by 2004, four more are to follow in 2005, and the reminder will be shut down as late as 2010 [WNIH 2004; NUCWEEK 41_04].

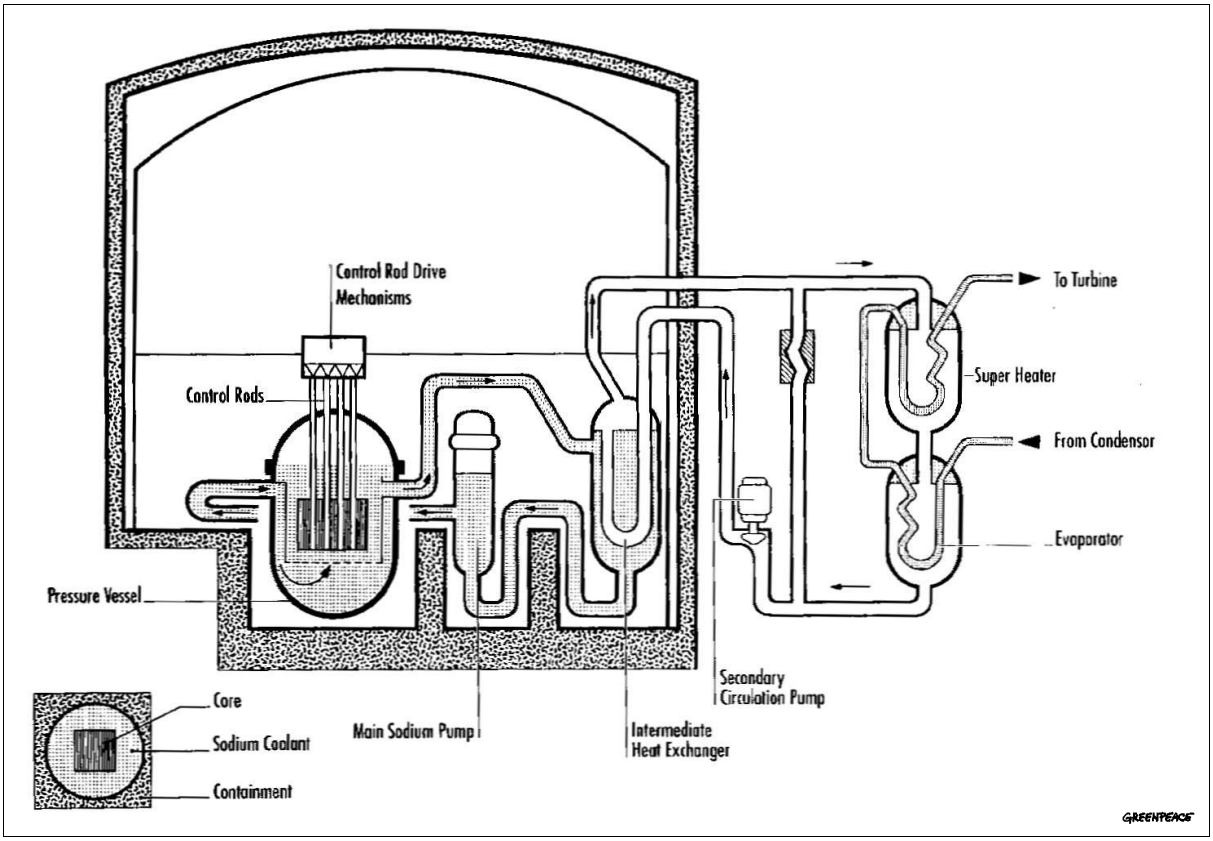
Due to increasingly severe ageing problems, the operating periods of the AGRs also might be limited. There are indications that their planned lifetime of 35 years may not be achievable, and that plans for life extensions beyond that point were wildly over-ambitious. A major problem that discovered late in 2004: A new type of graphite cracking in the reactor that might endanger the integrity of the core. This failure mechanism had not been anticipated by analytical methods applied before in safety analyses. It can have adverse consequences for the movement of control rod, the exchange of fuel and the flow of the coolant gas [NUCWEEK 50_04].

There are also ageing problems in connection with the boiler integrity at several AGR stations [NUCWEEK 32_04]. It is well possible that the era of the British gas-cooled reactor lines will be over within the next 10 years.

Sodium-cooled Fast Breeder Reactors (SFR)

Fast reactor designs with sodium cooling have been under development from the very beginning of nuclear power in the late 1940s. Indeed, for several decades they were heralded as the solution to all energy problems since, in theory, breeder reactors permit the complete use of natural uranium as nuclear fuel by systematically breeding fissile plutonium-239 out of non-fissile uranium-238. Thus, uranium reserves would theoretically be extended by about a factor of 100. On the other hand, proliferation problems of this reactor types are particularly severe, since large amounts of weapons-grade plutonium would be produced, separated and used for fuel production.

The reactor is typically arranged into the fission zone (20-30 % PuO_2 and UO_2) and the surrounding breeding zone ("blanket", UO_2), and this arrangement has to be as compact as possible to preclude high neutron leakage out of the core. A coolant with a high heat capacity and low neutron absorption is required for this design, hence the selection of sodium.

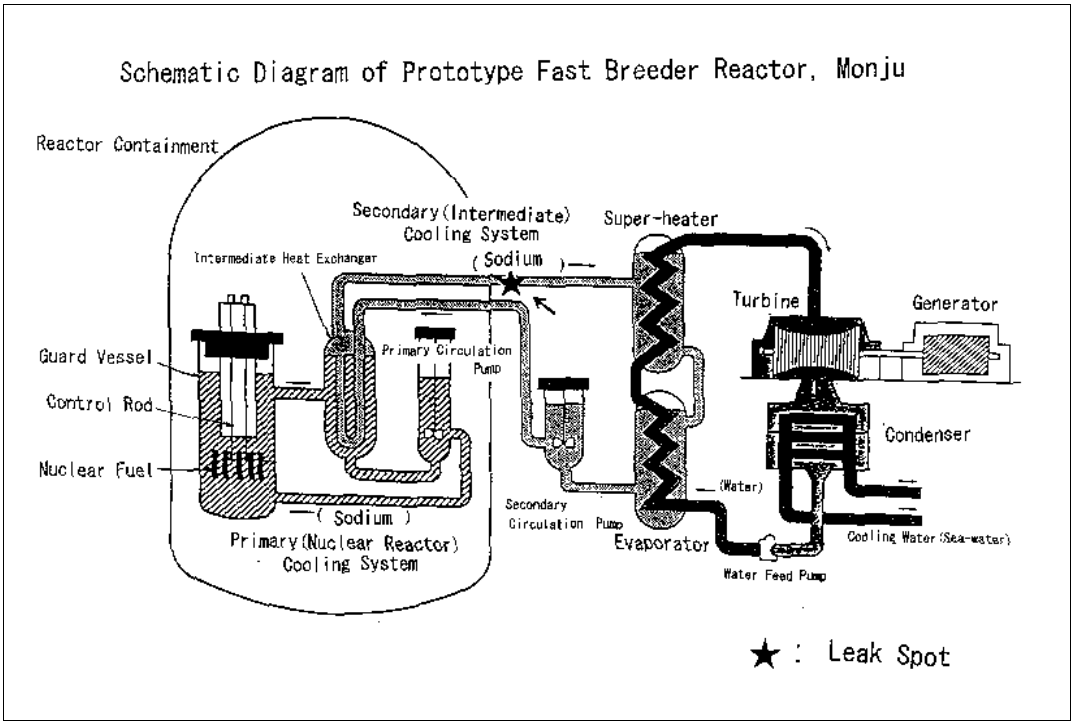


However, the core arrangement, in connection with the liquid sodium as coolant, exhibits some serious risk features. The sodium void coefficient is positive. As most cores are not in their maximum reactivity configuration, variations in core geometry can lead to higher reactivity. Rapid and uncontrollable nuclear power excursions (so-called Bethe-Tait accidents) are thus possible, with subsequent meltdown and vaporization of the fuel – for example if the coolant pumps shut down and the reactor fails to scram.

The large number of events that might trigger power excursions also includes absorber element ejection, propagation of local coolant blockages, geometrical instabilities due to earthquake or failure of the core-hold plate, failure of the grid plate, and fission gas or vaporized oil entering the core. The dangerous reactivity characteristics are particularly pronounced in large reactors. The catastrophic radiological consequences that could result from a release of fission products would be aggravated by the release of vaporized plutonium.

Other accidents are made possible by the strong chemical reactivity of sodium with air and water. Sodium fires, with the danger of subsequent common-mode failures, can be initiated by sodium leaks from either the primary or the secondary circuit. If water should find its way into a sodium circuit, the result would be an explosion, possibly, breaching the containment, and a major release of radioactivity. Steam generator tubes are especially vulnerable to rupture.

The disadvantages of sodium coolant were drastically illustrated in December 1995 when 700 kg of molten sodium leaked from the secondary circuit of the Japanese breeder Monju. A fire creating extensive destruction resulted.



Source: Kanazawa Institute[2]

Coolant temperatures in the primary circuit can considerably exceed 500° C. This can lead to problems of thermal embrittlement and thermally induced stresses. Corrosion constitutes another risk factor. It is impossible to build a containment capable of coping with the energy release from Bethe-Tait accidents, as no reliable upper bounds have yet been established. Even if these were determined, containment designs would have to be extremely strong and massively expensive. Lack of an adequate containment was one of the main safety concerns about the German SNR-300 fast reactor, which never obtained the required operating license.

The support and safety systems in fast reactors are generally less complex than in light water reactors. Due to the reactor's high susceptibility to rapid power excursions, its scram system is particularly important. Even though control rod injection is supported by accelerators and gravity, shut-down may be impossible in time in the event of very rapid reactivity increases, accompanied by sodium boiling and leading to uncontrollable, "run-away" conditions in the reactor.

Other fast reactor designs will be discussed in section B.2.

[2] http://www.**onlineethics.org/ cases/iino.html**

Conclusion

Key Characteristics Of Reactor Types Presented Here:

	PWR/VVER	BWR	RMBK	CANDU	AGR	SFR
Fuel enrichment	3 – 4 %	3 – 4 %	2 – 2.4 %	Natural U	2 – 3 %	20 – 30 %
Moderator	Light water	Light water	Graphite	Heavy water	Graphite	None
Coolant	Light water	Light water	Light water	Heavy water	CO_2	Liquid sodium
Core enclosure	Pressure vessel (steel)	Pressure vessel (steel)	Pressure tubes (steel)	Pressure tubes (steel)	Pressure vessel (concrete)	Vessel (steel)
Average power density (kW/l)	100	50	5	15	2.5	400
Maximum temperature (°C)	300 - 330	280 - 300	280 - 290	290 - 310	670	550 - 600
Cooling circuits[3]	2	1	1	2	2	2
On-load refuelling	No	No	Yes	Yes	Yes (at low power)	No
Typical burn-up (GWd/t)	40 – 50	30 – 45	10	8 – 10	18	100
Secondary containment	Yes (except old VVERs)	Yes	Confinement system only	Yes	None	Yes

[3] Between reactor and turbine

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B.1 Overview of New Reactors – "Generation III"

Introduction

Among commercial nuclear power plant types, four generations of reactors are commonly distinguished. Generation I were prototype commercial reactors developed in the 1950s and 1960s. Most, but not all of them have already been decommissioned; the Magnox plants some of which are still operating in Great Britain today belong to Generation I.

The vast majority of the reactors in commercial operation worldwide today belong to Generation II (PWR, VVER, BWR, RBMK, CANDU, AGR).

Generation III are the so-called "Advanced Reactors", three of which are already in operation in Japan, and more are under construction or planned. About 20 different designs for Generation III reactors are reported to be under development [IAEA 2004; WNO 2004a]. Most of them are "evolutionary" designs that have been developed from Generation II reactor types with some modifications, but without introducing drastic changes. Some of them represent more innovative approaches. Generation IV reactors are described as radically different designs. Partially, they have closed fuel cycle.

According to the World Nuclear Association, reactors of Generation III are characterized by the following points [WNO 2004b]:

- a standardised design for each type to expedite licensing, reduce capital cost and reduce construction time,
- a simpler and more rugged design, making them easier to operate and less vulnerable to operational upsets,
- higher availability and longer operating life - typically 60 years,
- reduced possibility of core melt accidents,
- minimal effect on the environment,
- higher burn-up to reduce fuel use and the amount of waste,
- burnable absorbers ("poisons") to extend fuel life.

It is quite clear that those goals mainly are directed towards better economics. Their addressing higher safety remains rather vague.

The distinction between Generation III and Generation IV is not always clear-cut. Generation III is generally considered to also include some reactor types which incorporate more than evolutionary innovations; the best known among those is the Pebble Bed Modular Reactor (PBMR), which is sometimes classified as "Generation III+".

Generations III and IV are more clearly separated, however, by their respective time horizons. While some reactors of Generation III are already operating, and demonstration plants for several concepts of this generation are to be operational by about 2010, Generation IV reactors will not, even from the most optimistic advocates be operational before 2020, with others suggestion 2045, if at all.

In this section, two examples for Generation III reactors will be presented and discussed with their main safety deficits and hazards. For both of them, concrete projects have already been started. They illustrate the variety of this generation: The European Pressurized Water Reactor (EPR) is not far removed from the French and German PWRs from which it was developed; it is a light-water cooled and moderated type with large capacity (1,600 MWe). The Pebble Bed

Modular Reactor (PBMR), on the other hand, is an offspring of earlier high-temperature reactors significantly developed further; it is helium cooled and graphite moderated and to be built in small modules (110 – 125 MWe).

Furthermore, examples for other Generation III reactors in operation, under construction or planned worldwide will be given.

The European Pressurized Water Reactor (EPR)

The EPR is a pressurized water reactor that represents a development from the French N4 and the German KONVOI reactor line, the latest Generation II reactors which went taken into operation in those countries. [HAINZ 2004]

The goals stated for EPR development are to improve the safety level of the reactor (in particular, reduce the probability of a severe accident by a factor of ten), achieve mitigation of severe accidents by restricting their consequences to the plant itself, and to reduce costs.

Compared to its predecessors, however, the EPR displays several modifications which constitute a reduction of safety margins: The volume of the reactor building has been reduced by simplifying the layout of the emergency core cooling system, and by using the results of new calculations which predict less hydrogen development during an accident. Furthermore, the thermal output of the plant was increased by 15 % relative to the N4 by increasing core outlet temperature, letting the main coolant pumps run at higher capacity and modifying the steam generators.

In some cases, the EPR actually has fewer redundant trains in safety systems than the KONVOI plant; for example, its emergency core cooling system has only 4 accumulators (pressure tanks) whereas the KONVOI plants' has 8 such tanks.

Several other modifications are hailed as substantial safety improvements:

- The <u>incontainment refuelling water storage tank</u> (IRWST) is located at the bottom of the reactor building and combines coolant storage and sump function. During a loss-of-coolant accident, switchover from safety injection to sump recirculation is thus avoided. In this way, some sources of failures are avoided. The overall safety gain, however, appears rather small.
- The <u>core catcher</u> has the function to control a core melt accident. In the EPR, the molten core collects in the reactor cavity below the pressure vessel. After melting through a bulkhead, it then passes through an outlet conduit and spreads in specifically designed area. By means of passive features, the water of the IRWST is then released for flooding and cooling the core melt in this area. The floor of the spreading area is provided with a cooling system to avoid excessive temperatures in the structural concrete of the reactor building. However, even before the melt reaches the core catcher, a violent steam explosion could take place in the reactor pressure vessel, possibly leading to containment failure. Furthermore, steam explosions can also occur later in the course of the accident, when the melt in the spreading area comes into contact with IRWST water. Even if this does not happen, it is not clear that effective cooling of the spread molten core will be possible. A solid layer on the surface of the melt could form, preventing heat removal, and the core could eat into the concrete below the spreading area.
- The <u>containment heat removal system</u> is taken from the N4 design. Its purpose is to lower containment pressure and thus, avoid overpressure failure. This system must remain operable over a long period of time to ensure cooling. No information on its failure probability is available.

- Hydrogen recombiners serve to reduce hydrogen concentration in the containment by passive, catalytic processes. Such recombiners are already employed in many PWRs worldwide. They probably are effective in reducing the hazard of hydrogen detonations, but cannot completely exclude it.
- The EPR is equipped with an instrumentation and control system on a digital basis. The use of such a system is very demanding on the developer and it is very difficult to verify its correct implementation. A similar system was installed at the German PWR Neckar-1 in 2000; the system failed and for a while the ability for fast reactor shutdown (scram) was blocked. A digital instrumentation and control system has been installed at the UK PWR Sizewell B from the beginning; in April 1998, it led to a severe degradation of the reactor protection system.

The protection of the plant against airplane crash is equivalent to that of the German KONVOI plants and hence does not reach a new, higher safety level.

In spite of the changes being envisaged, the EPR appears to be plagued by a problem which is widespread among PWRs of Generation II, and still not fully resolved for those (see section A): According to the Finnish regulatory authority, sump strainer clogging is an issue with the EPR, in spite of claims by French experts that this problem is not relevant due to design differences compared to with existing reactors. The issue has been identified by the Finnish authority many years ago, but still appears to be a big challenge for the EPR [NUCWEEK 11_04].

All in all, there is no guarantee that the safety level of the EPR does indeed represent a significant improvement compared to N4 and KONVOI; in particular, the reduction of the expected core melt probability by a factor of ten is not proven. Furthermore, there are serious doubts as to whether the mitigation and control of a core melt accident with the “core catcher” concept will actually work as envisaged.

The Pebble Bed Modular Reactor (PBMR)

The PBMR is a high-temperature gas-cooled reactor (HTGR). The HTGR line has been pursued until the late 80s in several countries; however, only prototype plants were ever operated, all of which were decommissioned after about twelve years of operation at most (Peach Bottom 1 and Fort St. Vrain, USA, in 1974 and 1989; Winfrith (UK) in 1976, Hamm-Uentrop, Germany, in 1988) [WNIH 2004].

Unlike light water reactors that use water and steam, the PBMR design uses pressurized helium heated in the reactor core to drive a series of turbines that attach to an electrical generator. The helium is cycled to a recuperator to be cooled down by a secondary helium circuit and returned to cool the reactor. Helium temperature at the core outlet is about 900° C, at a pressure of 69 bar. The secondary helium circuit is cooled by water [ESKOM 2005].

Designers claim there are no accident scenarios that would result in significant fuel damage and catastrophic release of radioactivity. These claims rely on the heat resistant quality and integrity of the tennis ball-sized graphite fuel assemblies or "pebbles," 400,000 of which are continuously fed from a fuel silo through the reactor to keep the reactor core. Each spherical fuel element has an inner graphite core embedded with thousands of smaller fuel particles of enriched uranium (up to 10 %), encapsulated in multi-layers of non-porous hardened carbon. The slow circulation of fuel through the reactor provides for a small core size that minimizes excess core reactivity and lowers power density, all of which is credited to safety. However, so much credit is given to the integrity and quality control of the coated fuel pebbles to retain the radioactivity that no containment building is planned for the PBMR design. While the elimination of the containment building provides a significant cost savings for the utility – perhaps making the design economically feasible – the trade-off is public health and safety [GUNTER 2001].

According to the prospective PBMR operator Eskom, the reactor is “walk-away-safe”. This is meant to imply that even should the plant personnel leave the site, the reactor would not get into a critical condition. It is claimed that fuel temperature will peak at 1600° C in any case, whereas fuel damage will not begin below 2000° C [ESKOM 2005].

However, the temperature limit of 1600° C is not guaranteed in reality. It depends on successful reactor scram as well as on the functioning of the passive cooling systems (which can be impeded, for example, by pipe breaks and leaks in coolers). Furthermore, fission product releases from the fuel elements already begin at temperatures just above 1600° C. In this context, it is irrelevant that severe fuel damage or melting only occurs above 2000°C. Massive radioactive releases can take place well below this temperature.

While it is true that core heating proceeds rather slowly after cooling failure, this thermal inertia causes its own problems: By the use of graphite as moderator and structural material. If air enters the primary helium circuit, a severe accident with graphite fire, leading to catastrophic radioactive releases, can be the consequence. Also, in case of water ingress through the secondary circuit, for example due to leakages in the heat exchangers, violent graphite-steam-reactions can occur. Burning of graphite is probably the most risk significant accident scenario possible for the PBMR [HAHN 1988].

Other “Generation III” Reactor Designs

Many different concepts bearing the label “Generation III” are in various stages of development and implementation today. A complete listing will not be attempted here. In the following, the most important examples as mentioned by the World Nuclear Association [WNO 2004b] and the International Atomic Energy Agency [IAEA 2004] will be provided. Usually, the concepts are classified into two categories: Large designs – above 700 MWe; small and medium designs – below 700 MWe.

Pressurized Water Reactors:

The principal large designs are APWR (Mitsubishi/Westinghouse), APWR+ (Mitsubishi), EPR (Framatome ANP), AP-1000 (Westinghouse), KSNP+ and APR-1400 (Korean Industry) and the CNP-1000 (China National Nuclear Corporation).

Regarding VVERs, an advanced VVER-1000 has been developed by Atomenergoproject and Gidropress in Russia.

The main small- and medium-size advanced PWR designs are the AP-600 (Westinghouse) and the VVER-640 (Atomenergoproject and Gidropress).

Boiling Water Reactors:

The main large concepts are the ABWR and the ABWR-II (Hitachi, Toshiby, General Electric), the BWR 90+ (Westinghouse Atom of Sweden), the SWR-1000 (Framatome ANP) and the ESBWR of General Electric.

The HSBWR and HABWR (Hitachi) are small- and medium-sized advanced BWR concepts.

Three ABWRs are already operating in Japan: Two at Kashiwazaki-Kariwa since 1996, a third started operating in 2004.

Heavy Water Reactors:

The ACR-700 is an evolutionary CANDU design (Atomic Energy of Canada Limited).

India is developing the AHWR (Advanced Heavy-Water Reactor), a heavy water moderated, boiling light water cooled evolutionary design.

Gas-cooled Reactors:

Apart from the PBMR (ESKOM/BNFL), a small gas turbine modular helium reactor (GT-MHR) is being developed in an international effort; this reactor is discussed in section B.2.

Fast Breeder Reactors:

No evolutionary breeder type is being developed. Several fast reactors are among the concepts under consideration for Generation IV.

During the last years, a "revival" of small nuclear power reactors is claimed to occur by nuclear promoters. The main driving force is the desire to reduce capital costs and to provide power away from main grid systems. Some of the concepts developed with this aim could also be classified as Generation III – for example the LWR designs CAREM (Argentina) and SMART (South Korea) [WNO 2005].

Conclusion

Generation III is characterized by a multitude of different designs on the one hand, and very few plants already under operation on the other. Indeed, of about 20 concepts worldwide, only one has so far reached the stage of operating power plants (three ABWRs in Japan).

It is obvious that the large number of Generation III concepts does not represent a corresponding number of really new reactor types. The effort of developing so many genuinely new reactor lines is disproportionate to the limited number of concrete projects with any chance of realisation. Indeed, the EPR is typical for many Generation III types in that it constitutes, as pointed out above, simply a slightly modified version of current reactor designs (in this case, French N4 and German Konvoi), with some improvements, but also with reduction of safety margins and fewer redundancies for some safety systems.

The same holds, for example, for the Korean KSNP+ which simply is the latest KSNP (Generation II) plant with optimisation and simplification of some features, some improvement in monitoring systems and some changes regarding building materials and construction concepts to save time and costs [JAE YOUNG YANG 2003] – hardly a new reactor generation.

Atomic Energy of Canada's ACR-700 combines features from current U.S. light water reactors with CANDU technology; a combination of well-known technology primarily aiming not at improved safety, but at better economics (for example, by requiring a smaller heavy water inventory, and extending fuel life) [FABIAN 2004].

Other Generation III concepts are still in rather early phases of implementation and appear as showpieces to be presented to the public and/or playground for under-employed reactor experts, rather than as serious concepts with chances of commercial implementation.

The U.S. AP-1000, for example, shows some more innovative characteristics than the EPR does or other purely evolutionary types. It is hailed as possessing passive safety features and really to belong to Generation III+; its safety systems, however, it actually rely on valves as well as on active heating, ventilation and cooling units. The AP-1000 is still some way away from being built. The U.S.NRC's design application process is in an advanced stage, but not yet completed, but expected by the end of 2005 [PAULSON 2002; NEI 2004]. (Other Generation III concepts are still in the pre-application review phase in the U.S., namely, ESBWR, ACR-700, SWR-1000

and PBMR – along with two Generation IV concepts (GT-MHR and IRIS, see section B.2) [NRC 2005].

The French/German SWR-1000 BWR concept is also characterized by a mixture of passive and active safety features, as well as design simplifications aiming at lower costs [KTG 1998]. It is probably even further removed from practical realisation than is AP-1000 and was developed in the 1990s mostly to keep German reactor experts busy and their know-how alive.

The PBMR is a special case within Generation III – being an evolutionary development from a reactor line which was intended to be part of Generation II, but never really made it into this generation. The PBMR still displays the main hazardous features of its forerunners, the prototype HTGRs.

All in all, "Generation III" appears as a heterogeneous collection of different reactor concepts. Some are barely evolved from the current Generation II, with modifications aiming primarily at better economics, yet bearing the label of being safer than current reactors in the hope of improving public acceptance. Others are mostly theoretical concepts so far, with a mixture of innovative and conventional features, which are being used to underpin the promise of a safe and bright nuclear future – while also not forgetting about simplification and cost-cutting.

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B.2 Generation IV

Introduction

The U.S. Department of Energy (DoE) launched the "Generation IV International Forum" (GIF) in 2000. Today, ten member countries are participating in this initiative (Argentina, Brazil, Canada, France, Japan, Republic of Korea, South Africa, Switzerland, U.K., U SA), as well as EURATOM. Their goal is to develop innovative nuclear systems (reactors and fuel cycles) likely to reach technical maturity by about 2030, but many suggest that this is optimistic. These Generation IV reactors are expected to be highly economical, incorporate enhanced safety, produce minimal amounts of waste, and be impervious to proliferation. Last but not least, Generation IV systems should address these issues in a manner that promotes greater public acceptance.

Goals for Generation IV are defined in four broad areas:

- Sustainability
- Economics
- Safety and reliability
- Proliferation resistance and physical protection.

Groups of international experts from industry, universities and national laboratories were organized to undertake the identification and evaluation of candidate systems, and to define research and development (R&D) activities to support them.

Some 100 different reactor designs were identified as candidates and evaluated. These designs ranged from concepts that really belonged to Generation III+ to a few that were radically different from all known technologies. At the end of the process, six concepts were recommended for further development (see below). The GIF noted that some of the concepts might ultimately not be viable or might not achieve commercial deployment.

A Roadmap describes the R&D required to develop each of the six selected Generation IV systems as well as the approximate time and cost for completion. Many of the technological gaps are common to more than one system and the Roadmap identifies several areas where crosscutting R&D will be required. The necessary R&D will be very expensive and no single country has the necessary facilities and expertise to carry it out alone, hence the need for international collaboration [DOE 2002].

The Generation IV program also established a separate effort to evaluate nuclear power plant designs that might be deployed as commercial operating units by 2010. Light water and gas-cooled systems are being considered.

To further encourage and strengthen research and development for Generation IV reactors, the United States, Canada, France, Japan and the U.K. signed the International Forum Framework Agreement on February 28, 2005 in Washington. Special emphasis appears to lie in developing systems for the generation of hydrogen as well as electricity [NNF 2005a; ANDERSON 2005].

In 2001, the IAEA has initiated a similar initiative, the International Projects on Innovative Nuclear Reactors and Fuel Cycles (INPRO). INPRO is likely to focus on more than one system depending on regional needs. It is funded through the IAEA budget. As of November 2004, 21 countries or entities[4] have become members of INPRO. GIF and INPRO have agreed to

[4] Argentina, Armenia, Brazil, Bulgaria, Canada, Chile, China, Czech Republic, France, Germany, India, Indonesia, Republic of Korea, Pakistan, Russian Federation, South Africa, Spain, Switzerland, Netherlands, Turkey and the

formalize cooperation at the technical level. (The U.S. has been reluctant to participate in INPRO because it was seen as a Russian-inspired initiative [NUCWEEK 14_02].)

Concepts Selected for Generation IV

As pointed out above, six concepts were selected for further development in the framework of GIF. They are briefly discussed in the following.

GFR – Gas-Cooled Fast Reactor System:

The GFR system is a helium-cooled reactor with fast-neutron spectrum and closed fuel cycle. It is primarily envisioned for electricity production and actinide management. The GFR is not intended for hydrogen production.

The reference reactor (288 MWe) is operating with an outlet temperature of 850°C, using a direct cycle gas turbine. The GFR reference assumes an integrated, on-site spent fuel treatment and re-fabrication plant, but the viability of the planned technology has yet to be demonstrated. Fuel cycle technology is the most comprehensive technology gap of the GFR.

The viability has to be demonstrated also in the area of safety, including decay heat removal systems, fuel forms and core design. Core configurations are being considered based on pin- or plate-based fuel assemblies or prismatic blocks.

It is hoped that the GFR may benefit from development of the HTGR technology (which is also beset with many problems; see discussion of VHTR below) as well as from development of innovative fuel and very-high-temperature materials for the VHTR.

In spite of large technology gaps, according to GIF, the GFR system is top-ranked in sustainability because of its closed fuel cycle and excellent performance in actinide management. It is rated good in safety, economics, as well as proliferation resistance and physical protection. The GFR is estimated to be deployable by 2025 [DOE 2002].

Several GIF members have a specific interest for a sequenced development of gas-cooled system: The first step of the ´Gas Technology Path` aims to develop a modular HTGR, the second step would be the VHTR, and the third step the GFR [CARRÈ 2004]. The gas-cooled systems VHTR and GFR are seen as the top priorities in Europe and the USA.

LFR – Lead-Cooled Fast Reactor System:

LFR systems are reactors cooled by liquid metal (lead or lead/bismuth) with a fast-neutron spectrum and closed fuel cycle system. A full actinide recycle fuel cycle with central or regional facilities is envisaged. A wide range of unit sizes is planned, from ´batteries` of 50–150 MWe, and modular units of 300-400 MWe to large single plants of 1200 MWe. The LFR battery option is a small factory-built turnkey plant with very long core life (10 to 30 years). It is designed for small grids, and for developing countries that may not wish to deploy a fuel cycle infrastructure. Among the LFR concepts, this battery option is regarded as the best, concerning fulfilment of Generation IV goals. However, it also has the largest research needs and longest development time.

The LFR is cooled by natural convection with a reactor outlet coolant temperature of 550° C. It is envisaged to reach a reactor outlet coolant temperature of 800°C. The fuel is metal- or nitride-based [DOE 2002]. Experience with the technology is restricted to seven Russian Alpha class submarines, which stopped operation in 1995, and on the advanced liquid-metal fast breeder reactor (ALMR), the design of which was withdrawn from NRC review at an early stage [WANO 2004b].

European Commission

Important LFR technology gaps are in the areas of system fuels and materials, with some gaps remaining for the 550ºC options, and large gaps for the 750–800ºC option (only the higher temperature makes the production of hydrogen possible), including: nitride fuels development, high-temperature structural materials and environmental issues with lead. Nitride fuel is clearly required for the higher-temperature option. It is estimated that 10–15 years will be necessary to qualify any new fuel. Therefore, the nearer-term options focus is on electricity production, and hydrogen production is envisaged for the longer-term options [DOE 2002].

Although Russia, where almost all the experience LFRs is concentrated, was not a part of GIF, this design corresponds with Russia's BREST reactor [NEI 2002a]. (BREST is a fast neutron reactor, of 300 MWe with lead as the primary coolant. A pilot unit is being built at Beloyarsk [WANO 2004b].) Among the GIF members, only Switzerland has a major interest in the development of LFR. The U.S. has initiated design explorations. Noteworthy among them is the Small Secure Transportable Autonomous Reactor (SSTAR).

The LFR system is top-ranked in sustainability because a closed fuel cycle is aimed at, and in proliferation resistance and physical protection because it employs a long-life core. It is rated good in safety and economics. The LFR system is estimated to be deployable by 2025 [DOE 2002].

MSR – Molten Salt Reactor System:

The MSR system is based on a thermal neutron spectrum and a closed fuel cycle. The uranium fuel is dissolved in the sodium fluoride salt coolant that circulates through graphite core channels. The heat, directly generated in the molten salt, is transferred to a secondary coolant system, and then through a tertiary heat exchanger to the power conversion system. It is primarily envisioned for electricity production and waste burn-down. The reference plant has a power level of 1,000 MWe. Coolant temperature is 700°C at very low pressure. The temperature margin to the salt boiling temperature (1400°C) is large.

Since the fuel is a liquid, it could be cleansed of fission product impurities without the need to shut down the reactor. Fission products are removed continuously, while plutonium and other actinides can be added along with U-238. Because of the liquid fuel, there is no need for fuel fabrication. The MSRs have a low inventory of fissile materials compared with other reactors. However, one of the major drawbacks of the technology is the highly corrosive nature of the salts.

During the 1960s the USA developed the molten salt breeder reactor as the primary back-up option for the conventional fast breeder reactor (cooled by liquid metal). A small prototype (8 MWth), the Molten Salt Reactor Experiment (MSRE), was operated for only four years. The next project planned, the Molten Salt Breeder Reactor (MSBR), was never built. The present work rests only on these projects. Detailed designs of a MSR have not been produced since the 1970s [FORSBERG 2002].

There are many technology gaps, for example: At the high temperatures in an MSR, the tritium can diffuse through the heat exchangers into the secondary system. Tritium control technologies need to be developed. Non-proliferation was not a concern when the MSR technology was first developed. Research is required to determine if design changes are required. No work was done to convert the wastes into acceptable forms for disposal. Nevertheless, according to GIF, the MSR system is top-ranked in sustainability because of its closed fuel cycle and excellent performance in waste burn-down. It is rated good in safety, and in proliferation resistance and physical protection, and it is rated neutral in economics because of its large number of subsystems. The MSR is estimated to be deployable by 2025 [DOE 2002].

The GIF selected the MSR as the most innovative non-classical concept. Of all six reactor systems, MSR requires the highest costs for development (1000 million US$). All in all, the interest of the GIF member states in the MSR is rather low. The high development costs and the required time frame could eliminate the MSR system from Generation IV altogether [NUCWEEK 02_05].

SCWR – Supercritical-Water-Cooled Reactor System:

The SCWRs are high-temperature, high-pressure water-cooled reactors that operate above the thermodynamic critical point of water (i.e. at pressures and temperatures at which there is no difference between liquid and vapor phase). The reference plant has a 1700 MWe power level, an operating pressure of 25 MPa, and a reactor outlet temperature of 550°C. Fuel is uranium oxide. Passive safety features similar to those of the simplified boiling water reactor (SBWR) are incorporated. SCWRs could be designed as thermal or as fast-spectrum reactors, but current worldwide efforts focus on the thermal design.

The thermal efficiency of a SCWR can approach 44%, compared to 33–35% for LWRs. Because no change of phase occurs in the core and the system utilizes a direct cycle (like the BWR), steam separators, dryers, pressurizes and recirculation pumps are not required, resulting in a considerably simpler and more compact system than traditional light-water reactors (LWR). SCWRs are expected to be more economical than LWRs, due to plant simplification and high thermal efficiency. The Governments of Japan, the U.S. and Canada are developing the SCWR. There have been no prototypes built so far.

The technology for the SCWR is based on the existing LWRs and supercritical-water-cooled fossil-fired power plants. However, there are important SCWR technology gaps in the areas of: materials and structures, including corrosion and stress corrosion cracking (SCC), safety and plant design. Only few data exist on the behaviour of materials in SCWR under irradiation and in the temperature and pressure ranges of interest. At present, no candidate alloy has been confirmed for use as the cladding or structural material. The main feasibility issues are the development of suitable in-core materials and the demonstration of adequate safety and stability.

The SCWR system is primarily envisioned for electricity production, with an option for actinide management. According to GIF, the SCWR system is highly ranked in economics because of the high thermal efficiency and plant simplification. The SCWR is rated good in safety, and in proliferation resistance and physical protection, and neutral in sustainability because of its open fuel cycle. (If the fast-spectrum option can be developed, the SCWR system will be ranked high in sustainability.) The SCWR system is estimated to be deployable by 2025 (with development cost of 870 million US$) [DOE 2002].

Almost all GIF members display a high interest in the development of the SCWR – almost as high as for the gas-cooled reactors.

SFR – Sodium-Cooled Fast Reactor System:

The SFR system consists of a fast-neutron reactor and a closed fuel cycle system. There are two major options: One is a medium size (150 to 500 MWe) reactor with metal alloy fuel, supported by a fuel cycle based on pyrometallurgical reprocessing in collocated facilities. The second is a medium to large (500 to 1,500 MWe) reactor with MOX fuel, supported by a fuel cycle based upon advanced aqueous reprocessing at a centralized location serving a number of reactors. The primary coolant system can either be arranged in a pool layout or in a compact loop layout. The outlet temperature is approximately 550 °C [DOE 2002; LINEBERRY 2002].

According to GIF, the SFR has the broadest development base of all the Generation IV concepts. The existing know-how, however, is based mainly on old reactors, which have already been

shutdown for various reasons (safety, economics, resistance from the population). Only three prototypes of sodium-cooled breeders were operating in 2004.

Because of its history, as well as because of the significant hazards of this reactor line (see section A), it is hard to understand why the SFR has been selected by GIF. According to GIF, research on both the fuel cycle and the reactor system is necessary to bring the SFR to deployment. Furthermore, there is important work to be done regarding safety. Key needs are to confirm reliability of passive feedback from heat-up of reactor structures and to establish the long-term ability to cool the oxide or metal fuel debris after a bounding case accident [DOE 2002].

The primary mission for the SFR is the management of high-level wastes. It is also hoped that the SRF becomes economically competitive as an electricity producer. The SFR system is top-ranked in sustainability because of its closed fuel cycle and potential for actinide management. It is rated good in safety, economics, and proliferation resistance and physical protection. The SFR system is estimated to be deployable by 2015 [DOE 2002].

VHTR – Very-High-Temperature Reactor System:

The VHTR system uses a thermal neutron spectrum and a once-through uranium fuel cycle. The reference reactor concept has a 600-MWth graphite-moderated helium-cooled core based on either the prismatic block fuel of the GT-MHR or the pebble bed of the PBMR. It is regarded as the most promising and efficient system for hydrogen production, either using the thermo-chemical iodine-sulphur process, or from heat, water, and natural gas by applying the steam reformer technology at core outlet temperatures greater than about 1000°C. The VHTR is also intended to generate electricity with high efficiency (over 50% at 1000°C). It is planned to drive the helium gas turbine system directly with the primary coolant loop. However, a high performance helium gas turbine has to be still developed. The VHTR requires significant advances in fuel performance and high-temperature materials [DOE 2002].

The VHTR is a next step in the evolutionary development of high-temperature gas-cooled reactors (HTGR). The technology is based on some decommissioned thermal spectrum HTGR pilot and demonstration projects, all of which had rather short and unsuccessful overall operating times, such as the small Dragon reactor experiment (20 MWth, 1966 -1975, U.K.), the AVR (15 MWe, 1967-1988, Germany), the THTR (308 MWe, 1986-1988, Germany) as well as the U.S. plants at Peach Bottom (42 MWe, 1967-1974) and Fort St. Vrain (342 MWe,1976-1989).

Furthermore, it is hoped that the concept could benefit from the experience gained with the Japanese HTTR and the Chinese HTR-10 projects, which are still in the test phase, as well as from the GT-MHR and the PBMR projects at present in the planning phase.

The Chinese HTR-10 is an experimental 10 MWth pebble bed HTGR. Full-power operation was achieved in January 2003. The test operation schedule was disturbed by the outbreak of the SARS epidemic in Asia in 2003 [NUCWEEK 38_03].

The HTTR (30 MWth), a prismatic core research reactor, is operating in Japan to demonstrate the feasibility of reaching core outlet temperatures of up to 950°C and to perform tests for hydrogen production. Under construction for eight years, the HTTR reached its first criticality in 1998. At the end of 2001, a core outlet coolant temperature of 850°C was reached for the first time. Operation at higher temperatures (950°C), however, will not be achieved in the short term. During operation in the last years, core temperatures reached higher values than anticipated. Therefore, it is to be feared that core temperature limits will be exceeded if outlet temperature is raised further [POHL 2002]. (For more information on GT-MHR,concerning PBMR, see section B.1.)

According to GIF, the VHTR system is ranked high in economics because of its high hydrogen production efficiency, and in safety and reliability because of the inherent safety features of the fuel and reactor. It is rated good in proliferation resistance and physical protection, and neutral in sustainability because of its open fuel cycle. The VHTR system is seen as the nearest-term hydrogen production system available, estimated to be deployable by 2020 [DOE 2002].

Other Projects Regarded as Generation IV

Apart from the concepts selected by GIF, and outside the framework of GIF proper, other concrete projects are under development which are labelled as "Generation IV" by their promoters. Two important examples for such projects will be briefly discussed here.

International Reactor Innovative and Secure (IRIS):

IRIS is a modular PWR (335 MWe) being developed by an international consortium including twenty-one organizations (industry, utilities, laboratories and universities) from ten countries. In 1999, Westinghouse formed the group to develop a reactor for deployment by 2015. It is intended to satisfy the Generation IV goals [NEI 2005; WESTINGH 2005].

IRIS has an integral configuration. The reactor vessel houses not only the nuclear fuel and control rods, but also all the major reactor coolant system components. This design enhances safety, because it eliminates external loop piping and thus, accidents involving a large loss of coolant (LOCAs) [NUCNEWS 2003]. On the other hand, the integral vessel is larger than a traditional PWR pressure vessel and flawless manufacture of such vessels, including inspection, is more difficult (see also section A, sub-section on BWRs).

The incorporation of the control rod drive mechanisms inside the vessel is eliminating the possibility of head failures. One of the improved features of the steam generator design is that the high-pressure primary coolant flows on the outside of the tubes. Thus, the probability of tube failure is reduced. On the other hand, inspection and maintenance of integral steam generators require new tools and methodologies.

IRIS was a candidate system under consideration in GIF. It was not selected as one of the six concepts for further development, however, because of the difficult inspection of the primary system [NERAC 2002]. It is remarkable that international efforts are nevertheless undertaken to further develop this "loser" system. This is probably for economic reasons as because of its simplified design requiring fewer pumps, valves, pipes, and other components, it is aimed to shut down the reactor for major maintenance only every four years. The entire reactor core is replaced after eight years; therefore the handling of spent nuclear fuel is not necessary at the plant.

The reactor requires uranium that is more highly enriched than that used in conventional reactors – 5 % for the first reactor core and 9 % for successive reactor cores. It is aimed to develop a 15-year operating cycle for the fuel, at 15 % enrichment [NEINST 2005]. With high burn-up fuel, the reactor uses up more of the fissionable material than does a conventional reactor, reducing the amount of nuclear waste produced. But currently, only fuel enrichment up to 5 % U-235 is licensed. Therefore, the IRIS first-of-kind-plant would use standard enrichment fuel (4.95%) in a fuel assembly that is practically identical to currently operating PWR.

The capability of employing high burn-up, long-life cores, which, together with the capability of operating four years without shutdown for maintenance, addresses the proliferation-resistance requirement, but, more important, increases the capacity factor and decreases the operation and maintenance costs [NUCNEWS 2003].

IRIS is offered in configurations of single or multiples modules. Within a multiple unit many systems and physical facilities are shared, including the control room. Thus, costs can be minimised; on the other hand, sharing of facilities creates new risks: The hazard of common

cause failures is increased, and the new hazard of critical situations arising because of mixing-up of units is created.

The core damage frequency for IRIS is estimated to be in the order of 10^{-8}/yr. It is clear that not all accidents necessitating an off-site emergency response are being completely eliminated. Nevertheless, it is intended to have IRIS licensed without any emergency response requirements, due to the claimed low probability of severe accidents – in spite of the fact that all probability estimates are beset with large uncertainties, particularly those below about 10^{-6}/yr. Thus, public acceptance is to be increased and the possibility to site IRIS closer to population centres opened. The NRC pre-licensing application will be completed by 2005. According to Westinghouse, no prototype is needed for design certification since IRIS does not represent a new technology, only new engineering [NUCNEWS 2003].

Gas Turbine Modular Helium Reactor (GT-MHR):

The GT-MHR is a modular gas cooled reactor, based on HTGR technology. It is being developed by General Atomics (USA) in partnership with Russia's Minatom, supported by Fuji (Japan) and is based on General Atomics' MHTGR (modular high-temperature gas-cooled reactor) concept that was developed in the 1980s. The initial application is intended to be the consumption of plutonium from dismantled weapons in conjunction with the generation of electricity. Future commercial deployment for electricity production using low enriched uranium fuel is anticipated. The current design is claimed to meet the Generation IV nuclear programme goals [LABAR 2004].

The GT-MHR module couples a gas-cooled modular helium reactor (MHR) with a modular gas turbine (GT) energy conversion system contained in an adjacent vessel. The reactor and power conversion vessels are located in a below ground concrete silo. The use of a direct gas turbine cycle instead of a steam-water circuit represents the only substantial point in which the GT-MHR has been developed further from the MHTGR. The main motivation behind this modification is to save costs – it increases the efficiency, simplifies the power plant and reduces the number of systems and components, in particular avoiding the need for steam generators [LABAR 2004].

The cylindrical core consists of 102 hexagonal fuel element columns of graphite blocks with channels for helium and control rods. Half the core is replaced every 18 months. The preliminary design stage was completed in 2001. Plant costs are expected to be less than 1000 US$ /kW. The development timeline is for a prototype to be constructed in Russia 2006-09, following regulatory review there [WANO 2005].

The GT-MHR will be built as modules of 285 MW_e each, four of which make up the full plant rated at 1140 MW_e. As in the modular IRIS concept, it is intended that systems be shared between modules, which gives rise to new, specific hazards (see above, sub-section on IRIS).

The GT-MHR produces less radioactive waste than other reactor options because of the plant's high thermal efficiency and high fuel burnup [LABAR 2004]. The difference, however, is not dramatic. (Compared to an LWR of the same size, a GT-MHR produces about 40 % of the actinides. Regarding waste handling, transport, and final disposal, this cannot be regarded as a significant difference.)

The proponents of the GT-MHR forward the same claims concerning safety, performance, and economy as do the proponents of the PBMR. As a matter of fact, however, the GT-MHR is beset by the same weaknesses as is the PBMR (see section B.1).

Regarding safety, key questions therefore remain unanswered, for example related to confinement requirements [HITTNER 2004]. As far as published information goes today, the GT-MHR appears to have no containment, only a "containment structure" the design of which is

not clear [GA 2005]. However, the feasibility of adding a containment to the MHTGR concept has already been studied by DOE in the 1980s – with the result that this would be possible, with the cost between 30 and 90 million US$ (1989), depending on the containment concept [MHB 1990]. If nevertheless no containment were indeed foreseen for the GT-MHR, this would clearly illustrate that cost cutting, not improvement of safety, is the main driving force behind the development of Generation IV reactors.

Evaluation of Generation IV; Conclusions

Unanticipated technical problems, accidents, the unsolved nuclear waste problem as well as the high costs of nuclear power, combined with lack of public acceptance, have lead to a decline of nuclear power. The nuclear share of electricity production is expected to decrease in most regions of the world [SCHNEIDER 2004].

This is the background for the Generation IV initiative of the U.S.DOE. A label is created which is to sell the illusion to the public that a completely new generation of reactors is being developed, which is free from all the problems which are plaguing current nuclear installations. DOE has been careful not to identify specific reactor designs that it considers compatible with Gen IV principles. It prefers to focus on general criteria [NUMARK 2000].

The message for the media, politicians and the population is: Generation IV means a safe, economically competitive, proliferation-resistant power source without the problem of increasing greenhouse gas emissions [INEEL 2003]. Generation IV is even presented as sustainable – a label which is usually, and with good reason, reserved for renewable energy sources, and conservation. The fact that none of the six reactor concepts selected for development fulfils all Generation IV aims is usually not mentioned.

Due to economic and political reasons, the era of a single company's or nation's developing and deploying a new type of nuclear plant has passed. Therefore, about two thirds of all countries with nuclear power plants are cooperating in the framework of GIF (other countries had been invited but declined to participate). The membership in this international forum commits participating countries to support long-term research efforts. This includes – via EURATOM – countries which are basically opposed to nuclear power.

Although the nuclear industry strongly supports the idea of a revitalization of nuclear power, there is considerable debate as to what reactor type should be used. In particular, among industry experts, there is a raging controversy between proponents of Generation III and Generation IV. Supporters of Generation III argue that there is no need to abandon today's mature LWR technology, in order to experiment with half-developed but "alternative" concepts [NUCWEEK 19_04].

In the end, it all comes down to selecting between two bad alternatives. The U.S. government is attempting a compromise and divides the (sparse) financial means devoted to the development of new generation reactors about equally between Generation III and Generation IV. Of the 300 million US$ for DOE's nuclear energy activities in fiscal 2004, the Nuclear Power 2010 initiative (concerning Generation III) received 20 million US$, and the Generation IV initiative 24 million US$ [NUCWEEK 46_03].

A primary goal of Generation IV lies in the securing of financial means for nuclear research. Today, nuclear power still receives a large amount of R&D money – half of the energy R&D budget (87.6 billion US$) spent by 26 OECD member states between 1991 and 2001 went to nuclear research; only about 8 % to renewables [SCHNEIDER 2004]. Gradually, however, a shift away from nuclear power is taking place. The Generation IV initiative attempts to reverse this shift by making nuclear energy attractive and presenting it as sustainable and CO_2-free – labels usually (and with justification) reserved for renewables.

This strategy will help the nuclear industry and nuclear research institutions to survive. Whether it will really lead to the development of new reactors remains highly doubtful. The estimated costs for the development of the six Generation IV concepts are about 6 billion US$ (about 600 to 1000 million US$ per system, plus about 700 million US$ for cross-cutting research) [DOE 2002]. It is more than likely that overruns will occur both for costs and for the time required. According to one of the strongest supporters of the GIF programme, the French government, Generation IV "will at best be ready for commercial deployment around 2045" [NUCWEEK 20_04], and not 2030 as officially envisaged by GIF.

This is to be seen before the background that nuclear energy is currently not cost competitive in the deregulated market; not with coal and natural gas [MIT 2003], and also not with wind energy. A recently published study demonstrates that for the same investment, wind generates 2.3 times more electricity than a nuclear reactor [GREENPEACE 2003].

As nuclear power generation has become established since the 1950s, the size of reactor units has grown from 60 MWe to more than 1300 MWe, with corresponding economies of scale in operation. Today there is a move to develop smaller units, which may be built independently or as modules in a larger complex, with capacity added incrementally as required. The driving forces for small NPPs are the reduction of the financial risk and the need for integration into smaller grids in many developing countries [WANO 2005]. The largest increase in nuclear generation is projected for the developing world, where a potential market for Generation IV is seen. However, an IAEA expert has voiced doubt concerning these prospects: Developing countries will not order new NPPs that have not demonstrated their constructability and operationability. They would not like to have completely new types of innovative NPPs unless they have been built and operated successfully elsewhere [NPJ 2002].

Furthermore, the opinion that the only way to make nuclear power cost competitive is the use of small modules is not shared by all nuclear industry experts.

Another attempt to improve the economics of nuclear power is to go into the production of hydrogen, which is envisaged for several of the Generation IV concepts. "Hydrogen is one of the three pillars of nuclear hopes for the future (the others are the need to phase out fossils fuels and the increased demand for power expected from developing countries)" [GORDON 2004].

According to GIF, a **closed fuel cycle** is celebrated as a major advantage of Generation IV concepts. A system with a closed fuel cycle is regarded as more effective, and sustainable.

However, not all of the six concepts selected for development employ a closed fuel cycle. The VHTR, most favoured, relays on an open cycle; and for the SCWR, once-through constitutes the nearer-term option. Furthermore, it is questionable whether it will actually be possible to successfully develop and implement the closed cycles.

Finally, the costs of such fuel cycle concepts would be very high. According to the recently published study "The Future of Nuclear" of the U.S. Massachusetts Institute of Technology [MIT 2003], a convincing case has not yet been made that the long term waste management benefits of advanced closed fuel cycles involving reprocessing of spent fuel are not indeed outweighed by the short term risks and costs, including proliferation risks. Also, the MIT study found that the fuel cost with a closed cycle, including waste storage and disposal charges, to be about 4.5 times the cost of a once-through cycle. Therefore it is not realistic to expect that there ever will be new reactor and fuel cycle technologies that simultaneously overcome the problems of cost, safe waste disposal and proliferation. As a result the study concludes that the once-through fuel cycle best meets the criteria of low costs and proliferations-resistance [NEI 2003c].

For thermal reactors, "sustainability" is to be achieved by higher enrichment. This, however, does not solve the waste problem. On the contrary – experts are pointing out that so-called high

burn-up fuel elements will lead to additional problems not only during reactor operation, but also during intermediate storage and final disposal [BORN 2002].

As was to be expected, short-term efforts will concentrate on thermal reactors. According to a recent announcement of the U.S.DOE, the GIF efforts have been divided into near-term Gen IV-A thermal systems that will use advanced high-burnup fuels and the longer-term Gen IV-B that will use fast reactors [FABIAN 2004].

All in all, Generation IV reactors are far away from the goal to successfully minimize and manage their nuclear waste.

In addition to not being economical, reprocessing separates plutonium, which is a serious proliferation concern. The Nuclear Control Institute (NCI) warned that transmutation of spent nuclear fuel is no guarantee against proliferation [ENS 2004]. Furthermore, the growing concerns about the safe and secure transportation of nuclear materials and the nuclear security of nuclear facilities from terrorist attacks is not adequately taken into account in any of the concepts.

Regarding proliferation, it is generally recognized that it is a practical impossibility to render civilian nuclear energy systems proliferation-proof. Thus, it cannot be expected that Generation IV will achieve a great leap forward in this respect [ANDERSON 2005].

The **basic concepts of the "new generation" have been around** as long as nuclear power, but they were forced out of the market in the early years by the light water reactors – not without reason, considering the experiences so far, which are dominated by technical and economic problems, and safety deficits.

In order to overcome these problems, materials, processes and operating regimes that are significantly different from those of currently operating systems or previous systems have to be developed.

Research and development are needed to confirm the viability and safety of new design approaches. Nuclear power plants are very complex systems that cannot be completely modelled in an accurate manner – in particular regarding passive systems that represent a marked departure from established concepts. The actual plant response to all conceivable events cannot be tested. The safety issues surrounding nuclear power are also especially difficult because of the potentially catastrophic and irreversible consequences of severe accidents.

Nuclear regulators in the U.S. are not enthusiastic about the new reactor concepts. New nuclear power plants should be based on evolutionary, not revolutionary, technology, according to an NRC commissioner. The commissioner cautioned against *"too much innovation"* which would lead to new problems with untested designs, and urged the industry not to *"overpromise"* the capabilities of new reactor systems [NNF 2005b].

Even nuclear industry representatives are very sceptical towards the Generation IV systems. "We now that the paper-moderated, ink-cooled reactor is the safest of all. All kinds of unexpected problems may occur after a project has been launched" [GÜLDNER 2003].

A closer look at the technical concepts shows that many safety problems are still completely unresolved. Safety improvements in one respect sometimes create new safety problems. And even the Generation IV strategists themselves do not expect significant improvements regarding proliferation resistance.

But even real technical improvements that might be feasible in principle are only implemented if their costs are not too high. There is an enormous discrepancy between the catch-words used to describe Generation IV for the media, politicians and the public, and the actual basic driving force behind the initiative, which is economic competitiveness.

The one concept that appears to be most innovative (the MSR) is not pursued with high priority, since the development costs are too high. Interest appears to focus on the VHTR, which is an evolutionary development of the HTGR line, with unresolved safety issues.

Another problematic aspect of Generation IV development is that it provides a reason to keep operating old and obsolete reactors – like, for example, the Phénix breeder reactor in France. This reactor is expected to play an important role in the development of the Generation IV international program. However, operating Phénix is controversial because of its age – it first achieved criticality in 1973 [NEI 2003a, b]. But it is the only European power reactor where experiments in the transmutation of long-lived radionuclides are conducted. The CEA is promoting the use of fast reactors for long-lived-waste transmutation in a future closed nuclear fuel cycle and intends to operate Phénix for another six cycles to 2009 [NUCWEEK 25_04].

What is the real motivation behind the Generation IV initiative? It seems that neither the nuclear industry, nor the electrical utilities believe in those new concepts. A revival of nuclear power is not to be expected – it will remain costly, is not competitive for hydrogen production and not suitable for developing countries. Is Generation IV a desperate attempt to get into hydrogen production in spite of all obstacles? Or is the goal simply to keep obsolete research installations running, which otherwise would be shut down due to safety concerns and lack of need? Is there a serious attempt to develop the HTGR technology, selling it as innovative while pursuing an evolutionary path? Or is it all only about an improvement of the image of nuclear energy, to be able to perform life extension of existing reactors while talking about Generation IV?

Fact is that substantial amounts of money are to be invested in an effort that does not solve at all the problems of nuclear power – money which could be to better uses.

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B.3: Problems of Fusion Reactors

Nuclear fusion was the subject of early dreams about safe and cheap energy: About 50 years ago, fusion proponents were convinced that it would be best to skip fission reactors because they are too dangerous, and go straight for fusion reactors. In the early 70s, it was expected that there would be "competitive fusion" by the year 2000 [HEINDLER 1995].

At the turning of the century, however, it was clear that an enormous development effort is still needed until a fusion power station can go into operation. Engineers aim at the middle of the 21st century for the first availability of an economical fusion reactor [SCHAPER 1999]. The EURATOM Scientific and Technical Committee prudently stated recently that it would take twenty years before it could be determined whether fusion is a viable option for electricity supply in the 21st century at all [STC 2003].

The timetable for the International Thermonuclear Experimental Reactor (ITER) reflects the scale of the efforts still required. ITER is merely a first step, still far away from a commercial fusion power plant: The feasibility of a self-sustaining fusion reaction is to be demonstrated. Present construction cost estimates for this effort lie in the range of 5 billion Euros, with operational costs expected to be similar. If construction begins in 2006, the plant is to be operational by 2014; about 20 years would then be required to complete the research work [NUCWEEK 08_04, 49_04]. It is questionable whether this schedule will be kept since the site for this project still has not been selected in early 2005 [NUCWEEK 06_05].

All in all, it is highly questionable whether fusion will be worth all the efforts planned for the next decades. Already, it is foreseeable that it will be neither safe, nor clean, nor proliferation-resistant.

The radioactive inventory of a fusion reactor is expected to be high, comparable to that of a fission reactor of the same size. Compared to a fission reactor, the fusion inventory is generally less toxic and shorter-lived because it consists of tritium and activation products, and contains no fission products and actinides. However, there are activation products with half-lives in the order of millions of years.

Accident mechanisms are different from those as fission power plants. Event sequences in a fusion reactor resulting from initiators like, for example, plasma disruption or loss of coolant flow are not completely understood [BÄHR 1995]. Reliable risk estimates will only be possible, if at all, when a concrete design for a commercial plant has been developed in detail. From the present viewpoint, accidents with catastrophic releases of radioactivity appear possible [SCHAPER 1999].

Tritium releases during normal operation are expected to be least ten times higher than those of pressurized water reactors; estimations vary by about three orders of magnitude [SCHAPER 1999]. Routine emissions of aerosols are could be higher than those of fission power plants by a factor of 100 to 10,000 [HEINDLER 1995].

Because of the high exposure to particle radiation and heat, the components of a fusion reactor which are facing the plasma must be replaced regularly. They constitute the main part of the radioactive wastes arising from fusion reactor and contain activated metal and tritium. Further radioactive wastes arise by contamination of other plant parts. The waste amounts correspond to the waste from a fission reactor or are even higher. As has been pointed out, on the other hand, the wastes are generally shorter-lived than those of nuclear fission. However, a particularly problematic nuclide often overlooked in analyses is beryllium-10, with a half live of 1.51 million

years. All in all, long-term radioactive waste management will be required for fusion plants [SCHAPER 1999]. They give rise to a waste problem comparable to that of fission power plants.

Regarding proliferation issues, a fusion reactor as such does not contain nor would produce materials usable for fission weapons. However, the neutron radiation could be employed to produce plutonium or uranium-233. The hard energy spectrum of fusion neutrons is particularly "favourable" to the production of high-quality weapon plutonium, much more so than the neutron spectrum of a light water reactor.

The fuel of a fusion reactor as envisaged at present consists of deuterium and tritium. Tritium is not required for simple fission bombs. However, advanced nuclear weapons ("boosted" weapons) employ tritium to increase the strength of the fission chain reaction by additional fusion neutrons. For states with an advanced military nuclear program, therefore, a fusion reactor could be interesting from the viewpoint of tritium supply.

Furthermore, some processes involved in fusion research are similar to those in a hydrogen bomb. Therefore, there are fusion experiments that could be "helpful" in the development of thermonuclear weapons.

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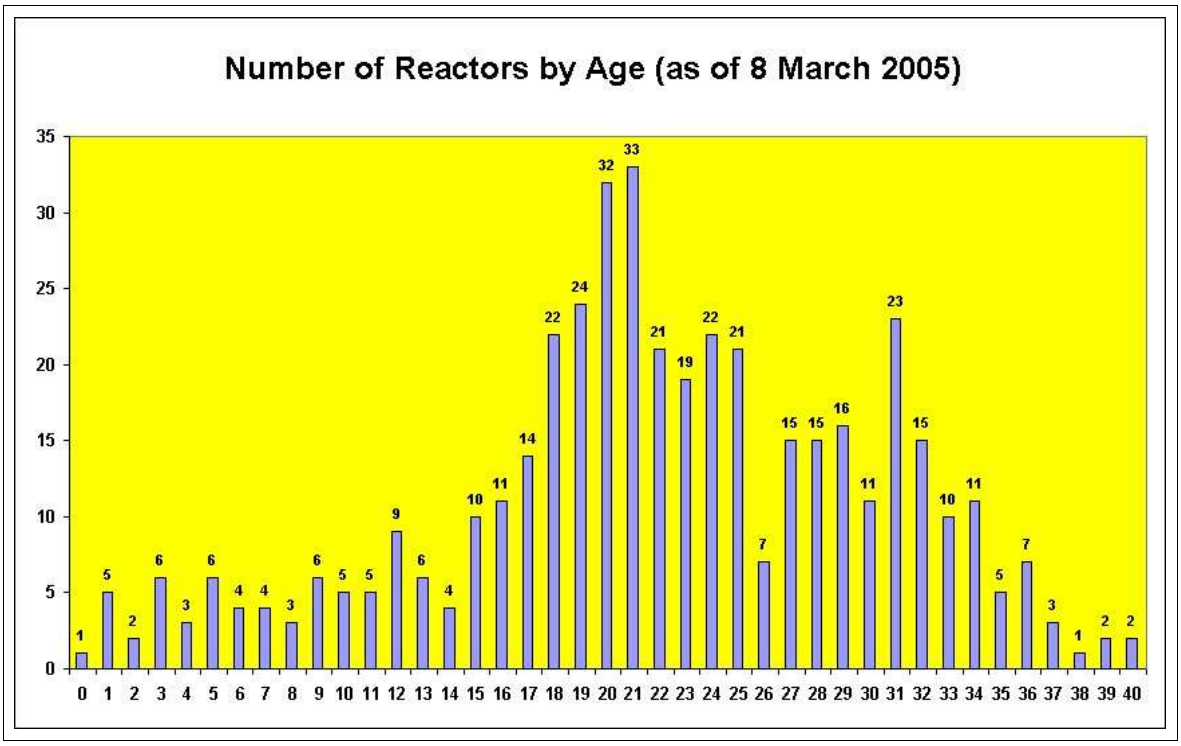
C: Ageing, PLEX and Safety

Introduction and Overview

There is general consensus that the extension of the life of reactor is of the foremost importance today for the nuclear industry. The International Energy Agency pointedly sums it up as follows [IEA 2001]: *"If there are no changes in policy towards nuclear power, plant lifetime is the single most important determinant of nuclear electricity production in the coming decade."*

Across the world over the last two decades there has been a general trend against ordering new reactors. This has been caused by a variety of factors; fear of a nuclear accident, following the Three Mile Island, Chernobyl and Monju accidents; historic over capacity of generation; increased scrutiny of economics and financing of nuclear power with the introduction of liberalised electricity markets; and environmental factors, such as waste management or radioactive discharges. As a consequence of this lack of orders the average age of nuclear reactors have increased year on year and is now 21 years old [Schneider 2004].



Source: IAEA, PRIS, 2005[5]

At the time of their construction it was usually assumed that the reactor would not operate beyond 40 years. However, now, in order to retain the nuclear share of the electricity supply and to maximise profits – with in theory the large construction and decommissioning costs paid for- life extension offers an attractive proposition for the nuclear operators.

What is Ageing?

In any industrial plant, material properties are deteriorating during operation due to the loads the components are subjected to. The International Atomic Energy Agency (IAEA) defines ageing as

[5] http://www-ns.iaea.org/conventions/nuclear-safety.htm

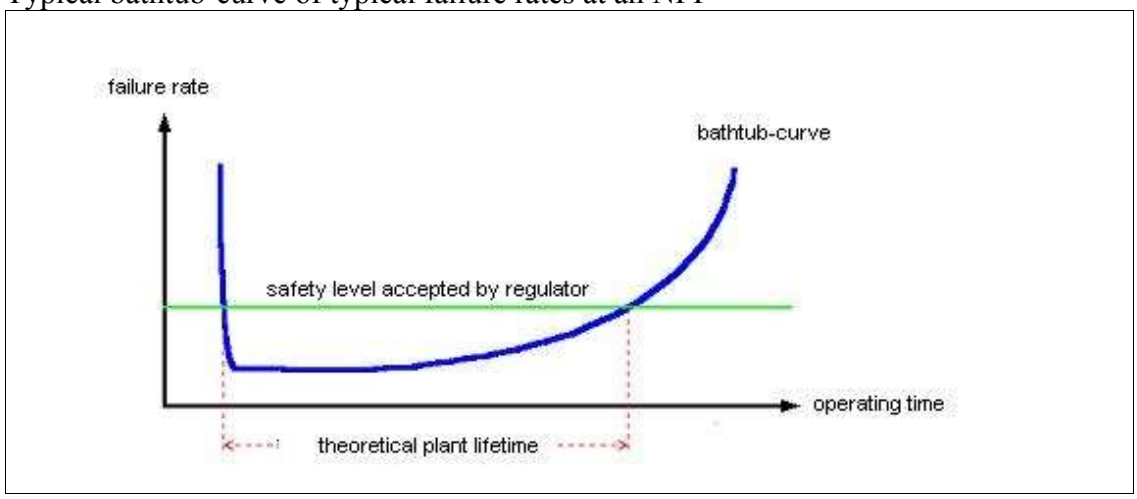
a continuous time-dependent loss of quality of materials, caused by the operating conditions [IAEA 1990].

Ageing processes are difficult to detect because they usually occur on the microscopic level of the inner structure of materials. They frequently become apparent only after a component failure, for example break of a pipe, has occurred.

Failure rates generally are higher after start-up of a plant, when construction errors or design shortcomings become evident. In this phase, considerable efforts are usually undertaken to correct all problems, since there is a high economic incentive to achieve smooth plant operation as soon as possible.

During the 'middle age' of a plant, problems tend to be at a minimum. Later, as ageing processes demand their due, there will be a gradual increase of failure rates. The result is a "bathtub-curve" as shown in the figure below:

Typical bathtub-curve of typical failure rates at an NPP



This is a process that is not always easy to recognise and to follow, and which increases plant risk considerably. For a nuclear power plant, whatever the reactor type, the ageing phase will begin after about 20 years of operation. This, however, is a rule-of-thumb number only and ageing phenomena can begin earlier.

As the world's nuclear power plant population gets older, there are efforts to play down the role of ageing. Those efforts include conveniently narrowing the definition of ageing. In a German study of the late 90s, ageing-related damages are limited to damages caused by unforeseen loads during operation, in spite of design and operation being in accordance with the requirements. Damages occurring later in plant life because design, manufacturing, commissioning or operation is not in accordance with requirements are not regarded as ageing-related [LIEMERSDORF 1998].

On this basis, according to a recent study, only a small percentage of failures in German nuclear power plants appear to be due to ageing. This restriction, however, is not acceptable. The restriction is artificial and arbitrary since it is not possible to completely avoid deviations from requirements in components of nuclear plants; their possibility always has to be taken into account during plant operation. On the other hand, failure to foresee a load in the design phase could well be regarded as a design error.

Thus, ageing will be understood in a comprehensive manner here, according to the IAEA definition quoted above.

PLEX and PLIM

PLEX (Plant Life Extension) appears to be a fairly straightforward and a clear concept. The definition of life extension, however, gets somewhat muddled if the definition of plant life time itself is less then clear.

This does not apply to the USA where operating licenses are granted for 40 years and life extension clearly begins after this time. In the UK, operating periods are likewise fixed (for example, originally 30 years for the Hinkley B AGRs). Similar rules apply in Russia and Eastern and Central European countries.

In many countries, on the other hand, operating licenses are not explicitly limited in their validity. Assumptions concerning the lifetime are usually contained in the proof of safety for a nuclear power plant, giving, potentially a considerable amount of flexibility. However, there often is a requirement for a decennial safety review of the plant.

In France, for example, 30 years appear have been generally recognised as the expected plant lifetime in the past; longer periods are now under consideration. The situation is similar, e.g., in Spain and Germany. In the latter country, there is a ceiling for the amount of electricity to be produced by each NPP, roughly corresponding to the commercial lifetime as generally envisaged. Amounts of electricity can be transferred, however, from older to newer plants, thus providing leeway for life extension.

The distinction between measures to make sure that the lifetime originally planned is reached, and measures aiming at prolonging the lifetime becomes unclear in many countries – particularly so since the measures are basically the same in each case. This complication has to be well kept in mind.

Nuclear power plant operators and their technical support organisations appear to take advantage of this situation by employing wording which is deliberately enigmatic. “Plant Operational Life” increasingly is not regarded as a pre-defined period, with the consequence that *“[the] term PLEX is now falling out of favour because, of ‘Plant Life’ or ‘Operational Life’ is undefined, it is then illogical to talk about plant life extension.”* Instead, the general term PLIM (Plant Life Management) is employed which can relate either to activities required to keep a plant in an adequate condition to meet the intended amortisation period, or to activities which aim to extend plant operation in time [LMD 2002].

Thus, the intention to operate a plant beyond the original design lifetime can be hidden in PLIM measures. For greater clarity, therefore, we will use the term PLEX in this report, even if it is ill favoured in some circles.

Phenomena of Ageing

Ageing already occurs during the period usually regarded as typical commercial lifetime (30 to 40 years). Naturally, with PLEX, ageing mechanisms will become increasingly important over the years, contributing significantly to overall plant risk.

The most important influences leading to ageing processes in a nuclear power plant are [MEYER 1998]:

- Irradiation
- Thermal loads
- Mechanical loads
- Corrosive, abrasive and erosive processes

- Combinations and interactions of the processes mentioned above

Changes of mechanical properties frequently cannot be recognised by non-destructive examinations. Therefore it is difficult to get a reliable, conservative assessment of the actual state of materials. In many cases, non-destructive examinations permit to monitor crack development, changes of surfaces and wall thinning. However, because of limited accessibility due to the layout of components and/or high radiation levels not all components can be examined one hundred percent. Therefore, it is necessary to rely on model calculations in order to determine the loads and their effects on materials. Those models can only be validated with the aid of simplified systems, samples or mock-ups. Non-quantifiable uncertainties remain. Not even the most complex calculations can cover all conceivable synergistic effects.

With increasing age of plants, damage mechanisms might occur which have not been foreseen, or which had even been excluded (for example, stress corrosion cracking in titanium-stabilized austenitic steels), exacerbating the ageing problems.

The measures to monitor and control ageing processes are known as ageing management. Ageing management consists of programmes with accelerated samples, safety reviews and also the precautionary exchange of components in case cracks or other damages have been found during inspections. Furthermore, it includes optimizing of operational procedures in order to reduce loads. In the United States, a specific ageing management programme for reactor pressure vessels (time limited ageing analyses) has been developed [RINCKEL 1998].

New, integral methods for the monitoring of NPP operation have been developed in the late 90s which attempt to predict the future behaviour of individual components on the basis of limited, known information. This was motivated by the increasing age of nuclear power plants worldwide as well as by the general trend towards life extension. The aim is, at the one hand, to arrive at inspection programmes which are economically more efficient and save time; on the other hand, actual failures are to be avoided to keep down-times short, improving economy and safety in parallel [ALI 1998; BARTONICEK 1998; BICEGO 1998; DUTHIE 1998; ESSELMANN 1998; HIENSTORFER 1998; ROOS 1998].

Ageing Effects at Specific Components

Ageing can occur in many different manifestations at different components. The most important ones are, for light water reactors (PWRs including VVERs, and BWRs):

Reactor Pressure Vessel:

- Materials close to the core: Embrittlement (reduction of toughness, shift of the ductile-to-brittle-transition temperature) through neutron irradiation. This effect is particularly relevant if impurities are present. Copper and phosphorus favour embrittlement, as well as nickel at very high neutron fluences as encountered at VVER reactor vessels. Neutron embrittlement is mostly relevant for PWRs. Because of a potential flow rate effect (higher damage at lower flow rates, for a given overall dose), it can also become relevant for BWRs.
- Welds: Crack growth because of changing thermal and mechanical loads. For PWRs, this occurs mostly in embrittled welds close to the core; for BWRs, in longitudinal welds.
- Vessel head penetrations: Crack formation and growth due to corrosion mechanisms; concerns PWRs [MEYER 1998].
- Penetrations of vessel bottom: Damages due to corrosion, abrasion and thermo-mechanical fatigue; concerns BWRs.

- Inner edge of nozzles: Strong concentration of stresses because of varying wall thickness, with changing thermal and mechanical loads as well as corrosion and erosion effects. This leads to the hazard of crack formation or growth of cracks which were formed during vessel production. The situation is exacerbated by the fact that inspections are hindered by geometric lay-out and high wall thicknesses. Relevant for PWRs and BWRs.
- Core internals, core shroud: Embrittlement due to high neutron fluences, as well as damages by corrosion and erosion. Can only be inspected visually. If materials containing cobalt are used, there is the additional hazard of activated cobalt getting into the cooling water – leading to contamination problems, for example at refuelling. Relevant for PWRs and BWRs.
- Bolts and nuts under pressure: Localised leakages of the borated coolant (for example, due to stress relaxation and accordingly, reduction of stretch elongation) can lead to corrosive damage at flange surfaces and screws' materials. Relevant for PWRs and BWRs.

Pipelines:

Cracks have been found in titanium-stabilized austenitic steel pipes of all Geman BWRs, which are mainly due to stress corrosion cracking [ERVE 1994]. In all pipelines which have not been exchanged, further damage is possible. Because of the more benign water chemistry, damage due to stress corrosion cracking is not expected in PWRs. However, strain-induced corrosion and erosion corrosion are possible at longer operating times. Apart from mechanical loads, there is increasing influence of thermal loads which are not sufficiently known (i.e. because of thermal layering) and which are higher than assumed in plant specifications [ZAISS 1994]. Wall thinning and material's fatigue because of resonance vibrations, water hammer etc. are very difficult to keep under surveillance. For all those reasons, damages become more likely with ageing of materials.

Particularly regarding materials' behaviour in austenitic pipes in BWRs, there are still open questions today. Not all damage mechanisms occurring are completely understood, for example crack propagation due to stress corrosion cracking in materials which have not been thermally sensitized [KILLIAN 2000]. This implies that the occurrence of new, unexpected phenomena cannot be excluded.

In connection with pipe failures, the leak-before-break criterion is increasingly relied upon. According to this criterion, leakages can be detected, before a dangerous break occurs. However, complete "guillotine" breaks have already occurred at nuclear power plants – for example, in Surry 1987 as well as in Loviisa 1990, where there was a break in the secondary circuit without leakage beforehand [AHLSTRAND 1991]. In February 1992, sudden break of the thermally embrittled feedwater pipe at the conventional power plant Kardia-1 (Greece) occurred [JANSKY 1993]. Therefore, it is to be feared that under unfavourable circumstances, breaks without preceding leak can occur.

Main Coolant Pumps:

Crack formation and crack growth can occur due to thermal and high-frequency fatigue processes, supported by corrosive influences. Inspections are difficult. This problem concerns PWRs and BWRs. In nuclear standards, ASME (USA) as well as KTA (Germany), corrosive influences seem to have been underestimated when determining the design curves for fatigue [RINCKEL 1998]. Therefore, in spite of assumed sufficient long-term strength, there have been breaks of pump shafts after comparatively short operating times (for example, four years) [SCHULZ 1987].

Steam Generators:

Corrosive and erosive damage as well as wall thinning in the steam generator tubes have lead to comprehensive ageing management activities worldwide. In the last years, this increasingly includes exchange of the whole component [MEYER 1998]. Of course, the problem only applies to PWRs and is particularly severe for VVER-1000 reactors (see section A).

Turbines

Ageing phenomena because of corrosion, erosion and thermo-mechanical fatigue are to be expected for the turbine casing, the turbine shaft and turbine blades. Large forged pieces always contain in-homogeneities (inclusions, segregations, small cracks), which can lead to damages due to the influences mentioned. Embrittlement has been observed at turbine shaft materials (12Cr-steel and stellite 6B) because of erosion due to liquid phase impact [LEE 1998].

Concrete Structures:

Structural components like the concrete parts of the containment, protective outer hulls of buildings, biological shields, basis structures and cooling towers are subject to thermo-mechanical loads, but also to effects of the weather, chemical attacks and partly also to high radiation doses. This is relevant for PWRs and BWRs.

Corrosive damage of steel reinforcements are difficult to inspect. Hence reductions in strength may occur unnoticed. The damage mechanisms to concrete through corrosive processes similar to high radiation doses are still largely unknown. It is particularly difficult to quantify the uncertainties of the models that were developed, and to validate those models with experimental data [NAUS 1996].

In the United States, a data bank (Structural Materials' Information Center) has been compiled in order to assess environmental influences and ageing factors for concrete. A comprehensive study on the ageing of French cooling towers lead to the conclusion that the design lifetime of 40 years is likely to be reached, safety margins, however, are considerably smaller than assumed [BOLVIN 1993]. In Switzerland, a systematic ageing surveillance programme for NPP structures was begun 1991 [ZWICKY 1993].

Seismic safety analyses generally are performed with design material parameters. So far, little notice has been taken of the weakening of structures through ageing in this context, in spite of the importance of this issue: *„The evaluation for seismic loading is particularly important because the degraded structures or components could be more vulnerable to the seismic loads. From a seismic analysis point of view, the aging or degradation may affect dynamic properties, structural response, resistance or capacity, failure modes, and locations of failure initiation."* [SHAO 1998].

Cables:

To begin with, the mechanical stability deteriorates when cables age, due to embrittlement of the isolating layers. At first, the electrical properties are not influenced, even if cracks have formed. However, an aged cable with cracked isolation constitutes a hazard in humid or chemically aggressive surroundings, particularly in case of accidents [SLITER 1993].

Electronic Devices:

In an NPP, many electronic devices are being used. Temperature and radiation are the main factors leading to ageing. Additional degradation can occur due to humidity and chemical attacks. Because of the great variety of different devices and the complex ageing phenomena, which have not been systematically investigated so far, reliable lifetime estimates are very

difficult. The possibility of flow rate effects, particularly in semiconductor elements, constitutes an additional hazard [IAEA 1990]. With increasing age of a plant, the reliability of electronic devices can thus be reduced – while at the same time, safety margins in the whole system are decreasing.

Most of what has been said above about light water reactors (PWRs and BWRs) also applies to other reactor types.

Beside those explicitly mentioned here, all other components of an nuclear power plant are also, in principle, subject to changes of their material properties due to ageing, and thus to a reduction in functionality. The operational hazard, which increases over time, is exacerbated by the combination of all those negative changes that defy quantitative modelling and estimation. In the course of maintenance and ageing management, NPP operators have reacted to damages by repairs and exchange of components. Nevertheless, experience shows that again and again, unexpected ageing-related damages occur – for example, the graphite cracking discovered in British AGRs in 2004, or the cracking of austenitic steel pipes in German BWRs in the early 1990s (see section A). Austenitic steel is a type of steel optimized for corrosion resistance.

Embrittlement is a particularly severe problem for pressure tube reactors like CANDUs and RBMKs, since the tubes' material is located inside the core and hence, subjected to a particularly high neutron flow. Extensive programmes of pressure tube exchanges have been implemented for both those reactor types (see section A).

Reactors with graphite moderator are subject to the specific problems of graphite ageing. Graphite cracking in AGRs has recently been observed, which can be a hazard to core integrity. In RBMKs, graphite swelling leads to gap closure (see section A).

Ageing constitutes a particularly severe problem for passive components, i.e. components without movable parts. Not only is it often difficult to detect ageing phenomena. Replacement usually was not expected for components like pipelines or graphite parts, and no provisions made for it.

Regarding active components like pumps and valves, deterioration usually manifests itself in an obvious manner, and exchange of components can often be performed during regular maintenance work. Nevertheless, ageing of active components cannot be neglected as a risk factor, as the possibility of catastrophic failures of main coolant pumps and turbines illustrate. In electronic and electric devices, too, damage can cumulate unnoticed until the point is reached when a dramatic failure occurs.

Various individual ageing-related problems have been studied in some detail in the past. A number of mechanisms are known; nevertheless, they are not completely understood.

For example, the dose rate effect in steel irradiation embrittlement, already mentioned several times above, has been known for many years. But it still cannot be described reliably and quantitatively today, giving rise to an increased risk of pressure vessel failure in older NPPs. Another problem not fully understood is the propagation of cracks in austenitic steel pipes.

The lack of complete knowledge in crucial areas is of course exacerbated when plant life is extended. For example, regarding the prediction of neutron embrittlement, there are standard surveillance programs for monitoring reactor pressure vessels during their design life (usually, up to 40 years of operation).

In Spain, where plant operators are considering to extent service life from 40 to 60 years, it has been shown that it is necessary to introduce modifications in the present surveillance programs in order to achieve a more precise reactor pressure vessel integrity evaluation [BALLESTEROS 2004]. This is highly problematical since surveillance programs require irradiation of samples

over years and, to yield most reliable information, have to be planned before start-up of a reactor and not after decades of operation.

Furthermore, ageing processes can have far-reaching effects into other fields that are not immediately obvious. For example, a U.S.NRC-initiated working group (working group an "Fire Induced Damage to Electrical Cables and Circuits) found that in ageing plants with deteriorating cable isolation materials, short-circuits and subsequent cable fires seem to appear with increasing frequency. This can lead, for example, to erroneous actuation of safety-relevant valves and creates higher demands regarding fire protection measures [RÖWEKAMP 2004].

Consequences of Ageing Processes

The consequences of ageing can roughly be described as two-fold. On the one hand, the number of incidents and reportable events at an NPP will increase – small leakages, cracks, short-circuits due to cable failure etc. In Germany, for example, the ten older plants (out of 19 NPPs in operation) are responsible for about 64 % of all reportable events in the time span 1999 – 2003 (severity of the events taken into account) [BMU 1999 – 2003].

On the other hand, there are effects leading to gradual weakening of materials which may never have any consequences until the reactor is shut down, but which could also lead to catastrophic failures of components with subsequent severe radioactive releases. Most notable among those is the embrittlement of the reactor pressure vessel, increasing the hazard of vessel bursting. Failure of the pressure vessel of a PWR or a BWR constitutes an accident beyond the design basis. Safety systems are not designed to cope with this emergency. Hence, there is no chance that it can be controlled. Furthermore, pressure vessel failure can lead to immediate containment failure as well, for example through the pressure peak after vessel bursting, or the formation of high-energy fragments. Catastrophic radioactive releases are the consequence.

Pressure tube embrittlement of RBMK or CANDU reactors also falls into the category of ageing processes with potentially catastrophic consequences. In case of failure of a single or a small number of tubes, there is a chance that the accident can be controlled – but not with large numbers failing.

Another example are corrosion processes which may be overlooked for years – as a recent event at the U.S. pressurized water reactor Davis Besse illustrates (see appendix of this section).

In probabilistic risk assessment studies (PRAs), which are increasingly used as a tool by nuclear regulators, ageing is usually not taken into account. PRAs assume that equipment failure rates are taken from the low center portion of the "bathtub curve". This leads to underestimation of the risk [LOCHBAUM 2000]. There are some attempts to include ageing in such studies, for example in a recent PRA of Beznau NPP (PWR, Switzerland). However, the consideration of ageing appears to be incomplete, and the available information is somewhat contradictory [FEA 2004]. Since some ageing mechanisms are still not completely understood, as has been pointed out above, a complete and satisfactory treatment of ageing effects in the framework of a PRA is not possible today and would require extensive further research.

Thus, it is clear that the risk of a nuclear accident grows significantly with each year, once a nuclear power plant has been in operation for about two decades. But it is not possible to quantitatively describe this continuous increase of risk. Increased vigilance during operation and increased efforts for maintenance and repairs have the potential to counteract this tendency, at least to some extent. However, in the age of liberalization and growing economic pressure on plant operators, the trend rather goes in the opposite direction, even as the reactor fleet is ageing.

Counter-Measures

In principle, the same measures are required to counter non-anticipated ageing phenomena during the planned lifetime of an NPP, and to extend this lifetime.

When discussing countermeasures to ageing, a distinction has to be made between replaceable and non-replaceable components. There is a wide consensus among plant operators that in principle, all components crucial for safety in PWRs or BWRs can be replaced except two: The reactor pressure vessel (RPV), and the containment structure. For the ex-Soviet reactor type VVER-440, steam generator replacement also does not seem to be feasible due to the so-called box system [LMD 2002].

The reactor pressure vessel mostly is regarded as the component decisive for limiting a nuclear power plant's lifetime. Therefore, in recent years, investigations have been performed whether RPV replacement could not be possible after all. Siemens studied this option [WISE 1998]; a feasibility study for a BWR was also undertaken in Japan [DAISUKE 1999]. The result of the latter was that an integrated judgment was needed for RPV replacement that lay outside the scope of the study, but technical feasibility was confirmed. All in all, however, RPV replacement is not an option seriously considered at the moment; pressure vessels are generally considered to be irreplaceable [LMD 2001].

RBMKs and CANDUs have an advantage in this respect since their pressure tubes can be exchanged; indeed, extensive refurbishment programmes have already taken place. They are, however, costly and time-consuming. The lifetime of a pressure tube is considerably shorter than that of the average pressure vessel, because tubes are subject to considerably higher neutron influences.

For the counter measures available, four levels generally can be distinguished:

- Exchange of components: This is the only option – apart from permanent shutdown – in case of obvious shortcomings, leakages developing and other problems that directly influence the power plant operation. Even large components like steam generators and reactor pressure vessel heads (as well as pressure tubes) can be exchanged.
  The costs of measures at this level usually are high. Exchange of components also includes the generation of additional radioactive wastes.
- Reduction of loads: This applies primarily to the reactor pressure vessel. To avoid thermal shock emergency cooling water can be preheated. To reduce neutron irradiation (and hence the progress of embrittlement), neutron fluency in the vessel wall can be reduced by putting dummy elements or highly burnt-up fuel elements in outer core positions. In principle, measures of this kind could also be applied to other components – however, they can run counter to the trend for power uprating. Costs are moderate at this level.
- Intensify inspections and plant monitoring: Ageing effects in materials can be 'compensated' by more frequent examinations and/or be intensification of plant monitoring, coupled with appropriate maintenance, on the optimistic assumption that cracks and other damage and degradation will be detected before they lead to catastrophic failure.
  The costs of such measures are relatively low, particularly regarding plant monitoring.
- Reduce safety margins: By reducing conservatism in proofs of safety, longer lifetimes result – at least on paper.
  Such measures are cheaper than all other options.

The option to repair components has not been included here since repairs are largely part of the measures required regularly during plant operation anyway, independent of PLEX. One noteworthy exception is the annealing of reactor pressure vessels as practised in Eastern and

Central Europe, a method to reduce embrittlement that is however questionable regarding the longer-term benefits, since there is no sufficient knowledge to date on the re-embrittlement behaviour of a vessel after annealing.

Most recent publications on ageing emphasize, on a general level, that the counter-measures practised are adequate to control the effects of ageing. On the other hand, this conclusion is strongly qualified, if not refuted, by frequent statements that further investigations into ageing issues are urgently required.

For example, a French/German publication [MORLENT 2001] states that according to international analyses, there is a trend towards more and more ageing-related events, requiring further investigation. Also, *"operating experience has shown that new insights concerning the assessment of the ageing behaviour of [structures, systems and components] may come to light in the course of time. It is therefore seen as a necessity that the investigations performed are continued in order to obtain indications of any safety-significant ageing-related changes at an early stage"*.

Under present circumstances, economic pressure is severe to the extent that even inspections are being reduced – the opposite of what would be required for ageing control. This is combined with general cost-reduction strategies of nuclear utilities because of the liberalization of the electricity markets, accompanied by deregulation and increased competition. It is claimed that intensification of plant monitoring can be a sufficient replacement for inspections [SCHULZ 2001]; however, this claim rather appears as an attempt to mask the reduction of safety margins, and is by no means reassuring.

Increasingly, on-site storage of spent fuel is practised or being implemented for lack of alternatives (in the USA, Germany, Central and Eastern European countries and others). In the countries concerned, a necessary precondition for PLEX, which has received very little attention so far, is the increase of storage capacity, leading to a corresponding increase of the radioactive inventory at the site.

PLEX Programmes World-wide

Country	No of reactors	Average Age	Original	Plans	Notes
Argentina	2	25			No information available
Armenia	1	24	30	30	Medzamore, VVER 440-230, unlikely to be life-time extension
Belgium	7	25	30	40	Political Agreement in 2003 limits operating life to 40 years.
Brazil	2	12			Not yet an issue
Bulgaria	4	20	30		Political agreement for closure of 1-4. To early to assess closure of 5 and 6
Canada	17	22	30		Degradation problems forced the temporary closure of 8 reactors in the late 1990s. How these will operate, and the other Candu reactors, will determine operating life.
China	11	5			Not yet an issue
Czech Republic	6	13		40	An extensive modernisation programme is underway to allow the Dukovany reactors to operate for 40 years
Finland	4	25	30	60	The Olkiluoto plant has already undergone technical changes to allow it operate for 40 years with plans being developed to enable it to operate an additional 20 years.
France	59	20	30	40	The are definitive plans to allow all reactors to operate for 40 years
Germany	18	25		32	A political agreement reached with the utilities will see the average operating life of reactors restricted to 32 years of operation.
Hungary	4	20	30	50	Measures are being introduced to allow the Paks facility to operate for 50 years.
India	14	17			It is reported that Plant life extension activities are progressively being implemented at some plants, although little specific information is available.
Japan	54	24		60	The utilities operating license has no definitive end point. MITI is current investigating proposals to allow reactor to operate for 60 years.
Korea, Republic of	20	13			Proposals are being developed to extend the operating live to upto 60 years
Lithuania	1	18			The remaining reactor is scheduled for closure in 2009, after 22 years of operation as part of its Accession Partnership Agreement.
Mexico	2	12			Not yet an issue
Netherlands	1	32		40	The Borsselle plant has undergone retrofitting and is now intended to operate until 2013
Pakistan	2	19	30	45	The Kanup reactor has undergone Plex to allow it operate an additional 15 years.
Romania	1	9			Not yet an issue
Russian Federation	31	24			The St Petersburg RBMK reactors are undergoing a second re-tubing exercise, which will allow them to operate for 45 years. Similar changes are expected in other similar reactor designs.
Slovak Republic	6	17			The oldest reactors at Bohunice V1 are scheduled to close by the end of 2008 as part of the Slovakian Accession Partnership Agreement.

Slovenia	1	22		40	No plans exist to operate the reactor beyond its 40 year expected life.
South Africa	2	20		40	No plans exist to operate the reactor beyond its 40 year expected life.
Spain	9	23	40	60	The oldest reactor, Jose Cabrera is scheduled for closure in 2006 after 37 years operation.
Sweden	11	26			All reactors were supposed to be closed by 2010 as a result of a referendum, however, this closure schedule is no longer likely and a reactor by reactor assessment is made.
Switzerland	5	30			Some reactors have indefinite licences to operate others have been granted 10 year licences, no operating life-times have been set.
Taiwan	6	23			
Ukraine	15	16	30		Plans have been developed to upgrade and extend the operating lives of all the VVER 1000s.
UK	23	26			All the Magnox reactors now have a fixed operating live time, of upto 50 years. The AGRs (second generation) are likely to have limited Plex (upto 5 years).
US	104	22			The first 40 years operating licenses will expire for three plants in the year 2009. Of the remaining 100 operating plants, 23 will have licenses expire by 2015[6]. Reactors that have received 20 year life extension: Calvert Cliffs (1&2); Oconee (1,2&3); Arkansas Nuclear One 1; Edwin I Hatch (1&2); Turkey Point (3&4); Surry (1&2); North Anna (1&2); McGuire (1&2); Catawba (1&2); Peach Bottom (2&3); St Lucie (1&2); Fort Calhourn; Robinson 2; Ginna; Summer; Dresden (2&3); Quad Cities (1&2)

Source: IAEA PRIS Database [7]and CNP Profiles[8]

As can be seen from the overview on PLEX programmes presented above life extensions are planned in most countries operating nuclear power plants, for many NPPs. The overview presented in the table is supplemented here by several important examples.

In the United States, licensing for life extension from 40 to 60 years is now well under way, after a difficult start-up period. At first, safety and licensing problems had interacted with economy in ways not anticipated by the applicants, as became apparent in case of the U.S. lead plant Yankee Rowe, where a license renewal procedure ironically lead to premature shut-down in 1992 (for details, see appendix to this section). This gave rise to considerable uncertainty among U.S. reactor owners [NUCWEEK 50_97]. There is increasing optimism, however, since the first extensions were granted in spring 2000 (for Calvert Cliff and Oconee) and license renewal appears to be speeding up in the USA [NUCWEEK 41_00].

In France, there are definite plans to extend the lifetime of the whole PWR fleet from 30 to 40 years [NUCWEEK 40_03]. Lifetime extension will begin at the earliest, 900 MWe-series PWRs for which the 30th year-outage could become the springboard to life extension. The backfits required are already being planned and will be implemented from 2008 onwards, when the head-of-series unit will have reached 30 operating years [NUCWEEK 45_03].

[6] Reactor License Renewal: FACT SHEET, US NRC, download March 2005.
[7] http://www.iaea.org/programmes/a2/index.html
[8] http://www-pub.iaea.org/MTCD/publications/PDF/cnpp2003/CNPP_Webpage/pages/countryprofiles.htm

For the Olkiluoto plant in Finland (two BWR units), life extension was performed, after a modernization programme 1994-1998, from 30 to 40 years. It is planned to gradually validate the units for a total lifetime of 60 years [RASTAS 2003].

Life extensions are also planned in South Korea, Sweden (to up to 60 years), India and other countries. Even for NPPs with Soviet-designed reactor types, PLEX is already under way or planned. The operating lifetime of Paks nuclear power plant in Hungary (four units of second-generation VVERs) is to be increased from 30 to 50 years, i.e. longer than for many western reactors [NUCWEEK 47_04]. The Ukrainian government recently has approved a comprehensive programme for life extension of the 13 nuclear units operating at four power plants. Lifetime is to be increased by 10 to 15 years [NUCWEEK 23_03; NUCWEEK 22_04].

Life extension plans do not even stop for the most hazardous and obsolete Soviet reactor types, first-generation VVERs and RBMKs. For example, in Russia, operating lifetime of the two Kola VVER-440/230s is to be extended by 15 years [NUCWEEK 33_04]. Leningrad-1 RBMK reached its design lifetime at January 2004. After a "modernization program" which was completed in October, lifetime was extended by 15 years, despite protests from scientists working at the All-Russian R&D-Institute for Atomic Power Engineering. Life extension is also planned for the three other RBMK units of the Leningrad NPP [NUCWEEK 05_04; NUCWEEK 45_04].

The Cost Angle

The consequences of ageing which become apparent as events and incidents tend to reduce the NPPs availability, and thus, the amount of electricity produced and sold. Therefore, there is – up to a point – a clear motivation for the plant operator to implement modernization and counter measures.

On the other hand, the consequences which "merely" increase the probability of some catastrophic failure, while this probability remains small compared to everyday experience, carry no direct economic penalty (as long as luck will have it). Therefore, there is no particular incentive, from an economic viewpoint, to invest in countermeasures against such ageing mechanisms, and operators will try to keep the costs involved as low as possible.

Accordingly, there is a tendency of NPP operators to remain at the two lower levels (reductions of load, and of safety margins), and restrict exchange of components to smaller parts.

Exchange of large components has been (and will be) practised extensively only whenever the remaining (possibly increased) lifetime was sufficient to amortise the investment. For example, steam generators have been exchanged in nuclear power plants in most Western countries with NPPs with pressurised water reactors, and reactor vessel heads are being exchanged in France and other countries.

The quantitative economic evaluation of PLEX measures is complicated and depends on the concrete circumstances for each plant. In several studies, substantial benefits are described. For example, a US analyst recently claimed that the costs of PLEX for a US nuclear power plant are about 10 – 50 US$/kW, whereas construction of the cheapest non-nuclear alternatives would cost 325 – 405 US$/kW. Life extension of a coal fired power plant, for 20 more operating years, would cost 100 – 250 US$/kW [MACDOUGALL 1998]. New nuclear capacity would be considerably more expensive than all those options (far above 1000 US$/kW).

Russian authors, quoting US sources from the late eighties, reported considerably higher costs for PLEX measures: up to 300 US$/kW [BARANENKO 1998].

A systematic study undertaken by the IAEA demonstrates the large spread of cost estimates for PLEX. Based on responses to a questionnaire, which were received from NPP operators in 12

countries, the range is given as 120 – 680 US$ per kW. However, this represents only the central part of the various estimates; the probability of the actual costs lying below the lower value given is 20 %, as well as the probability of it being above the higher value. The cost data are presented as ranges only in the IAEA report because of data confidentiality due to the competitive environment in the electricity sector [IAEA 2002].

French Industry Secretary Pierret, advocating life extension for French reactors, stated that each year of operation beyond the nominal 30 years-lifetime would bring a gain of about 70 million US$ [NUCWEEK 47_00]. For the whole French reactor fleet, 10 extra years of life are reported to represent a cumulative cash flow of € 15 to 23 billion [NUCWEEK 40_03].

Compared to this, the cost of PLEX for one NPP according to the lowest estimate quoted above (10 – 15 million US$) appears to be more than reasonable; the higher limit of the range given by the IAEA, however, corresponds to costs of 650 to 900 million US$ per NPP, representing a considerable investment which would not be worthwhile unless about two decades of additional operating time are guaranteed.

Taking into account that a complete steam generator replacement for a PWR alone costs about 150 to 200 million US$ [EPRI 2003; KLIMAS 2003], it seems plausible that the costs of PLEX measures in many cases will be closer to the higher estimates quoted above.

Apart from those general cost estimates, concrete cost figures have been published for some PLEX projects. For example, modernization of the two Olkiluoto BWRs for ten years' life extension is reported to have cost about € 130 million [RASTAS 2003]. At Paks NPP, 20 years' extension for the four VVER units will cost about € 700 million [NUCWEEK 47_04]. For the Ukrainian life extensions plans (by 10 to 15 years), it is claimed that they will be about three to four times less expensive than construction of new plants [NUCWEEK 23_03]. Life extension at the Kola first-generation VVERs by 15 years cost about € 150 million for both units [NUCWEEK 33_04].

Costs of license extension preparation and regulatory review fees constitute only a comparatively small, yet not negligible, part of PLEX costs. For example, for the two units of Nine Mile Point BWR (USA), they are estimated at about US$ 25 million [NUCWEEK 48_03].

There are some cases in which ageing mechanisms forced shut-down of an NPP, because the measures of life extension which would have been required were regarded as too costly by the plant operator. One example, Würgassen BWR in Germany, where shut-down even occurred considerably earlier than the design lifetime originally envisaged, is treated further in the appendix to this section.

Compared to new reactors like the Finnish EPR, which will cost the utility TVO € 3 billion, the costs of modernization measures for PLEX appear almost modest. Compared to other, cheaper alternatives like modern gas-fired power plants with investment costs in the order of only 20 to 25 percent of those of a new NPP, PLEX costs in the order of € 100 million or more, on the other hand, appear quite substantial.

It is clear that PLEX therefore creates an economic compulsion to really operate the plant for the whole additional lifetime envisaged, and possible beyond that – with all the hazards of increasing ageing effects this entails. Only in case of "PLEX light" which would correspond to the lowest cost estimates quoted above, economic pressure would be less overpowering. This implies, however, that no or only minimal refurbishment is undertaken and hence, ageing hazards will be still higher than in normal PLEX cases.

Operational hazards and economic pressure is increased further if PLEX is combined with power uprating and other measures to economize plant operation.

Power Uprating

Power uprating is an economically attractive option for NPP operators that usually goes largely unnoticed by the public. It pays off particularly well when combined with life extension.

Power uprating is practised in most countries where NPPs are operated. Upgrading turbines and steam generators yielded an additional 4 % of nuclear generating capacity in Spain between 1995 and 1997. During the last years, power uprating has continued in this country. Power output of the Cofrentes BWR has been raised by about 11 % at the beginning of 2003 [FORATOM 2004]. Capacity was increased by 600 MWe in Sweden [VARLEY 1998].

The output of the Finnish NPP Olkiluoto was boosted by 18,3 % [RASTAS 2003]. In Germany, output of a number of plants was increased. Until mid-2004, power uprates amounted to about 800 MWe, or 4 % of installed nuclear capacity. Another 450 MWe are planned [DATF 2003; ATW 2004]. Power uprating is also practised extensively in the USA. For example, the output of Ginna PWR (at present, 495 MWe), where life extension is also planned, is to be increased by 17 % within five years. This seems to be achieved with hardly any costs for safety systems' refurbishment, since the investment costs per kWh are reduced accordingly [NUCWEEK 48_03]. Uprating measures are also implemented at obsolete Soviet reactor types. For example, the for units of second-generation VVERs at Paks in Hungary are to be uprated from the (already slightly increased) power level of about 470 MWe to 510 MWe [NUCWEEK 47_04].

In order to uprate the electrical power of a nuclear power plant, there are two options (which are often combined):

- At constant reactor power, thermal efficiency of the plant is increased. This is mostly achieved by optimising the turbines. Operational safety of the plant remains on the same level. Also, replacement of the steam generators can increase efficiency if the new heat exchangers have higher efficiency.
- Thermal power of the reactor is raised, usually by increasing coolant temperature. Thus, more steam is produced and the reactor can produce more electricity via the turbines (which have to be modified as well). An increase of thermal power implies more nuclear fissions and thus increases operational risk. Also, higher loads to the reactor materials are unavoidable. There is general consensus, that an increase of reactor power reduces operational safety margins and at the same time accelerates ageing processes.

The possibilities for power uprates through improvement of the thermal efficiency have, to a large extent, already been realised in the last years. Thus, there is a trend towards uprates through raising the reactor power. For example, all uprates planned today in Germany fall into the latter category.

Furthermore, increasing the thermal power of a reactor is regarded as a particularly cost-effective way to increased electricity production [FRAMATOME 2004].

For PWRs, reactor power is increased by raising the average coolant temperature, accompanied by increasing the temperature rise in the core. This leads to decreasing safety margins: Corrosion of fuel element hulls becomes more likely and primary circuit pressure will reach higher peaks during transients. Furthermore, the radioactive inventory in the reactor core is increased proportionally to the power uprate. Measures to control or mitigate critical situations become more difficult. For example, in case of containment venting, the venting rate has to be increased [BORNEMANN 2001].

Similar problems arise for power uprates of other reactor types. For example, power uprating of Quad City 2 BWR in the U.S. lead to vibrations of the main steam line, which in turn damaged other components and necessitated several shut-downs and repairs [UCS 2004].

Increasing the fuel burn-up (i.e. getting more energy per ton of fuel) is another way in which NPP operators attempt to improve the economy of their plants. This requires a corresponding increase of the enrichment of the fresh fuel.

The trend towards higher burn-up has started very early; in the last years, the efforts to increase burn-up have been intensified. Several decades ago, typical burn-up of PWR spent fuel was around 30.000 MWd/t or slightly higher. Today, burn-ups of 50.000 MWd/t have been reached and 60.000 MWd/t are aimed for. The situation is similar for BWRs, although at a slightly lower level.

Increasing burn-up also increases the hazard of fuel hull failure and hence, radioactive contamination of the cooling water. Furthermore, the influence of high burn-up on the behaviour of fuel rods under accident conditions is not fully understood.

The use of high burn-up fuel can also reduce operational safety margins. For example, the hazard of neutron flux oscillations in BWRs is increased (see section A).

Increased burn-up reduces the mass of spent fuel produced annually by a power reactor. On the other hand, handling, transport, storage and disposal of spent fuel becomes more difficult and hazardous because of higher radiation intensity, higher heat development and higher content of long-lived actinide nuclides.

Regulators' Perspective

Although there is general consensus that the main responsibility for safe operation of a nuclear power plant lies with the operator, the regulatory authorities play a very important role regarding the safety standards upheld in different countries, and the level of hazard regarded as acceptable. Therefore, the regulators' perspective and the problems nuclear regulators are faced with regarding ageing and life extension deserve to be discussed here. Unless indicated otherwise, this section is based on a recent report by the OECD Nuclear Energy Agency's Committee on Nuclear Regulatory Activities, which primarily consists of senior nuclear regulators from many countries [CNRA 2001].

Nuclear regulatory practice varies considerably between countries. This holds particularly true concerning regulation of ageing and life extension.

To begin with, some countries (for example, the USA and Finland) issue operating licenses for a fixed period of time. In Switzerland, there are limited licenses for some power plants and not for others. Most countries, however, issue licenses that are basically indefinite, subject to continued safe operation of the plant.

Periodic safety reviews play an increasingly important role, particularly in countries with indefinite licenses, to justify further operation. In this respect, too, there are considerable variations between countries. There is divergence in the extent of documentation and other information that has to be supplied by the operator. There are also differences in the extent to which the regulatory authority carries out an independent evaluation of the safety case.

Practices also vary widely regarding development and updating of rules and regulations. In all countries, regulation is mostly based on deterministic methods and criteria. The importance of probabilistic methods, however, is growing. In some countries, they are already formally integrated into the licensing process, whereas regulators in other countries remain more sceptical.

One fairly common feature of regulatory approaches worldwide is that regulators usually review the entire design basis of a plant in order to decide which safety improvements can be required and expected from the operator. Even in this respect, however, there is a notable exception: The

license renewal process in the United States focuses on the detrimental effects of ageing and does not review the current licensing basis of a plant.

In spite of this heterogeneous picture, there are a number of problems which regulators are facing all over the world. The most basic and severe shortcoming of regulatory practice everywhere is that no country has a comprehensive set of technical criteria for deciding when further operation of a nuclear power plant can no longer be permitted.

A generally valid principle is that the licensing basis of a plant is to be maintained throughout its life. In addition, a few countries (for example, Switzerland) have the explicit requirement that nuclear plants should conform to the state-of-the-art in science and technology. In many other countries, this requirement is implicit in the regulatory approach. This criterion is regarded as potentially very onerous. The extent to which it is practicable for older plants generally requires a very difficult judgement from regulatory authorities.

In practice, backfitting of modern requirements to older NPPs is only demanded by the regulators to the extent that it is *"reasonably practicable"*, taking into account safety gains and costs, as the responses to a questionnaire circulated by the OECD Nuclear Energy Agency show. Of course, this formula leaves considerable leeway for interpretation and compromises. Generally, deviations from modern standards are evaluated by regulators on a pragmatic, case-by-case basis.

The trend towards increasing use of probabilistic methods also constitutes a problem for regulators. Probabilistic analyses are increasingly used as regulatory tools. However, regulators are mostly unwilling to accept that probabilistic arguments alone should be sufficient to reverse licensing decisions taken on deterministic grounds. This may become more and more contentious as plant operators attempt to make arguments, on the basis of probabilistic assessments, about what is reasonably practicable for them regarding backfitting of older plants.

Another difficult task for regulators is to contribute to ensuring that there is a continuing supply of competent personnel to operate and maintain older plants where design details, technical limits etc. may be less well documented than for modern ones. This problem can be exacerbated by the gradual retirement of plant designers as well as operators that were working at the plant from start-up.

Impact of Electricity Market Restructuring

The most severe problem regulators are facing, however, very likely results from the re-structuring and privatisation of electrical utilities occurring worldwide. This development exerts considerable pressure on the operators' safety culture and management of safety. Regarding this issue, it is well worth letting the conclusions of the nuclear regulators constituting the OECD NEA's Committee on Nuclear Regulatory Activities speak for themselves (emphasis added by the authors):

*"The rapidly-changing economic and industrial environments in many countries over the past few years had presented major challenges to both the nuclear operators and the regulators. In the privatised and deregulated situations that now existed (or were coming into being) in many countries**, the operators were acutely aware of the potential lifetimes and long-term profitability of their plants when making decisions about upgrading, undertaking PSRs [Periodic Safety Reviews] or applying for formal licence renewals. Plant management was under increasing pressure to find the correct balance between operating the plant safely and not spending money unnecessarily**. The regulatory challenge was to find appropriate ways to check and monitor this without interfering unduly in the licensee's business. The impact of safety culture, safety management, and the competence and level of staffing on the safety of NPPs will continue to demand considerable regulatory attention.*

Another challenge to regulators, in the new environment, is to find a logical way to deal ***with requests from operators for relief from some of the existing regulatory constraints. This is a sensitive regulatory issue that can easily generate public and political suspicion, if not handled openly and properly.*** *However, a selected increase in the apparent level of risk may be acceptable by some regulators if research or operating experience allow the uncertainty in the risk assessment to be reduced, to the extent that the realistic safety margins are still sufficient to cover the remaining uncertainty. The increasing use, and wider acceptability of, risk-informed regulations may help to clarify such decision-making. (It is recommended that CNRA should have an in-depth exchange of experience on this topic at a future meeting)."*

In view of all those problems, the goal set by the Swiss nuclear regulatory authority [TIPPING 2003] appears to be rather unrealistic:

"Generally, from a nuclear regulator's standpoint, plant lifetime management strategies practised by operators must at least contribute to maintaining presently existing safety levels by ensuring the suitability of all aspects of NPP operation (...). Ideally, lifetime management strategies should also be instrumental in increasing overall safety levels of NPPs."

Indeed, increasing safety is more likely to remain an dream, rather than an ideal which can realistically be achieved, in view of massive economic pressure and other problems facing nuclear regulators worldwide.

Conclusions

From the discussion above, it is clear that nuclear operators are faced with a dilemma which is gets more and more pressing the longer a nuclear power plant is operating and ageing mechanisms become virulent: Measures for life extension (as well as power uprating) on the one hand can be economically attractive and offer a chance to improve the overall economic balance of NPP operation. On the other hand, they exacerbate the hazards of ageing and increase the risk of a nuclear catastrophe with severe radioactive releases.

With very few exceptions, it appears the economy triumphs over safety and PLEX programmes are implemented. The situation is particularly grave since such a programme generally can only make economic sense for plant owners if the plant is operated for one or two more decades after its implementation.

PLEX measures certainly compare favourably, from an economic point of view, with the construction of new nuclear power plants. But this is by no means true for comparison with other alternatives, like the construction of modern gas-fired plants.

PLEX therefore creates strong pressure to keep an NPP on the grid, to get an adequate return on the investment, and to ignore or play down the hazards of ageing. This pressure is strengthened if further money has been spent on power uprating. In addition, there is pressure to keep expenses for PLEX programmes as low as possible.

All this is happening in an economic context of liberalisation of the energy economy, general cost pressure and increasing competition, which is leading to decreasing safety margins, personnel reductions and reduced efforts for inspection and maintenance – whereas the trend towards an ageing NPP population would require exactly the opposite.

At the same time, safety margins are further reduced by power uprating and increasing fuel burn-up.

Examples of Age Related Problems

In this appendix, some examples for nuclear plants with ageing problems which led to decommissioning, or to prolonged shutdown periods are presented.

Yankee Rowe PWR (USA, 185 MWe) was permanently shut down in February 1992, after only about 31 years of operation.

Ironically, Yankee Rowe was to be the U.S. lead plant for license renewal to obtain life extension from 40 to 60 years. The license renewal procedure was started early because it was the first attempt of this kind. Review of the safety case for the reactor pressure vessel showed that embrittlement of the weld near the core had already reached a critical stage, mainly because of high content of copper impurities.

The operator temporarily shut down the plant in October 1991, two days before the U.S.NRC would have ordered a shutdown. A petition of the Union of Concerned Scientists had prompted the NRC

A programme of reactor pressure vessel testing and analyses was under consideration, with the aim to demonstrate sufficient safety margins after all. This six month programme was expected to cost US$ 23 million. However, the plant's owner, Yankee Atomic Electric Company, declared that those costs were too great, particularly in view of the fact that the path leading to restart of the plant was not sufficiently clear. Yankee Atomic emphasized that the shut-down was exclusively motivated by economic, not safety concerns [WEISSMANN 1992].

This claim, however, does not appear credible in view of the comparatively low costs mentioned above. It is more than likely that the plant owners knew it would not be possible to construct a credible safety case, and hence decided on a shut-down without attempting further analyses.

Würgassen BWR (Germany, 670 MWe) was permanently shut down in May 1995, after less than 24 years of operation.

During the overall maintenance inspections beginning August 1994, special inspections of the core shroud were performed in addition to routine inspections, because cracks had been found in this component in 13 boiling water reactors in other countries. The Würgassen core shroud was found to be severely cracked.

Repairing the core shroud plus the necessary accompanying backfits and modernization measures would have cost DM 350 – 400 million (about € 220 million at 2004 value). The plant owner, PreussenElektra, decided against performing the repairs and decommissioned the plant [JATW 1995; NNI 1995].

Davis Besse PWR (USA, 925 MWe) was temporarily shut down after a hole was discovered in the reactor pressure vessel head on March 6, 2002. The reactor was off the grid for more than two years and restarted on April 8, 2004.

The hole went through the whole thickness of the vessel head; only the stainless steel liner welded to its inner surface was still intact. This liner (less than 5 mm thick) was the last remaining barrier to prevent a severe loss-of-coolant accident. It had already bulged by about 3 mm under the high pressure in the RPV.

Unnoticed, this incident had begun 1990, when a crack had developed in the nozzle of a control rod drive mechanism (CRDM). Over the years, the crack grew through the nozzle, boric acid leaked out and a corrosive attack on the outside of the vessel head begun. Once opened, the hole widened by nearly 50 mm per year. Nevertheless, it was overlooked at visual inspections in 1998 and 2000.

Cracks in CRDM nozzles have been observed worldwide since 1991. In 1993, Greenpeace International petitioned the U.S.NRC to require inspections of CRDM nozzles at all U.S. reactors. Those inspections very likely would have lead to an early identification of the problems at Davis Besse (as well as at other plants). However, it was denied in 1994 [UCS 2002].

Total costs for the shut-down, including replacement power costs, were about US$ 600 million. From the NRC, Davis Besse is now licensed until April 2017 in spite of the serious lack of safety culture which had become apparent at the plant, where a problem was allowed to grow for 12 years.

Stade PWR (Germany, 662 MWe) was permanently shut down in November 2003, after less than 32 years of operation, although the amount of electricity assigned to this plant by the German Atomic Law (revision of 2001) had not been produced yet.

Similar to the situation at Yankee Rowe, critical welds in the Stade reactor pressure vessel were particularly prone to embrittlement due to a high copper content. This had been known for many years, but experts from the TÜV, the technical support organisation working for the supervising authority, kept confirming that a sufficient safety margin existed for 40 years of operation.

In the early 90s, the supervising authority (Ministry for the Environment of the State of Lower Saxony) commissioned another technical organisation, Gruppe Ökologie Hannover, to look into the embrittlement situation. In 1994, the new experts submitted their results that showed that operation of the NPP was only safe, at most, for about one more year.

Under pressure from the plant operator PreussenElektra, the state authority did not further employ Gruppe Ökologie. Under leadership of TÜV (an organisation which had overlooked the critical situation for many years to begin with), an urgent investigation programme was started. Given the circumstances, it is not surprising that after hectic work 1995 to 1997, to TÜV experts managed to confirm the safety case of Stade RPV after all.

In 2003, Stade was shut down; so far, the only NPP which has been decommissioned since the red/green coalition came to power in Germany 1998 whose goal it is to gradually phase-out nuclear power. Under the German Atomic Law, Stade would still have been permitted to continue electricity production. The plant owner emphasized that the reasons for the shut-down were entirely economic – the capacity of Stade was too small, it was claimed, for economic operation. However, the older Obrigheim PWR with only half of Stade's capacity is still in operation. Owners even transferred an additional amount of electricity to this PWR from another, newer plant. Hence, it might be a good guess that Stade in fact was prematurely shut down because of embrittlement problems.

Those examples clearly show that there are significant safety problems due to ageing at "western" reactors, and that they are not always dealt with rigorously and efficiently by the responsible authorities.

There are very severe ageing problems at many "eastern" reactors, too. However, the situation is more complicated for the ex-Soviet reactor types. Ageing problems and general design and construction deficiencies are usually closely interwoven. Furthermore, apart from small plants, military production reactors and early prototypes, eastern reactors mostly were only decommissioned in case of severe pressure from the West (Chernobyl-1 to -3, Ignalina-1, Kozloduy-1 and -2 and Armenia-1) and there is no clear-cut case study for a plant being shut down because of one particular technical issue. The VVERs at Greifswald which came under the supervision of the Federal Republic of Germany after reunification constitute a notable

exception; these were shut down quickly by German authorities, again because of a multitude of shortcomings (see also section A).

It has to be noted, however, that pressure vessel embrittlement probably was the single most important safety issue in case of Greifswald-1 to -4 as well as the Kozloduy VVER-440s.

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D.1.i: Acts of Terrorism and War – Vulnerability of Nuclear Power plants

The Terror Threat

Long before September 11, 2001, numerous deliberate acts of terrorism have taken place in the 20th century. The terrorist threat appears to be particularly great, however, in the early 21st century. The overall situation, which is determined by economic, military, ideological and political factors, cannot be discussed and evaluated here.

It is important, however, to note the following: Although general attention is focussed on the threat from the direction of Islamic fundamentalism right now, there are, worldwide, many different ideological positions and organisations from which potential terrorists could be recruited. For example, the bombing of a building of the U.S. federal government in Oklahoma on April 19, 1995, which killed 169 people and injured more than 500, was committed by American extremists of the right [THOMPSON 1996]. At present, the growing threat of Neo-Nazi terrorism in Germany also illustrates this point [FR 2005b, c].

There are numerous potential targets for terrorist attacks. Industrial installations, office buildings in city centres or filled sports stadium can appear "attractive", if a terrorist group plans to kill as many human beings as possible in one attack. A nuclear power plant (NPP), on the other hand, could be selected as target for one of the following reasons, or a combination of those reasons:

1. Because of the symbolic character – nuclear power can be seen as the epitome of technological development, as typical "high-tech". Furthermore, it is a technology of an ambiguous civilian/military nature. Many people therefore regard it as potentially very hazardous – justifiably so. Therefore, attacks against nuclear power plants can have a particularly strong psychological impact.
2. Because of the long-term effects – an attack can lead to far-reaching radioactive contamination with long-lived radio-nuclides. The state that is being attacked will bear the mark of destruction for a long time. Furthermore, there will be economic damage for decades. Large areas (cities, industrial complexes) will have to be evacuated for an indefinite period, which could destabilize entire regions.
3. Because of the immediate effects on the electricity generation in the region affected – nuclear power plants are, wherever they are operated, large and centralised components of the electricity supply system. The sudden shutdown of such a large plant can possibly lead to a collapse of the local electricity grid.
4. Because of the longer-term effects on electricity generation, not only in the affected region, but also in other regions (possibly even in all states where nuclear power plants are operated) – a successful attack against a nuclear power plant in one country is also an attack against all nuclear power plants in the world [BRAUN 2002]. After such an attack has demonstrated the vulnerability of an NPP, it is possible that other NPPs will be shut down in the country affected, but also in other countries.

There are also conceivable reasons, from the point of view of a terrorist group, against a nuclear power plant as target: A nuclear installation can be less vulnerable than other targets; radiological damage could occur in large distances in non-enemy countries; and the attacked country could react with extreme violence [THOMPSON 2005]. There seems to be no chance, however, to estimate probabilities that certain targets would be attacked, or not. It is clear and undisputed that a terror attack against a nuclear power plant is possible; and also, that there are many types of other targets for such attacks as well.

Terror-attacks against nuclear power plants can be performed with a large variety of means. It is not possible to list all conceivable scenarios since it is absolutely impossible to anticipate all products of human fantasy. Since September 11, 2001, authorities have been focussing on airplane suicide attacks. However, totally different scenarios are also plausible.

In principle, attacks can vary with respect to the means being used, the concrete target, the organisation, number and effort of the attackers as well as other factors. For each of those variables, there are many possibilities of implementation. Even the attempt to completely list what is foreseeable therefore would lead to a matrix with a large number of different scenarios.

Therefore, some examples only will be presented here, to show the diversity of the threat. Those examples will include scenarios that, so far, have hardly received attention in the expert and public debates.

Terror attacks against nuclear plants are not purely theoretical. In the past, a number of such attacks have already taken place. Luckily, they did not lead to a catastrophic radioactive release so far. A few examples can illustrate the record [COEYTAUX 2001; THOMPSON 1996; NISSIM 2004; TMI 2005; NUCWEEK 46_94]:

- On 12 November 1972, three hijackers took control of a DC-9 of Southern Airlines and threatened to crash it on the Oak Ridge military nuclear research reactor. The hijackers flew on to Cuba after they obtained two million dollars.
- December 1977: Basque separatists set off bombs damaging the reactor vessel and a steam generator and killing two workers at the Lemoniz NPP under construction in Spain.
- December 1982: ANC guerrilla fighters set off four bombs inside the Koeberg plant under construction in South Africa, despite tight security.
- May 1986: three of the four off-site power lines leading to the Palo Verde NPP in Arizona were sabotaged by short-circuiting.
- February 1993: At Three Mile Island NPP (Pennsylvania), a man crashed his station wagon through the security gate and rammed the vehicle under a partly opened door in the turbine building. Security guards found him hiding in that building four hours later.
- In 1993, the terrorists behind the car bombing against the World Trade Centre, belonging to the terrorist networks that claimed to be part of the Islamic jihad, threatened to target nuclear sites in a letter received by the New York Times and authenticated by the authorities. In addition, the investigation is said to have revealed that the terrorist group trained in November 1992 in a camp near Harrisburg, in Pennsylvania 15 km away from the Three Mile Island nuclear power station.
- November 1994: Bomb threat at Ignalina NPP, Lithuania. However, no explosion occurred and no bomb was found in the power plant.

Acts of War

Military action against nuclear installations constitutes another danger deserving special attention in the present global situation. Since the fall of the Iron Curtain, there is an increasing tendency towards “small”, regionally restricted wars of long duration. Those wars can be connected with the falling apart of a large state; or with efforts of groups in a population to achieve independence [MÜNKLER 2003]. The reasons for terror attacks listed above could, in such a war, motivate one of the conflict parties to attack a nuclear plant.

Wars of intervention are another form of warlike conflict. They can occur as a consequence of a regional war of long duration, as mentioned above. In the course of such wars, countries attack a state from which emanates a real or alleged threat. The political goals and interests of the attacking states usually play an important role in this case. If there are nuclear plants in the attacked country, there is the risk that those could be damaged unintentionally during the fighting. Furthermore, an intervening power might attack power plants to paralyze electricity supply in the attacked country. There would be efforts to avoid radioactive releases. Because of the compact layout of the individual parts of a nuclear power plant, however, safety relevant installations might nevertheless be damaged. Also, in times of war, the electrical supply system might collapse without direct attacks against power plants. In combination with further destruction of infrastructure, this, too, could in the end lead to incidents or accidents in nuclear power plants, with consequences for the surroundings.

It is also conceivable that nuclear plants, which serve military purposes or are feared to serve such purposes, will be deliberately destroyed. In this case, the release of radioactive materials might be accepted by the attacker.

In June 1981, a large (40 MWth) research reactor under construction at the Tuwaitha research centre in Iraq was destroyed by the Israeli air force because of Israeli fears that the reactor could be used (directly or indirectly) for a nuclear arsenal. During the 1991 Gulf War, two smaller reactors at the same site were destroyed in a night attack by U.S. aircraft [THOMPSON 1996].

Threats through acts of war cannot be excluded in any region. During the Balkan conflicts in the early 90s, the Slovenian nuclear power plant Krško was endangered several times. In June 1991, three fighter bombers of the Yugoslavian air force flew over the plant. There was no attack; however, this act clearly constituted a warning. In September 1991, war again approached the Slovenian border. There was fighting in the surroundings of Zagreb, which could easily have spread to Slovenian territory [HIRSCH 1997].

In case of a warlike conflict, commando attacks might occur in combination with acts of war (performed by special forces active behind enemy lines, or by a “fifth column”). This danger is particularly high in case of an asymmetric war – where an enemy attacks a much weaker country, for example during a war of intervention-. Scruples about actions mostly directed against the enemy’s civilian population, might be drastically reduced if the attacked country has no other options of hitting back against an all-powerful enemy, and/or has already suffered severe civilian losses itself.

The use of nuclear weapons against nuclear power plants (through terrorist or military attack) will not be discussed here. However, it should be mentioned that the destruction of a nuclear power plant could significantly increase the radioactive contamination produced by a nuclear fission weapon – the fission product inventory of a commercial nuclear power plant is in the order of magnitude of 1000 times that released by a fission weapon.

Targets, and Their Vulnerability

Of all nuclear plants and other facilities with toxic inventories, such as chemical factories, nuclear power plants are probably the most “attractive“ targets for terrorist or military attacks. They are wide-spread (at least in a number of industrialized countries), contain a considerable radioactive inventory and are, as already pointed out, important components of the electricity supply system. Furthermore, they are large buildings with a typical structure, visible even over large distances.

The area of a nuclear power plant consists of several tens of thousands of square meters. The core piece of the installation is the reactor building, which, as the name indicates, contains the

reactor with the highly radioactive nuclear fuel (in the order of magnitude of 100 tonnes), as well as important cooling and safety systems.

It is likely that the reactor building will be the primary target in case of an attack. If the reactor is operating as the attack occurs, and if the cooling is interrupted, a core melt can result within a very short time (about 1 hour). Even if the reactor is shut down, the decay heat is still considerable, and the fuel will also melt – although somewhat slower.

In case of destruction of the reactor building with failure of the cooling systems, a core melt accident of the most hazardous category results: Rapid melting with open containment. The resulting radioactive releases will be particularly high, and occur particularly early.

The spent fuel storage pool is another vulnerable component with considerable radioactive inventory. In some plants, it can contain several times the amount of fuel (and thus more long-lived radioactive substances) than the reactor core itself. In some nuclear power plants, this pool is located inside the containment and is protected against external impacts by a concrete hull (for example in German pressurized water reactors). In many cases, however, the pool is installed in a separate building with less protection (this applies to many U.S. nuclear power plants). The pool in German boiling water reactors of generation 69 is located inside the reactor building, but above the containment, and protected to a considerably lesser degree than the reactor.

Apart from the reactor building and, if applicable, the building with the spent fuel pool, there are further buildings and installation of varying safety significance. The most important are, in case of a modern pressurized water reactor (PWRs, including VVERs, account for about 60 % of the world's operating plants):

- Switchgear building with control room and central electric and electronic installations
- Auxiliary building with installations for water purification and ventilation
- Machine hall with turbine and generator
- Transformer station with connection to grid and station transformer
- Emergency power building with emergency diesel units and chilled water system
- Emergency feed building with installations for emergency feeding of steam generators (i.e. cooling of reactor via the secondary cooling circuit), with remote shutdown station
- Off-gas stack
- Workshop building with staff amenities
- Cooling towers (if required)
- Building for cooling water intake and discharge

The situation is similar for a boiling water reactor. However, there is no emergency feed building in this case, since BWRs have only one cooling circuit and thus, no steam generators. Instead of the emergency feed building, some BWRs are supplied with an emergency standby building with an emergency control room permitting control of the most vital safety functions.

So far, not all nuclear power plants have been specially designed against external, human-made impacts (for example aircraft crash). In the case of those that have been, an impact in one spot only has been assumed (corresponding, for example, to the crash of a small military aircraft). Spatial separation of safety relevant installations was the most important counter measure. This should guarantee that only one installation vital for safety could be destroyed by an impact – a situation where compensation is possible. For example, in case of failure of the auxiliary power

supply via the corresponding transformer, the emergency power supply with diesel generators can be activated.

Even if the reactor building remains intact in case of an attack, it is still possible for the situation to get out of control, if more than one safety relevant installation of the plant is destroyed. This can happen even in case of spatial separation of important components, if the attack has effects that are spread over on the site.

For example, in case of the simultaneous failure of power supply from the grid (via station transformer) and emergency power supply, there are no more coolant pumps operable. In case of simultaneous destruction of control room and emergency feed building (emergency standby building), a situation could arise where the safety system required are still operable, but cannot be controlled any more. Far-reaching destruction on the plant area can furthermore have the effect that access by personnel, and thus emergency measures and repairs, are rendered impossible – at least not within the required time span of a few hours.

Destruction of the cooling water intake building alone already has the effect that all cooling chains of the power plant are interrupted. However, a critical situation is slow to develop in this case, since there are various water reservoirs available at the plant area. Thus, there is time for improvised measures – unless those are hindered by further destructions at the site.

Regarding other nuclear installations or nuclear transports as targets, see the other sub-sections of section D.1.

Conceivable Attack Scenarios

As pointed out above, the public debate tends to concentrate on suicide attacks with a commercial airliner since September 9, 2001. In fact, the threat is much more diverse and complex.

In the following, various possibilities for terror attacks on NPPs are listed, as examples. Almost all of them could also take place in times of war, committed by commando troops or a fifth column. Some of the scenarios could be implemented, with minor changes, in the course of military operations.

<u>Attack from the air:</u>

- Deliberate crash of commercial airliner, freight plane or one or several business jets (possibly loaded with explosives)
- Deliberate crash of a helicopter loaded with explosives, or dropping a bomb from helicopter
- Attack by military plane (with bombs and/or other weapons), possibly combined with deliberate crash of military plane
- Deliberate crash of a pilot-less aircraft (drone) loaded with explosives

<u>Attack from the water:</u>

- Crash of boat loaded with explosives into cooling water intake structures from sea or river;
- Deliberate explosion of gas tanker close to NPP

<u>Firing on plant from a distance:</u>

- Shelling with field howitzer, with explosive grenades (from ground or water)
- Firing with armour-piercing weapons (rockets), from ground, water or from the air

Intrusion of attackers onto plant area:

- Use of one or more car bomb(s)
- Intrusion of armed attackers, carrying explosives, from land or water
- Intrusion of armed attackers, carrying explosives, by helicopter or ultralight aircraft

Attacks involving insiders:

- Insiders support attack from outside, for example through creation of confusion, obstruction of counter measures or simultaneous attack from inside
- Explosives are being smuggled on the site and into buildings; are exploded in safety-relevant sectors
- A knowledgeable group of insiders directly intervenes in the operation of the plant, triggering a severe accident
- Insiders perform sabotage during repair and maintenance
- Armed members of the security personnel perform an attack from the inside or support an attack from outside

Attacks against installations located outside the plant perimeter:

- Attack against the cooling water intake building of a nuclear power plant from the water (with boats, possibly divers), using explosives
- Attack against the grid connection of a nuclear power plant (or other nuclear plants), for example by blowing up power connections, and against on-site power supply (emergency diesels etc.)

Not all nuclear plants are vulnerable to the same extent. Most attack options listed here can lead, in the worst case, to very severe releases. Some will have rather limited effects. Different parts of a plant can be vulnerable to different modes of attack to a different extent.

However, all these aspects will not be discussed in detail here. It is not the intention of the authors to provide "useful" information to terrorists or military planners, which could be used for the planning of attacks.

Consequences of an Attack on a Nuclear Power Plant

One example, from the long list of possible scenarios, will be discussed in more detail here – shelling of a nuclear power plant. Such an attack can lead to a reactor accident of the most severe category: Core melt with early containment failure. It would be more effective than an attack with armour- or concrete-piercing missiles.

A possible scenario would be shelling with a 15.5 cm-howitzer, transported by road, as part of military operations or as terror attack. Almost every army of the world today possess such a weapon; it is conceivable that terrorists are able to acquire one. A 15.5 cm-howitzer can be brought to the vicinity of an NPP under camouflage; it can be made ready to fire within minutes. If shelling takes place from a distance of 12 to 15 km, an area of about 50 m x 50 m on the site can be hit several times. If the distance is smaller and weather conditions are favourable, accuracy will be significantly increased. Multiple hits of the reactor building are possible.

If high-explosive shells are used, the reactor building will be partly destroyed. Severe damage will occur inside. Plant personnel will be killed or injured. At the site area, shots which are slightly off-target will create further devastation. This can be deliberately enhanced by the use of

fire shells and other types of munitions. It will be extremely difficult to implement effective and rapid counter measures.

Within a few hours, core melt will occur, with severe releases of radioactivity. The amount released to the atmosphere can be about 50 – 90 % of the radioactive inventory of volatile nuclides like iodine and caesium, plus a few percents of further nuclides like strontium-90. In case of a nuclear power plant with 1000 MW electric power, this corresponds, among others, to several 100.000 Tera-Becquerel (TBq) of Cs-137 [HAHN 1999], compared to about 85.000 TBq Cs-137 at Chernobyl [NEA 1996].

The consequences amount to a catastrophe with effects over a large region: Up to 10.000 km^2 would have to be evacuated in the short term. There would be up to 15.000 acute radiation deaths and up to 1 million cancer deaths, as well as uncounted cases of genetic damage. The area that would be contaminated in the long term to a degree necessitating relocation of the population can measure up to 100.000 km^2. The economic damage has been estimated at about 6.000 billion Euros [HAHN 1999].

For many reactors, the probability of destruction or severe damage of the spent fuel pool is high. In this case, releases can be several times those given above, with correspondingly more severe consequences.

During a certain period of time, intervention could be possible to provide cooling of the fuel. If the pool cooling system fails because of the attack and water gradually boils off, it will take between one and ten days (depending on amount and cooling times of the spent fuel in the pool) until the tops of the fuel elements are exposed. If the pool is damaged and the water drains off, this point, of course, can be reached much faster. Once the fuel is exposed, radiation shielding is completely lost and intervention becomes impossible because of the prohibitive radiation dose rates

Freshly discharged fuel would then reach the point where it burns in air (900° C) and very severe radioactive releases begin within hours [ALVAREZ 2003].

Countermeasures and Their Limits

Several measures are conceivable which could possibly provide a certain degree of protection for nuclear power plants, against acts of war and terror attacks. Regarding terror attacks, such measures are at present under examination by NPP operators and authorities. Some have already been implemented or are in a concrete planning stage.

The most important options are the following, which are, to some extent, also subject of public debate:

1. Preventive shut-down
2. Structural backfitting against deliberate aircraft crash and other hazards
3. Covering buildings with a smoke screen as protection against deliberate aircraft crashes
4. Additional personnel (and equipment) at the site, for the mitigation of the consequences of an attack
5. Strengthening the guard force
6. Implementing additional measures of accident management

Furthermore, the effects of a core melt accident induced by a terror or military attack could be somewhat mitigated by reducing the potential source term (e.g. by removing spent fuel from the storage pool near the reactor, or by stopping plutonium fuel manufacturing and use)

Potentially, all those measures can also increase protection against acts of war.

In connection with terror attacks, further measures are also under consideration, which belong to the military, police or administrative sector.

Preventive shutdown:

Preventive shutdown of a nuclear power plant in case of a threat can increase safety margins against all types of attacks. In particular, it can increase the time span available for counter measures after the attack (intervention time).

In order to achieve a significant safety gain, intervention times of about one day should be available. (In case the primary circuit is destroyed and thus, the barrier around the fuel elements does not remain intact, even this would not be sufficient for effective counter-measures.) The longer the spent fuel has cooled in the shut down reactor, the slower it will heat up after an attack and the longer the intervention time available will be. Since the thermal power of the fuel elements decreases rather slowly, however, after the reactor has been shut down, it would be necessary to shut down a nuclear power plant (light water reactor) several months before the attack, at the latest.

If barriers are compromised, in particular, if the reactor pressure vessel and/or the cooling circuit are damaged, even preventive shutdown cannot guarantee appropriate intervention times. Also, the potential advantages of preventive shutdown are mostly irrelevant if the spent fuel pool is in an exposed position in the reactor building – as is the case in many nuclear power plants.

Structural backfitting against deliberate aircraft crash and other hazards:

In principle, structural backfitting could be a protective measure against attacks of all kind from the air, but also against some other types of attacks. The following options are conceivable:

- Protective buildings against air attacks (e.g. towers)
- Obstacles on the ground against car bomb attacks

The construction of protective buildings around the reactor buildings, on the other hand, is under consideration – for example, in Germany. The construction of such buildings, however, would create specific new problems: If the buildings are placed at a greater distance from the reactor building, their height would have to be considerable. Thus, the buildings would be visible from a large distance. They could serve as orientation points in case of other attacks. If they are placed closely to the reactor building, on the other hand, they will create hindrances for traffic on the site.

The erection of massive reinforced concrete structures leads to another problem. The destruction of such a structure by aircraft attack leads to the formation of heavy concrete pieces that can create damage on the site.

The situation is different regarding the intrusion of attackers with vehicles on the ground. If such intrusion onto the site is effectively precluded, the options for terrorist are reduced. In particular, the use of car bombs in the vicinity of a nuclear plant can be made considerably more difficult. Such obstacles could hinder even a military attacker. However, the traffic frequency is usually high in the surroundings of nuclear plants, and to some extent also on the site itself. This, in practice, creates limits to the implementation of this measure.

Covering buildings with a smoke screen:

Concepts for covering nuclear power plants with smokescreens, mainly for the protection against deliberate crash of an aircraft, are being planned in Germany, where, smokescreens constitute the central element of the NPP operators' protective concept.

Adaptation of a military concept is envisaged. This is problematical, since military smokescreens are used under completely different circumstances. For example, a military smokescreen can protect a warship against attack by an automatic, target-seeking missile. Under cover of the smokescreen, the ship will then withdraw. In case of an attack against a nuclear power plant, the target is not movable. Furthermore, a human pilot who can circle for some time over the target until the smoke has dispersed guides the aircraft. Also, it will probably be more difficult to mislead a human pilot, than an automated rocket system.

The timely triggering of this measure constitutes a further problem. Many nuclear power plants are located close to large airports and air traffic routes. Thus, it is possible that an intention to attack cannot be recognised sufficiently early. Furthermore, even if a screen is successfully created, it will be rather easy to find the target nevertheless.

If, in times of peace, there are air attacks at low height, by helicopter or military aircraft, the smokescreen system will be completely useless. In this case, the attack will only be recognised as such when it is too late. Furthermore, the deliberate triggering of the smokescreen by terrorists (faking an air attack) cannot be excluded – possibly, to launch a ground attack during the resulting confusion.

In times of war, smokescreens probably give better protection, since the system has originally been tailored to military needs, and it is more likely that approaching enemies will be recognised in time. For the protection of an immovable target, the position of which is well known, however, a smokescreen alone will nevertheless not be sufficient.

Additional personnel (and equipment) at the site:

In order to mitigate the consequences of an attack, experts in various fields (medical personnel, fire fighters etc.) are needed on the site. The possibilities and chances for mitigation will be improved if the number of this personnel is increased – be it located directly on-site, or in installations in the vicinity. The corresponding equipment and materials could also be stored at the site.

Strengthening the guard force:

In principle, strengthening of the guard force at the site is a suitable measure to improve protection against a terror attack on the ground. The task of the guard force consists in repelling the attacks of small groups, as well as in delaying larger attacks at least until police and/or military forces arrive.

Strengthening of the guard force, however, can lead to other risks: Members of the guard force could be blackmailed or bribed into supporting attacks; and protective installations on the site (in particular, weapons) could be taken over by terrorists. In case of private guard services, there is also the issue of sufficient quality control and vetting of guards.

In a recently published report on the U.S.-firm Wackenhut, which is, among others, responsible for security at 30 U.S. NPP sites, many shortcomings are listed. This concerns, for example, poorly maintained weapons' inventories, inappropriate storage of explosives, inadequate control over access badges and improperly positioned guards [SEIU 2004]. Another investigation concludes that guard forces frequently are under-manned, under-equipped, under-trained, under-paid and unsure about the use of deadly force in case of a terrorist attack [POGO 2002].

Furthermore, in case of a stronger attack, the guard force is to use delaying tactics, while calling for reinforcements. However, a terror attack is likely to be "successfully" concluded within three to twenty minutes, and will not necessarily be noticed immediately – and help from outside will need about one or two hours to reach the nuclear power plant. In case of military attacks of large units, in particular if those are equipped with heavy weapons, the guard force is still less likely to be able to mount an effective defence.

Additional measures of accident management:

For many years, measures of accident management are planned in most nuclear power plants worldwide, to control a severe accident or to at least mitigate its effects. In connection with the protection against terror attacks, there have been new considerations since September 11, 2001 to further improve accident management. However, it is questionable to which extent the measures already planned could be expanded further. Concrete information on this issue has not been published so far.

Remark on military, police and administrative measures against terror attacks:

Concerning military, police, secret services and administration, the following measures, are conceivable and have, to some extent, already been implemented in some countries:

1. Protection of plants by military (including anti-aircraft defence and control of neighbouring waterways).
2. Measures to prevent hijacking of airplanes, for example improving control of passengers and improved protection of military airplanes.
3. Measures for the early recognition of a skyjacking, for example by improved control of air traffic, and for preventing attacks with skyjacked airplanes.
4. Intensifying measures for vetting and control of plant employees (including sub-contractors) – leading to better protection against insiders.

The first measure mentioned clearly could also improve protection against acts of war.

However, measures like the "militarization" of the energy economy or extensive control of flight passengers as well as intensified vetting and control of personnel must be severely limited, in order to remain compatible with an open and democratic society.

In Germany, the "Air Security Act" (Luftsicherheitsgesetz) of January 2005 permits shooting down of skyjacked planes by the German air force, if other measures are not successful and it has to be assumed with high probability that the lives of the passengers would be lost anyway. This Act is highly controversial and criticised by leading German politicians, among them the Federal President [FR 2005a].

If plants are protected by military units the protection measures themselves can lead to new risks, just as in the case of private guards as military personnel, too could be recruited by terrorists using bribes or blackmail. Furthermore, military installations at the site could be taken over by terrorists. Furthermore, military installations located directly at the plant site alone will be largely useless against certain kinds of attacks, if there is no timely warning – for example in case of a tree-height attack with helicopters.

The insider problem is of particular complexity. Generally, at present, qualified personnel for nuclear plants are scarce. Sub-contractors are extensively used. This considerably increases the "chances" for terror organisations to recruit insiders.

Conclusions

The threats to nuclear power plants form terror attacks and acts of war can be summarized as follows:

- Because of their importance for the electricity supply system, the severe consequences of radioactive releases as well as because of their symbolic character, nuclear power plants are “attractive” targets for terrorist as well as for military attacks.
- Nuclear power plants could be targets in case of war if a military use is suspected.
- The spectrum of possible modes of attack is very diverse. Attacks could be performed by air, on the ground and from the water. Different means/weapons can be used.
- An attack on a nuclear power plant can lead to radioactive releases equivalent to several times the release at Chernobyl. Relocation of the population can become necessary for large areas (up to 100.000 km^2). The number of cancer deaths can reach more than 1 million.
- Protective measures against terror attacks are of very limited use. Furthermore, a number of conceivable measures cannot be implemented in an open and democratic society.
- There is no protection against military attacks, in particular if heavy weapons are used.

Taking into account the vulnerability of nuclear plants as discussed above, it is particularly clear that nuclear power is part of the “hard” path of energy supply, not of the “soft”, sustainable path – and that phase-out of nuclear power would be the best answer to the terrorist threat.

The use of nuclear energy requires construction and operation of a relatively small number of large, centralised installations, with an enormous concentration of capital as well as economic and political power. “Hard” energy systems always have marked military implications. This centralisation also leads to a particular vulnerability against terror or warlike attacks.

The “soft” path, with a maximum of efficiency of energy use and the reliance upon renewables, implies the production of energy in many small decentralized plants. “Soft”, sustainable energy systems are, contrary to nuclear installations, not under the suspicion of being used for military purposes and, thus, will not be targets of attacks for this reason. Furthermore, they are generally less vulnerable to attacks than “hard” systems.

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D.1.ii Vulnerabilities of Reprocessing Plants and Spent Fuel Storage Pools to Terrorism Risks Reprocessing Plants

"It is hard to think of a nuclear terrorist attack which could, at least in theory, be more catastrophic than a successful attack on either the tanks at Sellafield that contain the liquid fission products separated from spent reactor fuel elements by the two reprocessing plants or on the stores holding the plutonium separated by the reprocessing plants."

Dr. Frank Barnaby

Introduction

Spent nuclear fuel reprocessing plants contain by far the largest inventories of radioactive and strategic nuclear materials of any of the elements of the nuclear fuel chain. The sites of these nuclear-chemical plants typically store significant quantities of spent fuel (the equivalent of several dozen nuclear reactor cores), several dozen metric tons of plutonium, thousands of tons of reprocessed uranium and tens of thousands of tons of nuclear wastes in various chemical and physical forms. Reprocessing plants also store significant amounts of traditional dangerous materials like fuel, gas and chemicals.

There is a high probability that a massive attack on such a facility would lead to catastrophic consequences [LARGE]. However, the extent of the damage and devastation caused largely depends on the point and type of impact. The following part of the study is intended to provide a rough overview of potential issues involved and does by no means provide a comprehensive review of the complex problem. Detailed analysis is particularly difficult because of obvious confidentiality issues.

The Spent Fuel Pools

Every nuclear power plant has a spent fuel pool where the nuclear fuel is cooling off at least for a few years. The fuel is kept under several meters of water that constitute an effective radiation barrier. Unshielded, radiation of spent fuel at short distance would be lethal within a few minutes. The minimum time the fuel stays in the on-site pools before it can be transferred into a shipping container depends on the type of fuel and on the burn-up. The theoretical maximum intermediate storage time is several decades and therefore likely extends beyond the operational life of the power plant.

In practice the storage time for spent fuel in on-site pools is limited by the capacity of the pools, safety and security considerations as well as the decision in favour of other spent fuel management options (dry storage, centralised away from the reactor storage or reprocessing). In many cases the quantities of spent fuel stored per pool have been increased through so-called re-racking, that is increasing the density by re-organising the placement of the fuel in the pools.

The quantities of spent fuel stored in pools at a given reactor site can reach several thousand metric tons and therefore constitute a very significant radioactive inventory. However, the largest quantities of spent fuel are stored at reprocessing plants and at few centralised away-from-reactor (AFR) intermediate storage facilities.

Away From Reactor (AFR) Spent Fuel Wet Storage Capacities in the World

Country	Location	Number of facilities	Capacity (in t HM)
France	La Hague	5	17,60
Russian Federation	Kursk,	6	0
	Leningrad,		14,96
	Novovoronezh,		0
	Mayak,		
	Krasnoyarsk,		
	Smolensk		
United Kingdom	Sellafield	4	
Sweden	Oskarshamn	1	
Japan	Fukushima,	3	10,30
	Rokkasho, Tokai		0
Ukraine	Chernobyl	1	8,000
USA	Hanford, Idaho,	4	3,140
	Savannah River		
	West Valley		2,518
Belgium	Tihange	1	2,127
Finland	Loviisa,	3	
	Olkiluoto		
Slovakia	Bohunice	1	1,760
Argentina	Atucha	1	1,694
Bulgaria	Kozlodui	1	
Germany	Greifswald	1	1,690
China	Lanzhou	1	986
India	Tarapur	1	600
			560
			550
			275

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As can be seen from the table above the largest AFR wet storage facilities are in the countries that operate spent fuel-reprocessing facilities (France, Russia, UK, Japan).

The only exception is Sweden with a centralised spent fuel storage facility at Oskarshamn with a capacity of about 8,000 tons where about 4,000 tons are already stored (as of February 05). However, the Swedish CLAB concept provides particular protection for the fuel. The eight spent fuel pools are in 30 m to 40 m depth in a rock formation. The site should withstand major external impact like aircraft crash or armed attack.

The opposite example is the main French plutonium separation facility. As of the end of June 2004, the La Hague reprocessing site stored about 7,900 tons of spent fuel of which close to 500 tons of spent MOX fuel (plutonium-uranium mixed oxide) containing a significantly higher toxic inventory than standard uranium fuel. Rather than decreasing the storage density as one possible post-9/11 precautionary measure, COGEMA was authorised to *increase* the La Hague spent fuel storage capacity by 26% to 17,600 t. None of the buildings at the site are specifically designed to withstand the crash of a large aircraft.

In order to simplify the evaluation of the risk potential, one can limit the consideration to potential release of radioactive caesium-137 that, in the case of the Chernobyl disaster, has contributed approximately 75% of the radiological impact. Spent light water reactor fuel with a burn-up of 33,000 MWd/t contains, after a typical cooling time of seven years, roughly 1 kg of caesium-137 per ton of fuel. At 45,000 MWd/t the ratio is about 1.5 kg/t. The French fuel currently reaches an average of 45,000 MWd/t with a licensed limit in the reactor of 52,000 MWd/t. The spent fuel in the cooling ponds of La Hague has probably an average burn-up of around 40,000 MWd/t and, at about 1.2 kg/t, contains around 10,000 kg of caesium-137, which is over 370 times the amount released at Chernobyl.

In fact, the new operating license of the La Hague facilities, granted in 2003 [JOURNAL OFFICIEL 1993], allows for burn-up levels for up to 75,000 MWd/t. Such high burn-up fuels would have much larger fission product contents than current burn-up fuels with the same cooling time.

In a widely commented study [SCHNEIDER 2001], in 2001 WISE-Paris had calculated the potential release of caesium from a single La Hague spent fuel storage pool. Industry and safety authorities have argued that the assumption that up to 100% of a given inventory of caesium could be released during a major accident would be "unrealistic". WISE-Paris had based this assumption on a 2000 NRC report that had analysed the potential risks of spent fuel pool accident [US NRC 2001].

The La Hague operator COGEMA has argued that even in the case of total loss of coolant "the risk of subsequent melt-down can be discarded". The emptying of the pools "would not be immediate and the heating of the radioactive materials would stretch out over several days leaving the time for the anti-fire brigade of the plant to effectively intervene". [COGEMA] The official Institute for Radiation Protection and Nuclear Safety (then IPSN, now IRSN) stated in a confidential note to the Secretary of State for Industry that the caesium release rate would rather be limited to "less than 10%". In various discussions with the author, IRSN representatives have argued that the heat output would be smaller than anticipated (because of the average age of the fuel stored) and therefore less fuel would be damaged and less caesium released in case of accident. However, it should be noted that IRSN comments on accident scenarios only and explicitly refuses to discuss specific terrorism scenarios. Therefore the statements do not cover any scenario that would either imply a significant external heat source, such as the impact of a fully loaded large aircraft on the site, or major impact on intervention possibilities, such as major destruction after a large explosion. Also, neither the COGEMA nor the IRSN statements have been backed up by any published technical reference document.[9]

In a recent statement on the situation at power plant decommissioning projects, the NRC has downplayed its own findings in the 2000 study. "From the studies completed thus far, it is clear that current decommissioning plant EP [Emergency Preparedness] programs are adequate given the age of spent fuel contained in their pools. Modestly aged fuel will be air cooled under a loss of spent fuel pool water accident. The age of spent fuel dictates the time it would take to heat up the fuel, potentially releasing radioactive nuclides. All spent fuel at the current fleet of decommissioning plants is older than five years and is therefore very slow to overheat even under these more challenging conditions. Regardless of the spent fuel age or configurations considered, the current analyses show that spent fuel heat-up time is longer than previously estimated by NRC in draft NUREG-1738 (…). Based on the analysis performed to date, the staff has not identified any spent fuel pool accident issues that would invalidate the EP planning basis."[US NRC 2004]

[9] Repeated specific documentation requests from the author to IRSN (letters dated 1 and 11 March 05), Thierry CHARLES, Director for plant, laboratory, transport and waste safety have not produced any noteworthy result.

While there are many possible scenarios for terrorist attacks on spent fuel pools and in spite of the new NRC position, at least two cases remain of particular relevance to spent fuel pool integrity, the deliberate aircraft crash and the deliberate explosion of a large gas tanker at the coast.

In the following example one out of five cooling ponds at La Hague would be severely damaged in the course of a plane crash scenario. The middle sized D pond with a capacity of 4,600 t has been taken as example and a 50% load factor is envisaged. The potential for a self-sustaining zirconium fire, following loss of water, increases with the packing of fuel pools to high densities. In 2003 COGEMA was authorised to increase the storage capacity of the pond by 1,110 t (+32%). Under the present scenario, the loss of water in the D cooling pond could lead, because of exothermic oxidation reactions of zirconium and other metals, to the release of caesium-137 and other fission products contained in 2,300 t of spent light water reactor fuels stored. If one considers a release of only 5% or 150 kg of the close to 3,000 kg of caesium in the pond, it would correspond to over five times the total amount of caesium-137 released as a consequence of the Chernobyl accident.

It should be stressed that this is a rather conservative estimate because:

- The spent fuel pool could contain twice as much fuel.
- The caesium content per ton of fuel increases almost linearly with burn-up, so the caesium content might be much higher because of increasingly higher burn-ups;
- The caesium release rate could be much higher than 5%;
- The caesium contained in 10 year old light water reactor spent fuel represents only about half of the fission product activity;
- Impact on La Hague could destroy more than one pool, damage safety equipment and make short term access to the pools impossible;
- In addition, the five La Hague pools contain a no longer published amount of non-irradiated waste from the MOX fabrication. The amount is certainly considerable, probably around 100 tons or more containing several tons of plutonium;
- The release pattern could be much worse and lead to much higher contamination levels than in the case of Chernobyl when initial explosions and a 10-day lasting fire dispersed most of the released radioactivity high into the atmosphere and over very large territories.

A group of US scientists has calculated that even for a limited release (equivalent to 1.75 times the Chernobyl caesium release) due to a spent fuel pool accident at a US nuclear power plant or the attack against the facility the areas calculated as “contaminated above 100 Ci/km2 are 5–9 times larger than the area contaminated to this level by the 2 million Ci release from the Chernobyl accident” [ALVAREZ].

The La Hague example above covers only light water reactor fuel. Other fuels have other particular problems. As the British nuclear consultant John Large points out: “For Magnox elemental metal fuel, both the cladding and fuel are pyrophoric and in-situ burning of the fuel could result in a very significant release of respirable-sized fission product particles[10] again released with emulsions of aviation fuel with greater efficacy of dispersion.”[11]

[10] The Magnox magnesium alloy cladding will ignite in air at about 600-700ºC and the Magnox uranium metal at about 220ºC or lower if hydrides have formed on the surface.
[11] Personal communication, 11 March 05

Precautionary Measures

In October 2002, the Attorneys General of 27 States of the USA, in an unprecedented move, wrote a letter to the key representatives of Congress: "An interagency Task Force—chaired by the NRC and working in concert with the Director of Homeland Security, should be created and tasked to recommend changes to enhance the physical security of nuclear plants, increase security measures, expand emergency response capability, and enhance protections for one of the most vulnerable components of a nuclear power plant—its spent fuel pools". [ATTORNEY GENERAL]

There are several ways to reduce the potential risks of accidental or voluntary impacts on spent fuel wet storage facilities:

- Increase physical protection;
- Lower density in spent fuel pools;
- Lower inventories of storage facilities.

While the significant increase in physical protection of an existing spent fuel pool beyond a certain level (access restriction, surveillance, physical barriers, etc.) is complex, the transfer of as much spent fuel as possible from pools into dry storage casks would allow to decrease at the same time the radioactive inventory at a given site and to decrease the storage density per pool. A group of US academics suggests the following for the case of US spent fuel stored at reactor sites, an estimated 96% in pools. [ALVAREZ]:

> "To reduce both the consequences and probability of a spent-fuel-pool fire, it is proposed that all spent fuel be transferred from wet to dry storage within five years of discharge. The cost of on-site dry-cask storage for an additional 35,000 tons of older spent fuel is estimated at $3.5–7 billion dollars or 0.03–0.06 cents per kilowatt-hour generated from that fuel. Later cost savings could offset some of this cost when the fuel is shipped off site. The transfer to dry storage could be accomplished within a decade. The removal of the older fuel would reduce the average inventory of 137Cs in the pools by about a factor of four, bringing it down to about twice that in a reactor core. It would also make possible a return to open-rack storage for the remaining more recently discharged fuel. If accompanied by the installation of large emergency doors or blowers to provide large-scale airflow through the buildings housing the pools, natural convection air cooling of this spent fuel should be possible if airflow has not been blocked by collapse of the building or other cause."

There might be some costing differences in the case of other countries, but it is obvious that the principle is valuable for any wet spent fuel storage facility, whether at reactor sites or AFR. The only noteworthy exception remains the Swedish example of the subsurface intermediate storage facility CLAB. However, centralised stores always need shipment between the reactor and the storage facility and therefore entail exposure to vulnerability during transport (see section D.1.iv).

In April 2005, the US National Research Council published a report which concluded that the spent fuel current stored in ponds may be at risk from terrorist attacks and called for additional analyses to obtain a better understanding of potential risks and to ensure that power-plant operators take prompt and effective measures to reduce the possible consequences of such attacks. The report identified several scenarios that could have serious consequences at some plants including that an attack which partially or completely drains a plant's spent fuel pool might be capable of starting a high-temperature fire that could release large quantities of

radioactive material into the environment and called for measures to counter the threat [UCNRC 2005]

The Plutonium Stores

The amount of plutonium in store is steadily increasing. While the US and Russia agreed to dispose each of 34 t of "excess" weapons grade plutonium, the world's "civil" plutonium stockpile exceeds 230 t. As of the end of 2002, the largest holder of plutonium is the UK with over 90 t, of which 20 t owned by foreign utilities, followed by France with 80 t, of which 32 t owned by foreign utilities and Russia with over 37 t. The plutonium in the UK is stored at the Sellafield site, most of the plutonium in France at La Hague and most of the Russian plutonium in Mayak.

It is remarkable that the bulk of the plutonium stocks in the world are at the same sites as the largest concentrations of spent fuel. Plutonium has two particular characteristics, it is of high strategic value as primary weapon ingredient and it is highly radiotoxic. A few kilograms are sufficient in order to manufacture a fission weapon, a few micrograms inhaled are sufficient to develop cancer.

In March 2002 the Royal Society of Edinburgh (RSE) stated: "What the September 11 events also necessitate is improved physical security of separated plutonium storage, sufficient to withstand direct impact by fully-fuelled large-capacity aircraft and ballistic missiles and capable of being policed and defended against all foreseeable forms of terrorist attack"

Unfortunately, none of the buildings at Sellafield or La Hague have been designed to withstand any of these extreme impacts. While this does not mean that the structures could not withstand such an impact under some circumstances, the probability that they would resist impact is limited. The worst release mechanism for plutonium, usually stored in oxide form, is a large fire that would render plutonium particles airborne in micron sizes that are inhalable.

According to Frank Barnaby, former director of SIPRI "if evenly distributed, a kilogram of plutonium in the Sellafield store will, on average, contaminate more than 300 square kilometres to the level at which the NRPB [National Radiation Protection Board] recommends evacuation. A terrorist attack on a plutonium store at Sellafield could contaminate a huge area of land." [BARNABY]

The Radioactive Waste Storage

The sites of large-scale reprocessing plants not only hold the largest inventories of spent fuel and separated plutonium but also the largest quantities of a great variety of conditioned and unconditioned radioactive wastes of any nuclear site.

As of 1 April 2001, the UK radioactive waste management agency NIREX included in the inventory for Sellafield the following wastes:

- 1,440 m3 high level waste in liquid form stored in 21 storage tanks;
- 340 m3 of conditioned (vitrified) high level waste;
- 51,000 m3 intermediate level wastes (ILW); only 15% of UK ILW is conditioned;

As of the end of 2002, the French radioactive waste management agency ANDRA presented the following inventory of high and intermediate level wastes for La Hague (excerpt) [ANDRA]:

- 1,162 m3 unconditioned liquid high level waste;

- 7,697 containers with vitrified high level waste (122 x 10^{18} Bq);
- 807 t of unconditioned hulls and nozzles;
- 3,740 drums of hulls and nozzles to be reconditioned;
- 969 t of unconditioned graphite sleeves;
- 9,301 m3 of unconditioned plutonium bearing sludges;
- 10,244 bitumen packages.

The presence of unconditioned wastes is particularly problematic because they are close to a dispersible state. The storage of high-level liquid waste stemming from the spent fuel dissolution represents the highest potential hazard because of the combination of the large inventories of radioactivity and the physical-chemical state. While the fission products stored in the form of liquids at Sellafield and La Hague contain several hundred times the amount of cesium-137 that was released at Chernobyl, unconditioned sludges often contain high levels of plutonium. The 9,300 m3 of sludges at La Hague contain an estimated 160 kg of plutonium. This situation leads also to severe safeguards difficulties. Faced with the impossibility to properly account for an estimated 1.3 t of plutonium in the B30 pond at Sellafield, in March 2004 the European Commission issued a directive against the UK for non-compliance with its safeguards regulations.

While the specific problem of the liquid high-level waste at Sellafield has been the subject of many publications and particular action by the UK safety authorities[12], the situation at La Hague has gone entirely unnoticed. While the original stocks from the reprocessing at the old UP2-400 should have been absorbed a long time ago, the situation seems to have deteriorated over the years or, at best, has been stable. The stock has increased by 50 m3 between 2001 and 2002. According to the French Nuclear Safety Authorities, at the end of September 2004 the stock stood at 1,145 m3 of which 230 m3 where stored at UP2-400, 760 m3 at UP2-800 and 155 m3 at UP3. [TERNEAUD]

At the middle of 1992, according to a representative of the Safety Authorities, the volume of liquid waste at UP2-400 stood at 930 m3, down from 1,200 m3 at the end of 1990. The volume at UP3 was at 180 m3 (as of the end of 1991). [CSPI] However, in a December 1992 note the Safety Authorities state that "after having culminated at 1,070 m3 up to 1990-91 (about have of the storage capacity), the volume of the stored fission product solution has been reduced to 724 m3 in June 1992. It is planned that the volume will be reduced to 445 m3 as of 1 July 93 in order to be stabilised at 400 m3 as of 1994." [SAFETY AUTHORITY] Ten years later, the volume is three times the level.

The representative of the Safety Authority has declined to indicate any data on the radioactive or thermal inventory or on the evolution of the stocks or any reasons for the current size of the stocks. [SCHNEIDER 2005] He simply indicated that the Authority would consider the storage facilities at La Hague "perfectly safe" and "not comparable" to the installations at Sellafield. However, he declined to indicate in what respect the design of the facilities would make them safer than the Sellafield storage. He also stated that the stock would have declined from the September 2004 figure, but could not indicate by how much.

[12] See for example NII, *The Storage of Liquid High Level Waste at BNFL Sellafield - An Updated Review of Safety*, February 2000 and NII, *The storage of liquid high level waste at BNFL, Sellafield - Addendum to February 2000 Report,* August 2001. In 2000 BNFL agreed to a reduction strategy that would limit the stock to a buffer quantity of 200 m3 by 2015. Instead, in September 2001, BNFL was ordered to shut down its reprocessing plants because, due to malfunctioning of the vitrification facilities, the stock had risen by another 100 m3 over the year

Conclusion

Spent nuclear fuel and central radioactive waste storage facilities contain by far the largest inventories of radioactive substances of any facility throughout the nuclear fuel chain. Spent nuclear fuel in cooling pools as well as unconditioned high level radioactive wastes in liquid and sludge form are particularly vulnerable to attack. The main reason for this is that they are present in readily dispersible form in storage facilities that are not designed to withstand large aircraft crash or an attack with heavy weapons. Storage facilities at reprocessing plants contain hundreds of times the radioactive inventory that was released as a consequence of the Chernobyl disaster.

In addition reprocessing facilities store dozens of tons of plutonium, some of which could be dispersed as a result of a major fire that could be triggered by accident or terrorist attack. The inhalation of a few dozen micrograms (millionth of a gram) can trigger a lethal lung cancer. Plutonium could also be diverted for weapons purposes. Several kilograms are sufficient for the manufacture of a crude nuclear device.

The situation at the La Hague facility raises many questions. Public attention has so far focussed on the risk potential of the spent fuel pools, while at Sellafield the storage of liquid high level radioactive waste has been the subject of major concern. However, the La Hague site also stored very significant amounts (> 1,100 m3 as of September 04) of unconditioned liquid high level radioactive waste, a fact that has not been the subject of any independent expert review or any public attention so far.

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D.1.iii: Terrorist Attacks on Spent Fuel Storage Sites with Cask Storage

Introduction

Increasingly, dry storage in casks is being practised worldwide for longer-term intermediate storage of spent fuel. After a decaying period of several years in the nuclear power plant's storage pool, the spent fuel is loaded into massive containers that are then placed in a storage facility, which is usually located at the power plant site. Casks are cooled by passive airflow.

Cask storage facilities are in use in many countries. They are planned, and to some extent already built, at all but one of the sites of nuclear power plants operating in Germany; in addition, there are three central cask storage facilities in that country. In the United States, about 30 facilities of this type were operational by mid-2004 [REUTERS 2004]. Belgium, the Czech Republic and Japan, among other countries, are also using this concept [WNIH 2004].

The casks are designed to provide protection against external impacts. For example, the CASTOR V/19, a modern German cask type, is a cast iron cylinder with a length of about 6 m and a diameter of about 2.5 m. Wall thickness is 0.42 m. The weight of the cask, loaded with 19 spent fuel elements, is about 125 t [EON 2000].

In most countries, casks are set up in a light building designed to give shielding against the weather. In the United States, casks are usually put on concrete foundations in open air.

Spent fuel in casks is generally less vulnerable to attacks than is fuel stored in pools. In a pool, once the water cooling circuit is interrupted, a severe accident can develop leading to melting of all the fuel in the pool, and accordingly, to very large releases (see section D.1.i). In a cask storage facility, there are no central cooling systems the destruction of which could lead to comparably all-encompassing effects.

However, casks in a storage facility at the reactor site can be better accessible for attacks, in particular if they are standing in the open. There are several modes of attack that could lead to destruction of one or several casks. While releases would tend to be smaller than releases resulting from the attack of a pool storage facility of comparable capacity, they could be no means be regarded as negligible or trivial.

Conceivable Attack Scenarios

Most of the attack scenarios listed in section D.1.i also apply for cask storage facilities. Some examples are given here which are particularly relevant for cask storage.

The scenarios here are discussed with the "massive" cask type in mind, i.e. a cask type the walls of which are thick enough to provide shielding. They basically are also valid for the other type of cask presently in use, consisting of thin-walled canister plus overpack. Worldwide, the "massive" cask type is predominantly used; in some countries, particularly the United States, however, the other type is also currently employed.

<u>Attack from the air:</u>

The deliberate crash of a large airliner like the Boeing B-747 or the Airbus A-380 which will be commissioned for commercial flights 2006 can lead to considerable radioactive releases from a cask storage facility. The large volumes of kerosene those airliners can carry in their tanks (more than 200 m^3 for the B-747, more than 300 m^3 for the A-380) constitute the main risk factor.

An investigation of possible accident scenarios for the cask storage facilities licensed at NPP sites in southern Germany led to the following results: If about 150 m^3 kerosene get into the

storage area, a hot fire with a duration of several hours can result. A large number of casks – in the order of 20 – would be subjected to this fire. They would be heated up, and the seals of the cask lids would eventually fail. Volatile radionuclides would then be released into the atmosphere [UBA 2002]. The fire is likely to be shorter in case of an open-air facility, since in that case, the kerosene will be more easily dispersed than in case of a fire in a (partly destroyed) storage building. This is the only case, however, were storing casks without a protective building actually can be advantageous.

Also conceivable is a bombing attack with a military airplane the pilot of which has been bribed, blackmailed or "converted" by a terrorist group. A 2000 pound bomb like the BLU-109 that can be carried by fighter-bombers around the world could be used; the BLU-109 is specifically designed to penetrate hardened targets [BGT 2004; GLOBAL SECURITY 2004]. A light building would be no obstacle; the bomb could enter deeply into the cask wall. The subsequent explosion would severely damage the cask. In case of light water reactor fuel, zircalloy fragments would be produced from the fuel element hulls that would start burning since air can ingress into the cask. Eventually, all or most of the zircalloy inventory of the cask would participate in the fire. A significant amount of the volatile radionuclide inventory of the cask would be released.

Attack with armour-piercing weapons from outside the facility:

Armour-piercing weapons like the U.S. Javelin, the French/Canadian Eryx or the German Panzerfaust 3 can go through 600 mm of steel and more and hence are more than powerful enough to penetrate spent fuel storage casks. They can be carried and fired by one or two people and are widely used in all armies over the world.

The attack can be performed from a distance of 50 to several 100 m. If the casks were located in a building, the building's wall would have to be pierced first. This could be achieved, for example, by firing a different type of warhead with the weapon used or by shelling with a mortar. If a cask is hit and its wall penetrated, part of the fuel content will be vaporized, and a further part will be broken up into small particles. If air enters the cask, a zircalloy fire will result, further increasing releases.

Attack by entering the storage facility:

Attackers entering the storage area could severely damage casks with explosives and other equipment that could be carried by a group of people. It would be difficult in this case for the attackers to escape unharmed; however, suicide attacks cannot be excluded.

Using shaped charges, or an oxygen lance to pierce holes into the wall of a steel or iron cask which are then filled by explosives, large openings could be created in the wall(s) of one or several casks. They would allow air intrusions into the cask(s), and hence a zircalloy fire. Furthermore, shielding of the radiation from the spent fuel elements would be significantly reduced; large parts of the storage area would be subject to very high dose rates. Counter measures like attempting to cover the openings in the casks wall thus would be practically impossible. Significant releases, as in the case of an air attack, can result.

Consequences of an Attack on a Cask Storage Facility

Releases will be discussed here using caesium-137 as lead nuclide. Cs-137 is radiologically important due to its hard gamma-radiation; and it is one of the most volatile radionuclides.

The quantity released will depend on the detailed circumstances of the attack, the configuration of the storage area as well as the damage done to the building if the storage is not in open air. Furthermore, it depends on the cask inventory – i.e. the mass and burn-up of the spent fuel

elements. Only the possible orders of magnitudes of releases can be given here for orientation. A cask with an inventory of 10 t of spent PWR fuel with typical burn-up is assumed.

In case of the crash of a commercial airliner with kerosene fire, releases can be in the order of 1,000 TBq of Cs-137 (possibly less for an open air facility).

For the other attack modes, releases from 500 TBq (attack with an armour-piercing weapons and damage of one cask, no zircalloy-fire) up to about 100,000 TBq (bombing attack from the air, two casks hit, zircalloy fire as well as fire from incendiary bombs) must be expected, with somewhat higher values in case of an open air facility.

At the Chernobyl accident, 85,000 TBq of Cs-137 were released [NEA 1995]. Thus, releases in case of an attack at a cask storage facility can be in the order of magnitude from about one percent of Chernobyl releases to about 100 percent or even somewhat more. These releases do not reach the extent possible for accident at large spent fuel pools, but nevertheless they have to be regarded as catastrophic.

Countermeasures

There is little scope to reduce the vulnerability of the casks as such. Further thickening of the cask walls hardly seems feasible, with cask masses already around 100 tons. Handling in the storage facility would become difficult, and transport a near impossibility, if the mass was significantly increased. (It should be noted that many storage casks are dual-purpose and are also used for transports.)

Thick-walled, smaller casks with smaller fuel inventory are conceivable. However, the vulnerable surface of a storage facility would be thus increased. Also conceivable would be the provision of additional overpacks like a concrete hull for each cask that could go a little way towards better protection.

The most promising countermeasure appears to modify the layout of the whole storage facility towards a “robust and dispersed” concept which has already been proposed in the United States [THOMPSON 2002]. This concept implies spacing of the individual casks at greater distances and surrounding each cask with a massive concrete structure (bunker), as well as a conical mound of earth, gravel and rocks.

Siting the storage facility below ground would provide further protection; on the other hand, it would raise the danger of becoming a final repository by default (which also, to some extent, has to be seen in connection with the robust and dispersed storage concept). The robust and dispersed storage concept requires a significant area per cask and hence appears best feasible and secure for small inventories of spent fuel. Large inventories would require spreading the facility over a large plot of land, rendering guarding and securing against attacks or infiltrations on the ground difficult.

Some experts request protection measures for dry storage facilities which are still going further. Arjun Makhijani, President of the Institute for Energy and Environmental Research (IEER) considers that “dry cask storage of spent fuel in present day systems licensed by the Nuclear Regulatory Commission also does not meet the criteria of secure, hardened storage”. [MAKHIJANI 2003]

Makhijani defines his criteria for Hardened On-Site Storage (HOSS) of spent fuel against terrorist attack as follows:

1. It should not result in catastrophic releases and should be able to resist almost all types of attacks. The estimated amount of radioactivity that would be released in even severe attacks should be small enough that the storage system would be unattractive as a terrorist target.

2. It should be able to withstand a direct hit by a large commercial airliner full of fuel or anti-tank weapons without catastrophic offsite releases.
3. The individual canister locations should not be easily detectable from offsite. This means that it must not be visible from offsite and the infrared signature should be obscured enough to prevent a direct hit in case of attack with infrared guided munitions."

One possibility envisaged by Makhijani is the construction of silos "resembling small hardened missile silos. Spent fuel could be put in large casks that are then emplaced in these silos. A building would cover the entire set of silos."

Conclusion

Storage of spent fuel in casks is vulnerable to terrorist attacks, like other forms of storage. Radioactive releases from attacks are likely to be smaller than those that would result from attacks of storage pools. On the other hand, accessibility of casks appears to be greater than of spent fuel pools located in massive buildings.

Improvements of the storage concept are conceivable. However, they likely only have a chance of being implemented if the inventories are not too large; for example, in case of a rapid phase-out of nuclear power.

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D.1.iv: Terrorist Attacks on Nuclear Transports

The commercial use of nuclear power is dependent on a worldwide system of transports of radioactive materials. Many transports occur year by year, between uranium mining and processing, enrichment, fuel element fabrication, nuclear power plants, intermediate waste storage facilities, reprocessing plants and other installations.

During transport, radioactive substances are a potential target for terrorists. Of the numerous materials being shipped, the following are the most important:

1. Spent fuel elements from nuclear power plants and highly active wastes from reprocessing (high specific inventory of radioactive substances)
2. Plutonium from reprocessing (high radiotoxicity, particularly if released as aerosol)
3. Uranium hexafluoride – uranium has to be concerted into this chemical form in order to undergo enrichment (high chemical toxicity of released substances, resulting in immediate health effects in case of release)

Since the amounts transported with one shipment are about several tonnes at most, the releases to be expected will be smaller by orders of magnitudes than those that result from attack of a storage facility – even if the transport containers are severely damaged. On the other hand, the place where the release occurs cannot be foreseen, as attacks can occur, in principle, everywhere along the transport routes. Those routes often go through urban areas; for example at ports or during rail transport.

Thus, releases can take place in densely populated regions, leading to severe damage to many people, even if the area affected is comparatively small.

The hazards of terrorist attacks against nuclear transports will be illustrated in this section by two examples: Spent fuel or highly active reprocessing waste transports, and uranium hexafluoride transports. In the first case, the hazard is due to the intense radiation of the released substances; in the second, to their high chemical toxicity.

Attack of a Spent Fuel or Highly Active Waste Transport [HIRSCH 2001]

Transport casks for spent fuel or highly active waste from reprocessing typically are massive cast iron or steel container with a weight of around 100 tons. (For more details, see sub-section D.1.iii) They are transported on a heavy truck or special rail wagon.

A transport cask for spent fuel or highly active reprocessing waste can be penetrated by an armour-piercing weapon, which can be fired from a distance of several hundred meters (see sub-section D.1.iii). Releases can be in the order of 500 TBq of caesium-137 (which is taken as the lead nuclide here, due to its radiotoxicity and volatility) or more.

Most of the caesium released will settle over an area of a few 1000 m^2 around the location of the damaged cask, creating an intense radiation field which will render the rescue of people extremely difficult and hazardous, and countermeasures (such as sealing the opening in the cask wall) all but impossible.

Part of the caesium will spread further, in wind direction, as a radioactive cloud of small particles. Ground will be contaminated by fall-out (and wash-out in case of rain). If there are unfavourable weather conditions, ground contamination can reach values to necessitate permanent resettlement of the population in distances of 5 km downwind, or more. Thus, villages and small towns at the transport route might have to be evacuated to a large extent, and

cease to exist as functioning social units. Effects can still be more severe in case of spent fuel transports if cask damage is such to permit sufficient air ingress so that fuel hulls, consisting of inflammable zircalloy, can start burning.

Attack of a Uranium Hexafluoride Transport

Uranium hexafluoride is transported in containers of the type 48”Y, if it is material yet to be enriched, or depleted uranium. Those are steel containers with a wall thickness of merely 16 mm; they can be loaded with up to 12.5 t UF_6. One container can be transported on a truck; in case of rail transport, up to three on a wagon [URENCO 2001].

If a road transport of uranium hexafluoride is attacked, a tanker with petrol or liquid gas could be used as a “weapon”, particularly in case of a suicide attack. After a violent collision with the uranium hexafluoride transport, the tanker will be severely damaged (particularly if it also carries an explosive load). At the site of the accident, a hot fire lasting several hours will result.

A container of the type 48”Y will fail after about 50 minutes in a fire with a flame temperature of 800° C. Failure will occur earlier in case of higher flame temperatures (1000° C and more could in fact be reached). The steel cylinder will burst. Part of the UF_6 will be ejected high into the air, the remainder will be thrown piecewise in the nearer surroundings. Chemical reaction with the humidity of the air will produce, among others, HF (hydrofluoric acid). HF is a very severe respiratory poison as well as contact poison.

It is questionable whether the fire brigade will be able to extinguish such a severe fire before the container bursts. Attempts to extinguish the fire with water, after the container has failed, can increase formation and release of HF.

In the immediate vicinity of the site of the accident (up to about 100 m distance), there is acute mortal danger. In a distance of up to 500 m, people will suffer severe poisoning and burning from HF. In case of longer exposure times, there is mortal danger also in this region. Even in distances of more than 1 km, there is the risk for health damage for sensitive people [ALBRECHT 1988].

The short-term consequences of such an attack, regarding health effects and deaths caused by HF, can be drastic – in particular, if the attack takes place in a densely populated region. It is possible that thousands of people will be killed and injured. Additional effects will result from uranium contamination. Uranium is a metal of relatively low specific activity, but considerable chemical toxicity. If it is the product of reprocessing, it could contain further toxic radio-nuclides. If the attack takes place in a rural area, on the other hand, there will be severe damage to plant and animal life.

Conclusion

Terrorist attacks against transports of radioactive materials can occur almost everywhere in industrialized countries. The consequences, in terms of radiological land contamination and/or chemical poisoning, can be dramatic.

This hazard is one more reason why nuclear materials’ transports should be avoided as far as possible, and the materials stored at the place of their origin. To make storage facilities less vulnerable, robust and dispersed storage concept should be implemented, as described for spent fuel casks in sub-section D.1.iii.

Regarding uranium hexafluoride, uranium should be neither transported, nor stored in this chemical form. Conversion to more chemically inert and physically stable compounds is possible.

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D.2 Climate Change and Nuclear Safety

Introduction

Global climate change is a reality. There is a broad consensus among scientists regarding this issue. The exceptionally hot summer of 2003 in Europe will not remain an exception. On the contrary, such extreme weather events may become the norm [MUNICH 2005].

In this section, extreme weather events and their effects on the hazards of nuclear power plants will be discussed. The discussion will concentrate on those parts of the world where most nuclear power plants are operating.

Regarding climate change and the consequences to be expected, the section is based mainly on publications of the Intergovernmental Panel on Climate Change (IPCC) and of the Munich Re re-insurance company. The IPCC has been established in 1998 by the WMO (World Meteorological Organisation) and the UNEP (United Nations Environment Programme) to assess scientific, technical and socio- economic information relevant for the understanding of climate change, its potential impacts and options for adaptation and mitigation. The IPCC's *Third Assessment Report* was completed in 2001 [IPCC 2001]. The Geo Risks Research team at Munich Re has been analysing and documenting natural hazards and the effects of climate change throughout the world for about 30 years now. For risk managers in the insurance industry it is important not only to look at the individual events in isolation but also and above that to consider identifiable trends.

In order to demonstrate the beginning climate change, natural extreme events which have occurred in the years 2003 and 2004 will be briefly reported. Of particular importance are storms and floods, which also may occur in combination. Furthermore, incidents that have already taken place at nuclear power plants during extreme weather conditions and which can be regarded as the harbingers of future events will be discussed. Those incidents clearly show how complex the situations created by extreme natural events can be. Site-specific problems became apparent in most cases. Furthermore, it became clear that the personnel of the plants concerned were not prepared for this type of emergency and that the access to the site from outside, and hence the possibilities to bring in outside help, were severely restricted.

For a realistic assessment of the hazards, the possibility of two extreme events occurring at about the same time also has to be taken into account. Two extreme events can have the same cause (as in the case of a tropical cyclone which is accompanied by strong rain, giving raise to floods). Furthermore, because of the increasing frequency of extreme events, it cannot be excluded that an NPP site will be hit by two independent events within a short time – the second event occurring while the damage from the first has not yet been repaired.

Extreme weather conditions can lead to failure of the electricity grid. In this situation, emergency power systems are required which are not necessarily sufficiently reliable and the failure of which can lead to a severe accident. Therefore, blackout situations are specially considered in this section.

Finally, counter-measures and their limits are briefly discussed.

Climate Change – an Overview

Observed Changes in the Climate System

It is more than likely that human activities have already caused a discernible impact on observed warming trends. There is a high correlation between increases in global temperature and increases in carbon dioxide and other greenhouse gas concentrations during the era of rapid industrialization and population growth, from 1860 to the present. The IPCC describes the observed changes in the climate system as follow [IPCC 2001]:

- The global average surface temperature has increased by 0.6 ±0.2°C since the late 19th century. More hot days and heat waves over nearly all land areas are expected.
- Annual precipitation on land has continued to increase in the middle and high latitudes of the Northern Hemisphere (in the order of 0.5% to 1%/decade), except over Eastern Asia.
- A decrease of the cover of land by snow and ice continues to be positively correlated with increasing land-surface temperatures. Satellite data show that, very likely, there have been decreases of about 10% in the extent of snow cover.
- Based on tide gauge data, the rate of global mean sea level rise during the 20th century is in the range 1.0 to 2.0 mm/yr, with a central value of 1.5 mm/yr. Deep ocean temperatures change only slowly; therefore, thermal expansion would continue for many centuries. After thermal expansion, the melting of mountain glaciers and ice caps is expected to make the largest contribution to the rise of sea level over the next hundred years.

Based on the temperature forecasts, the IPCC has produced a list of likely effects of climate change, most of which are negative for human living conditions. These include more frequent heat waves; more intense storms (hurricanes, tropical cyclones, etc.); increased intensity of floods and droughts; warmer surface temperatures, especially at higher latitudes; rising sea levels, which could inundate coastal areas and small island nations.

Extreme Events

The results of research performed by climate scientists indicate that even slight changes in temperature have a tremendous impact on the corresponding extreme values (values within the upper or lower ten percentiles). It is to be feared that extreme events which can be traced to climate change will have increasingly grave consequences in the future. However, in general, the analysis of extreme events in both observations and coupled models is not well developed. Trends in severe weather events are notoriously difficult to detect because of their relatively rare occurrence and large spatial variability [IPCC 2001].

Precipitation

Precipitation amounts increase if the earth gets warmer because more water evaporates and the atmosphere can hold more water vapour. The water cycle becomes more intense as a result and the probability of torrential rain rises substantially. It is likely that total atmospheric water vapour has increased several percent per decade over many regions of the Northern Hemisphere (where 99 % of all NPPs are located). New analyses show that in regions where total precipitation has increased, it is very likely that there have been even more pronounced increases in heavy and extreme precipitation events. In some regions, however, heavy and extreme events have increased despite the fact that total precipitation has decreased or remained constant. More intense and more frequent precipitation events increased flood, landslide, avalanche und mudslide damage, and also soil erosion.

Storms

Analysis of and confidence in extreme event simulations within climate models are still emerging, particularly for storm tracks and storm frequency. “Tropical-cyclone-like” vortices are being simulated in climate models, although considerable uncertainty remains over their interpretation to warrant caution in projections of tropical cyclone changes.

The regions affected by rare and exceptional events in 2004 were regions where the exposure to tropical cyclones was well-known from historical time series. What made 2004 special was the fact that tropical cyclones had reached regional frequencies and – in the case of Hurricane Ivan – intensities that were observed for the first time in the time ever since meteorological data are available. To attribute this to the natural variability of storm activity is no longer really plausible. The results of climate simulations indicate that the storm hazard in the Atlantic will continue to increase in the long term. 2004 was a year of storm records not only in the Atlantic but also in the West Pacific. The number of tropical cyclones in Japan was the highest ever since the systematic recording of tropical cyclone track data first began [MUNICH 2005].

Projections of Future Changes in Extreme Events

Precipitation extremes are projected to increase more than the mean values. The frequency of extreme precipitation events is projected to increase almost everywhere.

There is little agreement yet among models concerning future changes in mid-latitude storm intensity, frequency, and variability. There is little consistent evidence concerning the changes in the projected frequency of tropical cyclones and areas of formation. However, some modelling studies suggest that the upper limit of the intensities could increase. Mean and peak precipitation intensities from tropical cyclones are likely to increase appreciably.

An increase of hot days and heat waves is very likely over nearly all land areas.

Very small-scale phenomena such as thunderstorms, tornadoes, hail, and lightning are not simulated in global models, but an increase is also feared [IPCC 2001].

Uncertainties

A full assessment of the range of climate change consequences and probabilities involves a cascade of uncertainties regarding emissions, carbon cycle response, climate response, and impacts. Furthermore, there are uncertainties associated with the probabilities generated with carbon cycle modelling, and, equally important, uncertainties surrounding climate response sensitivity estimated from climate models. The IPCC also suggested that, particularly for rapid and substantial temperature increases, climate change could trigger “surprises”: rapid, non-linear responses of the climate system to anthropogenic influences expected to occur when environmental thresholds are crossed and new equilibriums are reached.

Furthermore, unexpected feedback effects and irreversible phase transitions in the complex “atmosphere – earth – ocean – ice” system (e.g. changes in ocean currents) can make all prognoses worthless. However, in accordance with the precautionary principle, humankind would be well-advised to be prepared for dramatic changes [SCHNEIDER 2005]

Examples of Natural Hazards in 2003

Together with 2002 and 1998, 2003 was one of the warmest years ever recorded. About 700 natural hazardous events were registered in 2003. 300 of these events were storms and severe weather events, and about 200 were major flood events [MUNICH 2004].

a. Two devastating series of severe weather events and tornadoes destroyed buildings and infrastructure in the U.S. Midwest. More than 400 single tornadoes were counted in just one tornado outbreak in May.

b. With wind speeds of up to 215 km/h and record rainfall of 500 mm, Typhoon Maemi was one of the strongest typhoons in the history of South Korea. Thousands of buildings and bridges were destroyed; whole streets turned into raging torrents in next to no time and became impassable.

c. Heat waves with temperatures of up to 50°C were followed by severe floods in many Asian countries in May and June.

d. Following the heat wave in the summer, December 2003 brought extreme torrential rain to the South of France, affecting the entire southern part of the Rhone valley. 300 mm of rain fell within 24 hours.

e. 2003 was also a year of fires. There were extensive forest fires in Australia, North America, and Europe.

Examples of Natural Hazards in 2004

At the end of the year 2004, South Asia was hit by one of the most devastating natural catastrophes of recent decades. In a very dramatic way it demonstrated the power and unpredictability of nature. This catastrophe was due to a seismic event and was unrelated to global warming. By and large, however, 2004 was dominated by extreme atmospheric events and weather-related natural catastrophes. The year 2004 thus confirmed the fear that has long been expressed by Munich Re: global warming is leading not only to an increase in the frequency and intensity of exceptional weather events but also to new kinds of weather risks [MUNICH 2005].

f. A Tropical Storm/Hurricane formed off the coast in southern Brazil for the first time since observations began – this area had been considered hurricane-free.

g. Japan was hit by ten tropical cyclones between June and October – a record number that was unequalled throughout the previous century. Heavy rainfall and numerous landslides caused severe damage to buildings and infrastructure in large parts of the country.

h. Florida was hit by four hurricanes in the space of a few weeks. Severe damage was generated by Hurricanes Charley, Frances, Ivan, and Jeanne. Between 1850 and 2004, there was only one similar accumulation of four hurricane hits in one US state. Ivan was one of the strongest and most destructive hurricanes since meteorological recordings began.

i. Brazil experienced its worst flood catastrophe of the past 15 years. Heavy rain led to massive flooding in the north and east of the country, which destroyed important infrastructure installations.

j. Major rivers in China flooded their banks from June to September after heavy rain. Hundreds of thousands of buildings were destroyed.

k. 2004 was the fourth warmest year since temperature recordings began (following 1998, 2002, and 2003).

l. Windstorms accounted for almost half of the 650 registered events worldwide.

m. Floods and flash floods accounted for almost a quarter (150) of all natural hazard events in 2004.

Consequences of Climate Change for NPP Hazards

The potential threat climate change constitutes for nuclear power plants can be illustrated by looking at events that have already taken place at NPPs. Such events are getting more frequent with the beginning climate change and must be regarded as precursors of worse incidents and accidents yet to come.

Examples of Flooding

- USA, 1993: In July 1993, the operator of the Cooper NPP on the Missouri River, Nebraska, was forced to shut down the reactor as dykes and levees collapsed around the site closing many emergency escape routes in the region. Below grade rooms in the reactor and turbine buildings had extensive in-leakage with rising water levels. The NRC inspectors noted that plant personnel *"had not established measures to divert the water away from important components"*. For example, water levels rising inside the reactor building impinged on electrical cables and equipment, for example in the reactor core isolation cooling (RCIC) pump room. The RCIC system is critical to plant safety in the event of loss of offsite power [GUNTER 2004].
- Ukraine, 2000: in summer 2000, reactor 3 at the Chernobyl NPP was shut down due to flooding caused by a strong storm. Workers had to pump water out of the reactor building [STATER 2000].
- France, 2003: Electricité de France (EDF) shut two PWRs at Cruas in December 2003 in response to torrential rainfall along the lower Rhone River, prompting French nuclear safety authority DGSNR to activate its emergency response centre for only the second time up to this date. Filters on heat exchangers between the component cooling system and the essential service water system at Cruas-3 and -4 were clogged, hindering operation of the residual heat removal system. At the nearby Tricastin site, clogging of filters on the conventional site caused two more 900-MW PWRs, Tricastin-3 and -4, to scram [NUCWEEK 49_03].
- India, 2004: Kalpakkam-2, also known as unit 2 of Madras Atomic Power Station (MAPS), was operating at nominal power when the giant wave of December 2004 sent seawater into its pump house. Operators brought the unit to safe shut-down. The tsunami swept away 59 people from Kalpakkam town, including five employees of the nuclear installation. Nuclear authorities are now talking about factoring tsunamis into the design of any new nuclear power station to be located near the sea coast [NUCWEEK01_05].

Examples of Storm Events

- USA, 1992: In August 1992, Hurricane Andrew passed directly over Turkey Point NPP (Florida), with a sustained wind speed of 145 – 175 mph (230-280 km/h). There are two nuclear reactors and two oil-fired plants at the site. The plant lost all offsite power during the storm and the following five days. Fortunately, about two years ago the operator was forced to install two new generators; before, they had had only one for each unit. All four generators were working. All offsite communications were lost for four hours during the storm and access to the site was blocked by debris and fallen trees. The nuclear power station's fire protection system was also destroyed. The nuclear power station is one of the few US reactors with important electrical power cables installed on the exterior of the reactor containment buildings. These cable trays and conduits were coated with a fire-resistant material. The hurricane force winds stripped much of the fire resistant coating off these exterior applications, exposing them to any subsequent fire. This was very significant

because the Turkey Point site includes two fossil-fueled units. The fuel oil storage tank of one unit was ruptured by a wind-generated missile spilling over a large amount of combustible fuel oil onto the site. Fortunately, the fuel oil did not ignite with the passing storm [GUNTER 2004; WISE 1992a].

- USA, 1998: In June 1998, the Davis-Besse nuclear power station (Ohio), while at 99% power was hit by a tornado with winds between 113 and 156 mph (180 and 250 km/h). Lightning strikes to the station's switchyard and winds caused a loss-of-offsite power automatically shutting the reactor down. Three independent offsite power lines were knocked down along with the station's telephone fibre optic system. The emergency diesel generators (EDG) A and B were started to power priority safety systems. EDG A had to be started locally because bad switch contacts in the control room prevented a remote start. Then, problems due to faulty ventilation equipment arose, threatening to overheat the emergency diesel generators. Even with the EDGs running, the loss of offsite power meant that electricity supply to certain equipment was interrupted, including the cooling systems for the onsite spent fuel pool. Water temperature in the pool rose from 43° C to 58° C. Offsite power was narrowly restored to Davis-Besse safety systems after 23 hours just as diesel generator B was finally declared inoperable [GUNTER 2004].

- France, 1999: The French electricity grid was hit hard by storms on December 27: About 180 high-voltage towers broke down and nine million people found themselves cut off from the grid. The NPP at Blayais suffered the loss of auxiliary 225 kV power supplies for the four units at the site, as well as a loss of the 400 kV power grid at units 2 and 4. The load shedding design that allows the units to self-supply with electrical power after disconnection from the grid failed. This led to an automatic shut-down of these two units. The diesel generators were started and functioned until the connection to the 400 kV power grid was restored, after about three hours. Furthermore, a flood caused by the confluence of the rising tide with exceptionally strong winds resulted in the partial submergence of the Blayais site. The flood started two hours before the tidal peak. At 10:00 pm, a high water alarm for the Gironde was transmitted to unit 4. It is noteworthy that the information concerning the high level was not transmitted to units 1, 2, and 3.
  The winds pushed the water over the protective dyke. Invading the site through underground service tunnels, the water flooded the pumps of unit's 1 essential service water system (ESWS), and one of the two trains (with two essential service water system pumps each) was lost because the motors were flooded. Furthermore, other facilities were flooded; most notably:
  - Some utility galleries, particularly those running in the vicinity of the fuel building linking the pump house to the platform;
  - some rooms containing outgoing electrical feeders. The presence of water in these rooms indirectly led to the unavailability of certain electrical switchboards;
  - the bottom of the fuel building of Units 1 and 2 containing the rooms of the two LHSI pumps and the two containment spray system pumps. The nuclear operator considered that the pumps were completely unavailable. The systems to which these pumps belong are vital for safety and are designed mainly to compensate for breaks in the primary system.
  The French standard safety rule contains two criteria for flood protection: (1) placing the platform that supports safety-relevant equipment at a level at least as high as the maximum water level; and (2) blocking any possible routes through which external waters could reach reactor safety equipment located below the level of the site platform. At Blayais, both criteria were not met: the concrete platform was 1.5 meter too low; and the resistance of the fire doors in the tunnels to the underground safety equipment was miscalculated: the waters surged into the tunnels and simply broke through the doors. Before the incident, EDF

declared that the underground tunnels were perfectly safe. Before the floods, EdF had been planning to raise the dike around Blayais by 50 cm, to 5.70 m, as required by the 1998 safety analysis report. This work had been delayed. Furthermore, the waves on December 27 rose to more than a meter above the dike level of 5.20 m [GORBATCHEV 2000; WISE 2000a, b].

- USA, 2003: The Hurricane Charley led to a shut-down at Brunswick-1 NPP in North Carolina due to loss of off-site power because of a trip of the station auxiliary transformer. The transformer trip was due to an electrical fault on a transmission system line. Operators manually shut down the reactor [NUCWEEK 34_04].
- South Korea, 2003: As typhoon Maemi approached, there was concern that the storm might cause salt deposits to build up on power lines and lead to a short circuit that could cripple off-site power supplies. Therefore, all four Kori PWRs have been shut down for two days. (A build-up of salt deposits on power line insulators had led to problems at the Maanshan PWR station during the typhoon season in Taiwan, in 2001.) The typhoon, which struck southeastern Korea in September 2003, was a 50-year event [NUCWEEK 39_03].
- Germany, 2004: On February 8, both Biblis PWRs (A and B) were in operation at full power. Heavy storms knocked out power lines in the station vicinity. Because of an incorrectly set electrical switch and a faulty pressure gauge, the Biblis-B turbine did not drop, as designed, from 1,300 to 60 megawatts, maintaining station power after separating from the grid. Instead the reactor scrammed. When Biblis-B scrammed with its grid power supply already cut off, four emergency diesel generators started. Another emergency supply, over four trains from Biblis A, also started but, because of a switching failure, one of the lines failed to connect. These lines would have been relied upon as a backup to bring emergency diesel power from Biblis-B to Biblis-A, if Biblis-A had also been without power. The result was a partial disabling of the emergency power supply from Biblis-B to Biblis-A for about two hours. Then, the affected switch was manually set by operating personnel [IPPNW 2004; NUCWEEK 04_04].
- Sweden, 2005: In January 2005, four reactor units in Sweden were forced off line by a storm meteorologists characterized as the worst in almost 40 years. Hurricane-force winds, torrential rain and high waves battered the entire Baltic. Western and southern Sweden, as well as eastern Finland, were particularly hard-hit. It was the first time Swedish NPPs had been forced to shut down because of the weather [NUCWEEK 02_05].

Vulnerability of Atomic Power Plants in the Case of Grid Failure

Nuclear power plants generate electric power and supply it to the offsite grid. On the other hand, the plants themselves are dependent on a continuous electric power supply to operate, particularly for the instrumentation and safety systems, even when they are shut down. A typical nuclear power station is connected to the electric grid through three or more transmission lines. Heavy storms can lead to multiple damage of the transmission lines, and hence to loss of off-site power. Also, there can be grid failures even if transmission lines in the vicinity of the NPP remain intact. The probability of general grid failures will also increase due to present development trends not related to the climate (liberalisation, cost pressure).

Should the power lines to the NPP be cut-off or a regional electrical grid collapse occur, onsite emergency generators are designed to automatically start. Every nuclear power plant has emergency power supplies, which are often diesel-driven. These generators provide power to special electrical safety distribution panels. These panels in turn supply power to those

emergency pumps, valves, fans, and other components that are required to operate to keep the plant in a safe state.

If the emergency diesel generators (EDG) fail, the situation at the plant becomes critical ("station blackout"). A natural disaster that disables the incoming power lines to a nuclear power station coupled with the failure of on-site emergency generators can result in severe accident. Apart from the diesel generators, there are also batteries that supply direct current in case of an emergency; however, the batteries cannot provide electricity for large components such as pumps and have only very limited capacity (typically for about 2 hours).

NRC reviews in recent years have shown that a station blackout at a nuclear power station is a major contributor to severe core damage frequency. According to NRC studies, over 50% of all postulated accidents leading to a core melt accident begin with a station blackout [PORZTLINE 2001]. Without electricity the operator looses instrumentation and control power leading to an inability to cool the reactor core. Counter measures (accident management) are practically impossible. If the blackout lasts for a long time, not only the reactor, but also the fuel in the spent fuel pool can overheat, contributing to radioactive releases.

Every nuclear power plant has at least two emergency diesel generators. These generators are typically tested one or two times per month, when they are run for about one to four hours. Several times per year, the diesels may be run for up to 24 hours to ensure that the equipment would function during a loss of offsite power.

However, emergency power systems with diesel generator are notoriously trouble-prone. Disturbances in the emergency power system are responsible for a considerable number of reportable events in nuclear power plants. Emergency diesel generator defects and problems at US nuclear plants as reported to occur on a weekly basis. In 1999, there were 32 reports affecting virtually half (49.5%) of all US nuclear plants [PORZTLINE 2001]. Over 40% of U.S. nuclear power plant emergency diesel generators (EDG) are obsolete. EDG voltage regulators, typically of 1950-60 vintage, have recently experienced ageing and obsolescence problems that have created a heightened awareness among nuclear utilities because of the threat to overall EDG performance [EPRI 2004].

In Germany, 24 % of all reportable events in 2003 occurred in the emergency power supply system – about half of them concerning the Diesel generators, and half other components [BFS 2004].

In addition, the fuel in store for the EDG is limited. If a failure of the off-site electricity supply is connected with adverse effects on the traffic infrastructure, which appears probably in the event of natural hazards, it is questionable whether additional fuel can be brought to the site in time.

All in all, there is reason for concern regarding the precautions for emergency power supply at NPP sites. Regulations and practices governing these precautions still reflect the conditions of the 1980s and are not appropriate for the present situation of increasing hazards to the electricity grid due to climate change as well as due to the liberalization of the electricity markets and the increased threat of terrorist attacks.

Indeed, the grid failures and blackouts that have occurred in 2003 clearly show the increasing danger. The grid failure in the USA and Canada in August led to the shut down of 22 NPPs [WISE 2003].

In the same year, two major electricity blackouts also occurred within one week in Europe. Both might have been connected to weather conditions. The first big blackout occurred September 23 in Denmark and Sweden. After a scram at the Swedish NPP Oskarshman-3, Ringhals-3 and -4 also scrammed due to technical problems at a relay station, possibly caused by high winds [NUCWEEK 47_03].

Italy was hit by a blackout on 28 September. The failure started due to problems in a high-voltage connection line between Switzerland and Italy, possibly because of storms.

In all three cases, problems at one single line or station resulted in a cascading event when more and more lines and stations started to trip and disconnect.

The blackout in North America possibly also caused damage to some of the affected NPPs. Restart of Indian Point-3 was delayed, as repairs were needed on electrical cables in the control rod mechanism. Fermi-2 went back to power after damage was repaired at turbine equipment, pumps and circuit boards. The components got damaged when they were overheated in the sudden power loss. Davis-Besse suffered unexpected damage during the blackout. A metal bellow in a containment air cooler was deformed, apparently by a surge in water pressure (bellows are attached to the water piping leading to each cooler to allow it to expand and contract in response to temperature changes) [NUCWEEK 38_03].

The Oskarshamn-3, reactor scrammed Sept. 23, contributing to grid failure and a massive blackout in southern Sweden. When operators attempted to restart the unit, a thermal transient occurred, with coolant temperature increase exceeding operating limits. Both regulators and plant operators were concerned that the incident, rated Level 1 on INES (International Nuclear Event Scale) could have caused damage to the vessel, when hot and colder water met [NUCWEEK 47_03].

Those events illustrate that the direct initiation of a nuclear accident is not the only hazard associated with grid failure. Such failures can also have indirect effects affecting plant safety.

Furthermore, in case of a grid failure of longer duration, traffic and communication infrastructures will be massively impaired. Thus, emergency measures that might be required in case of an accident with radioactive releases (for example, information of the population, evacuation) will be hindered. Monitoring and alarm systems might not be operational.

Vulnerability of Atomic Power Plants in the Case of Flooding

Cooling needs of nuclear reactors dictate a location at the sea or at a large river. All reactors on sea coasts are endangered by sea level rise. Over the next hundred years there will be significant rises, while many sea coasts, for example in England, are gradually sinking. Many closed nuclear reactor sites could be flooded, including the stored nuclear waste. That could contaminate the coast lines for decades. Back in 1992 a study was performed in the U.K. on flood threats to U.K. nuclear reactors. All but one of the U.K. reactors are located on sea coasts at or near sea level. By 2025, several nuclear sites are predicted to be under water. Until now, no protective measures around nuclear sites in the U.K. or anywhere else have been taken [WISE 2000b]. It also seems likely that natural land movements along the south-eastern coastline of China (currently sinking), where the Chinese NPPs are located, would exacerbate the effects of sea level rise [WISE 1992b].

New measurements show that the world's oceans have heated up just as predicted in computer models, and, more ominously, that massive amounts of freshwater from melting Arctic ice are seeping into the Atlantic Ocean [BORENSTEIN 2005].

Recently, an IAEA Safety Guide was published which is to provide recommendations relating to the evaluation of the flood hazard for a nuclear power plant on a coastal or river site so as to enable the identification of hazardous phenomena associated with flooding events to the site [IAEA 2003].

According to this IAEA Safety Guide, the region shall be assessed to determine the potential for flooding due to one or more natural causes such as runoff resulting from precipitation or snow melt, high tide, storm surge, seiche and wind waves that may affect the safety of the nuclear

installation. The possible combinations of the effects of several causes shall be examined. The potential for instability of the coastal area or river channel due to erosion or sedimentation shall be investigated. Information relating to upstream water control structures shall be analysed to determine whether the nuclear installation would be able to withstand the effects resulting from the failure of one or more of the upstream structures.

According to IAEA, the effects of flooding on a nuclear power plant site may have a major bearing on the safety of the plant and may lead to a postulated initiating event that is to be included in the plant safety analysis. The expected main effects of flooding on NPP are as follows [IAEA 2003]:

- The presence of water in many areas of the plant may be a common cause of failure for safety related systems, such as the emergency power supply systems or the electric switchyard, with the associated possibility of losing the external connection to the electrical power grid, the decay heat removal system and other vital systems.
- Considerable damage can also be caused to safety related structures, systems and components by the infiltration of water into internal areas of the plant, induced by high flood levels caused by the rise of the water table. Water pressure on walls and foundations may challenge their structural capacity. Deficiencies in the site drainage systems and in non-waterproof structures may also cause flooding of the site.
- The dynamic effect of the water can be damaging to the structure and the foundations of the plant as well as the many systems and components located outside the plant. In such cases there could also be major erosion at the site boundary.
- A flood may transport ice floes in very cold weather or debris of all types which may physically damage structures, obstruct water intakes or damage the water drainage system.
- Flooding may also affect the communication and transport networks around the plant site. The effects may jeopardize the implementation of safety related measures by operators and the emergency planning by making escape routes impassable and isolating the plant site in a possible emergency, with consequent difficulties in communication and supply.
- Flooding can also contribute to the dispersion of radioactive material to the environment in an accident [IAEA 2003].

Isolation of the site is a specific consequence of an external event which must be taken into account when defining the required emergency provisions.

All in all, it appears likely that in the decades to come, the hazards associated with flooding will increase for many nuclear power plants world-wide and could even become dominant in some cases. It is highly questionable whether NPP operators an regulatory authorities are fully aware of this problem.

Vulnerability of Nuclear Power Plants by Other Natural Hazards

In the summer of 2003, the highest temperatures occurring so far were recorded in France. The heat was exceptional in both intensity and duration. A potential impact on the safety level of French NPP units was seen through high air temperature, high cooling water temperature and low cooling water level [THUMA 2004].

Long-lasting and repeated heat waves can also lead to unexpected acceleration of ageing processes, increasing the probability of safety system failure in case of an accident.

Other risk factors are the possibility of increased frequency and intensity of hailstorms and sleets. Also, heat and dry weather have led to an increased occurrence of forest fires in the last years.

Possible Counter-measures

Nuclear power plant structures, systems, and components important to safety are to be designed to withstand the external effects of natural phenomena such as tornadoes, hurricanes, or floods without loss of capability to perform their safety functions. Extreme values for wind, precipitation, snow, temperature and storm surges, based on empirical data from the weather statistics, are used for calculating the design parameters and estimating the impact load from severe weather conditions.

The apparent increase of frequency and intensity of extreme weather conditions in the past few years has resulted partially in a re-assessment of potential consequences of such effects, and heightening of the standards for NPP design. For example, regarding flooding of nuclear power plants in Germany, the plants now have to be designed against an event with a probability of 1:10,000 per year, while it was 1:1,000 years before [KTA 2004].

The estimation of probabilities for extreme events resulting from climate change, however, is extremely difficult due to fact that there is no sufficient database for such estimates. Furthermore, because the situation is constantly evolving, any data that can be acquired may be outdated by the time their evaluation is concluded.

The time lag is still more drastic for the drafting of new rules and regulations by the authorities, and their implementation by the NPP operators. It seems hardly possible to win this race against time – particularly in the face of economic pressure that might lead to the result that only low-cost measures are realised.

The inadequate protection against floods at the Blayais site illustrates the problem of delayed backfitting (however, in this case, even the backfit would not have prevented flooding).

In spite of the fact that the hazards of climate change are becoming more and more obvious, safety reassessments and improvements generally are only implemented – if at all – after an event occurred. This practice is aggravated by the fact that an event in one NPP does not necessarily lead to backfits in another plant, let alone to backfits worldwide.

Regarding the new reactor generations (Generation III and IV), the increasing hazards due to climate change have not been taken into account in their design, as far as can be seen today.

Apart from improving design, advance warning in case of extreme events can contribute to safety. For example, the U.S. NRC is now observing the development of storms. In connection to tropical cyclones, factors like extreme wind speed, pressure and precipitation are of importance. About 12 hours before expected hurricane-force winds, NRC will enter one of its response modes and begin receiving continuous status updates from all of the nuclear facilities in the hurricane's path [NRC2005]. According to NRC, severe tornadoes can produce winds and tornado missiles that can badly damage steel reinforced concrete structures. (Ageing mechanisms can aggravate such effects- see section C.) It was assumed that a tornado could also significantly damage support systems for onsite irradiated fuel storage ponds. Furthermore, tornadoes may induce floods and consequently may be the cause of additional indirect damage. In Central Europe, too, tornadoes have received increased attention in the last years [NRC 2005].

An advance warning system can permit the implementation of protective measures at NPP sites before the hurricane arrives. It is not conceivable, however, to avoid large-scale grid failures

with the aid of protective measures. The main problem is that most grid lines are above ground and thus, very vulnerable. Their masts are not designed to withstand severe storms.

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