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Nuclear Reactor Hazards

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Introduction

This report is based exclusively on Greenpeace International's report "Nuclear Reactors Hazards, Ongoing Dangers of Operating Nuclear Technology in the 21st Century," published in April 2005 (GREENPEACE 2005). The sections reproduced here look at the characteristics and inherent flaws of the main reactor designs in operation today; the second part assesses the risks associated with new designs, and discusses the "ageing" of operational reactors; and the third part looks at the terrorist threat to 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.
- 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 around twenty-one 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.
- Deregulation (liberalization) 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, operational temperature, and the burnup of the fuel. This accelerates ageing and decreases safety margins. Nuclear regulators are not always able to fully cope with this new regime.
- Reactors cannot be sufficiently protected against a terrorist threat. There are several scenario's—aside from a crash of an airliner into the reactor building—which could lead to a major accident.

1. Commercial reactor types and their shortcomings

At the start of 2005 there were 441 nuclear power reactors, operating in 31 countries. Although there are dozens of different reactors designs and sizes there are four broad categories for nuclear reactors currently deployed or under development:

Generation I were prototype commercial reactors developed in the 1950s and 1960s and are modified and enlarged military reactors, originally either for submarine propulsion or plutonium production.

Generation II are the classification of the vast majority of the reactors in commercial operation worldwide.

Generation III reactors are now being built in some countries, notably in Japan.

Finally, **Generation IV** reactors are currently being developed with an objective of commercialization around twenty to thirty years from now.

Generation I

The early Soviet designed reactors, the **VVER 440-230**s, are classified as Generation I. These are reactors which use pressurized water as a coolant and have the same basic design as the Pressurized Water Reactor (PWR) which is the most common reactor design world wide—see Generation II. However, the VVER 440-230s have significant and serious design flaws and consequently, the G8 and EU believe that they cannot economically be brought up to an acceptable safety standard. All of these reactors operating in Central Europe will be closed by the end of this decade but others in Russia are likely to continue to operate. The lack of a secondary containment system and adequate emergency core cooling system are of particular concern.

The other Generation I reactor design still in operation is the United Kingdom's **Magnox** design—air-cooled, graphite-moderated natural uranium reactor. Magnox reactors have very low power density and consequently large cores. Carbon dioxide gas circulates in the primary circuit.

The reactor core is located inside a large pressure vessel. Some of the Magnox fleet have older steel pressure vessels and 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, however, others will operate until 2010 allowing around forty years of operation.

These reactors do not have a secondary containment system—which protects the reactor core from external events and helps to contain radioactivity in the event of a core-related accident—and thus the reactors 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.

Generation II

Probably the most notorious reactor design in the world is the **RBMK** reactor and is a Generation II design. It 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 has some fundamental design problems, 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 (1,693 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 (Donderer 1996; Butcher 2001). 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 shutdown systems, and hence the remaining ten do not conform to IAEA safety requirements (IAEA 1999).

RBMK reactors also contain more zirconium alloy in the core than any other reactor type (about 50 percent more than a conventional BWR). They also contain a large amount of graphite (about 1700 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 ruptures 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 forty channels. A catastrophic destruction of the whole reactor core could follow (Butcher 2001).
The fundamental design flaws of these reactors have lead to the international community classifying these reactors as "non-upgradable" and to seek 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 most prevalent design in operation is the **Pressurized 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 have 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. The reactors are fueled with low-enriched uranium.

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 United States. The most serious example discovered to date occurred at the Davis Besse reactor in Ohio, United States. 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. Despite this, new problems arise, a process which may continue as components no longer perform as expected due to age-related problems.

Of similar design and history to the PWR is the Russian **VVER** reactor. There are currently fifty-three of these reactors deployed in seven countries in Eastern Europe in three main reactor designs. The oldest, the VVER 440-230, has been mentioned above and is classified as Generation I.

The second-generation of VVERs, the 440-213s, has introduced a more effective emergency core cooling system that still does not deploy a full secondary containment system, but has a system designed to capture released radioactivity in the event of a release—through a bubble condensing tower—although it does not protect the reactor core from external events.

A third design of VVER, the 1000-320s, introduced further design changes, in addition to increasing its size to 1000 MW, 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 was abandoned. Both safety and economic considerations were given for these decisions.

The second most prevalent reactor design is the **Boiling Water Reactor** (BWR) (there are ninety 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 had been detected in a number of German BWRs, in piping of a material that was regarded as resistant to so-called stress corrosion cracking.

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 **Pressurized Heavy Water Reactor**, of which there are thirty-nine currently in operation in seven countries. The main design is the Canadian CANDU reactor, which is fueled 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 programs 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." The operation of eight of Ontario Hydro's CANDU reactors was suspended or indefinitely deferred in the late nineties—although some have now restarted.

The **Advanced Gas Reactor** (AGR), is only operated in the United Kingdom and is a modified and updated version of the Magnox reactors. However, some of the inherent problems of the earlier reactor remain, notably, the lack of a secondary containment system and age-related degradation. Most recently, cracking in a number of graphite bricks that make up the reactor core was discovered. It is thought that this problem, if replicated across the reactor fleet, might result in the premature closure of reactors (NUCWEEK50_04).

Generation III

The 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 twenty 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. According to the World Nuclear Association, reactors of Generation III are characterized by the following points (WNO 2004b):

- A standardized 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 sixty years
- Reduced possibility of core-melt accidents
- Minimal effect on the environment
- Higher burnup 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 toward better economics. Their addressing higher safety standards remains rather vague.

The European Pressurized Water Reactor (EPR)

The EPR is a pressurized water reactor that has developed from the French N4 and the German KONVOI reactor line, the latest Generation II reactors which went 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 including:

- 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.
- The thermal output of the plant was increased by 15 percent relative to the N4 by increasing core-outlet temperature, letting the main coolant pumps run at higher capacity and modifying the steam generators.
- The EPR actually has fewer redundant trains in safety systems than the KONVOI plant; for example, its emergency core cooling system has only four accumulators (pressure tanks) whereas the KONVOI plant has eight such tanks.

Several other modifications are hailed as substantial safety improvements:

- The **incontainment refueling water storage tank** (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 **core catcher** 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 a 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 **containment heat removal system** 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 crashes is equivalent to that of the German KONVOI plant’s 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 Generation II PWRs, and still not fully resolved for those: 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 with existing reactors. The issue had 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 represents 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 was pursued up until the late eighties 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, United States, in 1974 and 1989; Winfrith, United Kingdom, in 1976; and 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 percent), encapsulated in multilayers of nonporous 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' ability 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.

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-sized 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 (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;

Fast Breeder Reactors

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

Generation IV

The US 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, United Kingdom, United States), 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 heralded as highly economical, incorporate enhanced safety, produce minimal amounts of waste, and as being 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.

To further encourage and strengthen research and development for Generation IV reactors, the United States, Canada, France, Japan and the United Kingdom 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 had 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, twenty-one countries or entities[1] have become members of INPRO. GIF and INPRO have agreed to formalize cooperation at the technical level. (The United States has been reluctant to participate in INPRO because it was seen as a Russian-inspired initiative) (NUCWEEK 14_02).

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[1] 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 European Commission.

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.

It is hoped that the GFR may benefit from development of the HTGR technology (which is also beset with many problems; see discussion of the 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 United States.

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 to 150 MWe, and modular units of 300 to 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 (ten to thirty 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 fulfillment of Generation IV goals. However, it also has the largest research needs and longest development time.

Although Russia—where almost all the experienced LFRs are 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 United States 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 burndown. The reference plant has a power level of 1000 MWe. Coolant temperature is 700°C at very low pressure. The temperature margin to the salt boiling temperature (1400°C) is large.

The GIF selected the MSR as the most innovative non-classical concept. Of all six reactor systems, MSR requires the highest costs for development (US$1,000 million). 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 percent, compared with 33 to 35 percent 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 United States, and Canada are developing the SCWR. There have been no prototypes built so far.

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-sized (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 1500 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 shut down 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, 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).

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-sulfur 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 percent 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 still has to be 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, United Kingdom), the AVR (15 MWe, 1967–1988, Germany), the THTR (308 MWe, 1986–1988, Germany) as well as the US plants at Peach Bottom (42 MWe, 1967–1974), and Fort St. Vrain (342 MWe,1976–1989).

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 led to a decline of nuclear power. This is the background for the Generation IV initiative of the USDOE. 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.

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 ($US87.6 billion) spent by twenty-six OECD member states between 1991 and 2001 went to nuclear research; only about 8 percent 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 US$6 billion (about $600 to $1000 million per system, plus about $700 million 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 program, 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 doubts 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)" (Gorden 2004).

According to GIF, a **closed fuel cycle** is celebrated as a major advantage of Generation IV concepts. This requires the reprocessing of spent fuel to extract the plutonium and then using plutonium as a fuel. This has significant proliferation implications, in particular if these types of reactors are widely deployed around the world. The reprocessing of plutonium has been widely criticized for its negative impact on the environment as well as its costs and security implications. The widespread introduction of the closed fuel cycle requires a reversal of current anti-proliferation policy in a number of countries, including the United States, and a revision of current industry policy in most nuclear countries. A movement toward the deployment of Generation IV reactors utilizing the closed fuel cycle would require large-scale investment to construct reprocessing plants.

Finally, the costs of such fuel cycle concepts—the use of reprocessing—would be very high. According to the recently published study "The Future of Nuclear" of the 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 burnup 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 USDOE, 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).

Nuclear regulators in the United States 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 skeptical toward the Generation IV systems. "We know 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.

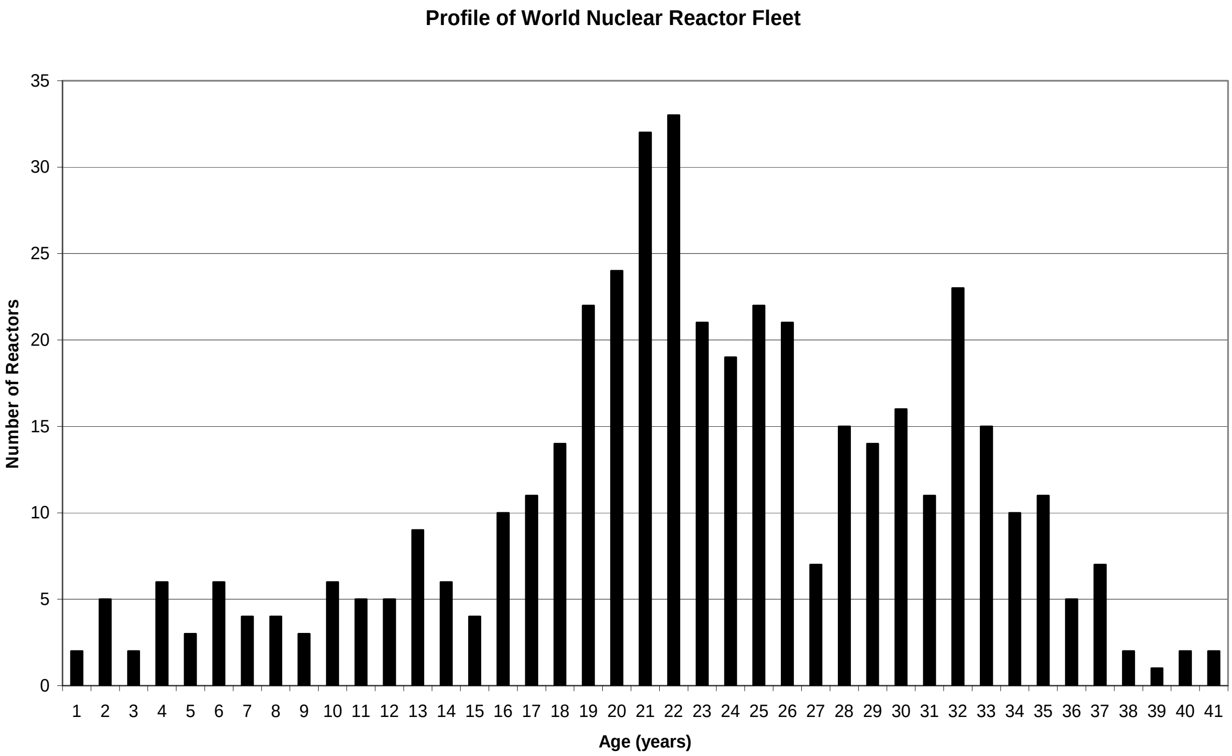
But even real technical improvements that might be feasible in principle are only implemented if their costs are not too high. There is an enormous discrepancy between the catch-words used to describe Generation IV for the media, politicians, and the public, and the actual basic driving force behind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

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

2. Ageing, PLEX and safety

There is general consensus that the extension of the life of the reactor is of foremost importance today to 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 overcapacity of generation; increased scrutiny of economics and financing of nuclear power with the introduction of liberalized 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 has increased year after year and was, in 2004, twenty-one years old (Schneider 2004).



Source: IAEA, PRIS, 2005

At the time of their construction it was often assumed that reactors would not operate more than forty years. However, now, in order to retain the nuclear share of the electricity supply and to maximize 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?

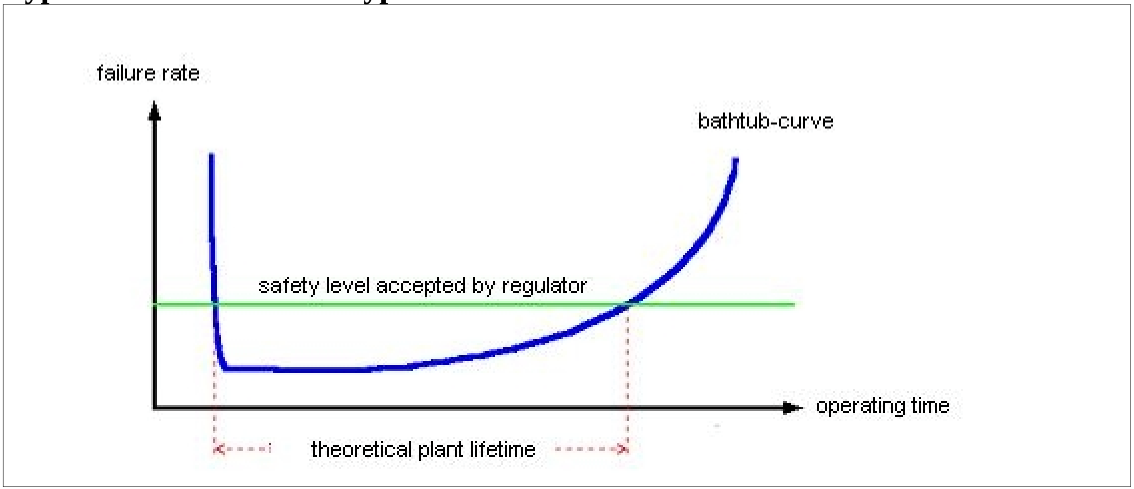
In any industrial plant, material properties are deteriorating during operation due to the loads the components are subjected to. The IAEA defines ageing as a continuous time-dependent loss of quality of materials, caused by the operating conditions (IAEA 1990).

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 the 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 occur 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 recognize 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 twenty 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 nineties, 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.

Phenomena of ageing

Ageing already occurs during the period usually regarded as typical commercial lifetime (thirty to forty years). Naturally, with Plant Life Extension (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 recognized 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 the monitoring of 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 100 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 the increasing age of plants, damage to mechanisms might occur which have not been foreseen, or which might even have 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 programs with accelerated samples, safety reviews, and also the precautionary exchange of components in case cracks or other damages that have been found during inspections. Furthermore, it includes optimization of operational procedures in order to reduce loads. In the United States, a specific ageing management program for reactor pressure vessels (time limited ageing analyses) has been developed (Rinckel 1998).

New, integral methods for the monitoring of NPP operation were developed in the late nineties which attempt to predict the future behavior 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 toward life extension. The aim is, on the one hand, to arrive at inspection programs which are economically more efficient and save time; on the other hand, actual failures are to be avoided to keep downtimes short, improving economy and safety in parallel (Ali 1998; Bartonicek 1998; Bicego 1998; Duthie 1998; Esselmann 1998; Hienstorfer 1998; Roos 1998).

Ageing effects on specific components

Ageing can manifest itself in many different ways in different components. All components of a 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 modeling and estimation. In the course of maintenance and ageing management, NPP operators have reacted to damages by repairs and exchanging of components. Nevertheless, experience shows that, time 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. 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 programs of pressure tube exchanges have been implemented for both of those reactor types.

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.

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 were made for it.

Regarding active components like pumps and valves, deterioration usually manifests itself in an obvious manner, and exchanging 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 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 forty years of operation).

In Spain, where plant operators are considering extending service life from forty to sixty 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 the 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 USNRC-initiated working group (“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 be appearing 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).

Some of the most important age related problems, primarily in light-water reactors (PWRs including VVERs, and BWRs) are outlined below:

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 accelerates 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.
- Core internals, core shroud: embrittlement due to high neutron fluences, as well as damages from 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 refueling. 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). Austenitic steel is a type of steel optimized for corrosion resistance. 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.

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 unfavorable circumstances, breaks without a 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 regarding ASME (United States) 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 (e.g., 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.

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 forty years is likely to be reached; safety margins, however, are considerably smaller than assumed (Bolvin 1993). In Switzerland, a systematic ageing surveillance program for NPP structures was begun in 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.

Consequences of ageing processes

The consequences of ageing can roughly be described as twofold. 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 nineteen NPPs in operation) are responsible for about 64 percent of all reportable events in the time span from 1999 to 2003 (severity of the events taken into account) (BMU 1999–2003).

On the other hand, there are effects leading to a 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 subsequently 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 a large number of failings.

Other examples are the corrosion processes which may be overlooked for years—as a recent event at the US pressurized water reactor Davis Besse illustrates.

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.

Countermeasures

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 decisive component 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 programs 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 countermeasures 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 are usually 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 by 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.

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 practiced 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 behavior of a vessel after annealing.

Most recent publications on ageing emphasize, on a general level, that the countermeasures practiced 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 toward 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 practiced or being implemented for lack of alternatives (in the United States, 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.

As can be seen from the overview on PLEX programs presented above, life extensions are planned in most countries operating nuclear power plants.

PLEX programs worldwide

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 forty years.
Brazil	2	12			Not yet an issue.
Bulgaria	4	20	30		Political agreement for closure of 1 to 4. To early to assess closure of 5 and 6.
Canada	17	22	30		Degradation problems forced the temporary closure of eight 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 modernization program is underway to allow the Dukovany reactors to operate for forty years.
Finland	4	25	30	60	The Olkiluoto plant has already undergone technical changes to allow it to operate for forty years with plans being developed to enable it to operate an additional twenty years.
France	59	20	30	40	The are definitive plans to allow all reactors to operate for forty years.
Germany	18	25		32	A political agreement reached with the utilities will see the average operating life of reactors restricted to thirty-two years of operation.
Hungary	4	20	30	50	Measures are being introduced to allow the Paks facility to operate for fifty 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 sixty years.
Korea, Republic of	20	13			Proposals are being developed to extend the operating life to up to sixty years.
Lithuania	1	18			The remaining reactor is scheduled for closure in 2009, after twenty-two 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 fifteen 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 forty-five 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 forty-year expected life.
South Africa	2	20		40	No plans exist to operate the reactor beyond its forty-year expected life.
Spain	9	23	40	60	The oldest reactor, Jose Cabrera, is scheduled for closure in 2006 after thirty-seven 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 licenses to operate, others have been granted ten-year licenses, 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.
United Kingdom	23	26			All the Magnox reactors now have a fixed operating live time, of up to fifty years. The AGRs (second-generation) are likely to have limited Plex (up to five years).
United States	104	22			The first forty years operating licenses will expire for 3 plants in the year 2009. Of the remaining 100 operating plants, 23 will have licenses expire by 2015[2]. Reactors that have received twenty-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 2005

[2] Reactor License Renewal: FACT SHEET, US NRC, download March 2005.

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 countermeasures.

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) practiced extensively only whenever the remaining (possibly increased) lifetime was sufficient to amortize the investment. For example, steam generators have been exchanged in nuclear power plants in most Western countries with NPPs with pressurized 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 US$10–50/kW, whereas construction of the cheapest non-nuclear alternatives would cost US$325–405/kW. Life extension of a coal fired power plant, for twenty more operating years, would cost US$100–250/kW (Macdougall 1998). New nuclear capacity would be considerably more expensive than all those options (far above US$1,000/kW).

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 twelve countries, the range is given as US$120–680 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 percent, 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 thirty year lifetime would bring a gain of about US$70 million (NUCWEEK 47_00). For the whole French reactor fleet, ten extra years of life are reported to represent a cumulative cash flow of €15 to 23 billion (NUCWEEK 40_03).

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, twenty years’ extension for the four VVER units will cost about €700 million (NUCWEEK 47_04). For the Ukrainian life extensions plans (by ten to fifteen 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 of fifteen years costs 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 (United States), they are estimated at about US$25 million (NUCWEEK 48_03).

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.

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 practiced in most countries where NPPs are operated. Upgrading turbines and steam generators yielded an additional 4 percent 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 had been raised by about 11 percent 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 percent (Rastas 2003). In Germany, output of a number of plants was increased. Until mid-2004, power uprates amounted to about 800 MWe, or 4 percent of installed nuclear capacity. Another 450 MWe are planned (DATF 2003; ATW 2004). Power uprating is also practiced extensively in the United States. For example, the output of Ginna PWR (at present, 495 MWe), where life extension is also planned, is to be increased by 17 percent 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 optimizing 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 realized in the last years. Thus, there is a trend toward 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 United States led to vibrations of the main steam line, which in turn damaged other components and necessitated several shutdowns and repairs (UCS 2004).

Increasing the fuel burnup (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 efforts to increase burnup have been intensified in recent years. Several decades ago, typical burnup of PWR spent fuel was around 30,000 MWd/t or slightly higher. Today, burnups 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 burnup also increases the hazard of fuel hull failure and hence, radioactive contamination of the cooling water. Furthermore, the influence of high burnup on the behavior of fuel rods under accident conditions is not fully understood.

The use of high burnup fuel can also reduce operational safety margins. For example, the hazard of neutron flux oscillations in BWRs is increased.

Increased burnup 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 United States 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 skeptical.

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 state-of-the-art 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 toward 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 and operators that were working at the plant from start-up.

3. The terror threat

Long before September 11, 2001, numerous deliberate acts of terrorism had taken place in the twentieth century. The terrorist threat appears to be particularly great, however, in the early twenty-first century.

There are numerous potential targets for terrorist attacks. Industrial installations, office buildings in city centers, or filled sports stadiums 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 a target for one of the following reasons, or a combination of those reasons:

1. Because of the symbolic nature—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 centralized 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 not only 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 a 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 focusing on airplane suicide attacks. However, totally different scenarios are also plausible.

Terror attacks against nuclear plants are not purely theoretical. In the past, a number of such attacks have already taken place. Luckily, they have not led 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):

- November 12, 1972: Three hijackers took control of a DC-9 of Southern Airlines and threatened to crash it in 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.
- 1993: The terrorists behind the car bombing of the World Trade Center, 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, Pennsylvania, fifteen km away from the Three Mile Island nuclear power station.
- November 1994: A bomb threat was reported at Ignalina NPP in 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 toward “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 a real or alleged threat emanates. 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 center 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 US aircraft (Thompson 1996).

Threats through acts of war cannot be excluded in any region. During the Balkan conflicts in the early nineties, 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 widespread (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 from 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 one 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 US nuclear power plants). The pool in many German boiling water reactors 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 installations of varying safety significance. The most important are, in case of a modern pressurized water reactor (PWRs, including VVERs, account for about 60 percent 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 to protect 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 countermeasure. 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 the 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 the case of spatial separation of important components, if the attack has effects that are spread over on the site.

For example, in the 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 the case of the simultaneous destruction of the control room and emergency feed building (emergency standby building), a situation could arise where the safety systems required are still operable, but cannot be controlled any more. Far-reaching destruction in 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 destruction at the site.

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 armor- 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 a terror attack. Almost every army in the world today possess such a weapon; it is conceivable that terrorists would be 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 favorable, 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 countermeasures.

Within a few hours, core melt will occur with severe releases of radioactivity. The amount released to the atmosphere can be about 50 to 90 percent of the radioactive inventory of volatile nuclides like iodine and caesium, plus a small percentage of further nuclides like strontium-90. In case of a nuclear power plant with 1000 MW electric power, this corresponds to, among other things, 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 (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 would begin within hours (Alvarez 2003).

Further information

NGOs

Anti-Atom (Russia): HYPERLINKhttp://www.antiatom.ru/eng.htm
Atom Stop (international): HYPERLINKhttp://www.atomstop.com
Citizens Nuclear Information Centre (Japan): HYPERLINKhttp://cnic.jp/english/
Earthlife Africa: HYPERLINKhttp://www.earthlife.org.za/
Friends of the Earth Europe:
HYPERLINKhttp://www.foeeurope.org/activities/Nuclear/nuclear.htm
Korean Federation for Environment Movement:
HYPERLINKhttp://www.kfem.or.kr/engkfem/
Greenpeace International:
HYPERLINKhttp://www.greenpeace.org/international/campaigns/nuclear
Nuclear Information Resource Service (United States):
HYPERLINKhttp://www.nirs.org/
Public Citizens (United States): HYPERLINKhttp://www.energyactivist.org
Sortir du Nucleaire (France): HYPERLINKhttp://www.sortirdunucleaire.org/
WISE Amsterdam (International): HYPERLINKhttp://www10.antenna.nl/wise/
WISE Paris (International): HYPERLINKhttp://www.wise-paris.org/

Nuclear sector

World Nuclear Association: HYPERLINKhttp://www.world-nuclear.org
International Atomic Energy Agency: HYPERLINKhttp://www.iaea.org
Nuclear Energy Agency: HYPERLINKhttp://www.nea.fr
United States Department of Energy, Office of Nuclear Energy, Science and Technology:
HYPERLINKhttp://gen-iv.ne.doe.gov/
Generation IV International Forum: HYPERLINKhttp://gif.inel.gov/

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