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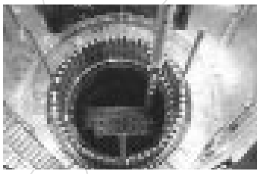
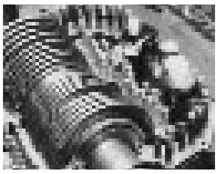
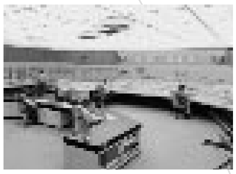
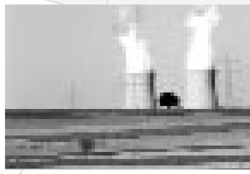

ILK Statement

on the Safety of Nuclear Energy Utilization in Germany

Für deutsche Fassung bitte umdrehen!

July 2000
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ILK - Geschäftsstelle beim Bayerischen Landesamt für Umweltschutz

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Summary

The Federal Government believes it to be imperative to phase out nuclear energy use as soon as possible in a financially compensation-free way. The government justifies this stance with a reassessment of the risk-benefit relationship that is not as favorable as before due to recent findings and the "failed" disposal concept. Based on this state of affairs, the ILK has dealt with this assessment from a scientific and engineering point of view and arrives at the following conclusions:

- The German safety philosophy has worked well. It is based on conservative safety principles that make core damage extremely improbable. In more than 30 years of nuclear energy use, corresponding to an operational experience of 590 reactor years in Germany at the end of 1999, hazards to public health and the environment through ionizing radiation have been prevented. No radiological releases to the general public occurred that exceeded the permissible values for the normal operation of a nuclear power plant.

- This very positive overall safety record is not just a German feature: among the more than 350 light water reactors built according to Western design and operating practice, only one accident with core damage (Three Mile Island) occurred, and, even in this case, no serious release of radioactivity into the environment occurred.

- The continuous evolution of the measures within the safety philosophy and their implementation has led to additional requirements on the plants. In so doing, a new set of safety measures was added to the original design that goes beyond the previous design basis, e.g., internal plant accident management measures (AM-Measures) such as primary and secondary side pressure relief and feeding (bleed-and-feed) and filtered containment venting. These measures can prevent core melt in the unlikely event of failure of safety systems or can mitigate substantially the consequences of a core melt.

- The feedback cycle of experience, research findings, as well as the implementation of findings from probabilistic safety analyses, have further raised the safety level of the facilities over time and have increased safety margins. This applies to both older and newer plants.

- Aging effects relevant to safety are given consideration in the structural design and are monitored during operation. The monitoring results form an essential basis for maintenance, replacement of components, and backfits that are carried out in addition to the periodic maintenance activities. These measures, which

involve considerable annual expenditures, serve the purpose of maintaining the safety margins. On the whole, operating experience to date does not provide any indication of an age-induced reduction in the safety level.

- During the last two decades, the frequency of reportable events and incidents has been reduced significantly. The same observation applies to the number of unscheduled reactor scrams, a further important indicator of the quality of a plant.

- If surrogate risk metrics such as the frequency of core melts or large releases of radioactivity are used as the basis for assessing the evolution of the risk to the public from German NPPs, a decrease in risk over time can be demonstrated. This improvement towards smaller risk values is consistent with the experience in other Western countries.

- Concerning the radiation risk coefficient, both the epidemiological data as well as the adjustment of the extrapolation model point towards a decreasing tendency, i.e., it can be expected that the recommended by the ICRP (in 1990) increase by a factor of four will be partly reversed.

- Because of the numerous improvements in NPPs, the proper way to evaluate the influence of new factors, such as the increase in the radiation risk coefficient, is to place it in the context of probabilistic safety assessment.

- First comparative investigations of important energy supply systems show that the use of non-nuclear energy forms is also associated with accident risks and that these systems have led to major accidents. Long term effects of extremely unlikely damage incidents that cannot be entirely ruled out appear to be a special feature of nuclear energy. Today, they need to be weighed against the anticipated climate changes due to greenhouse gas emissions with unlimited reach in terms of space and time.

- According to status-quo forecasts on long term energy demand and associated emissions, the worldwide output of CO_2 until the year 2020 will grow by between 50 - 80% in comparison to 1990. While renewable energy sources in Germany are expected to have the highest proportional growth, their contribution will remain at a low level. They will not be able to even begin to compensate for the loss of nuclear energy in the near future. A new EU-study concluded that at least an additional 100 nuclear GW of electrical generating capacity would need to be installed in the next 25 years if the EU countries wanted to come anywhere near to their CO_2 -reduction goals.

- An essential argument for nuclear phaseout is the "failure" of waste disposal in the eyes of the government. The ILK will deliver separate statements on this central claim. It should be noted here that facilities for safe transportation, conditioning, and interim disposal of radioactive waste already exist and, in terms of their final disposal, appear largely technically feasible in today's view. The ILK sees no factual argument to support the claim that the German disposal concept has failed.

The ILK concludes that the safety of German nuclear facilities for energy production is guaranteed in an internationally exemplary way. It does not perceive any technical and/or scientific reason for a renunciation of nuclear energy production, but to the contrary, especially with regard to the CO_2 issue, sees no ecologically adequate technical alternatives in the foreseeable future.

The Chairman

Prof. Dr.-Ing. Josef Eibl
9th of July 2000

1 Introduction

The Federal Government deems it imperative to phase out nuclear energy as soon as possible without financial compensation. It justifies this stance with an unfavorable assessment of the risk-benefit relationship based on more recent findings [2], [3]. The prior positive assessment of nuclear energy can be traced back to the legislation passed in 1959. It was not called into question when the eight amendments to the Atomic Energy Act were passed, although the Federal Government views the amendments made to the Atomic Energy Act to date only as selective replies to specific questions and has never regarded them as risk-benefit considerations.

The revised assessment by the Federal Government should be viewed against the background that, in its view, less risk-afflicted energy production forms, which are sustainable in the Agenda 21 sense of the term, are currently available. Additionally, the Federal Government sees signs for the worsening of the safety standard of German nuclear power plants since their licensing. Further arguments for nuclear phaseout are the changed assessment of the risk development for radiation exposure among the population and the "failure" of the waste disposal concept in the eyes of the government.

The Federal states of Baden-Württemberg, Bavaria, and Hesse have commissioned the International Nuclear Technology Commission (Internationale Länderkommission Kerntechnik, ILK) to scrutinize the above-mentioned assessment by the Federal Government [1], [2], [3] as well as its underlying arguments and to include the international state of knowledge and scientific discussion in so doing.

Of decisive significance for the risk assessment is the so-called Kalkar-resolution of the Federal Constitutional Court (Bundesverfassungsgericht, BVerfG) dating from 1978. It lays down that infringements of constitutional law which might arise from the licensing and the operation of technical plants cannot be excluded with absolute certainty; Uncertainties beyond this so-called threshold of practical reasoning, as it has been labeled by the court, are attributed to the limits of human thinking, are considered unavoidable and thus should be shouldered by all citizens as a socially tolerable burden.

The following Statement concentrates on the assessment of the safety of nuclear energy from a scientific and engineering point of view. The aim is to assess and present the facts in the best possible way. Neither higher order, evaluative questions and conclusions on energy political issues are considered nor is a systematic consideration of alternatives.

The safety or risk assessment rests on the quality of and adherence to conservative design requirements as well as the inclusion of results from probabilistic risk analyses. The major steps in the developments and insights into risk are presented and consideration is also given to empirical findings and measures that have been implemented in the meantime. The ILK also comments on the significance of the aging of German NPPs in terms of safety and on safety-engineering further developments.

The risk of lethal accidents by operating nuclear power plants is compared to those of other energy producing systems. Since in the meantime, the risk of environmental and especially climatic changes through energy converting systems have considerably increased in significance, statements on this aspect will also be made.

The ILK will deliver separate statements on another central claim of the Federal Government regarding the failure of waste disposal. The current paper will merely include the essential findings.

2 Evolution of safety in domestic Nuclear Power Plants

2.1 Basic safety philosophy

The design criteria for the current generation of light water reactors (**PWR** = Pressurized Water Reactor and **BWR** = Boiling Water Reactor) were developed in the 1960s and 1970s. The protection of individuals, society, and the environment were of paramount importance [4], [5]. These concerns led to the conservative principles of a multistage concept (defense-in-depth) and safety margins that formed the cornerstones of the reactor safety philosophy which represented the technical implementation for the Atomic Energy Act [4].

The defense-in-depth concept requires the employment of successive, staggered measures that can be assigned to four levels as shown in Table 1.

Safety level			Objectives	Measures
1	operating conditions	normal operation	prevention of abnormal occurrences	quality of operating systems and procedures as well as safety consciousness at work
2		abnormal operation	prevention of design-basis accidents	inherently safe facility behavior; limitation systems
3	design basis accidents		control of design basis accidents	inherently safe plant behavior; passive and active safety equipment
4	(beyond design-basis) major accidents	specific, very rare events	control of specific, very rare events	specific precautionary measures
		beyond- design-basis conditions/ emergencies	prevention of core damage and, if this is not possible, limitation of the impact on the environment	on-site accident management measures

Table 1: Implementation of the staggered conservative safety philosophy [7]

The features of the first two levels (see table 1) have both operating and safety functions. They are intended to avoid malfunctions or limit their consequences. This reduces the stresses on the equipment, lowers output reductions and avoids demands on safety systems. One example for these measures is that the pressure retaining boundary of the reactor coolant is designed in such a way that its failure is not anticipated according to engineering standards.

The features of level 3 determine the safety engineering design of the plant. Their function is control design basis accidents (DBAs) whose occurrence is stipulated despite the measures of the first two levels. Examples include the residual heat removal system as well as the containment. Their purpose is to control leaks and bursts in the pressure retaining boundary and to safely contain any escaping coolant and the radioactivity it carries.

Even though nuclear plants are very conservatively designed against accidents, measures are still taken at level 4 against events whose occurrence is deemed extremely unlikely or where a failure of the safety systems is assumed. These additional features are intended to prevent or substantially mitigate the potentially grave consequences of a major accident (beyond design basis major accident) and to reduce the residual risk as far as possible using appropriate measures. Initially, only single, selective measures were taken. The evolution of the safety philosophy led to an extension to an additional safety level when compared to the original design.

Uncertainties and probabilities for the failure or components cannot be quantified for this deterministic approach. Therefore systems, structures and components (SSCs) are designed in such a way that the stresses remain far below the limits at which damage can occur. The measures formulated at all levels rest on technical and organizational means. In particular, staff receive training in all measures. For design-basis accidents (level 3), a mastery of incidents is demanded even under the stipulated condition that the staff does not intervene during 30 minutes after begin of the accident.

This conservative safety philosophy, which satisfies and even surpasses international requirements [6], has essentially remained unchanged throughout the use of nuclear power. The technical means for its implementation, however, have advanced significantly.

2.2 Improvements over time

Safety has been continuously improving over the course of time:

- The demands placed on facilities constructed at a later date have been continuously increased and include the degree of redundancy, independence of trains and avoidance of dependent failures. Plants that were already operating were comprehensively readjusted to the new standard. A large part of the backfits mentioned in table 2 -mostly improvements of incident control (level 3, see Table 1)- can be traced to these developments.

- A further focus of the increased safety reserves concerns level 4 measures (see Table 3). On the one hand, they include precautions against extremely rare incidents such as a plane crash. On the other, on-site accident management measures (AM-Measures) have been introduced particularly over the last decade. They enable the prevention of core melts even in the case of extensive outages of safety systems or considerably limit their consequences. Special emphasis was placed on taking measures for avoiding failures of the containment that might lead to large releases of radioactive substances .

- Improvements based on operating experience cover a great area. They range from diverse detailed improvements such as optimized tests and maintenance measures to changed system operating modes, use of more reliable components, up to large and small backfits, e.g., the exchange of piping in BWR. In so doing, use is made of domestic and international experiences, e.g., for measuring the fill level in the RPV itself, the fill level sensor was backfitted in the American Three Mile Island nuclear power plant a direct result of the TMI-accident.

- A variety of improvements were derived from probabilistic analyses that were carried out as supplementary measures for power plants.

- Operating manuals used by staff during normal operations and during operational malfunction were continually improved. Their technical content was adapted to new findings on plant behavior, their layout improved in terms of clarity and ease of use. In addition to the incident-oriented descriptions, protection goal oriented instructions were added in order to provide instructions whenever an incident-oriented procedure should not lead to the desired result. Following the introduction of AM-measures, these were also included in the emergency manual as instructions. Care was given to ensure that the criteria for inducing measures were described simply and clearly.

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- Similarly, the training of staff also evolved, in particular the training given to shift staff who require authorized licensing for carrying out their activity. Simulator training plays an important role since it provides experience in how to proceed during accidents that staff would be unable to gain given the reliability of plant operation. Next to the existing 5 full simulators that currently exist, 9 more were procured in recent years for optimizing simulator training and 1 more is under construction. The training covers normal and malfunctioning operating conditions but also beyond-design basis conditions that necessitate the use of preventive AM-measures. Not only the content but also the method of training has been elaborated. The focus on technical competence has expanded to also give significance to skills in cooperation and leadership (safety culture).

- Both old and new facilities profit equally from insights derived from science, the evaluation of national and international incidents, etc. The level 4 measures, for instance, do not constitute a licensing requirement for existing plants. The operators of all German nuclear power plants nevertheless subsequently implemented such measures.

Type and extent of subsequent changes to nuclear power plants are plant-specific. A variety of changes which were requested for the Biblis nuclear power plant have thus far not been implemented.

The safety of German nuclear power plants is assessed regularly. On the one hand, proof must be provided that the essential system functions necessary for the safety of the plant are given. On the other, it needs to be shown that the quality features derived from the quality requirements are maintained over the course of time. An extensive examination is conducted for all German nuclear power plants within the framework of the periodic safety assessment (Periodische Sicherheitsüberprüfung, PSÜ) that consists of the deterministic part, the safety status analysis (Sicherheitsstatusanalyse, SSA), and a probabilistic part, PSA (see section 2.5). The SSA includes an examination of whether a facility with its safety systems is able to meet the protection goals "control of reactivity", "cooling of fuel elements", "confinement of radioactive materials" and "limitation of radiation exposure" when applying conservative assumptions in the event of any type of design-basis accident (level 3). The safety goals of old and new plants are identical.

Purpose of improvement / Improvement measures	PWR design generation 1	2	3	4	BWR construction line 69	72
1. Enhanced reliability of normal operation						
Additional off-site power supplies	x	x	●	●	x	●
2. Enhanced effectiveness and reliability of safety equipment						
Additional emergency diesel generators	x	x	●	●	x	●
Additional high pressure and low pressure emergency core cooling systems (PWR)	x	●	●	●		
Extension of emergency core cooling systems / additional injection lines (PWR)	x	x	●	●		
Technical improvement of the high-pressure/low-pressure interfaces	x	x	x	x	x	x
Self-supporting emergency core cooling systems/ new diversified emergency core cooling system (BWR)					x	x
Additional emergency feed water systems	x	x	●	●	●	●
Technical improvement of components important to safety to withstand design-basis accidents	x	x	●	●	x	●
Additional valves for containment isolation (BWR)					x	●
Diversified pilot valves for safety and pressure relief valves (BWR)					x	●
Diversified pressure relief valves (BWR)					x	x
3. Control of specific emergency situations						
Emergency systems	x	x	●	●	x	●
4. Mitigation of fire consequences						
Physical separation by installing new systems in separate buildings	x	●	●	●	x	●
Additional fire fighting systems	x	●	●	●	●	●
Backfitting of fire fighting systems	x	●	●	●	●	●
Technical improvement of fire dampers and fire partitions	x	x	●	●	●	●
Additional fire dampers	x	●	●	●	x	●
5. Improvement of barriers						
New pipes of improved materials for main steam, feed water, and nuclear auxiliary systems (BWR)					x	●
Optimised materials for steam generators (PWR)	x	●	●	●		
Removal of the former pressurised bearing water system with its connections outside of the containment (BWR)					x	●
6. Emergency preparedness						
Improvement of technical equipment for damage prevention	x	x	x	x	x	x
Improvement of technical equipment for damage mitigation	x	x	x	x	x	x

x improvement through backfitting measures
● already covered by the design

Table 2: Backfits and Safety Improvements in Nuclear Power Plants - According to Design Generation (PWR) and Construction Line (BWR)[1] [7]

[1] 1. Design Gen. PWR: KWO, KKS; 2. Design Gen. PWR: KWB A, KWB B, GKN 1, KKU
3. Design Gen. PWR: KKG, KWG, KKP 2, KBR; 4. Design Gen. PWR: GKN 2, KKI 2, KKE, KMK
BWR 69: KKB, KKI 1, KKK; BWR 72: KRB B, KRB C

Preventive measures PWR:
- secondary side bleed and feed,
- primary side bleed (pressure reduction) and feed,

Mitigating measures PWR:
- assured containment isolation,
- primary side bleed,
- filtered pressure relief of the containment vessel,
- H_2 countermeasures,
- supply-air filtering for the main control room.

Preventive measures BWR:
- an independent injection system,
- additional possibility for injection and refilling of the reactor pressure vessel,

Mitigating measures BWR:
- assured containment isolation,
- (diversified) pressure relief of the reactor pressure vessel,
- filtered pressure relief of the containment vessel,
- inertisation of the atmosphere of the containment vessel or of the pressure
- suppression pool air volume,
- supply-air filtering for the main control room.

Auxiliary measures supporting the preventive and mitigating measures in both reactor types:
- emergency power supply from neighbouring plant unit (if applicable),
- sufficient* capacity of the batteries,
- possibilities for a prompt restoration of the off-site power supply,
- an additional off-site power supply (underground cable),
- sampling system in the containment vessel.

* Sufficient for the increased demand when carrying out AM-measures

Table 3: Measures of plant-specific Accident Management in German Nuclear Power Plants

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2.3 Aging phenomena

The aging of nuclear power plants is both conceptually and technologically determined. Technological aging refers to changes in the characteristics of technical equipment over the course of operating use until their decommissioning. These changes in characteristics that are generally not positive have already been taken into account in a conservative way in the design and operation of a plant. As stated in section 2.1, a fundamental principle of the conservative safety philosophy is that of sufficient safety margins, i.e., the design limits of systems, structures, and components (SSCs), are considerably lower than the damage thresholds. In order to both verify that sufficient margins are always available and to thereby maintain safety, aging effects are monitored. Examples include:

- Putting radiation samples inside the reactor pressure vessel to determine in advance the impact of neutron radiation on the material characteristics and to respond with corresponding measures.
- Periodic tests of SSCs that are important from a safety engineering point of view.
- Continuous determination and assessment of the mechanisms influencing aging (such as temperature, transients, water chemistry, vibrations).

The results of these supervisory measures provide the basis for preventive maintenance, repairs or the replacement of SSCs.

- According to [7], a total of approximately DM 3 billion are spent every year on maintenance, periodic tests, replacement of components and backfits for all currently operating German nuclear power plants. These expenditures ensure that even those aging effects that cannot be anticipated for the entire operating life do not lead to a decrease in safety.
- Characteristics that are not easily amenable to periodic tests, especially the Loss of Coolant Accident (LOCA) environmental resistance of components, are ensured by qualification of these SSCs. This includes tests of artificially aged SSCs under accident conditions. The reliability of the qualification over long time periods is verified by testing representative SSCs.

For several years, numerous studies have been conducted worldwide on the investigation of the aging of SSCs. This has resulted in a plethora of scientific papers and a lively international information exchange [8].

The experience gained from the supervision of plants and from investigations on dismantled SSCs and the resulting optimization of plants contributed to a continuous decline in the frequency of incidents. There are no significant differences between old and new plants in this respect.

The technical run-times of nuclear power plants is determined by the safety and reliability required of SSCs. The actual run-time of a nuclear power plant depends on its profitability which is decisively influenced by the costs required to maintain the safety standards and to implement upgrading measures.

The US Nuclear Regulatory Commission (USNRC) sponsored a major study, the Nuclear Plant Aging Research Program, that collected data and assessed the impact of aging on major SSCs [9]. The results of this program were a major input to the formulation of the License Renewal Rule in the United States. Nuclear power plants in the USA are licensed for a period of 40 years. The license can be renewed for an additional 20 years if the requirements of the License Renewal Rule are satisfied. This entails an exhaustive review of SSCs and their aging mechanisms, as well as an assessment of the adequacy of aging management programs.

The first plants to request such a license renewal are the Calvert Cliffs Nuclear Power Plant, units 1 and 2 and Oconee, units 1 to 3. The Commission granted this renewal recently on the basis of an evaluation by the NRC staff and a recommendation by the NRC Advisory Committee on Reactor Safeguards [10]. The 5 units which started commercial operation between 1973 and 1976 are now licensed to operate up to dates between 2033 and 2036!

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2.4 Operating experience

According to the International Atomic Energy Agency (IAEA), a worldwide cumulative operating experience of 9,384 reactor years was determined at the end of 1999. In Germany, the experience with light water reactors at the end of 1999 amounted to 590 reactor years [11] which allows for the following statements to be made:

- The conservative safety philosophy has proven successful in that accidents in German light water reactors leading to public health and environmental damage have been prevented. No radiological loads arose for the general public that exceeded the permissible values for the normal operation of a nuclear power plant. All incidents have been controlled by the original design. Thus, the backfit represent preemptive increases of the safety margins.

- Statements on the development of the safety standard over time can be made due to the availability of detailed evaluations [12]. According to these, a clear decrease in the number of occurrences of important transients, namely the loss of the main heat sink as well as malfunctions of the feedwater and auxiliary power supply has been recorded in the last two decades. The number of unscheduled reactor scrams has been lowered during this time by a factor of around 3. The declining number of reportable events over the years as well as the very high availabilities of the plants also point in the same direction. This development can largely be attributed to the steady rise in quality both in terms of technology and staff training (1st level of the safety philosophy).

- The unavailability [12] of single trains of safety systems during periodic tests have been at a very low level following a distinct drop in the 80s.

When comparing German operating experience with those of other countries that pursue reactor safety according to fundamentally similar principles such as Belgium, Finland, France, UK, Japan, Netherlands, Sweden, Switzerland, Spain, and the USA, then it can be established that the good overall safety balance is not unique to Germany. In the more than 350 light water reactors of Western design and operating practice, there has been one accident with core damage at Three Mile Island which did not result in a serious release of radioactivity beyond the plant. The accident, which took place more than 20 years ago, gave rise to measures for further increasing the safety margins of nuclear power plants in many countries, including Germany. One ensuing measure was the founding of an efficient international information exchange on operating experiences. The accident in Chernobyl has not been included in this comparison since this plant is of a reactor type which would not qualify for licensing in Western countries because inadequate standards were applied both in terms of design and operation

The developments over time that have been discussed for German plants are mirrored by many countries. The plant availability over the last years has increased distinctly on an international level.. American plants publish quantitative code numbers, so-called performance indicators. They have been reflecting a rise in the quality of operation for years (see also section 2.5, figure 1).

2.5 Risk Assessment

2.5.1 Probabilistic safety studies in Germany

As stated in section 2.1, the nuclear industry and regulatory authorities initially handled unquantifiable uncertainties in deterministic reactor safety analyses by implementing the principles of defense in depth and safety margins [6].

Starting in the 1970s, the methodology of Probabilistic Safety or Risk Assessment (PSA or PRA[2]) has provided the capability to actually quantify the uncertainties thus leading to a more rational approach to safety management. This methodology has changed the approach to reactor safety in two essential ways:

- The plant is analyzed as an integrated system consisting of hardware and plant personnel.
- Quantitative values characterizing the risk are defined and calculated. The most commonly used metrics are the core damage frequency (CDF); the large, early release frequency (LERF); the probability of death of an individual living near the plant; and the probability of a number of deaths in society at large.

Depending on their scope, we define the following three PSA levels [13]:

A **level 1 PSA** consists of an integrated analysis of plant design and operation focused on the accident sequences that could lead to core damage, their basic causes, and their frequencies. Over and above this, a **level 2 PSA** consists of an analysis of the physical processes of accident courses and the response of the containment building up until the emission of radioactive substances in addition to the analysis performed in a level 1 PSA. A **level 3 PRA** analyzes the transport of radionuclides through the environment and assesses the public-health and economic consequences of the accident in addition to performing the tasks of a level 2 PSA.

Within the framework of PSR (periodic safety reviews), mainly **level 1+ PSAs** are performed in Germany; these are level 1 probabilistic analyses that include the active safety functions of the containment, i.e., elements of level 2 PSAs. Another special feature of level is their definition of the final states of accident sequences. These states are categorized as controlled states and/or hazard states; the latter are defined as states whereby a protection goal, like *core cooling*, is threatened. For the definition of these states, only design-basis safety systems and human

[2] In Germany the term Probabilistic Safety Assessment (PSA) is more commonly used than the term Probabilistic Risk Assessment (PRA).

actions anticipated in the operating manuals are taken into account. This means that the possible beneficial effect of AM measures described in the emergency manual is neglected in the accident frequencies.

These level 1+ analyses are conducted every 10 years within the framework of the periodic safety reviews and follow a binding guideline [18] and are based on a recommendation made by the Reactor Safety Commission (Reaktor-Sicherheitskommission, RSK [19]) for all German NPPs. They rest on a voluntary self-commitment of the operators and supplement the continuous supervision by the authorities.

The results of these analyses include the non-availabilities of safety systems as well as the conditional probabilities for outages of the active functions of safety containment. This information is used for evaluating the safety level and balancedness of the plant design.

Amongst studies conducted in Germany going beyond this level, the German Risk Study, Phase A (DRS A) [14] as well as the German Risk Study, Phase B [15] for the NPP Biblis B should be mentioned. In 1979 the DRS A-Study determined core melt frequencies, type and frequencies of large releases as well as damages resulting from releases.

The risk study reflected the knowledge available of that time. Especially regarding failure modes of the containment and the associated probabilities, it had to rely very much on assumptions.

Phase B of the Risk Study (DRS B), finished in 1990 [15], broadened the scope of initiating events and performed in-depth analyses of a number of specific issues. The frequency of failure of the safety systems was calculated and demonstrated that Accident Management Measures were able to prevent core melt in most of these cases.

The AM-measures have been considered in a preliminary evaluation. Investigations conducted by the GRS in the meantime have led to a more favorable result [16]. The possible failure modes of the containment were analyzed but their probability of occurrence could not be quantified at the time. Yet the investigations revealed that large releases could only arise when there was an early failure of the containment or by an overpressure failure of the containment. Extensive assessments of these scenarios have been carried out since then and measures for their prevention were developed. These measures have largely been implemented in the PWRs or are in the process of being implemented.

Similar to the Risk Study for the PWR Biblis, a probabilistic analysis for a BWR was carried out with Gundremmingen as a reference [17]. Again, the non-availability of the safety systems was calculated and AM measures were identified which could prevent core melt. As in the case of PWRs, measures were also introduced for BWRs to prevent early containment failure.

2.5.2 Results of the PSA analyses

A quantified statement on the development of public health risk, e.g., the expected number of health effects, requires level 3 PRAs performed at different times. Since these are not available in Germany, one can gain insights regarding trends in public health risk by investigating the trends in core damage frequency (CDF) and the frequency of large, early releases (LERF) of radionuclides. These are good surrogate metrics because significant public health effects can only occur when the core is damaged and large amounts of radionuclides are released from the containment in a short period of time. These frequencies are evaluated in DRS A [14] and DRS B [15] for Biblis B. Additional insights can be obtained from level 3 PSAs performed in other countries such as the US PRAs [20-21].

A PSA represents a "snapshot" of the risks in time that is based on the understanding of the plant behavior (plant model), data available, consideration of uncertainties or - if need be - conservative assumptions. When the NPP under consideration continues operation, plant modifications or changes are implemented as a result of operating experience and improved understanding of phenomena. The extended research and development in the nuclear domain have led to an improved understanding of the complex interrelations and phenomena that are relevant to accidents. Many plant modifications have been effected as a result of this continuous evolution (see also section 2.2).

In Germany, the evolution of risk can partially be appreciated by comparing insights and results of DRS A and DRS B. DRS A gives a core damage frequency of $9x10^{-5}$/a whereas DRS B (10 years later) results in $4{,}5x10^{-6}$/a. The difference is partially due to various system improvements but the major part is the result of the introduction of two accident management measures, i.e., secondary and primary side feed and bleed.

DRS A gives a frequency of $2x10^{-6}$/a for early containment failure. The frequencies for core melts determined by the DRS B study result in the following current state of knowledge:

- Core melts that bypass the containment as well as an overpressure failure of the containment are extremely unlikely (as shown in DRS B).
- Due to hydrogen-reduction measures, this also applies to a failure of the containment as a result of hydrogen combustion [22], [23].
- In order to avoid core melts under high pressure, the AM-measure "primary side pressure relief" was introduced. The DRS B evaluates the availability of this AM-measure and yields a core damage frequency in the so-called high pressure path for the entirety of all investigated initiating events of $4{,}5x10^{-7}$ per reactor year. Given a more realistic assessment of the AM-measure according to [16], the core damage frequency in the high pressure path drops to $3{,}6x10^{-7}$ per reactor year[24]. Figures for the probability for a damage to the containment under these circumstances are not available. A corresponding frequency (a value of $3{,}6x10^{-7}$ per reactor year) for the failure of the containment due to core melt under high pressure and an associated early release of large quantities of radioactive substances can thus be assumed as a conservative upper-bound estimate.

From the above it can be seen that the core melt frequency has been decreased significantly. The probability of a large release has also been reduced substantially by implementing measures to prevent early containment failures. The frequency of large releases is smaller than it was assumed to be in DRS A of the risk study.

These considerations are representative for all PWRs because their containment failure modes are the same. The secondary and primary feed and bleed as well as the measures taken to present early containment failure are essentially the same for all PWRs.

For BWRs measures were also taken to prevent early containment failure:

- To cope with any kind of overpressure, a filtered containment vent was introduced.
- To cope with hydrogen, the containments were inerted, fully in the construction line 69 and part of the rooms in the construction line 72.

Similar positive developments in risk are observed worldwide. A recent presentation of the US NRC [25] demonstrates very clearly the continual improvement in performance that the operating experience indicates. In particular, an analysis of initiating-event (i.e., abnormal event) frequencies shows that:

- summated initiating event frequencies for all initiators are lower than the frequencies used in prior PSA studies, e.g., NUREG-1150 [21], by factors of 4 to 6,
- most risk-significant initiator frequencies decreased at a faster rate than the overall initiating event frequencies,
- loss of coolant accident frequencies are lower than those used in NUREG-1150 [21].

The frequency of common-cause failures, i.e., failures that defeat redundant systems, has decreased distinctly over the years, as figure 1 shows.

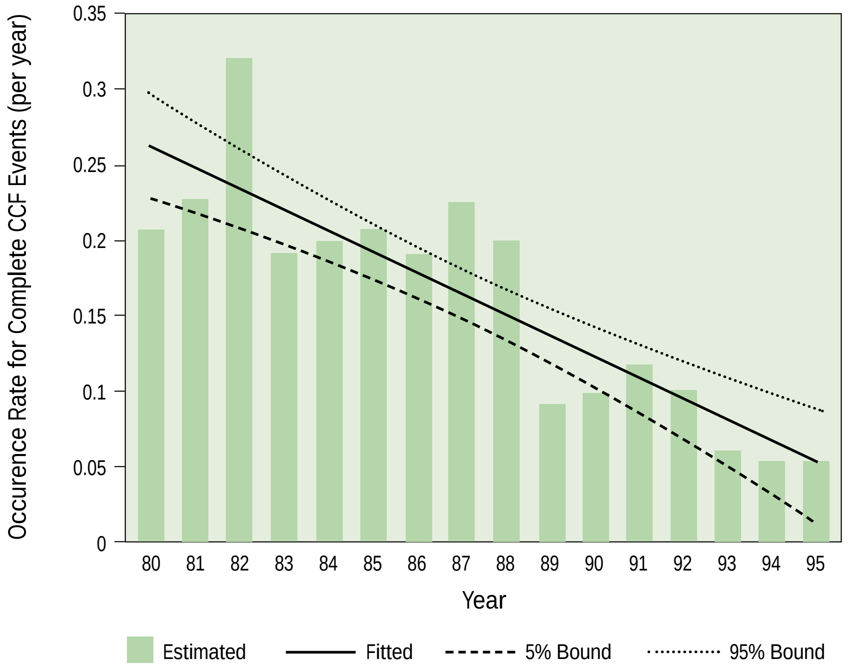

Figure 1 Evolution of the frequency of common-cause failures over time [25].

The American Surry NPP was studied by both the Reactor Safety Study [20] (published in 1975) and the NUREG-1150 Study [21] (published in 1987). The Reactor Safety Study mean estimate of core damage frequency at that plant is 4.6×10^{-5} /a and the mean estimate from NUREG-1150 is 2.6×10^{-5} /a, a reduction by a factor of 1.8. As pointed out in [26], NUREG-1150 *added* reactor coolant pump seal failures as a new initiator to the small loss-of-coolant-accident sequence, thereby increasing its frequency by a factor of 10. However, plant modifications (e.g., cross-ties between units) and improved analytical models (e.g., more realistic core thermal hydraulics) resulted in a total *decrease* in core damage frequency. That means that *individual changes* should *not* be evaluated in isolation.

Similar comparisons can be made for the severe accident source terms. Figure 2 shows the frequency of releasing an important radionuclide, iodine. The full uncertainty range of the NUREG-1150 results is shown (5th and 95th percentiles, mean, median), while only median values are available from the Reactor Safety Study. The shaded area in the figure illustrates the reductions in median frequencies. For example, the frequency of releasing 10% or more of the iodine inventory is reduced by more than a factor of 10. The reasons for these reductions are plant modifications (lower core damage frequency, as discussed above) and better understanding of severe accident phenomena.

The *consequences of nuclear accidents* are complex but can be studied and largely quantified. In fact, level 3 PSAs quantify a number of consequences. As an example, figure 3 shows the frequencies of early and latent fatalities following an analysis of internal events. The uncertainty in NUREG-1150 estimates is displayed by the "high" and "low" curves. The Reactor Safety Study estimates are, as before, found to be conservative, i.e., close to the NUREG-1150 upper bound. Note that for early fatalities greater than about 200, the frequencies estimated by the Reactor Safety Study are higher than even the upper bound estimated by NUREG-1150. This comparison confirms, once again, that, on the one hand, risks in the NPPs considered are decreasing with time and that, on the other hand, a better understanding of the phenomena enables the use of less conservative assumptions.

In summation, it can be said that the state of knowledge on the risks of nuclear energy (risk for the population, surrogate metrics such as CDF and LERF) has continuously advanced. The plants have undergone constant improvement. On the whole, a reduction of the risk can be demonstrated.

While theoretical models, as described in [3], that extrapolate and accumulate the risk resulting from operating NPPs 50 years into the future on the basis of a current core damage frequency of 10^{-5} per reactor year applied to 20 plants are valid,

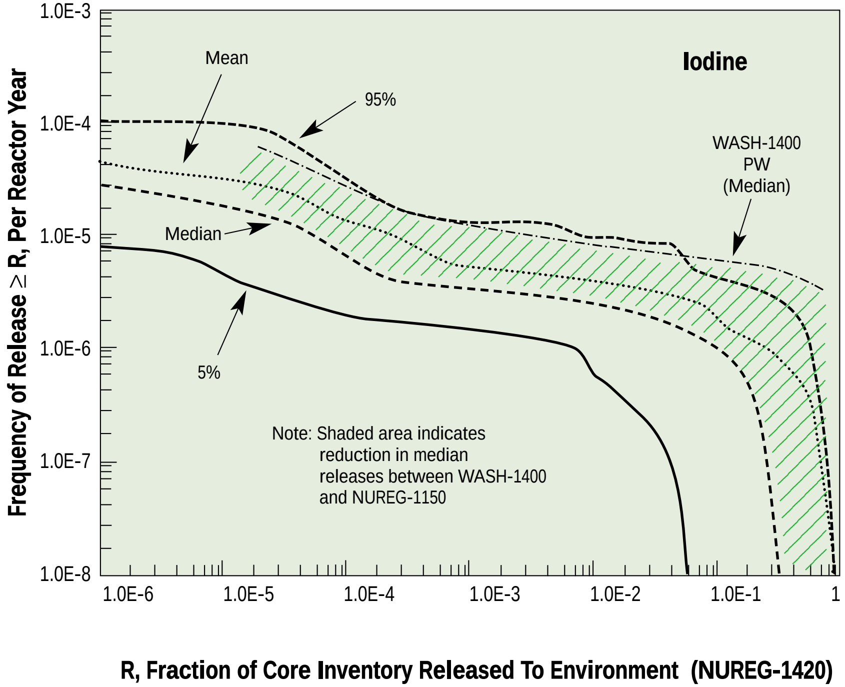

Figure 2 Frequency of Exceeding Iodine Release Fractions in NUREG-1150 and RSS (Wash-1400) Analysis of Surry, [26]

they are not very meaningful. One could arrive at a probability of 1% for an accident within the next 50 years if one assumes the core damage frequency to be constant over this long period of time, which is highly questionable. It should also be pointed out that a core damage event does not inevitably lead to serious releases and public consequences (cf. TMI-accident).

The question of whether a realistically calculated risk of a conceivable future accident that takes these developments into account can lastly be judged to be acceptable can only be answered on the basis of criteria and comparisons with the risks of other energy producing sources (see Chapter 3) and with internationally acknowledged guidelines. In this context, it should be noted that a core melt frequency of 10^{-5}/a is significantly lower than the IAEA recommendation for plants

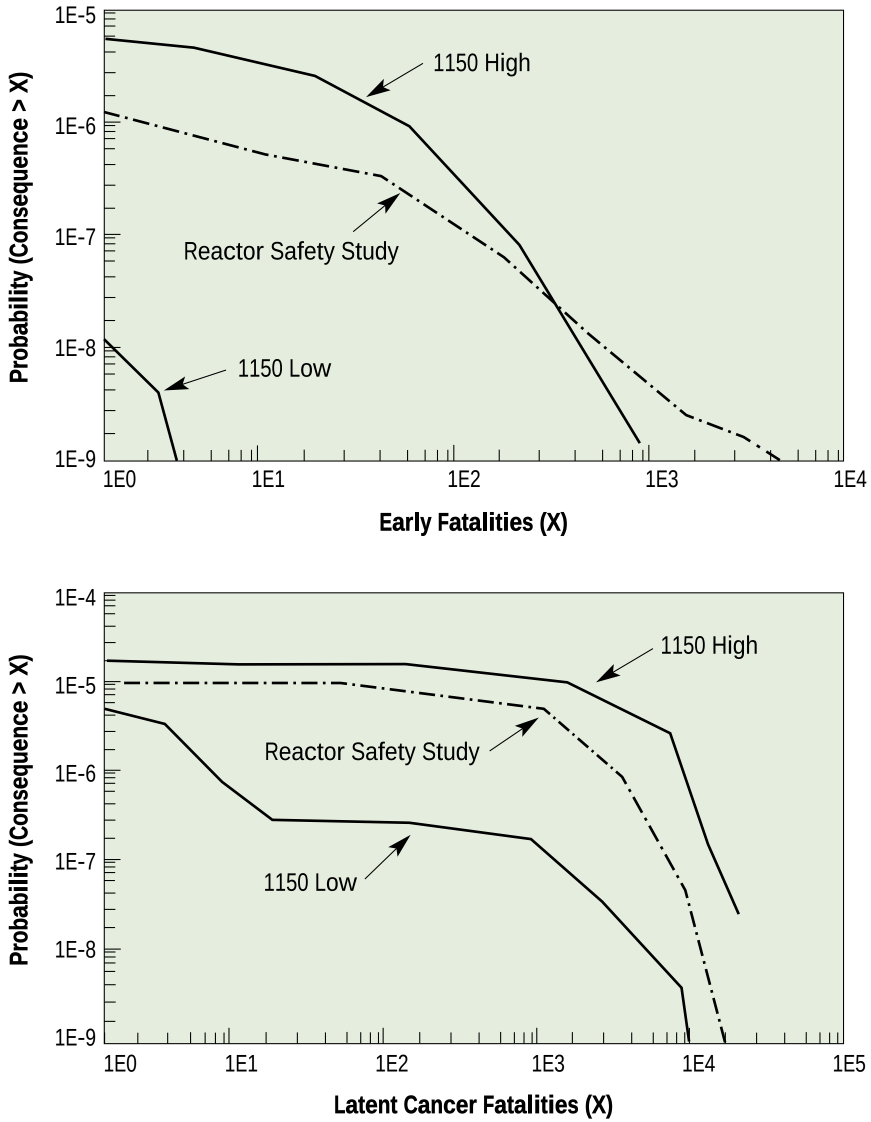

Figure 3 Comparison of the cumulative frequency of occurrence damage function between NUREG-1150 and RSS (Wash-1400)

already in operation and also below the safety goal of the US NRC (10^{-4}/a) by a factor of 10. Furthermore, it is important to point out that the US NRC's Quantitative Health Objectives demand that the risk of death or cancer from NPPs be less than 0.1% of the corresponding risks from all other causes to which residents of the USA are subjected. The goal for core damage frequency of 10^{-4}/a is consistent with this objective. This confirms that a core damage frequency of 10^{-5} per reactor year represents a high nuclear safety.

2.5.3 Assessment of risks of radiation exposure

The evaluation of the health risk of exposure to ionizing radiation is a further important element in the risk assessment of nuclear energy. Between 1972 and 1988, a risk coefficient of 1.25% per Sv was unanimously used for calculations. This value was anchored in the ICRP Recommendation 26 (1977) [27]. The coefficient signifies that an additional dose ΔD results in an additional cancer risk ΔR regardless of a person's prior radiological exposure:

$$\Delta R / \% = 1.25 \times \Delta D / \mathrm{Sv}.$$

This linear approach represents a practical simplification. The risk was considered to be real for $\Delta D > 200$ mSv and to be hypothetical for smaller doses.

The reevaluation (published in 1985) of the epidemiological data from the Hiroshima and Nagasaki atomic bombs resulted in an unexpectedly steep rise in the number of radiation-induced cancer cases. As a result of this fact and a change of the extrapolation model, the ICRP Recommendation 60 (1990) [28] determined the risk coefficient to be 5% per Sv; the validity of this relationship held to a value as low as approx. $\Delta D = 100$ mSv and has subsequently been confirmed [29].

Today, a deceleration of the growth in radiation-induced cancer cases for survivors of the bombing of Hiroshima and Nagasaki can be observed [30]. The model used by the ICRP 60 appears to be too pessimistic. Additionally, subsequent determination of doses using activation analyses show that the proportion of neutrons in "Hiroshima" was significantly greater than thus far assumed so that a greater number of the radiation-induced cancer cases can be explained via the neutron dose [31]. As a result, less cancer cases should have followed from "Hiroshima" that are attributable to the γ-dose. Consequently the risk coefficient for γ-radiation should be noticeably smaller than 5% per Sv.

It also becomes increasingly more apparent that the additional risk ΔR depends to a much greater degree on the dose rate. The same dose received with a high dose rate (such as in Hiroshima and Nagasaki) leads to a noticeably higher risk. The more precisely determined risk should be a function of the following variables:

$$\Delta R = f\,[\,\Delta D, (\Delta D)^2, \Delta D/\Delta t\,].$$

The influence of $(\Delta D)^2$ and $\Delta D/\Delta t$ was considered in a simplified way via the "Dose and Dose Rate Effectiveness Factor" (DDREF). A radiation exposure that is relatively small or that occurred with a low dose rate is less efficient by the factor DDREF in causing late damage. In its publication 60 [28], the ICRP introduces the factor DDREF and gives it the value of 2, which is regarded as conservative. The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) arrived at the conclusion in 1993 that a DDREF = 3 would be prudent and a value of 4 would be compatible with the existing data [32].

Thus a development is clearly underway towards a reduction of the "5% per Sv-coefficient". A short to mid-term reduction of the nominal radiation risk coefficient can be expected for the reasons given above, i.e. the past increase by a factor of 4 could at least be partially reversed soon.

2.5.4 Occupational risk in German nuclear power plants

The EURATOM-guidelines were adopted by German law for regulating radiation protection of occupational groups exposed to radiation. An effective dose of 50 mSV per year for workers is laid down as the limiting value for the whole-body-dose of persons exposed to radiation in today's radiation protection ordinance. This value will be lowered to 20 mSv in the future as a result of the reassessment in [28] (see section 2.5.3).

Continuous improvements in the radiation protection and maintenance fields and in plant operation have contributed to the continual decline in the past years of person doses among staff of German nuclear power plants so that an adherence to the new limiting value should not present a problem to the operators. This trend towards lower person doses is also mirrored in the collective doses. Figure 4 shows the course of the mean annual collective dose of nuclear power plants per year and plant.

For pressurized water reactors, this decrease is due mainly to the use of steel alloys low in Cobalt; for boiling water reactors, the reduction is due to the modification of internal circulation pumps and the replacement of piping with a reduced test effort for weld seams. In contrast, this extensive replacement was the reason for the dose peaks registered in the early 80s.

In all nuclear power plants, an increased use was made of local shields (lead mats). Additionally, the ALARA (As Low as Reasonably Achievable) principle was implemented with greater consistency.

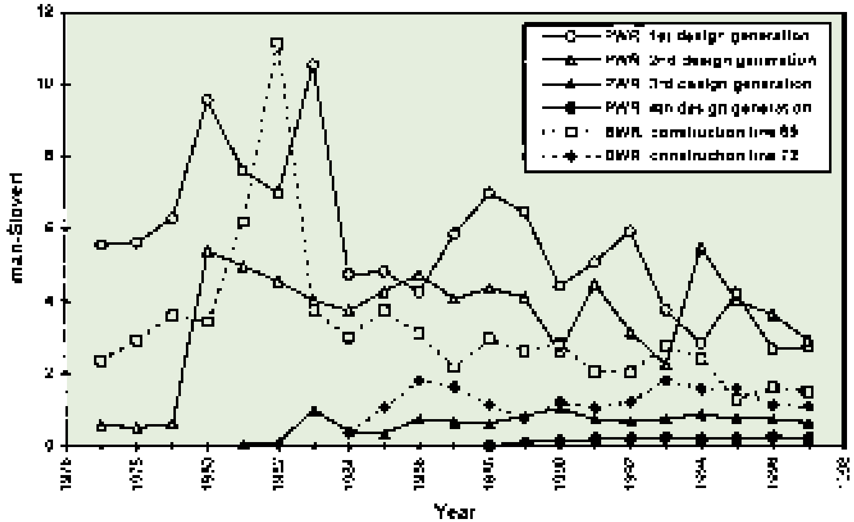

Figure 4: Mean annual collective doses of nuclear power plants per year and plant [7]

3 General categorization of nuclear energy and associated risks

Next to its risks, the assessment of nuclear energy must also include its benefits. According to the international standards that are customary today, it should not be viewed in isolation but must be compared with realistic alternatives. Furthermore, risk is "only" one evaluation criterion. If the compatibility with the precept of sustainability is being investigated, other criteria also gain relevance. Of these, the emission of greenhouse gases is the most important. The ILK specifies aspects and first partial results of such a perspective and will expand and explain these over the course of its further advisory activities.

For harnessing the energy from nuclear fission, 433 reactor blocks provided an installed capacity of 349.1 GWe at the end of 1999; 4 units (total capacity 2,7) began operations and two were shut down over the course of the year. Construction began on 7 new nuclear power plants in China (1), Japan, South Korea and Taiwan (2 each), so that currently there are 37 nuclear power plants in construction worldwide [11]. Nuclear energy has a 17% share in worldwide electricity production but only covers a little more than 6% of the primary energy demand. In Germany, 19 units provide around 33% of domestic electricity production [33].

The share of fossil energy carriers in the worldwide primary energy turnover lies beyond 80%. This will also be the case in the year 2020[3], which, extrapolated for an average annual growth of the primary energy demand from 2.2 to 2.5%, will increase the CO_2-emissions from the current level of 21,3 billion t to 39.5 billion t, i.e. by about 80% [34]. In Germany, a decline in the use of nuclear power solely through a regular expiration of plant service life would largely be compensated by an increase in the consumption of natural gas; while the proportion of renewable energy carriers will increase from 2.2 (1995) to 4-5% (2020), their contribution will remain fairly modest nonetheless. The German CO_2-reduction goal (-25% compared to 1990) cannot be achieved in this way; a premature exit from nuclear energy use would further aggravate this state of affairs [35].

A flagrant contradiction also exists on the international level between the guidelines provided by international agreements on considerably reducing[4] the emission of greenhouse gases and the actual or feared considerable increase of its output.

3 see Environmental Expert Opinion 2000 [35]; its statements are based on forecasts made by the European Energy Institute,99; US DOE-EIA,98; IEA, 98, WEC-IIASA, 98

4 "Kyoto" demands a reduction of worldwide CO_2 emissions by 8% (from the 1990 level) until 2008-2012.

Based on current insights, a decarbonization of energy systems seems called for, yet a carbonization is "pre-programmed" in Western countries given the greater use of natural gas and use of coal (doubling in India and China from 1990 to 2010 [34]).

A greater use of nuclear energy cannot fully prevent this development but can play its part in extenuating it. A new EU-study shows that, even if 100 nuclear GWe were to be installed over the next 25 years, their emission would lie 4% below the 1990 levels but the CO_2-reduction goals would not be achieved [36]. The changed stance by the US towards the future role of nuclear energy is remarkable: Together with eight other countries Argentina, Brazil, Canada, France, Japan, South Korea, UK, i.e. the majority of the G-7 countries without Italy and Germany, they have drafted a joint declaration and initiated new activities aimed at developing an innovative nuclear technology and giving special consideration to competitiveness, safety, disposal, supply guarantee and prevention of proliferation [37].

These aspects are included in a set of indicators with whose help the aim of a sustainable global development - which has recently been adopted by many countries- should be operationalized (e.g. [38]). A general consensus currently exists with regard to

- including economic, ecological and social as equivalent aspects when considering sustainability as well as
- analyzing energy supply systems using many criteria based on the entire cycle ("from cradle to grave")
- making a comparison with alternatives.

According to the current state of knowledge, the different energy carriers and chains can be assessed in terms of their benefits and disadvantages. The benefits of nuclear energy lie in its largely emission-free electricity production (see Appendix A), the safe availability of resources and - with reservations - in its competitiveness. High and intermediate level waste accrue in amounts worth mentioning only for nuclear energy. Yet on the whole, these amounts are small. The large damage consequences of "worst possible reactor accidents" without consideration of their extremely low probability have contributed to negative public perceptions. Finally, the economic competitiveness of current-generation nuclear plants could be significantly improved by a more rational management of their resources.

In the meantime, the state of knowledge concerning risks has progressed to such an extent that, at least on the basis of accident-related health damages (fatalities as damage indicator), a comparative classification of the risks of different energy supplying options can be undertaken [39].

One possibility of doing this is via an evaluation of the statistics on major accidents for different energy chains in terms of acute fatalities and also their relationship to the relevant total amount of electricity produced (table 4). "Chernobyl" is included in the worldwide evaluation as the only nuclear accident with acute fatalities. In order to give adequate consideration to the general substantial differences in the quality of facilities and the safety level, figures for OECD and non-OECD countries are given separately in the table below. The risk values arrived at in this way illustrate the superiority of nuclear power over other energy chains.

The statistics give an answer to the question on the greatest damage involved. According to the given statistics, over 1000 acute fatalities resulted from oil fires and dam bursts in each of a total of four individual events (in non-OECD countries). The results from probabilistic analyses (PSA level 3) are needed to provide comparative figures from Western nuclear engineering (see figure 3, section 2.5.2). According[5] to these, the number of immediate fatalities to be reckoned with would range between very few and up to several thousand, depending on the severity of the accident sequence, the weather conditions and the population density in the

Energy Chain	Number of severe* accidents	Number of acute fatalities [per GW_e and year]		
		Worldwide	OECD	Non-OECD
Coal	187	3.4×10^{-1}	1.4×10^{-1}	5.1×10^{-1}
Oil	334	4.2×10^{-1}	3.9×10^{-1}	4.6×10^{-1}
Natural gas	86	8.5×10^{-2}	6.6×10^{-2}	1.1×10^{-1}
Nuclear	1	8.4×10^{-3}	0	5.3×10^{-2}
Hydro	9	8.8×10^{-1}	4.0×10^{-3}	2.2

* Individual accidents with 5 or more fatalities, worldwide between the years 1969 and 1996

Table 4: Empirical frequency of major accidents and acute fatalities for complete energy chains, normalized for the amount of electricity produced as an utilization indicator, source [39]

vicinity (<20 km). However, the frequency of occurrence is extremely small ($<<10^{-7}$ per reactor year[6]). The question of whether small frequencies of occurrence comparable to those found for nuclear power would also lead to higher damage figures for conventional energy technologies is still unanswered today.

In order to pay tribute to the special nature of nuclear risk, acute and also latent fatalities due to late radiation cancer must be included in the model. Current analyses indicate that, in the worst case, up to several 10,000 fatalities must be reckoned with, again taking into account the extremely low frequency level which, when multiplying these two magnitudes, leads to a corresponding accident risk in the area of 10^{-3} to 10^{-1} of latent fatalities per reactor year. In comparison, according to the cancer atlas of the Federal Republic of Germany, more than 210,000 cancer deaths are recorded in Germany annually.

In the assessment of accident risks, long lasting and considerable land loss due to contamination must be considered especially in the case of nuclear energy. Its extent and duration is determined both by the decay time of the key nuclides (half-life I-131: 8 d, Cs-173: 30.2 a) as also by the ability to finance decontamination measures. Social damages such as those that were manifest as a result of the Chernobyl-accident but also in the wake of the Bophal-accident, have a greater significance today in the assessment of major accidents.

Of significance in the discussion on nuclear phaseout in Germany is the fact that these problems cannot be solved by giving up nuclear power plants domestically and continuing operation of nuclear power plants in neighboring countries. Imports of CO_2-emission-producing electricity are a step in the wrong direction due to their contribution to the worrisome climate changes that are an increasing cause for alarm given the current state of knowledge on their unlimited reach in terms of geography and time. A global conceptual approach also includes considering that the German safety culture and technology has had a positive influence especially in those countries willing to construct and operate nuclear power plants which thus far still have a reputation of low existing safety standards. This positive influence would be lost should the technology be abandoned domestically.

5 The German Risk Study Phase A (Deutsche Risikostudie Phase A, 1979) provided the basis for these data. Newer studies are also taken into consideration as well as trends that have developed in the meantime and that have been described in chapter 2.5.2. Since a more recent German study (PSA Level 3) is not available, the values given here are estimates.

6 A reactor year corresponds to one $GW_e a$ for a reactor of the 1300 MW_e type.

4 Waste disposal

A further major feature of the argument for nuclear phaseout is the failure of waste disposal in the eyes of the government. The ILK has issued a separate statement on this central claim of the Federal Government [40], [41].

The disposal of radioactive waste is regulated in Germany by the disposal concept for radioactive waste [42] that was negotiated between the Federation and the individual Federal States next to legislative requirements laid down in the Atomic Energy Act [4] and the Radiation Protection Ordinance [5]. The disposal concept for radioactive waste [42] is deemed to be a failure by the coalition agreement between the Social Democratic Party and the Greens [2]. Furthermore, the Environmental Expert Council (Umweltrat) voices doubts in its Expert Opinion 2000 that the final disposal of radioactive wastes can be implemented from a safety engineering point of view [35].

In the operation of nuclear power plants but also in the use of radioactive materials in industry, research and medicine, residual materials accrue that either need to be fed into non-hazardous re-use or are to be disposed of in an orderly manner as radioactive waste according to the stipulations of § 9a of the Atomic Energy Act [4]. Tried and tested procedures and containment vessels are available for the treatment and packaging appropriate to final disposal of radioactive wastes accruing during the operation of nuclear power plants.

One step in the disposal concept pursued thus far, was the treatment of spent fuel elements handled exclusively by reprocessing mainly in the UK (BNFL) and France (COGEMA) up until a corresponding change in the Atomic Energy Act in the year 1994.

The so-called direct final disposal represents an alternative disposal pathway to the reprocessing of spent fuel elements and has become an option following the amendment of § 9a AtG in 1994. Until a corresponding final repository has been made available, the spent fuel elements are currently stored in the tested containment vessel - e.g. of the type CASTOR - in interim storage sites in Ahaus and Gorleben (with a capacity of 420 vessel spaces each). The interim site Gorleben accommodates the reprocessing wastes returned from abroad in accordance to contractual obligations. To prepare the fuel elements for direct final storage, these are conditioned and packaged in the pilot conditioning plant (PKA) in Gorleben. The plant is technically complete and is only awaiting the last partial licensing by the federal state of Lower Saxony for begin of operations.

The aim for disposing of long-lived intermediate- and high-level waste (including spent fuel elements) in Germany [43] as well as on a world-wide basis is the final storage in deep geological formations. Apart from one final repository for low- and intermediate level alpha-emitters from the production of nuclear weapons in the USA currently there is no corresponding operating final repository anywhere in the world. The final storage projects in the individual countries are in different stages of development. While in some countries (e.g. Germany, USA), exploratory activities at a selected site have already been conducted, other countries (e.g. Sweden, France) have not yet completed their search for an appropriate site. The individual countries are examining different rock formations (e.g. salt, granite, clay, tufa) for their suitability as final repositories. Rock salt is frequently preferred as a host rock for heat-generating waste due to its physical characteristics The suitability of the salt deposit Gorleben as a final repository for radioactive waste, especially for waste with heat generation, can, however, only be conclusively assessed once the underground exploration currently in progress has been completed.

Due to the diverging safety engineering requirements and the different requirements in terms of temporal availability (substantially less High Active Waste (HAW) accrual than Low Active Waste (LAW) / Medium Active Waste (MAW)), the site Konrad was explored next to the planned final repository Gorleben in a comprehensive land use planning inquiry with regard to its suitability as a final repository for radioactive waste with negligible heat generation. The suitability of the shaft Konrad as a final repository for waste with negligible heat generation was already confirmed within the framework of investigations on the planning inquiry which also took into account extensive objections.

As the above-mentioned considerations show, the disposal of radioactive waste in Germany is based on the following steps:

- Conditioning of radioactive waste with negligible heat generation with tested conditioning procedures applied worldwide,
- Reprocessing of spent fuel elements in France and the UK and re-use of the isolated Pu-fiss in MOX-fuel elements,
- As an alternative to reprocessing, the interim storage of spent fuel elements (Gorleben and Ahaus) with the aim of direct final storage after completing conditioning appropriate to final repositories (PKA Gorleben)
- Final storage of radioactive waste with negligible heat generation in the site Konrad,

- Continuation of the underground exploration of the salt deposit Gorleben in terms of its suitability as a final repository, especially for waste with heat generation.

In summary, it can be stated that facilities for safe transportation, conditioning and for interim storage of radioactive waste already exist and their final disposal appears to be largely technically feasible in today's view.

In summary, the ILK sees no fact-based support for the claim that the German disposal concept has "failed".

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Appendix A:
Selected indicators (examples) for current systems [38]

	Fuel-reserves a)	Material consumption b) (example Bauxite)	Greenhouse-gases b)	Sulfur-oxide b)	Inorganic waste in chemical landfill b)	HAW and MAW b)c)	Production-costs d)	External (environmental) costs e)
	Years	kg/GWh$_e$	t($CO_{2\text{-eq}}$)/GWh$_e$	kg(SO_x)/GWh$_e$	kg/GWh$_e$	cm^3/GWh$_e$	Rp./kWh$_e$	Rp./kWh$_e$
Hard coal	160 - 2 300	50 - 94	954 - 1177	947 - 24516	5800 - 54000	67 - 100	5.7 - 7.4	3.1 - 15.8
Natural gas	70 - 170	55	528	259	1500	20	4.7 - 5.8	0.8 - 5.5
Nuclear power	120 - 400	50 - 70	8 - 29	56 - 152	650 - 1200	4500 - 5500	5.1 - 7.5	0.2 - 1.3
Hydro power	∞	9	4	8 - 10	30	3	3 - 14 (high head) 4 - 21 (pumped storage)	low (high head) 0 - 1.2 (pumped storage)
Photovoltaics (PV)	∞	2500 - 4000	106 - 257	698 - 3636	4900 - 10000	310 - 580	70 - 140	0.1 - 1.5

a) The values are based on secured and foreseeable resources given current consumption. Only direct consumption is considered for hydro power and PV.

b) All data are based on the LCA-method. The data for coal and nuclear power apply to countries of the Union for the Co-ordination of the Transmission of Electricity (UCPTE). Data for natural gas apply to a specific plant type (valid for the German energy chain). The data for hydro power and PV are based on Swiss conditions.

c) Radioactive wastes to be stored in geological repositories.

d) The cost spectrum of currently operating Swiss plants (hydropower, nuclear power, PV). For the fossil chains, the expected costs for constructing plants in Switzerland are estimated (the values depend on the technology employed and the fuel prices)

e) Figures for external costs apply to Western Europe. The range reflects the differences in the technologies employed, dependence on plant location (population density, meteorology) and uncertainties in the assumptions (especially for global warming).

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Objectives of the International Nuclear Technology Commission established by the States Baden-Württemberg, Hesse and Bavaria [Internationale Länderkommission Kerntechnik] - ILK -

Mission

Independently and objectively advising the states Baden-Württemberg, Hesse and Bavaria at the highest, internationally acknowledged scientific level on questions relating to the safety of nuclear installations, the regulated disposal of radioactive waste and the peaceful utilization of nuclear energy against the background of a sustainable energy supply.

Goals

1. Maintenance and improvement of the high safety standard of the German nuclear power plants and further development of the waste management concept for radioactive waste according to the internationally recognized state-of-the-art in science and technology.

2. Application of an holistic system approach to man-technology-organization.

3. Timely detection of safety defects against the background of competition in the liberalized European electricity market and development of countermeasures.

4. Inclusion of internationally acknowledged practice into the German safety philosophy and safety concept for improving state supervision and for increasing the safety standard of installations.

5. Treatment and evaluation of selected safety issues with regard to new scientific insights and development of recommendations on the harmonization of nuclear engineering standards on a European level.