An Overview of Differences in Nuclear Safety Regulatory Approaches and Requirements Between United States and Other Countries

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ABSTRACT

This report has been prepared for use by the NRC Advisory Committee on Reactor Safeguards (ACRS) in support of its ongoing effort to inform the Commission on significant differences in regulatory approaches and requirements between the United States and other countries. This report, which is based on review of a number of documents issued by various international organizations, provides an overview of regulatory approaches and discusses differences in specific regulatory requirements of current interest in the United States.

The views expressed in this report are solely those of the author and do not necessarily represent the views of the ACRS.

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ACRONYMS

Acronym	Definition
ABWR	Advanced Boiling Water Reactor
ACRS	Advisory Committee on Reactor Safeguards
AGR	Advanced Gas-cooled Reactor
ALARA	As Low As reasonably Achievable
ALARP	As Low As reasonably Practicable
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CSNI	Committee on the Safety of Nuclear Installations
CNRA	Committee on Nuclear Regulatory Activities
CSS	Containment Spray System
DCH	Direct Containment Heating
ECCS	Emergency Core Cooling System
EPR	European Pressurized Water Reactor
GCR	Gas-Cooled Reactor
GE	General Electric
IAEA	International Atomic Energy Agency
LOCA	Loss-of-Coolant Accident
LWR	Light Water Reactor
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Agency
NII	Nuclear Installations Inspectorate
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Cooperation and Development
PHWR	Pressurized Heavy Water Reactor
PWR	Pressurized Water Reactor
PRA	Probabilistic Risk Assessment
RCS	Reactor Coolant System
SAM	Severe Accident Management
SRM	Staff Requirements Memorandum
SSCs	Systems, Structures, and Components
TMI-2	Three Mile Island Unit 2
U.K.	United Kingdom
U.S.	United States

1 INTRODUCTION

The purpose of this report is to provide an overview of differences in nuclear safety regulatory approaches and requirements between United States (U.S.) and other countries.

In an April 28, 2003 Staff Requirements Memorandum (SRM) [1], resulting from the April 11, 2003 meeting with Advisory Committee on Reactor Safeguards (ACRS), the Commission stated that "In the course of its routine activities of reviewing and advising the Commission on reactor issues, the Committee should explore and consider other international regulatory approaches. Where there are significant differences in regulatory approaches and requirements, the Commission should be informed." This report has been prepared for use by the ACRS in responding to the Commission request.

This report focuses on regulatory requirements pertinent to western-designed light water reactors (LWRs). It does not address requirements relating to nuclear materials and waste safety, or safeguard and security issues.

A number of documents issued by various international organizations, in particular the European Commission and the Organization for Economic Cooperation and Development/ Nuclear Energy Agency (OECD/NEA), were reviewed for the preparation of this report.

The European Commission has sponsored many studies to support its activities toward harmonization of safety requirements and practices in an enlarged European Union. The results of these studies [2-7] on review of safety philosophies and practices in European Union member states were the major source of information for developing this report.

The OECD/NEA reports [8-11] on the scientific and technological background of nuclear safety criteria, rules and guidelines, and applied assessment methods were reviewed to identify the safety issues for which there may not yet be a common technical position among international communities.

The adoption of the Convention on Nuclear Safety in 1994 legally binds the participating countries to maintain a high level of safety. The Convention obliges parties to submit reports on the implementation of their obligations for "peer review" at regular meetings of the parties held by the International Atomic Energy Agency (IAEA). The National Reports on the Convention of Nuclear Safety [12] were also utilized for the preparation of this report.

The report begins with a general overview of regulatory approaches in various countries. It then discusses differences in the specific regulatory requirements in the areas of current interest in the U.S.. They are:

- Design-basis assessment
- Periodic safety reviews
- Protection against severe accidents
- Risk-informed regulations and practices
- Materials degradation issues and aging management

2 GENERAL OVERVIEW OF REGULATORY APPROACHES IN THE WORLD

Regulatory policies differ from country to country. These differences reflect the differences in culture, social, economic, and governmental systems between countries [2]. Regulatory regimes fall broadly within two categories, prescriptive or otherwise. In a prescriptive regime, the requirements on methodologies, standards, and quality assurance are prescribed by the regulatory authority. The licensee must demonstrate that the plant complies with these regulatory requirements. In addition regulatory guidelines, describing methods acceptable to the regulatory authority, may be provided which the licensee can follow for implementing specific portions of regulations. The U.S. Nuclear Regulatory Commission (NRC) regulations fall in this category.

In a less prescriptive regime, the emphasis is on principles which are largely qualitative (except perhaps for certain parameters, e.g., dose limits). The licensee must comply with these principles, but may choose its own methodology of meeting them. European regulations are generally less prescriptive than those in the U.S. There is, however, some degree of variation among the European countries [2].

The interactions between regulator and utility vary from country to country. Generally such interactions are formal with respect to licensing, but less formal from the point of view of safety research. However, there are differences within the licensing relationships. Some regulators encourage a collaborative approach and a continuing dialog through the stages of a licensing application, others adopt a very formal approach [2]. There is a non-adversarial relationship between the plant operator and the regulatory authority in many countries. This is, in part , due to the fact that the plants are owned by the government institutions in these countries.

The safety approaches and practices in the western countries have been largely open to public knowledge and scrutiny. This has encouraged a collaborative safety consciousness over many years. The Eastern European and the former Soviet countries are moving toward more open safety practices. This has been facilitated through the influence of various IAEA, OECD, U.S., and European initiatives.

There is a strong influence of the U.S. regulatory system in setting the basis for licensing requirements in many countries. This is because a large number of plants in operation in other countries are of U.S. design or derived from U.S. designs, which must be licensable in their country of origin. Some countries (e.g., Spain, Holland, and Belgium) completely follow the regulations of the country from which their nuclear power plants were purchased. They follow the U.S. NRC regulations for their Westinghouse pressurized water reactors (PWRs) and the General Electric (GE) boiling water reactors (BWRs), and the German regulations for their Siemens (KWU) plants [2].

Operating reactors by country and by type are presented in Table 1, and Table 2 respectively. The LWR technology was initially developed in the U.S., with GE pioneering the BWRs and Westinghouse developing the PWRs. The main nuclear electricity production in Europe and the Far East, in common with the rest of the world, now comes from LWRs. The exception is the United Kingdom (U.K.) where advanced (oxide fueled) gas-cooled reactors (AGRs) provide a large fraction of the nuclear-generated electricity. The U.K. also subsequently elected to follow the LWR route. The LWR also provides the basic concepts for the WWER reactors developed

Table 1 Operating Reactors in Various Countries[1]

Country	No. of Operational Units				Total MW(e)
	PWR	BWR	Other	Total	
United States Of America	69	35	0	104	98298
France	58	0	FBR 1	59	63363
Japan	23	27	ABWR 3 FBR 1	54	45464
Russian Federation	0	0	LWGR 15 WWER 14 FBR 1	30	20793
United Kingdom	1	0	AGR 14 GCR 12	27	12052
Republic of Korea	15	0	PHWR 4	19	15850
Germany	12	6	0	18	20643
Canada	0	0	PHWR 17	17	12113
India	0	2	PHWR 12	14	2550
Ukraine	0	0	WWER 13	13	11207
Sweden	3	8	0	11	9451
Spain	7	2	0	9	7584
China	7	0	PHWR 2	9	6587
Belgium	7	0	0	7	5760
Taiwan	2	4	0	6	4884
Czech Republic	0	0	WWER 6	6	3548
Slovak Republic	0	0	WWER 6	6	2442
Switzerland	3	2	0	5	3220
Bulgaria	0	0	WWER 4	4	2722
Finland	0	2	WWER 2	4	2656
Hungary	0	0	WWER 4	4	1755
Republic of Lithuania	0	0	LWGR 2	2	2370
Brazil	2	0	0	2	1901
South Africa	2	0	0	2	1800
Mexico	0	2	0	2	1310
Argentina	0	0	PHWR 2	2	935
Pakistan	1	0	PHWR 1	2	425
Slovenia	1	0	0	1	656
Romania	0	0	PHWR 1	1	655
Netherlands	1	0	0	1	449
Armenia	0	0	WWER 1	1	376

[1] Based on information in IAEA (PRIS) database [13], last updated on May 19, 2004

Table 2 Operating and Under Construction Reactors by Type[2]

Type	Operational		Under Construction	
	No. of Units	Total MW(e)	No. of Units	Total MW(e)
PWR	214	204335	6	6111
BWR	90	78025	1	1067
WWER	50	33040	8	7534
PHWR	39	19972	8	3135
LWGR	17	12589	1	925
AGR	14	8380	0	0
GCR	12	2484	0	0
ABWR	3	3955	3	3904
FBR	3	1039	0	0

in Russia and used in other Eastern European countries and Finland.

The LWRs in other countries are quite similar to the designs developed in the U.S.. The French PWRs are very similar to the Westinghouse PWRs since France had bought the license for their design from Westinghouse. The PWRs and BWRs designed by the KWU (Siemens) are similar to the U.S. designs but are different in the configurations of their containments. The BWRs designed by ABB Atom are similar to the Mark-II BWR plants of GE, except, with some modifications (e.g., internal pumps).

The LWRs in other countries, having been commissioned after the U.S. LWRs, were designed and constructed in accordance with the design criteria and safety philosophy developed in the U.S.. The U.S. safety philosophy of defense in depth was adopted by the regulatory authorities in western Europe, Japan, and Korea, not only for the barriers to the release of radioactive substances, but also in the design, construction, quality assurance, inspection, and operational practices. However, there may be differences in the implementation of the defense-in-depth principle, e.g., in levels of diversity and redundancy required from the safety systems. Requirements for three trains of safeguard in France and four trains of safeguard in Germany (because of on-line maintenance) and the requirement for diversity of instrumentation for all safety-related measurements in Germany are examples of such differences in implementation of the defense-in-depth principle. There are also some country-

[2] Based on information in IAEA (PRIS) database [13], last updated on May 19, 2004

specific regulatory requirements regarding the effectiveness of the various barriers. One such example is the requirement in Germany to design the containment to withstand the crash of a light fighter plane.

The 1979 accident at Three Mile Island Unit 2 (TMI-2), led to the reexamination of the design basis and the consideration of regulations for protection against severe accidents. The reexamination of the design basis was prompted by the fact that the TMI-2 accident initiated with a small-break loss -of-coolant accident (LOCA), whose consequences should have been bounded by those of a large-break LOCA, but became much more severe due to misunderstanding of the event by the operators. The event-based procedures have been modified to symptom-based procedures in most Western plants [2].

The Chernobyl accident in 1986 affected opinion in Western Europe about the safety of nuclear power plants in general, contributing to the decisions of some countries (e.g., Germany) to tighten safety requirements for new plants, implying design modifications, or to phase out nuclear power stations, either immediately (e.g., Italy), or over a period of time (e.g., Sweden, Germany).

In most countries, the principles of traditional deterministic approach have been accepted over many years to demonstrate the reliability and safety of design. (deterministic approach refers to an approach that specifies certain design and operational conditions and applies bounding criteria to demonstrate acceptable plant performance.) Systems, structures, and components (SSCs) are designed and manufactured to accepted standards, regulations, codes of practice etc. to ensure that the SSCs can perform their intended functions. The single-failure criterion has been commonly adopted, as has the 30 minutes rule, i.e., that the safety objectives can be met without operator intervention within the first 30 minutes into an accident.

The majority of licensing submittals have been based on the evaluation model (EM) methodology. This was established on the premise that deliberate modeling conservatisms are included to compensate for lack of knowledge of the governing phenomena. This methodology was based on the Appendix K of the U.S. Code of Federal Regulations (10CFR Part 50). However, with improved understanding of the phenomena, there have been moves to change the conservative biases and assumptions of the evaluation model methodology, allowing the licensee to move further toward best-estimate methodologies. Within the U.S. this led to a revision of the emergency core cooling system (ECCS) rule (10CFR 50.46) in 1988 enabling licensees to apply best-estimate methodologies, with the provision that due allowance is given to any remaining uncertainties in code, data, or modeling. The move toward best-estimate methodologies is also a common trend in most countries [2].

In light of increased realization of the impact of human factors on plant safety, regulatory authorities now require the utilities to consider human factor engineering concepts in the design and operational aspects of plants. There is an international recognition of the importance of safety culture and management. There is an evolving consensus on what constitutes good performance on the part of an organization but less on how it can be measured.

The ALARA (or ALARP) principles are generally adopted to ensure that risks are reduced to a level acceptable to the regulatory body to be “as low as reasonably achievable (or practicable).” Most countries follow this approach in qualitative terms. In principle risk may be quantified via a cost-benefit analysis, whereby the costs to industry are compared with the benefits to society. The extent to which cost-benefit analysis is encouraged or allowed varies from country to country. In most European countries, safety improvements are generally introduced without the requirement for formal cost-benefit analysis [2]. Nevertheless, cost/benefit is informally considered by regulatory authorities in all of these countries. The issue of cost may become more important as competition grows in Europe.

Basic deterministic safety assessments are now generally complemented by probabilistic risk assessments (PRAs) to verify the overall design and system of operation. PRAs are conducted by many countries to demonstrate that there are no sudden increase in risk for accidents that are outside of the design basis. Most countries with nuclear power plants have performed PRAs and have found that such assessments often lead to the identification of plant vulnerabilities. However, there is not much support, so far, in many other countries for formally considering risk information in regulatory decisionmaking as it is in the U.S..

3 DIFFERENCES IN REGULATORY REQUIREMENTS

Despite considerable similarities in the objectives and actual implementation of nuclear safety regulatory approaches, there are differences in regulatory requirements across the world. Indeed, efforts to harmonize safety requirements and regulatory practices within the European Union have been unsuccessful so far.

Reasons for the differences in regulatory requirements relate to national energy policy (mainly in support of public acceptance); national industrial tradition (e.g., giving more credit to redundancy or diversity, or crediting a software-based system as opposed to hard-wired controls); consistency with national regulatory or legislative system (e.g., compliance with probabilistic safety criteria on individual and societal risk as applicable to the environmental policy); country-specific conditions (e.g., differences in geography such as flooding for Netherlands and seismic for Japan); and uncertainties associated with the severe-accident phenomena.

Some of the areas where differences in safety requirements exist are discussed below.

3.1 Design-Basis Assessment

There is an internationally accepted rule that the licensee should provide a comprehensive safety assessment to confirm that the design of an installation fulfils the safety objectives and requirements. This assessment is submitted in a safety analysis report. Specific approval by the regulatory body is required before the start of operation. The U.S. NRC regulation (10CFR50.71) requires the licensee to update periodically (the interval between updates should not exceed 24 months) the final safety analysis report (FSAR) originally submitted as part of the application for the operating license. The update should include the effects of: all changes made in the facility or procedures as described in the FSAR; all safety analyses and evaluations performed by the licensee in support of approved license amendments, and all analyses of new safety issues performed by or on behalf of licensee at Commission request. In many other countries, a safety analysis report is updated every 10 years as a part of periodic safety reviews (see section 3.2). These reviews must take account of existing operational experience and any other information relevant to safety that is currently available.

The accident sequence groups and the accidents to be analyzed in the safety analysis report may be prescribed by the regulator (e.g., U.S. NRC), but if not, are defined by the licensee as part of its safety case submission (e.g., United Kingdom). The implementation of either approach is similar. There are, however, some differences in certain acceptance criteria and the licensing calculations due to various degree of conservatism made at each step of the calculation. Some of these differences are summarized below.

<u>3.1.1 Acceptance Criteria for Emergency Core Cooling System</u>

Most countries use acceptance criteria for ECCS that are based on those specified in Appendix K to 10CFR Part 50. Germany has also established an additional acceptance criterion to limit the fraction of failed fuel clad under LOCA conditions. The 10% fuel failure criterion in Germany was originally established to limit the radiological consequences in case of a LOCA (see section 3.1.2). The original intention of this criterion has since been broadened. Beside the radiological aspects, this criterion has been used for the evaluation of core loading. If the core is loaded with new fuel rods or new loading strategies are applied, the compliance with the 10% fuel failure criterion has to be demonstrated again by the applicant [6].

There is a common understanding among the German licensing authorities that if the compliance with all these acceptance criteria can be proven, there is no need to limit the fuel burn-up or to restrict the core loading[6].

3.1.2 Extent of Fuel Failures that is Assumed in Radiological Assessment

The extent of fuel failure that is assumed in radiological assessments varies from country to country. In performing the design-basis accident analyses, the commonly applied practice includes the use of conservative assumptions regarding system performance and components failure. Following a large LOCA, it is assumed that a fraction of the fuel is failed allowing release of the radionuclides from the fuel into the containment atmosphere. This release of fission products into the containment ("in-containment source term") has a wide range of regulatory applications, including the basis for (1) the adequacy of the leaktightness of the containment, (2) the performance requirement of fission-product cleanup systems such as sprays and filters, (3) post-accident habitability requirements for the control room, and (4) the radiation environment for qualification of safety-related equipments.

The determination of source term inside the containment involves assumptions corresponding to to various physical stages in the release of fission products, including fraction of core failure, release from damaged fuel, airborne part of release and release into reactor coolant system and sumps, chemical behavior of iodine in the aqueous and gas phases, and natural and spray removal in the containment atmosphere.

Some countries (Belgium and Spain) follow the U.S. and assume a source term corresponding more to a core-melt accident decoupled from the LOCA thermal-hydraulic calculations, while other countries take into account the physical phenomena during a LOCA still with conservative assumptions.

Table 3 shows the extent of fuel failure that is assumed in radiological assessments in different countries. Many countries (e.g., Belgium, United Kingdom, Spain) follow the U.S. and assume 100% fuel failures during a large LOCA. Some European countries (e.g., Germany, Switzerland, Netherlands) assume 10% of fuel failure during a LOCA. In France, a 100% fuel failure assumption is used for the radiological consequences evaluation of the 900 and 1300 MWe plants. However, for N4 plants, a 33% fuel failure assumption has been proposed by the utility and is under assessment by the regulatory body (IPSN). The utility position is that this value is sufficiently conservative to constitute a decoupling assumption avoiding a specific safety demonstration for each core refueling, taking into account a previous Framatome study for the 1300 MWe French nuclear power plant design for which 7% of clad failure was predicted [6].

Table 3 The Extent of Fuel Failure That Is Assumed In Radiological Assessments

Country	Extent of Fuel Failures in Radiological Assessment
Belgium	100%
France	100% (33% proposed for N4 plants)
Germany	10%
Netherlands	10%
Spain	100%
Switzerland	10%
United Kingdom	100%
United States	100%

3.1.3 Strainer Blockage Issue

The 1992 clogging of intake strainers for containment spray water in Barsebäck-2, a BWR in Sweden, renewed the focus of regulators around the world on safety questions associated with strainer clogging which, until then, had been considered as resolved.

Although the Barsebäck incident in itself was not very serious, it revealed a weakness in the implementation of defense-in-depth concept in the design, which under other circumstances could have led to the failure of the ECCS and containment spray system (CSS). The Barsebäck-2 event also demonstrated that larger quantities of fibrous debris could reach the strainers than had been predicted by models and analysis methods developed for the resolution of the strainer blockage issue[8,14].

The Barsebäck-2 incident prompted action on the part of regulators and utilities in other countries. Research and development efforts of varying intensity were launched in many countries. Extensive studies have been performed to assess the amount of insulation materials that could be dislodged during pipe break events inside the containment. In many countries, the analyses were based on the double cone model developed by the NRC [14]. The analyses have also included specific studies of the transport of insulation materials and other debris in the containment, and of strainer pressure drops. Such efforts resulted in a number of corrective actions being taken in BWRs and some PWRs around the world. For a number of plants, actions were taken as direct responses to requirements issued by regulatory authorities, while for other plants back-fitting measures were introduced voluntarily or because of anticipated requirements [9].

The modifications of the ECCS and/or CSS suction strainers carried out in different countries are summarized in Tables 4 and 5 for BWRs and PWRs respectively. The modifications have resulted in new strainer designs with significantly enlarged filtering area. Most of the new strainers have good self-cleaning properties. In some BWRs, the

Table 4 Summary of the BWR Strainer Modifications in Different Countries After the Barsebäck-2 Event

Country	BWR Strainer Modifications	Comment
United States of America	34 out of 34 units modified	New strainers with significantly increased area Established a schedule to remove particulate and other debris from the suppression pool
Japan	None of 28 units modified	More than 95% of the insulations are replaced by non-fiber type ones (e.g., Reflective Metallic Insulation). The absence of foreign materials in the suppression pool is ensured through inspection and maintenance practices
Sweden	9 out of 9 units modified	New strainers with 15 to 40-fold area increase
Germany	4 out of 6 units modified	Stainers were enlarged
Spain	2 out of 2 units modified	New strainers with significant area increase
Switzerland	2 out of 2 units modified	New strainers with 7 to 30-fold area increase
Finland	2 out of 2 units modified	About 10-fold area increase

Table 5 Summary of the PWR Stainer Modifications in Different Countries
After the Barsebäck-2 Event

Country	PWR Strainer Modifications	Comment
United States of America	One out of 69 units modified	Davis-Besse is the only U.S. PWR plant that its licensee voluntarily enlarged its sump screen area - by 30 fold
France	EDF plans Backfits to sump strainers at all 58 units	EDF is finishing design studies for the backfit, which is expected to consist of replacing the current filters with a system of pierced piping that provides more strainer surface.
Japan	None of 23 units modified	More than 95% of the insulations are replaced by non-fiber type ones (e.g., Reflective Metallic Insulation). The absence of foreign materials in the recirculation sumps is ensured through inspection and maintenance practices.
Germany	2 out of 12 units modified	Stainers were enlarged.
Spain	None of 7 units modified	
Belgium	2 out of 7 units modified	6-fold area increase
Switzerland	None of 3 units modified	
Sweden	1 out of 3 units modified	New strainers with > 7-fold area increase
Netherlands	1 out of 1 unit modified	New strainer installed (50% area increase) There is no mineral or fiberglass insulation of noteworthy importance around the primary components as well as in the sump area.
Finland	2 out of 2 WWER-440/213 units modified	New strainer design with significant area increase

design includes the capability to back-flush the strainers [9].

Large fractions of the thermal insulation materials utilized on piping and other components inside the containment have also been replaced. The newly installed insulation materials vary both within and among countries. They are primarily reflective metallic insulation, nuclear grade fiberglass, mineral wool, and calcium silicate. The same insulation material (e.g., mineral wool) are installed differently in different countries (i.e., Jacketed or encapsulated in cassettes). The administrative measures taken in other countries include a periodic cleanup of the suppression pool and containment sumps, with the aim to minimize the presence of foreign materials, and the control and eventual improvement of the containment coating.

In U.S., all BWR licensees were required to implement appropriate measures to ensure the capability of the ECCS to perform its safety function following a LOCA. The U.S. nuclear industry addressed the NRC requirements by installing large capacity passive strainers in each BWR plant and establishing a schedule to remove particulate and other debris from the suppression pools. Most U.S. BWR licensees followed the guidance prepared by the U.S. BWR Owners Group during the development of their corrective actions.

As a result of research findings related to resolving the BWR ECCS strainer blockage safety issue, the NRC conducted further research to determine if the transport and accumulation of debris in a containment following a LOCA would impede the operation of the ECCS in operating PWRs. The research program included debris transport tests, debris settling tests, debris generation tests, computational simulations, and various engineering analyses. The results of these studies indicated the need for accurate plant-specific assessment of adequacy of the recirculation function of the ECCS and CSS for each operating PWR. The Nuclear Energy Institute (NEI) also recognized this need and has developed guidance for such plant-specific assessment, which is under review by the NRC staff.

The issue of strainer blockage in PWRs have been particularly troublesome. Continuing research revealing new modes of blockage has shown that the prompt actions taken by some European plants may not have completely alleviated the problem of strainer blockage. Indeed, redesign may be required of these plants. There is a strong evidence that plant owners throughout the world do not have a definitive solution to the issue.

3.2 The Periodic Safety Reviews

In contrast to U.S. NRC, most regulatory authorities in the world have a requirement that the nuclear power plants be subject to an overall assessment on a periodic basis, in addition to the permanent supervision the regulatory body exerts on these plants. Table 6 presents a comparison of international practices with respect to periodic safety review activities.

The periodic safety review is a safety concept mainly developed in the European countries and was introduced later in the IAEA documents [16]. The periodic safety reviews are complementary to the routine reviews of nuclear power plant operation (including modifications to hardware and procedures, significant events, and operating experience) and special safety reviews following major events of risk significance. The frequency of review varies from country to country; typically every ten years (see Table 6). The periodic safety review necessitates licensees to take into account advances in technology unconstrained by licensing basis as in U.S..

Table 6 Periodic Safety Review Requirements in Various Countries

Country	Periodic Safety Review Frequency	Comment
United States Of America	None Required	Requiring the licensee to maintain the licensing basis for the facility or activity
France	Every 10 years (normally)	Linked to the statutory 10 year outage program interval, but the in-depth safety assessment performed on request , at the regulator's discretion
Japan	Every 10 years	Limited scope, concentrated mainly on aging behavior, without the evaluation of the overall plant design
United Kingdom	Every 10 years	Comprehensive safety reviews Requirements stipulated in conditions attached to licenses
Republic of Korea	Every 10 years	Comprehensive safety reviews
Germany	Every 10 years	Comprehensive safety reviews
Canada	None required	License renewed every 2-5 years subject to satisfactory safety performance
Sweden	Every 10 years	Comprehensive safety reviews
Spain	Every 10 years	Comprehensive safety reviews
Belgium	Every 10 years	Comprehensive safety reviews
Czech Republic	Every 10 years	Comprehensive safety reviews
Switzerland	Every 10 years	Comprehensive safety reviews (Regulatory requirement for facilities to comply with the state-of-the-art in science and technology)
Finland	Every 10 years	Comprehensive safety reviews
Hungary	Every 10 years	Comprehensive safety reviews
Mexico	Every 10 years	Comprehensive safety reviews
Netherlands	Every 10 years	Comprehensive safety reviews

The objective of these periodic safety reviews are to assess the cumulative effects of plant aging and plant modifications, operating experience, technical developments, and siting aspects. The reviews include an assessment of plant design and operation against current safety standards and practices in order to propose any eventual improvement. The reviews also examine an extension of the original design basis of the plant, in particular postulated initiating events (internal and external) not considered earlier. The reassessment of the original design basis in Europe is strongly linked to the research on severe accidents and their management strategies [3].

Deterministic safety analyses are used in safety reassessments made in the periodic safety reviews. However, it is now common to complement the deterministic analyses with a PRA (level 1 or 2), in particular to determine the modifications that significantly improve the safety [3].

3.3 Protection Against Severe Accidents

The desire for protection against severe accidents is shared by all of the regulatory authorities in the Western World. It has also been argued that the severe accident is a very low probability event; it deserves a response, but the cost/benefit should be a factor. This argument has been accepted by the U.S. NRC and it is a part of the regulatory practice (backfit rule, 10CFR50.109). However, most regulatory authorities of the European Union Member States do not formally accept this argument. Nevertheless, cost/benefit is silently considered by all of these authorities[2].

The first significant regulatory action for severe accident mitigation was the hydrogen rule (10CFR50.44) issued by U.S. NRC soon after the TMI-2 accident. The rule required control of the hydrogen that is produced in a severe accident. Decisions were made to inert the BWR Mark-I and Mark-II containments and install igniters for hydrogen control in BWR Mark-III and the ice condenser containments. The PWR plants with large dry containments (including those operating with a sub-atmospheric internal pressure) were exempted from hydrogen control, because of the large volume of their containments. Regarding hydrogen control, the BWR Mark-I and Mark-II plants in European countries followed suit in inerting containment atmosphere. The PWRs in Europe have gone through a long evaluation process and most of them (except the Westinghouse-designed plants) have decided to install catalytic hydrogen recombiners of sufficient capacity to address severe accident hydrogen production [2].

The phenomenology of the severe accident is extremely complicated. The severe accident evaluation methodologies are associated with large uncertainties. In fact, such uncertainties have led different parties to reach to different conclusions from research results obtained for several severe accident phenomena. For example, there is a large uncertainty associated with the coolability of a melt/debris attacking the concrete basemat, by flooding with water. This has introduced different approaches for severe accident management strategies. For example, U.K., Spain, Belgium, Sweden, and Finland will add water to their PWR cavities and their BWR lower dry-wells in order to fragment the melt, to facilitate its cooling, and possibly delay the basemat melt-through. On the other hand, the Germans do not have either the facility, or the desire to add water to their PWR cavities in order to avoid the possibility of steam explosions.

The European plant owners, with the encouragement of the regulatory authorities, have developed severe accident management measures. An excellent example is the containment filtered vent, which has been installed in the Swedish BWRs. Containment vents are being considered for installation in several European BWRs and PWRs [2]. Sand filters have been installed in French PWRs, as a backfit. The U.S. plants on the other hand, have not been partial to containment venting. Hard vents are being installed in U.S. BWRs with Mark-I containments, but no U.S. PWR is installing a filtered vent system. Correspondingly, some of the Westinghous-designed plants in Europe (e.g., in U.K., Spain, Belgium) are not considering installation of vents on their containments [2].

A severe accident management measure of very wide acceptance by the PWR plants in Europe and in the U.S. is that of reactor coolant system (RCS) depressurization in the event of a severe accident, in order to avoid the potential for early failure of the containment by direct containment heating (DCH). RCS depressurization was included in the design of the U.K. PWR, primary for accident prevention, and has been introduced in French PWRs as a backfit following high pressure melt ejection and DCH studies. This severe accident management measure, however, cannot be accomplished on some plants whose safety valves do not have sufficient relief capacity.

Another example of severe accident management measures is that of cooling the vessel from outside in order to retain the core debris inside the vessel. With the reactor intact and debris retained in the lower head, phenomena such as ex-vessel steam explosion and core-concrete interaction, which occur as a result of core debris relocation to the reactor cavity, could be prevented. This is the so-called severe accident management strategy of In-Vessel Melt Retention which has been approved for the Loviisa plant in Finland and has been incorporated in the design of the AP600 and AP1000 passive plants. Reactor vessel integrity is assumed if RCS is depressurized and the cavity adequately flooded. Cooperative, international research programs, RASPLAV and MASCA are producing results that suggest this approach may not work for plants with power densities higher than that in the Loviisa plant.

Future reactors are expected to have greater provision against severe accidents. The extension of the design to cover severe accidents, as proposed in Germany and being adopted by the French, would represent a significant departure from currently accepted safety practices in many countries. Whether such an objective becomes a regulatory requirement or not in a particular country will clearly have a major impact on different national approaches to safety [2].

In Europe, there is now a desire to extend the design basis to deal specifically with severe accidents, but the ways to achieve this have not been defined. Much of the current capability for severe accident mitigation arises from the strength of the containment. However, if (some) severe accidents are to be included in the design basis, there is a case for the containment to be designed for higher loads, possibly with a smaller safety margin. This is an area in which standards have yet to emerge, although current documents imply that “best estimate” should be sufficient for severe accident assessments [2].

Inclusion of severe accidents in the design basis poses technical challenges in other areas, such as steam explosion assessments. There are questions on how conservative should the loading be, or whether it is possible to show that the “design loading” is always conservative. Currently this is an area where probabilistic arguments, supported by deterministic analyses, have been accepted [2]. Current proposal for new reactor systems have focused on "evolutionary designs". These designs are essentially modifications to existing LWRs, usually with some advanced safety features. Examples of such safety features are more passive systems, the use of ex-vessel flooding in AP600 and AP1000, and the provision of a debris retention device in the European Pressurized Water Reactor (EPR) design and the WWER-1000 reactors currently under construction in China. However, there is no general agreement on what additional features should be included in future designs, what the design basis should be, and how the improvements to safety should be quantified [2].

3.4 Risk-informed Regulations and Practices

The U.S. NRC has led the development of the quantitative risk analysis for nuclear power plants. Though PRAs have been used extensively in the past, they were usually limited to a variety of applications on a case by case basis as deemed necessary or useful. The NRC is now moving toward a much expanded use of PRAs in what is termed risk-informed regulatory approach. In 1995, the NRC adopted a policy that promotes increasing the use of probabilistic risk analysis in all regulatory matters to the extent supported by the state-of-the-art to complement the deterministic approach. The current regulatory framework is based largely, but not entirely on a deterministic approach that employs safety margins, operating experience, accident analyses, and probabilistic assessment of the risk, and relies on a defense-in-depth philosophy.

The NRC has applied information gained from PRAs extensively to complement other engineering analyses in improving issue-specific safety regulation, and in changing the current licensing bases for individual plants. Using risk insights, the NRC has modified its oversight process and its requirements for maintenance (10CFR 50.65). The NRC is considering further revisions to its reactor regulations (10CFR Part 50) to focus requirements on programs and activities that are most risk significant. However, these revisions would provide alternatives, that are strictly voluntary, to current requirements. The agency is also considering changes to 10 CFR Part 50 that could lead to incorporating a new set of design-basis accidents, revising specific requirements to reflect risk-informed considerations, or deleting certain regulations. The main driving force behind the move toward risk-informing the current regulations and processes is the expectation that the use of risk insights can result in both improved safety and a reduction in unnecessary regulatory requirements, hence allowing both the NRC and licensees to focus resources on equipment and activities that have the greatest risk significance.

Within Europe, deterministic safety assessments are now often complemented by PRAs to verify the overall design and system of operation. An example is that of specifying as a safety target (goal) the core melt frequency of 10^{-5} or 10^{-6} and the conditional probability of containment failure of 10^{-1} or 10^{-2}. The use of safety evaluation based on probabilistic arguments is, so far, confined to resolution of severe accident safety issues. This trend is not uniform across the European Union. For Example, the German regulatory and technical support organization views of PRA are not as favorable as those of the comparable Spanish organizations[2].

There is not much support, so far, in Europe for formally considering risk-informed regulations and practices, as it is in the U.S.. The exception is the U.K. where the current Nuclear Installations Inspectorate (NII) licensing guidelines adopt a risk-based approach. However, most regulatory authorities in Europe declare that they consider risk information informally [2].

The aim of the most European regulatory authorities is to improve safety, not just to maintain it. Therefore, they encourage the development and the use of PRAs for improving safety, and not for reducing regulatory requirements. There is a considerable reluctance to use the results of quantitative risk assessment to reduce regulatory requirements regardless of the calculated risk significance of these requirements. Uncertainty in the quantitative results, concern over the completeness of the analyses, and lack of properly dealing with organizational or safety culture issues are usually cited as the bases for this reluctance [17].

3.5 Materials Degradation Issues and Aging Management

Safe control of aging of nuclear power plants is an important concern for plant owners and safety regulatory authorities in the world. The optimal ageing management of nuclear power plants require knowledge on materials degradation phenomena and evaluation techniques.

Contrary to U.S., there is no expiration time for the operating license in many countries (see Table 7). The periodic safety review (typically every ten years) is the principal method applied to reactors to ensure that the plant is adequately safe for a further period of operation. However, according to different countries, the operating authorization given by the regulatory authority to the plant operator is not associated with the same formal process. Formal aging management evaluation processes exist in some countries, for quite short periods (i.e., one year in Spain, two in the U.K.); in others, it appears through a requirement of ability for safety demonstration at any moment (in France and Belgium). In practice, safety aging management is implemented through the periodic safety review approach, widely accepted in many countries (see section 3.2).

The material degradation issues have been the subject of numerous studies in different countries and by several international organizations [4]. These studies have led to the establishment of various programs or projects specifically dedicated to the management of aging of SSCs.

Aging management begins with plant design. Many design criteria explicitly or implicitly address aging. The "long-lived" SSCs in a nuclear plant, for example, were originally designed with sufficient margins to meet minimum lifetime requirements. Current aging management programs aim essentially at managing the gradual degradation of SSCs as a result of their physical aging in order to ensure permanently satisfying the safety criteria.

The various aging aspects leading to slow degradation of SSCs are evaluated during periodic safety assessment. However, aspects related to more quick changes (in particular those affecting active components) are managed on a continuos basis through an appropriate maintenance and component qualification.

In the U.S., the original plant life is established by the regulatory process. The Atomic Energy Act and NRC regulations limit the initial operating licenses of nuclear power plants to 40 years, but also permit such licenses to be renewed. The original 40-year term was selected on the basis of economic and antitrust considerations, rather than by technical limitations. However, the selection of this term may have resulted in individual plants being designed on the basis of an expected 40-year service life. 10 CFR Part 54, known as the "license renewal rule," establishes the technical and procedural requirements for renewing operating licenses. Under the license renewal rule, the applicant must perform a screening review of all SSCs within the scope of the rule to identify "passive" and "long-lived" structures and components. The applicant must demonstrate that it will manage the effects of aging such that the SSCs will function as intended throughout the 20-year period of extended operation.

Table 7 Operating License Periods in Various Countries

Country	License Period Approach
United States Of America	Fixed term (40 years, with 20-year renewal option)
France	Lifetime
Japan	Lifetime
United Kingdom	Lifetime
Republic of Korea	Lifetime
Germany	Lifetime
Canada	Fixed term (2-5 years)
Sweden	Lifetime
Spain	Variable (5-10 years) Case-by-case, no fixed term but moving to 10-year standard for nuclear facilities that complete periodic safety reviews
Belgium	Lifetime
Czech Republic	Lifetime
Switzerland	Lifetime (except for 2 plants with term licenses based on historical technical concerns)
Finland	Fixed term (10-20 years)
Hungary	Lifetime
Mexico	Fixed term (30 years)
Netherlands	Lifetime

4 SUMMARY AND CONCLUSIONS

Despite considerable similarities in the objectives and actual implementation of nuclear safety regulatory approaches, there are differences in nuclear safety regulatory requirements between the United States and other countries.

There is a strong influence of the U.S. regulatory system on setting the basis for licensing requirements in many countries. This is because a large number of plants in operation in other countries are of U.S. design or derived from U.S. designs. The U.S. safety philosophy of defense in depth was adopted by the regulatory authorities in Western Europe, Japan, and Korea, not only for the barriers to the release of radioactive substances, but also in the design, construction, quality assurance, inspection, and operational practices. However, there may be differences in the implementation of the defense in depth principle, e.g., in levels of diversity and redundancy required from the safety systems.

In most countries, the principles of traditional deterministic approach have been accepted over many years to demonstrate the reliability and safety of design. Systems, structures, and components are designed and manufactured to accepted standards, regulation, codes of practice etc. to ensure that the SSCs can perform their intended functions.

There is an internationally accepted rule that the licensee should provide a comprehensive safety assessment to confirm that the design of an installation fulfils the safety objectives and requirements. The accident sequence groups and the accidents to be analyzed in the safety analysis report may be prescribed by the regulator (e.g., U.S. NRC), but if not, are defined by the licensee as part of his safety case submission (e.g., U.K.). The implementation of either approach is similar. There are, however, some differences in certain acceptance criteria and the licensing calculations due to various degree of conservatism made at each step of the calculation. Some of these differences were discussed in this report.

Basic deterministic safety assessments are now generally complemented by PRAs to verify the overall design and system of operation. However, there is not much support, so far, in many other countries for formally considering risk information in regulatory decisionmaking as it is in the U.S..

The desire for protection against severe accidents is shared by all of the regulatory authorities in the Western World. It has also been argued that the severe accident is a very low-probability event; it deserves a response, but the cost/benefit should be a factor. This argument has been accepted by the U.S. NRC and it is a part of the regulatory practice (backfit rule, 10CFR50.109). Most regulatory authorities of the European Union Member States do not formally accept this argument.

The Barsebäck-2 incident prompted a number of corrective actions being taken in BWRs and some PWRs around the world. Actions were taken as direct responses to requirements issued by regulatory authorities for many plants, while for other plants back-fitting measures were introduced voluntarily or because of anticipated requirements. The issue of strainer blockage in PWRs have been particularly troublesome. Continuing research revealing new modes of blockage has shown that the prompt actions taken by some European plants may not have completely alleviated the problem of strainer blockage. Indeed, redesign may be required of these plants. There is a strong evidence that plant operators throughout the world do not have a definitive solution to the issue.

In Europe, there is now a desire to extend the design basis to deal specifically with severe accidents, but the ways to achieve this have not been agreed. Future reactors are expected to have greater provision against severe accidents. The extension of the design to cover severe accidents, as proposed in Germany and being adopted by the French, would represent a significant departure from currently accepted safety practices in many countries. Whether such an objective becomes a regulatory requirement or not in a particular country will clearly have a major impact on different national approaches to safety.

Contrary to U.S., there is no expiration time for the operating license in many countries. The periodic safety review (typically every ten years) is the principal method applied to reactors to ensure that the plant is adequately safe for a further period of operation. However, according to different countries, the operating authorization given by the regulatory authority to the plant operator is not associated with the same formal process. Formal aging management evaluation processes exist in some countries, for quiet short periods; in others, it appears through a requirement of ability for safety demonstration at any moment.

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