Transactions of the 15th International Conference on
Structural Mechanics in Reactor Technology (SMiRT-15)
Seoul, Korea, August 15-20, 1999 S02/6

Development Status on the Safety and Regulatory Requirements and Guidance for the Korea Next Generation Reactors

J-H. Lee[1], W.S. Kim[1], Y.G. Yune[1], J-S. Lee[1] and H.S. Chang[2]

1) Korea Institute of Nuclear Safety, Korea
2) Ministry of Science and Technology, Korea

ABSTRACT

As a part of the Korea Next Generation Reactors (KNGR) Development Program, the regulatory side has carried out the Safety and Regulatory Requirements and Guidance (SRRG) development program from 1992. The SRRG development program has been performed in the following three categories: establishment of SRRG hierarchy, development of technical requirements and guidance, and consideration of new licensing system. A comprehensive and systematic SRRG hierarchy has been established which consists of six tiers from Safety Objectives to Safety Review Procedures. Drafts of technical requirements and guidance for the KNGR have been developed nearly up to 100%. As a complementary measure to the current licensing system, new processes such as Pre-application Safety Review and Standard Design Approval have been reviewed.

I. Introduction

Based on the lessons learned from TMI and Chernobyl accidents, some leading countries in the nuclear industry try to develop advanced nuclear power plants (NPP). The major goal of the development has been focused on the enhancement of safety level of the NPP, which will contribute to assuring the safety of the NPP and increasing the public acceptance of the NPP.
To keep abreast with this, the Korea Next Generation Reactors (KNGR) development program was launched by both government and utility in 1992. Its final goal is to complete the design of the KNGR on utility side, and establishment of regulatory positions, i.e. safety and regulatory requirements and guidance (SRRG), for KNGR licensing and regulation on the regulatory side until December of 2001.
The Korea Institute of Nuclear Safety (KINS) has performed the SRRG development program, which consists of three categories: establishment of a comprehensive and systematic SRRG framework, development of technical requirements and guidance for safety enhancement and reasonable regulation, and improvement of licensing system for effective regulation.
This paper introduces the overall aspects of the development of the SRRG, especially focusing on the development of General Safety Criteria review on the new licensing system.

II. Development of the SRRG Hierarchy

The basic approach to the development of the SRRG for the KNGR is to reflect operating experiences obtained from the existing plants and adopt enhanced safety concepts for improving safety level of the KNGR. In order to identify candidate elements of the requirements and guidance to be taken into account in the SRRG, an in-depth study has been performed based on the following information: safety requirements and guidance for future reactors established by foreign regulatory bodies and the IAEA, design requirements and features considered by utility to enhance the safety of advanced reactors, and safety issues raised by regulatory bodies during licensing review.

From this study, a systematic SRRG hierarchy has been established, which consists of the following six tiers: Safety Objectives, Safety Principles, General Safety Criteria, Specific Safety Requirements, Safety Guides, and Safety Review Procedures. Table 1 represents comparison of the tiers of the SRRG with regulatory documents of domestic and foreign countries.

Table 1. Comparison of Level of Each Tier with Other Regulatory Documents

SRRG for KNGR	Korea	USA	Japan	France	Germany	IAEA
Safety Objectives And Principles	AEA, Enforcement Decree	AEA, Policy Statements	AEA, Law of Rx	Loi, Decree	AEA	Safety Fundamentals
General Safety Criteria (GSC)	Enforcement Decree	10 CFR 50, App. A,GDC	MITI Orders, NSC Guides	Decree, Arrete	BMI Announ-cement	NUSS, Codes
Specific Safety Requirements (SSR)	MOST Notice	10 CFR (Main Text)	NSC Guides (MITI Note)	RFS, RCC-series	RSK Guidelines (KTA Rule)	NUSS, Safety Guides
Safety Guides (SG)		Regulatory Guides	NSC Guides	Circulaire (Instruction)	RSK Guidelines	NUSS, Safety Practices
Safety Review Procedures (SRP)		SRP	NSC Guides		RSK Guidelines	-

AEA: Atomic Energy Act; CFR: Code of Federal Regulation; BMI: NPP Safety Criteria
NSC: Nuclear Safety Commission; GDC: General Design Criteria; RFS: Basic Safety Rules
RSK: Safety Guidelines for NPP; RCC: Design and Construction Rule; KTA: Nuclear Safety Standards
SRP: Standard Review Plan; MITI: Ministry of Trade and Industry; MOST: Ministry of Science and Technology

1. Safety Objectives

The Safety Objectives are the top-tier of the SRRG hierarchy and provide an ultimate aim which has to be achieved for assuring the safety of the KNGR. They consist of three components: general safety objective, radiation protection objective, and technical safety objective. In particular, the technical safety objective specifies that radiological impacts be minimized under all severe accidents addressed in design as well as design basis accidents, and the likelihood of any severe accidents that would result in serious radiological consequences be ensured to be extremely low. It is required to enhance the safety level of the KNGR and to apply the latest advanced technology.

2. Safety Principles (SP)

The Safety Principles are essential elements which have to be observed in achieving the Safety Objectives and provide fundamental philosophies for the development of the SRRG. The SP consist of four areas: general SP, siting-related SP, design-related SP, and operation-related SP. The general SP deal with principles which must be addressed commonly in all licensing phases of the NPP from siting to decommissioning. Total of 13 SP, including reflection of operating experience, use of proven technology, strengthening of defense-in-depth concept, and consideration of human factors, have been established.

3. General Safety Criteria (GSC)

The GSC provide fundamental criteria to be complied with in implementing the Safety Principles in general aspects. They are constituted by criteria that have been established based on general and implicit concepts for ensuring the safety of NPP and provide fundamental bases for establishment of the Specific Safety Requirements. They are categorized into five areas: siting/environment, design, operation, decommissioning, and quality assurance. Comparing with the regulatory framework of the USA, the GSC correspond to General Design Criteria in the Appendix A of 10 CFR Part 50. However, they additionally provide general criteria in siting/environment, operation, decommissioning, and quality assurance, which are not established in the Appendix A to 10 CFR Part 50. Total of 61 items has been developed and the contents and characteristics of the GSC will be discussed in detail in the next chapter.

4. Specific Safety Requirements (SSR)

As the last tier of mandatory requirements in the SRRG hierarchy, the SSR provide designers with detailed guidelines necessary for their design activities and also present specific rules for determining the safety of the KNGR to regulators in licensing process. Thus, the SSR are strongly linked with Safety Guides and industrial codes & standards, where necessary. The SSR are composed of 23 chapters, which represent site, reactor, reactor coolant systems, engineering safety systems, instrumentation and control system, severe accident assessment, operational requirements and so on, under the same five categories as those of the GSC. Table 2 shows the SSR chapters and nearly 100% of draft SSR in each chapter have been developed until now.

5. Safety Guides and Safety Review Procedures

The Safety Guides provide acceptable methods or specifications, which are applied to implement the mandatory requirements. About 147 items have been identified as candidate Safety Guides and 90% of them has been developed. The Safety Review Procedures represent an internal guidance of the KINS on the safety review of the KNGR. They provide the reviewer with review scope, technical rationales, relevant requirements and guides, review method, and etc.

Table 2. Structure of the Specific Safety Requirements

I. Site/ Environment	Chap. 1	Site	II. Design (continued)	Chap. 6	Reactor Coolant System
	Chap. 2	Radiological Environment		Chap. 7	Engineering Safety System
				Chap. 8	Containment System
				Chap. 9	Instrument and Control System
II. Design	Chap. 3	Design Common Requirements		Chap. 10	Electric Power Supply System
	3.1	Classification of Systems		Chap. 11	Auxiliary Systems
	3.2	Codes and Standards		Chap. 12	Power Conversion System
	3.3	Classification of NPP conditions		Chap. 13	Radioactive Waste System
	3.4	External events		Chap. 14	Radiation Protection System
	3.5	Equipment qualification		Chap. 15	Human Factors Engineering
	3.6	Reliability		Chap. 16	Accident Analysis
	3.7	Test, inspection and maintenance		Chap. 17	Severe Accident Assessment

	3.8 Material	III. Operation	Chap.18 Conduct of Operation
	3.9 Intersystem LOCA		Chap.19 Initial Test Program
	3.10 Station Blackout		Chap.20 Technical Specifications
	3.11 Low power, shutdown operation		Chap.21 Emergency Preparedness
	3.12 Building design and Arrangement	IV. Decom-missioning	Chap.22 Decommissioning
	3.13 ALARA		
	3.14 Fire protection	V. Quality Assurance	Chap.23 Quality Assurance
	Chap. 4 Structural Design		
	Chap. 5 Reactor		

III. DEVELOPMENT OF GENERAL SAFETY CRITERIA

III.1 Items of General Safety Criteria

A draft of the GSC was developed based on the key items and major contents derived from review of relevant requirements, with reflection of safety enhancement features as well as Korean regulatory environments. The draft has been developed in four areas among five areas as shown in Table 1, except decommissioning.

The items of the GSC have been developed based on the enforcement decree of AEA in Korea, Safety Series of the IAEA, [1-3] 10 CFR 50 Appendix A General Design Criteria of the U.S.A.[4] and so on.[5-7] (Refer Table 1) New items which are essential to the safety improvement of the KNGR have been derived through the evaluation of the aforementioned references. The examples of such items are "severe accidents", "severe accident management", "human factors", "startup, shutdown and low power operation", "reliability", "ALARA and dose limits" and so on. These are the newly reflected items in the GSC in order to enhance the safety of KNGR. The major perspective of the new items, in the viewpoint of safety enhancement, are presented in section III.2 of this paper.

The GSC have been developed as measures for compliance of Safety Principles and have provided fundamental bases and directions for the development of Specific Safety Requirements(SSR). These draft GSC will be modified and finalized through public comments and discussions with related organizations

Table 3. The structure and items of the GSC

I. Site/Environment		
1. Evaluation of External Factors 2. Derivation of Input Data for Site-related Design Bases	3. Assessment of Radiological Impacts 4. Emergency Preparedness	5. Ultimate Heat Sink 6. Construction of Multiple Units
II. Design		
1. Safety Functions 2. Design Basis Accidents 3. Severe Accidents 4. Quality Standards 5. Design Bases for External Events 6. Protection against Fire and Explosion 7. Environmental and Dynamic Effects Design Bases 8. Sharing of Structures, Systems, and Components	Inspectability/ Maintainability 13. Emergency Response Facilities 14. Design in preparation of Decommissioning 15. ALARA and Dose Limits 16. Reactor Design 17. Integrity of Reactor Core 18. Inherent Protection of Reactor 19. Suppression of Reactor Power Oscillations 20. Instrumentation and Control	23. Reactor Coolant Pressure Boundary 24. Design of Reactor Coolant Systems 25. Containment Design Bases 26. Residual Heat Removal 27. Emergency Heat Removal 28. Heat Transfer to Ultimate Heat Sink 29. Electric Power System 30. Control Room

9. Human Factors 10. Reliability 11. Startup, Shutdown and Low Power Operations 12. Testability/Monitorability/	21. Reactivity Control System 22. Protection System	31. Radiation Protection Facilities and Equipment 32. Treatment, Control, and Storage of Radioactive Materials 33. Fuel Handling and Storage
	III. Operation	
1. Operating Organization 2. Qualification and Training 3. Initial Testing 4. Technical Specifications 5. Operating Procedures 6. Core Management and Fuel Handling 7. Shutdown and Low Power Operations	8. Control in Human Factors 9. Operational Experience Feedback 10. Design Modification and Document Revision 11. Maintenance, Test, and Inspection 12. Periodic Safety Review 13. Radiation Protection Program	14. Radioactive Waste and Effluent Management 15. Emergency Preparedness 16. Security 17. Fire Protection 18. Records and Reports 19. Severe Accident Management
	IV. Decommissioning (to be developed later)	
	V. Quality Assurance	
1. Establishment and Implementation of Quality Assurance Program	2. Structure of Quality Assurance Program	3. Quality Assurance Requirements

III.2 SAFETY ENHANCEMENT OF GENERAL SAFETY CRITERIA

1. Severe Accidents

The draft GSC require that all reasonable steps be taken to prevent the chances of occurrence of severe accidents and to mitigate consequences of any such accidents. Selection of severe accidents to be considered in design shall be done by performing a comprehensive and systematic evaluation of postulated initiating events and accident sequences beyond design basis which could lead to severe accidents. Operating organization is required to establish an implement an AM (Accident Management) program in order to prevent severe accidents, to terminate further proceeding, and to minimize tbeir consequences. The AM program shall include accident management strategy, instrumentation & control systems and information supporting systems required for accident management, allocating decision making responsibility witbin operating organization, guidelines and procedures of accident management, and training and its implementation program.

2. Reliability Assurance

For safety enhancement of the KNGR, the draft GSC require that a reliability target should be assigned to safety systems or functions, and the related provisions shall be made for testing and inspection to verify that the target is maintained during operation, in addition to the current requirements which specify that the structures, systems and components (SSC) important to safety shall be designed to have diversity, redundancy, and functional independence and physical separation considering the structure, operative principles, and safety functions to be performed.

The general approach to assure the reliability of safety functions or safety systems is to establish a specific reliability target for each safety function or safety system based on the safety objective during the design phase, and to verify and maintain it continuously during the operational phase.[8]

3. Shutdown and Low Power Operation

Many safety related events have occurred during shutdown and low power operation, and a significant risk during shutdown mode has also been identified from some Probabilistic Safety Assessment studies. The GSC require that SSC used during those operations be designed to maintain or quickly restore the reactivity control capability, decay heat removal capability, and the containment integrity, so as to prevent the release of radioactive materials resulting from accidents initiated during those operations. Also, it requires that limiting conditions for shutdown operation, including safety function limits such as water level, be established, monitored, and complied with. Sufficient mitigation capability should be also available to provide proper core cooling and protection against uncontrolled release of radioactive materials following a loss of residual heat removal capability. In addition, to minimize undesirable situations challenging the safety functions, such as fire, the GSC require that the possibility of fires and available fire protection features be evaluated, and the loss of RHR as a result of such fires be prevented.

4. Human Factors

The GSC require that human factors associated with plant workers and man-machine interface shall be taken into account systematically in the design of nuclear facilities from the beginning stage of design to the final design validation. This requirement specifies that human factors engineering design shall not require performance exceeding the limits of human capabilities but require exercise of human capabilities as much as practicable within the range not to induce human errors.[9] Furthermore, the effects of human errors shall be evaluated, minimized, and mitigated through verification and validation at design stage. To minimize human errors during operation stage of nuclear facilities, the GSC also require proper consideration of the lessons learned from accidents and near-misses, maintaining clarification and efficiency of operation procedures, and systematic operation management to prevent degradation of task performance capabilities due to adverse working environment.

5. Environmental Effects Design Bases

It is necessary that the equipment and the instrumentation installed to monitor or to mitigate severe accidents are required to survive environmental conditions during severe accidents in order to enhance the capability to cope with severe accidents. Also, the effects of aging should be considered as environmental conditions and required to be addressed in design and equipment qualification in order to assure that SSC important to safety can retain the capability to perform their safety functions during plant lifetime.

The GSC require that the equipment and the instrumentation to monitor or mitigate severe accidents shall be designed to survive the environmental conditions where their functions are needed for a required period. Since the design lifetime of the KNGR is 60 years, it is expected that the performance and the structural integrity of the SSC of the KNGR will be more degraded by the aging effects than the existing plants. Therefore, the GSC also require that the aging effects shall be addressed in the design stage so that the SSC important to safety can accommodate the effects of aging effects during plant lifetime.

6. Radiation Protection

In order to keep radiation exposure as low as reasonably achievable (ALARA), the GSC require that the KNGR designers verify that KNGR design meets the dose constraints of workers and public or, in the case of not verifying this, they implement an estimation such as cost-benefit analysis to show that the design follows ALARA principle. For occupational dose limit, the effective dose of 100mSv for five years on the conditions with not exceeding 50mSv in any year shall be applied based on the ICRP 60 recommendations.[10]

IV. Consideration of New Licensing System for the KNGR

At present, Korean regulatory system employs a two-step licensing approach based on the prescriptive regulation. However, the KNGR is being designed with the concept of standardized plant in such a way that the succeeding units are constructed with the standardized design. Under this circumstance, the introduction of the following new licensing system is currently being considered as a complementary measure for the KNGR to the current licensing system: pre-application safety review, standard design approval, and combined construction/operation licensing.

1. Pre-application Safety Review

The utility desires to secure early the licensibility of proposed design with advanced design features such as passive safety features, digital instrumentation and control system, etc., so that the development of the KNGR could be progressed under the stabilized environment. In this regard, the Government(MOST) and KINS have proposed the introduction of a Pre-application Safety Review (PSR) system to nuclear industries. It will encourage an advanced interaction of applicants with regulatory body not only to provide for early identification of regulatory requirements, but to provide for more timely and effective regulation. In addition, the PSR system enables to reflect the comments of interested parties including the public on the view of regulatory body concerning the desired characteristics of advanced reactor designs. Such licensing interaction and guidance in the early design stage will contribute to minimizing complexity and adding stability in the licensing process.

2. Standard Design Approval

A Korean vendor attempts to standardize nuclear power plant designs, which significantly enables to enhance the safety, reliability and competitiveness of the NPP. Therefore, the Government is willing to reduce complexities in the regulatory process of the standard NPP by introduction of standard design approval system. The use of approved standard design would benefit the public health and safety by concentrating resources on a resolution of plant specific safety issues, by stimulating standardized programs of construction practice and quality assurance, and by fostering more effective maintenance and improved operation. Accordingly, it is expected that a standardization of the NPP could further improve the safety performance of future plants, and promote to develop a more efficient review process. A standard design approval review could be performed prior to a construction of a nuclear power plant in accordance with an improved licensing process to be introduced later.

3. Combined Licensing System

A development of combined construction/operation licensing process is in progress for minimizing complexities and increasing an effectiveness of licensing system for the standard NPP. The current two-step licensing system will be improved into an one-step construction/operation license and a fuel loading permit. However, intensive care should be taken in the introduction of such combined licensing system because it may have an adverse impact on the Korean regulatory environment. In-depth evaluation on the merits and drawbacks will be performed before its introduction into Korea. It is expected that the new licensing systems, when introduced, will be applied to the licensing of the KNGR.

V. Conclusion

The SRRG development program for the KNGR was introduced based on the interim study results. Basic development approaches for the regulatory requirements and some new items caused by the adoption of the enhanced safety concept were presented. Some selected enhanced safety aspects were also reviewed and discussed in the GSC level of regulatory requirements for the KNGR.

The introduced licensing systems will be considered as a complementary measures to the current licensing system. These new licensing systems are expected to help the applicant to secure the licensability of the KNGR design through an early interaction with regulatory body, to enhance the safety, reliability and competitiveness of the NPPs by the use of standard design, and to improve the effectiveness of licensing process.

The draft SRRG and the improved licensing system will be incorporated into the legal system after refinement through the review of domestic and foreign experts, and efforts devoted to the development of the SRRG for the KNGR will be a Korean contribution to the enhancement of nuclear safety for the global nuclear family.

REFERENCES

1. IAEA, "Code on the Safety of Nuclear Power Plants: Siting", *Safety Series No.50-C-S*, Rev.1, IAEA, 1988
2. IAEA, "Code on the Safety of Nuclear Power Plants: Design", *Safety Series No.50-C-D*, Rev.1, IAEA, 1988
3. IAEA, "Code on the Safety of Nuclear Power Plants: Operation", *Safety Series No.50-C-O*, Rev.1 draft, 1997
4. Office of Federal Register, "Appendix A to Part 50 – General Design Criteria for Nuclear Power Plants", *10 CFR Part 50*, January 1997
5. IAEA, "Development of Safety Principles for the Design of Future Nuclear Power Plants", *TECDOC-801*, 1995
6. BMI, "Nuclear Power Plant Safety Criteria", *Safety Codes and Guides Edition 13/78*, Germany, October 1977
7. "Guides of Nuclear Safety Commission", Rev.6, Japan Daesung pub, January 1991
8. USNRC, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs", *SECY-93-087*, April 1993
9. USNRC, "Human Factors Engineering Program Review Plans", *NUREG-0711*, July 1994
10. ICRP, "Recommendations of the International Commission on Radiological Protection", *Annals of the ICRP, ICRP Publication 60*, Pergamon Press, 1990