Transactions of the 17th International Conference on Structural Mechanics in Reactor Technology (SMiRT 17)
Prague, Czech Republic, August 17 –22, 2003
Paper # D01-5

Development of Periodic Safety Review Technology in Korea

Tae-Ryong Kim
Korea Electric Power Research Institute

ABSTRACT

Periodic safety review is being performed in Korea for the confirmation of comprehensive safety of an operating nuclear power plant. For the successful review, it is necessary to develop some technologies related to the PSR. Accordingly a national research project for the development of PSR technologies was started. Among 11 safety factors to be evaluated in PSR per IAEA guideline, some factors were selected as the objects of technology development. These factors were grouped into 5 areas such as, management of ageing, safety analysis, safety performance, equipment qualification, radiological impact on environment, organization and human factors. The technologies developed in this project were directly used in PSR on Kori Nuclear Unit 1, and will be used in the implementation of corrective items recommended to enhance the plant safety. The results of this research and PSR on Kori Unit 1 can be also utilized for the continued operation of Kori Unit 1 beyond its design life.

KEY WORDS: nuclear power plant, periodic safety review, safety factor, management of ageing, safety analysis, equipment qualification, safety performance, radiological impact on environment, organization and human factors.

INTRODUCTION

Periodic safety review (PSR) is considered to be an effective and widely used method to obtain an overview of actual plant safety, to determine reasonable and practical modifications that should be made in order to maintain a high level of safety and to improve the safety of older nuclear power plants to a level approaching that of modern plants. The performance of PSR was recommended to the member states by International Atomic Energy Agency (IAEA) safety convention held in 1994[1]. Currently, there are 18 nuclear power plants (NPP) in operation in Korea. The total years of operation are 189 reactor-years as of 2002, which will grow very rapidly in the future. At the moment, Korea has no specific regulation of the lifetime of NPP. Electric utilities can operate their NPP as long as safety of the plant is maintained. Even though Korea has maintained an excellent operating record of NPP so far, it is very important to continuously improve safety of the operating NPP. Accordingly, Korean Government decided to adopt PSR in the legal system [2~4] and established the PSR performance plan in 1999. At present, PSR on Kori Unit 1 (the oldest PWR plant in Korea), Wolsong Unit 1 (the oldest PHWR in Korea) and Kori Unit 2,3,4 are being performed.

A comprehensive assessment of overall plant safety is a complex task. Experience shows that it can be facilitated by dividing it into a number of elements. These are so called ‘safety factors’ as listed in Table 1. IAEA guide has recommended 11 safety factors [5]. But the safety factors to be reviewed in PSR are widely varied according to countries even though they adopted PSR. Some states prefer alternative methods (e.g. a systematic safety assessment program). Each alternative should be able to achieve the PSR objective. Korea adopted all the 11 safety factors [4].

For the successful PSR in Korea, it is necessary to develop some technologies related to PSR. Accordingly Korean Government decided to start a national research project for development of PSR technologies. Among 11 safety factors to be evaluated in PSR per IAEA guideline, some factors were selected as the objects of technology development. These factors are grouped into 5 areas, such as management of ageing, safety analysis, safety performance, equipment qualification, radiological impact on environment, organization and human factors. These safety factors were selected based on their state of the art, urgency and importance in PSR, applicability to other areas, etc.

The national research project for development of PSR technologies and some results of the project are introduced in this paper.

Table 1. Safety factors in PSR [4,5]

Number	Title	Number	Title
SF 1	Actual condition of the NPP	SF 7	Procedures
SF 2	Safety analysis	SF 8	Organization and administration
SF 3	Equipment qualification	SF 9	Human factors
SF 4	Management of ageing	SF 10	Emergency planning

SF 5	Safety performance	SF 11	Radiological impact on environment
SF 6	Use of experience from other plants and research findings		

NATIONAL RESEARCH PROJECT FOR PSR

As mentioned earlier, a national research project sponsored by Ministry of Science and Technology was started for the development of PSR technologies. The outline of the project is as follows:

- Project Title: Development of PSR technology for nuclear power plant
- Project Period: September 16, 2000 ~ February 29, 2004 (4 fiscal years)
- Project Areas: 5 sub-projects (refer to Table 2)

Table 2. PSR research project areas

No.	Title of sub-projects	Related safety factors
1	Development of ageing management technology	Management of ageing, Actual condition
2	Development of technologies for seismic margin assessment of equipment	Equipment qualification
3	Development of operational transient analysis and new safety analysis methodologies for NPP	Safety analysis, Safety performance
4	Development of environmental radiation monitoring technology	Radiological impact on environment
5	Development of assessment technology of effect of human and organizational factors on NPP safety	Human factors Organization and administration

- Project Objectives:
  - Establishment of basic technologies related to PSR on the safety factors selected
  - Application of developed technologies to actual PSR on nuclear power plants
- Scope of research: refer to Table 3
- Project Budget: $9.86 Million (refer to Table 4)

Table 3 Scope of PSR research project

No.	Title of sub-projects	Scope of the research
1	Development of ageing management technology	- Applicability of ageing-related technologies to PSR - Environmental fatigue management technology for low-alloy steels and stainless steels - Thermal ageing management technology for cast stainless steels - Environmentally assisted cracking management - Localized nickel plating technology - Ageing management for concrete structure - Creep life evaluation technology using small disc specimen
2	Development of technologies for seismic margin assessment of equipment	- Concurrent assessment of seismic qualification based on both requirements of seismic IPEEE and USI A-46 - Re-evaluation of seismic response for the USI A-46 plants - Construction of database of equipment seismic qualification by testing - Test for seismic capacity of cast-in-place anchor system - Seismic isolation system - Procedure of equipment seismic qualification
3	Development of operational transient analysis and new safety analysis methodologies for NPP	- New non-LOCA analysis methodology - Operation transient analysis methodology - ATWS analysis methodology - New accident analysis methodology for equipment qualification - Database for safety analysis and best estimate analysis - Application of the developed technologies to operating NPP

4	Development of environmental radiation monitoring technology	- Establishment of the assessment technology for contamination and accumulation trends of radionuclides - Technology how to setup alarm setpoint for environmental radiation monitoring - Safety performance indicators for radiation - Evaluation technology for performance of radiation monitoring system
5	Development of assessment technology of effect of human and organizational factors on NPP safety	- State of the art report on assessment technology of effect of human and organizational factors on NPP safety - Application of system dynamics methodology - Development of causal loop diagram - Development of stock and flow diagram

Table 4. Budget of the Project (unit: million $)

No.	Title of sub-projects	2000	2001	2002	2003	Total
1	Ageing management	0.66	0.74	0.78	0.75	2.93
2	Seismic margin assessment	0.48	0.60	0.54	0.48	2.10
3	Safety analysis methodologies	0.63	0.65	0.84	0.72	2.84
4	Radiation monitoring	0.27	0.29	0.33	0.28	1.17
5	Human and organizational factors	0.19	0.21	0.23	0.19	0.82
Total		2.23	2.49	2.72	2.42	9.86

SOME RESULTS OF THE RESEARCH

Development of Ageing Management Technology

Ageing management is one of the key items in PSR. For the review on ageing management, the potential ageing mechanisms and their effects should be identified and evaluated for system, structures and components (SSC) of NPP. There have already been many researches on the subject of ageing and its management. But it is still necessary to develop methodologies regarding to practically applicable ageing management technology to operating NPP. In the development of ageing management technology, ageing-related R&D results up to date were reviewed focusing in the ageing management mechanism identification, the residual life assessment, and ageing management program evaluation. Subsequently, ageing management technologies directly applicable to PSR were selected and studied.

The SSC's to be assessed in Kori Unit 1 and Wolsong Unit 1on which PSR's are being performed were identified. Then the ageing-related technologies applicable to these specific plants were systematically classified and stored in a format of database. Guidelines for ageing evaluation for the SSC were also prepared and used in the PSR.

For the environmental fatigue management of reactor vessel material (SA508 Class 3) and piping materials (cast stainless steel, CF8M) used in Kori Unit 1, fatigue tests with several load ratio in high/low dissolved oxygen environments were done. Data of fatigue crack growth rates were collected and used as input data in the ageing evaluation of Kori Unit 1[6].

For cast stainless steel (CF8M) that is known to be most susceptible to thermal ageing, the test results and evaluation procedures were collected and reviewed (including Charpy impact test, J-R curve, and σ-ε curves of the materials). Based on the review, a procedure to evaluate piping integrity for thermal ageing was developed. For thermal ageing of pump materials, the fracture toughness was also estimated from the Charpy impact energy at end of life.

Probabilistic life prediction models in environmentally assisted cracking (EAC) condition were developed for stainless steels (SS 304 & 316) widely used as nuclear materials. Using the developed model, the failure probability of control rod driving mechanism (CRDM) in Kori Unit 1 was evaluated. Using in-situ Raman spectroscopic experimental apparatus, the effect of dissolved hydrogen on oxide layer stabilization in Alloy 600 was investigated as one of EAC control methods [7].

Nickel plating technology is being also developed as an alternative repair method when cracking was initiated in CRDM, pressurizer and stainless piping. It can be used for the improvement of plant safety in post-PSR stage.

To identify the creep characteristics of zirconium alloy (pressure tube material of Wolsong Unit 1, CANDU reactor), a method using small punch creep test was developed. It was found that the steady state creep constant and stress indices derived from the small punch test were relatively in good agreement with those from the uni-axial creep test.

Accelerated ageing test method for cables in NPP was investigated [8] and a portable ageing tester was developed.

Development of Technologies for Seismic Margin Assessment of Equipment

It was found that seismic qualification of equipment in Kori Unit 1was insufficient comparing with other NPP constructed later where the latest codes and standards were applied. To solve this kind of issue in PSR, seismic qualification technologies for operating NPP should be developed.

A particular seismic qualification procedure was suggested to meet individual plant examination for external events (IPEEE) requirement and unsolved safety issues (USI) A-46 requirement simultaneously for old plants in Korea. Since the seismic qualification data for these old plants did not exist, generic equipment ruggedness spectra (GERS) according to current seismic requirement were newly generated and used in seismic analysis. Seismic qualification database was also developed for the seismic margin assessment on Kori Unit 1, by expanding the SQUG (seismic qualification users group) database of EPRI.

The seismic capacity evaluation of anchor system of equipment is the most important and fundamental factor in the evaluation of 'equipment qualification' for seismic event in PSR. The design code for anchor system of equipment in NPP, ACI 349, was revised in accordance to many researches and experiments performed in Europe and USA. Test for seismic capacity of cast-in-place anchor system was done according to the code. The results have shown that the code estimates seismic capacity conservatively in case of single anchor system. However, in multiple anchor system, the original code had the tendency of over-estimation on concrete pull-out capacity. Furthermore the original code could not manage the edge effect. In the contrary, the revised code gave conservative results in all cases. It was also found that the effect of prying action should not be ignored because prying factors in various anchorage types were considerably large value [9].

Confirming the seismic qualification of new equipment replaced is also very important issue in operating NPP. A procedure based on seismic test database and experience was proposed.

Development of Operational Transient Analysis and New Safety Analysis Methodologies for NPP

Safety analysis in the scope of PSR requires that the analysis codes, methods, assumptions, etc described in the original FSAR be satisfactorily valid comparing with the latest safety standards, and that the operational experiences and recent R&D results be included. According to the requirement, it is necessary to develop new safety analysis methodologies to reflect the operational experiences and recent R&D efforts.

A new method using RETRAN code applicable to Westinghouse type PWR (Kori Unit 1~4, Yonggwang Unit 1&2) and CE type PWR (Yonggwang Unit 3&4) was developed for the operation transient analysis [10,11]. A new non-LOCA analysis method was also developed and applied to Kori Unit 1. Safety analysis database and best estimate analysis database for Kori Units 1&2 were developed for PSR and verified by analyzing turbine trip and loss of flow accidents.

For the environmental qualification of equipment for Kori Unit 1, a new LOCA mass/energy release, containment pressure and temperature analysis methodology was developed based on RELAP5/MOD3 and CONTEMPT4/MOD5.

For the development of anticipated transient without scram (ATWS) analysis method, three accident cases were chosen, that is, loss of normal feedwater, loss of load/turbine trip and rod withdrawal at power. It was found that the acceptance criteria, DNBR and the calculated peak pressure were satisfied in all cases investigated.

Development of Environmental Radiation Monitoring Technology

The past 10 year records of tritium concentration in the environments around three NPP sites were reviewed. A dynamic compartment model for the prediction of tritium behavior around Kori Unit 1 was developed [12]. The tritium concentration of the three sites was found to be 1~4 Bq/L which was a little bit higher than the natural background concentration in Korea. The Cs-137 concentration and the ratio of Cs-134/Cs-137 along the depth of soil at Kori NPP site were measured. As a result, it was found that Cs-137 at Kori site was not from the NPP but from the global fallout.

Radiation data measured at twelve monitors of environmental radiation monitoring system (ERMS) installed around Kori site was analyzed for setting up alarm point of the monitoring system. The alarm point suggested in abnormal state of ERMS was three times of the maximum value in normal state.

Three indicators were suggested for the evaluation of 'safety performance' factor in PSR, such as the effluent amount of tritium, the radiation exposure dose of plant workers and the concentration of Co-60 in the primary coolant.

Development of Assessment Technology of Effect of Human and Organizational Factors on NPP Safety

The effects of organization /administration and human factors on plant safety should not be overlooked in PSR, since they play a significant role in safety culture of the plant. There are many research results on the safety culture, but no specific safety standard and the assessment technology are established yet. The assessment technology of effect of human and organizational factors on NPP safety has been studied in this project by using System Dynamics.

A system dynamics model was developed including causal loop diagram based on the interview with employees in order to describe the cause and effect relationship of the organization/ human factors. Stock and flow diagram was also developed to quantify the relationship. Simulation of the model was done for several scenarios, that is, adjustment of human resource, leadership, frequency of regulatory activities and occurrence of unexpected work. Validation and verification of the system dynamics model is on going.

APPLICATION TO ACTUAL PLANTS

There are presently 18 nuclear power plants being operated by Korea Hydro & Nuclear Power Cooperation (KHNP) in Korea. In accordance with Government decision on the adoption of PSR system, KHNP established a strategic plan to do PSR of the operating NPP's in May of 2000 and set to the first PSR on Kori Unit 1, the oldest PWR in Korea [13]. Description of Kori Unit 1 under the PSR is shown in Table 5.

Table 5. Description of Kori Unit 1 under periodical safety review

Reactor Type / Power	PWR / 587 MWe
NSSS Supplier	Westinghouse
Turbine Generator	General Electric Company
Total Electricity Generated (at Dec. 31, 2000)	89.2 billion KWh
Capacity Factor	95% (2001)
Design Life of Components (in FSAR)	30 years
Terms of Operation License	Unlimited
Period to be Reviewed	First Criticality (June 19, 1977) ~ Dec. 31, 2000
Period of PSR	May 30, 2000 ~ Nov. 30, 2002

11 safety factors of the review were fully evaluated in accordance with the domestic legal system, following the PSR procedure as shown in Fig. 1. Since PSR is a very complicated task and many organizations/persons are involved, it is necessary to setup a project management system on computer network for the real time communication and easy access to common plant data. A PSR project management system was therefore established for Kori Unit 1.

The technologies and procedures developed in this study have been successfully applied to the assessment of each safety factors in PSR. The EAC database and life prediction model provided the basis for lifetime prediction and integrity evaluation of the major equipment. Thermal ageing evaluation procedures was used for integrity evaluation and ageing management of cast stainless steel components like RCS piping and pumps. The newly generated floor response spectra and the database can be used as input in seismic adequacy verification for which a separate project already started. New safety methodologies developed here was used in the additional transient analyses of PSR on Kori Unit 1, which were originally not performed. The additional transient analyses were single rod cluster control assembly withdrawal accident, boron dilution accident during operation modes 3, 4, 5 and 6, and failure of small lines carrying primary coolant outside containment.

Although it is the oldest nuclear power plant in Korea, Kori Unit 1 was found to have maintained good operating conditions and continuously enhanced its safety by implementing post-TMI action plans and other safety issues, such as replacing steam generators and process/control system. It can be therefore confirmed that safe operation of Kori Unit 1 is guaranteed until next periodic safety review. Nevertheless, some corrective action items were recommended to further enhance its safety level, such as equipment qualification, additional ageing management program, strengthening of some procedures related to administration and human factor. The results of this research can be also used in the implementation of the corrective action items For instance, nickel plating and oxide layer analysis techniques can be used to control and mitigate the EAC-related damage for enhancing the plant safety.

As mentioned earlier, Korea has no specific regulation of the lifetime of NPP. Electric utilities can operate their NPP as long as safety of the plant is guaranteed. Therefore the results of PSR on Kori Unit 1 can be utilized for its continued operation beyond the design life. Experiences of the PSR on Kori Unit 1 can be also applied to PSR on other plants.

CONCLUSION

A national research project for development of periodic safety review technologies has been performed. Among 11 safety factors to be evaluated in PSR, some factors were selected as the objects of technology development based on their state of the art, urgency and importance in PSR, applicability to other areas, etc. These factors were grouped into 5 areas such as, management of ageing, safety analysis, safety performance, equipment qualification, radiological impact on environment, organization and human factors.

The technologies and procedures developed in the above five areas were directly used in PSR on Kori Unit 1 and other operating nuclear power plants. In addition, they will be afterward used in the implementation of corrective actions to enhance plant safety. The results of this research and PSR on Kori Unit 1 can be also utilized for the continued operation of Kori Unit 1 beyond its design life. Furthermore, experiences of the PSR on Kori Unit 1 can be applied to PSR on other plants.

ACKNOWLEDGEMENT
This paper is financially supported by Ministry of Science and Technology.

REFERENCES
[1] IAEA, Convention on Nuclear Safety, Vienna, 1996.
[2] Korean Atomic Energy Act, Paragraph 3 of Article 23 (PSR), Jan. 16, 2001
[3] Presidential Decree of the Act, Paragraph 3 of Article 42 (Content of PSR), July 16, 2001
[4] Ministry of Science and Technology Decree of the Act, Paragraph 2 of Article 19 (Details of PSR), July 25, 2001
[5] IAEA, Safety Series 50-SG-012, Periodic Safety Review of Operational Nuclear Power Plants, 1994.
[6] I. S. Kim et. al., "Strain Rate Effects of Fatigue Crack Growth of SA508 Cl.3 Reactor Pressure Vessel Steel in High Temperature Water Environment," J. of Pressure Vessel Technology, vol.123, No. 1, 2001
[7] J. H. Kim, et. al., "In-situ Raman Spectroscopic Investigation of Alloy 600 in Simulated PWR Water," Int. Symp. on NPP Life Management, IAEA, Budapest, Hungary, Nov. 4~8, 2002
[8] J. S. Kim, et. al., "Accelerated Ageing & Life Evaluation of Cable Jacket based on the Plant Condition Monitoring," 3rd Int. Conf. on AMDP, Oct. 16~19, 2002.
[9] J. B. Chang, et. al., "An Evaluation of Tensile Capacity of Anchor System by Actual Model Tests in NPPs," Symposium on Connections between Steel and Concrete, RILEM, ACI, 2001.9
[10] Y. H. Kim, et. al., "Research Activities Related to RETRAN in KEPRI," RETRAN-VIPRE User Group Meeting, EPRI, 2000
[11] A. J. Chung, et. al., "Feedwater System Malfunction Analysis with RETRAN," 3rd Korea-Japan Symp. on Nuclear Thermal Hydraulics and Safety, Kyongju, Korea, 2002
[12] H. J. Choi, et. al., "Development of a Dynamic Compartment Model for the Prediction of Tritium Behavior around Kori Unit 1," Int. Symp. on Transfer of Radionuclides in Biosphere, 2002
[13] KHNP, The Periodic Safety Review of Kori Unit 1 (Summary Report), November 30, 2002.

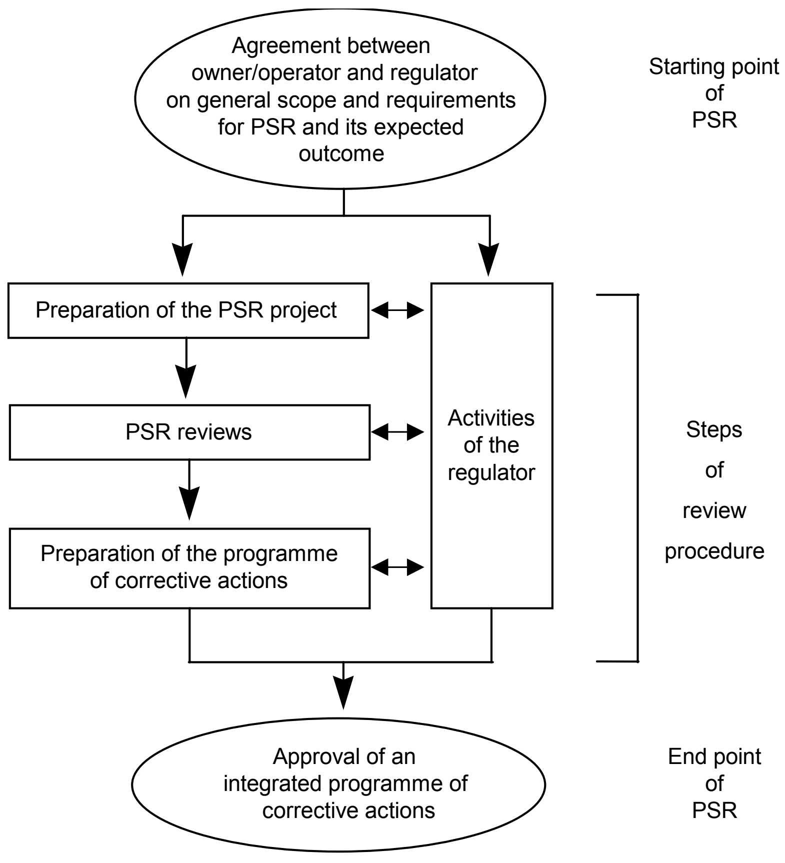

Fig. 1 Typical procedure of Periodic Safety Review (recommended by IAEA[5])