Transactions of the 15th International Conference on
Structural Mechanics in Reactor Technology (SMiRT-15)
Seoul, Korea, August 15-20, 1999 D01/2

Major Components Life Evaluation for Nuclear PLIM Feasibility Study

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ABSTRACT

The Nuclear Plant Lifetime Management(PLIM) program is one of the major areas of interest for utilities operating old nuclear power plants. To pursue the extended operation of nuclear power plant, technical, economic, and regulatory feasibilities have to be examined. A comprehensive PLIM program has been being underway to ensure intended functions of systems, structures, and components(SSCs) and safe operation of nuclear power plant throughout its lifetime. It is consisted of integrated plant assessment(IPA), residual life evaluation, identification of aging mechanisms, and development and implementation of proper monitoring, mitigation, and maintenance schemes. The current and projected conditions of plant were analysed by understanding various aging mechanisms(such as fatigue, irradiation embrittlement, thermal embrittlement, corrosion, stress corrosion cracking, erosion-corrosion, and wear etc.) of structures and components during plant lifetime. Through feasibility study, it was concluded that life extension of nuclear power plant was a viable option and that more detailed evaluation should be conducted to implement PLIM program. In this paper, the method and result of screening of SSCs are presented. The results of aging mechanism analyses and life evaluations of SSCs necessary for the PLIM feasibility study are summarized and discussed. Finally, recommendations for the next phase of the program are presented

INTRODUCTION

The first phase of nuclear plant lifetime management program, or PLIM(I) has been carried out to cope with plant aging and degradation of Kori Unit 1 since 1993.[1] The PLIM program provides a plan for preventive maintenance and aging management throughout plant lifetime with analyzing aging phenomena and residual life of systems, structures, and components(SSCs).

The PLIM feasibility is evaluated in terms of technical, economic, and regulatory aspects of extended operation. For the life extension of nuclear power plant which is the objective of the PLIM program, components replacements, plant refurbishments, and upgrade strategies are determined based on the results of the PLIM feasibility study. Major components evaluation, understanding of ageing phenomena and residual life evaluation are the key parts of the technical feasibility study.

The technical aspect of life extension feasibility of one of the oldest nuclear power plant in Korea is presented. This paper covers the screening of SSCs, aging mechanism analyses and life evaluations of SSCs. Also, the recommendations for the next phase of the program are presented..

SCREENING AND PRIORITIZATION OF MAJOR COMPONENTS

The objectives of screening are to identify systems and structures important to assuring safe and economical plant operation throughout the desired plant life, to identify Kori Unit 1 major components within these systems and structures, to prioritize major components with respect to their impact on plant life, and to support PLIM program goal of extending plant lifetime.

The screening process applies safety-related criteria which are based upon the U.S. NRC's license renewal rule(LR, 10CFR54)[2] and maintenance rule(MR, 10CFR50.65)[3]. Additionally, the screening process applies power production(PP) related criteria which are based on plant availability. Sixty-two out of the 71 systems and structures listed in the Final Safety Analysis Report of Kori Unit 1 met either the LR, MR, or PP criteria. Fifty-one of them were identified important to LR, 55 to MR, and 32 to PP, respectively. From the screened components, 47 critical components within following categories were selected.

○ Category 1: Plant life non-limiting but justification required.
○ Category 2: Long life with potential for high impact on plant life.
○ Category 3: Long life with nominal impact on plant life.

Prioritization of Kori Unit 1 critical components was based upon ten attributes which were selected to assess the impact that either the replacement or refurbishment of these critical components would have on the decision to improve design life. These attributes are

○ Cost to replace or refurbish
○ Impact on plant availability
○ Radiation dose
○ Regulatory importance
○ Modification required
○ Replacement precedent
○ Generic applicability
○ Mode of failure
○ Consequences of failure on plant safety
○ Consequences on plant operations

Prioritization result of Kori Unit 1 was compared to those of the U.S. nuclear plant experiences.[4,5] The result of Kori Unit 1 showed a similar result to the previous results as shown in Table 1.

In order to evaluate the aging status of Kori Unit 1, following major components are selected in the beginning of the PLIM(I). Steam Generators, though ranked high in the list, were excluded due to their planned replacement in 1998. All major components below were enlisted within top twenties of the component prioritization ranking

○ Reactor Pressure Vessel
○ Reactor Pressure Vessel Internals
○ Control Rod Drive Mechanism

Table 1 Comparison of Prioritization Results

Rank	Kori Unit 1	Another U.S. Plant	NRC	YNPS
1	Reactor Pressure Vessel	Reactor Pressure Vessel	Reactor Pressure Vessel	Reactor Pressure Vessel
2	Containment(Liner, Basemat, and Shield Bldg.)	Containment and Basement	Containment and Basement	Reactor Internals
3	Steam Generators	RPV Supports	Reactor Coolant Piping	Neutron Shield Tank
4	RCS Piping, Large Valves, Nozzles	RCS Piping(Cat. 1 & 2)	Steam Generators	HP Turbine
5	Reactor Coolant Pump Casing	Steam Generators	RCS Pump Bodies	LP Turbines
6	Pressurizer(Nozzles, Surge, Spray Piping)	Emergency Diesel Generators	Pressurizer	Generator
7	RPV Internals	RPV Internals(Upper and Lower)	CRDMs	Steam Generators
8	Cables(in Containment)	RCS Pump Body	Cables and Connectors	Pressurizer
9	CRDMs	Pressurizer	Emergency Diesel Generators	CRDMs
10	MS Piping(Stop, Control, Intercept Valves)	Neutron Shield Tank	RPV Internals	Condenser
11	HP & LP Turbines	CRDMs	RPV Supports & Biological Shield	Service Water System

○ Pressurizer
○ Reactor Coolant System Piping
○ Reactor Coolant Pump
○ Reactor Pressure Vessel Supports
○ Pressurizer Nozzle
○ Reactor Coolant System Nozzle
○ Turbine
○ Generator
○ Containment
○ Cable

LIFE EVALUATION PROCEDURE AND AGING MECHANISMS

In the PLIM feasibility study, aging mechanisms and the lifetime of 13 major components were analyzed by the process shown in the Figure 1. Recommendations from the result of major components life evaluation will be accommodated in the next phase of PLIM program.

As a prerequisite to the evaluation of the plant aging status, a huge amount of design and field data of Kori Unit 1 that had been accumulated since the start of commercial operation were surveyed

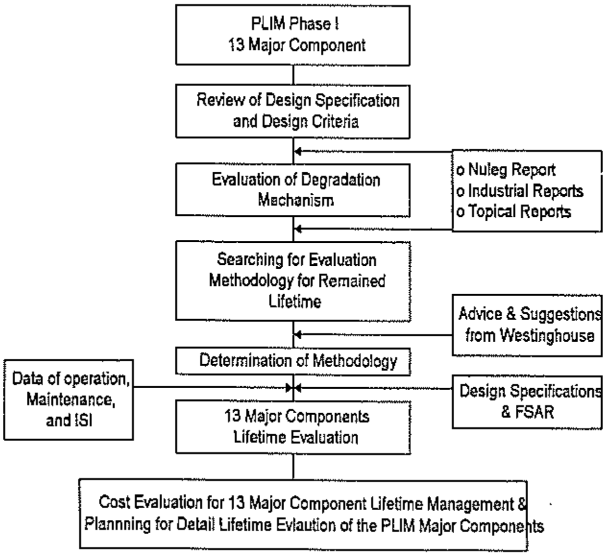

Figure 1. Component Lifetime Evaluation

and reviewed. Even though tremendous man-power was required in the process of compiling operating transient numbers to re-produce useful data from the raw materials, data survey was the most important job that should be done. Such data required for the PLIM(I) can be classified as follows.

- ○ General methodology and technical references
- ○ Operating transient history
- ○ Component design specifications and manufacturing data
- ○ Maintenance and in-service inspection data

MAJOR COMPONENTS LIFE EVALUATION

The stressors and the potential degradation sites and mechanisms in conjunction with the resulting failure modes and their operating history were identified through the appropriate tests and technical evaluations. The results are summarized in Table 2. In consequence, the task ended up with qualitative and/or quantitative evaluations of the plausible age-related degradation mechanisms, with the aid of proven technical papers and the generic technical procedures published by the foreign organizations who performed life management study before. Figure 2 shows a component life evaluation methodology of the reactor pressure vessel as an example.

Table 2. Potential Aging Mechanisms of Structures and Components

Plant System	Thermal Fatigue	Thermal Embritlement	Cracking, Fracture	Irradiation Embrittlement	Erosion/ Corrosion	Wear	Vibration
RPV	o		o	o	o		
Pri. and Aux. Piping	o	o			o		
Reactor Internals	o			o	o	o	o
Pressurizer	o				o		
S/G			o		o	o	o
Reactor Coolant Pump	o	o				o	o
Valves		o	o		o	o	
CRDM	o	o				o	
RPV Supports	o			o			
TBN, Generator	o		o			o	o
2ndary System Piping	o		o		o		o
Heaters, H/Xs, Condensers, MSRs			o		o	o	o
Pumps, Motor Fans		o	o		o	o	o
Containment Vessel					o		
Concrete					o		
Cable		o		o			

Reactor Pressure Vessel

In terms of plant safety, the reactor pressure vessel (RPV) is one of the most critical pressure boundary components in the nuclear power plant. The potential degradation sites and mechanisms are shown in Table 3. Of them the primary degradation mechanisms for reactor pressure vessels are irradiation embrittlement and thermal fatigue. The effects of irradiation and fatigue damage should then be considered in determining the overall lifetime of the reactor pressure vessel.

Table 3. Degradation Mechanisms of PWR Reactor Pressure Vessel[5]

Rank	Sites	Stressors	Mechanism	Failure	ISI Methods
1	Beltline region	Neutron irradiation, mechanical and thermal stresses	Irradiation embrittlement Environmental fatigue	Brittle fracture (i.e., PTS), Ductile low-energy tearing(LUST) Crack leading to a leak, possible brittle fracture if PTS occurs	100% volumetric inspection, Surveillance program for assessing irradiation damage
2	Outlet/inlet nozzles	Mechanical and thermal stress	Fatigue crack initiation and propagation	Crack leading to a leak, possible brittle fracture if PTS occurs with some irradiation embrittlement	Volumetric inspection of all nozzle welds and inside radius sections at each interval
3	Instr. nozzles, penetrations, and CRDM housing nozzles	Mechanical and thermal stresses, residual stress, water chemistry, temperature	Fatigue crack initiation and propagation PWSCC	Crack leading to a leak Intergranular cracking leading to a leak	Visual inspection of the external surface, 25% of nozzles inspected during each interval
4	Flange closure studs	Mechanical and thermal stresses Leaking borated water	Fatigue crack initiation and propagation(possibly corrosion assisted) Boric acid corrosion	Ductile overload failure (can be replaced) SCC leading to a leak	Volumetric and surface inspection of all studs and threads in flange stud holes at each interval

Irradiation Embrittlement

Special attention was paid to the RPV for its significant importance in the PLIM program. Fracture toughness test results of the WOL specimens of the Kori Unit 1 surveillance capsule irradiated for 34 EFPY show that the fracture toughness properties of Kori Unit 1 beltline weld material are similar to those of other plants that used Linde 80 flux weld metals and better than the fracture resistance predicting curve proposed by the U.S. NRC regulatory guide 1.161.[6] Figure 2 shows the procedure for RPV Integrity Evaluation with regard to irradiation embrittlement.

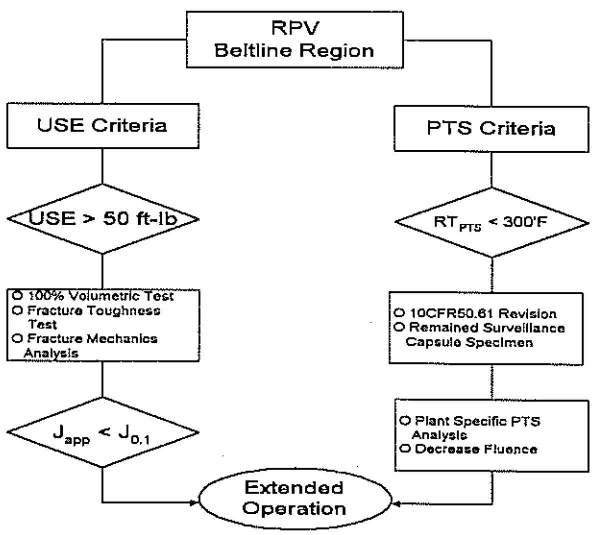

Figure 2. RPV Integrity Evaluation

As far as upper shelf energy criteria are concerned, the fracture toughness test proved that 40 years operation of Kori Unit 1 poses no significant penalty on the RPV integrity.[7] However, PTS screening indicated that it is necessary to perform plant specific PTS analysis for ensuring the RPV integrity beyond 34 operating years. To cope with the PTS issues, the Kori unit 1 plant-specific PTS analysis program was initiated in 1996.

Fatigue Life Evaluation

The fatigue evaluation was performed based on the design and operating history according to reactor pressure vessel final design documents.[8] The evaluation results for the critical region of RPV sub-components are presented in Table 4. In Table 4, the second column is cumulative usage factor based on the design transient numbers, and the third column is cumulative usage factor based on the actual transient numbers taken from operating history. The fourth column is predicted cumulative usage factor in 40-year design life assuming similar operating condition.. And, the last columns is the number of years of expected fatigue life based on limit 1.0 using the following equation.

$$\text{Expected fatigue life (yrs)} = \frac{1.0}{CUF_{40}} \cdot 40$$

According to these study, the unique section which requires close attention in view of fatigue damage is the closure studs.

Table 4. Fatigue Analysis of RPV

Sub-components \ Items			Design C.U.F (A)	Operating History C.U.F(B)	B/A	C.U.F (40Yr)	Fatigue Life (Yr)
Inlet Nozzle	Mag. Moly Section	Inside	0.269	0.046	0.171	0.108	370.0
		Outside	0.198	0.040	0.202	0.094	425.0
Outlet Nozzle	Stainless Steel Section		0.040	0.010	0.250	0.024	1666.6
	Mag. Moly Section		0.291	0.071	0.244	0.167	239.5
Closure Studs	Inside		0.909	0.178	0.196	0.419	95.4
	Outside		0.765	0.154	0.201	0.362	110.4
Vessel Shell			0.211	0.035	0.166	0.082	485.7

Pressurizer

The residual life assessment of the pressurizer were performed in similar procedure as RPV. Among plausible age-related degradation mechanisms shown in Table 5, the fatigue damage at the locations of the upper shell/head and the seismic lug were analyzed in detail. Specifically, the analysis procedures and the cumulative fatigue usage factors at the original design stage were reviewed on the basis of the up-to-date design codes which is currently in use. The actual plant transient data that had been compiled were subsequently categorized after compared with the design transients.[8] As a result, the consumed and residual pressurizer lifetime in terms of fatigue failure were obtained and shown in Table 6.

Table 5. Degradation Mechanisms of Pressurizer[5]

Rank	Degradation site(s)	Stressors	Degradation mechanism(s)	Potential failure mode(s)	Inservice inspection method(s)
1	Vessel shell near steam-water interface	Plant operational transients, water level changes(due to insurges, outsurges and heater actuations), sloshing, subcooled spray impact, and hydrotests; PWR coolant	Fatigue (possibly corrosion assisted)	Cracking leading to leak	Volumetric
2	Surge line nozzle	Plant operational transients, insurges, outsurges, and hydrotests; PWR coolant	Fatigue (possibly corrosion assisted)	Cracking leading to leak	Volumetric surface
3	Spray line nozzle	Plant operational transients, insurges, outsurges, and hydrotests; PWR coolant	Fatigue (possibly corrosion assisted)	Cracking leading to leak	Volumetric surface

Summary of Major Components Lifetime Evaluation

So far, we have demonstrated lifetime evaluation results of RPV and Pressurizer. The qualitative results of other major component lifetime evaluations are summarized in Table 7. Degradation mechanisms and lifetime evaluation results of each component are also illustrated with recommendations for next phase PLIM Program.

Table 6. CUF and Residual Fatigue Life of Pressurizer and Nozzles

	CUF			CUF Consumed (%)		Residual Fatigue Life (yr)	
	DSR	1993	2018	DSR	1.0	DSR	1.0
Upper Head/Shell	0.849	0.165	0.413	19.4	16.5	70	86
Seismic Lug	0.888	0.165	0.413	18.6	16.5	74	86
Surgeline Nozzle	0.139	0.030	0.076	21.9	3.1	57	507
Spray Nozzle	0.765	0.103	0.258	13.49	10.32	102	139

Table 7. Major Component Lifetime Evaluation and Recommendations

Component	Deg. Mechanisms	Lifetime Evaluation	Recommendations
RPV	Irradiation Embrittlement, Fatigue	○ PTS Criteria RT_{PTS}=300 °F @ about 27.4 EFPY ○ Upper Shelf Energy Criteria Safe to 34EFPY ○ Fatigue, O.K.	○ Plant specific PTS evaluation ○ Consider additional loads in DSR ○ Monitor boric acid corrosion @ head adapter seal welds & instrumentation port Joints ○ Chemistry variability of weld material
RPV Internals	Irradiation Embrittlement, Fatigue, SCC, IASCC, Stress Relaxation, Wear	○ Low cycle fatigue, O.K. ○ Control rod inspection & replacement every 15 years	○ Fatigue evaluation of design change ○ High cycle fatigue evaluation ○ Irradiation & Thermal Aging evaluation ○ Advanced ISI methodology application
CRDM	Therm.Embr'nt, Fatigue, Fret'g, Wear, Insul'n Breakdown, Elec. Shorting	○ Total stepping No., O.K. to 60 years	○ Life evaluation of coil stacks ○ Crack monitoring ISI required ○ Periodic check of CRDM wear
Pressurizer	Fatigue, Crack Growth, SCC	○ Detailed life eval. of high CUF upper head & shell, seismic lug	○ Detail review for CUF calculation method ○ Stress analysis considering cladding effect ○ Continuous monitoring ISI flaw @ manway ○ Review Size Effect of 1800 & 1000cuft PZR
RCS Piping	Thermal Embrittlement, Fatigue, SCC	○ Fatigue, O.K. ○ Thermal Embrittlement after 40 years	○ Life evaluation of other pipings ○ Generate DSR ○ Thermal embrittlement evaluation ○ Embrittlement data verification of SS
RCP	Thermal Embrittlement, Fatigue, Corrosion	○ Verification of Ferrite Content ○ Thermal Embrittlement O.K. to 40 years ○ ASME III fatigue waiver calculation to 40 years, it's O.K.	○ ASME III fatigue waiver calculation ○ Fatigue, O.K. to 40 years, but FEM analysis req'd for 60 yrs ○ Detail eval. for Ferrite Content and thermal embrittlement
RPV Supports	Irradiation Embrittlement, SCC, Corrosion, Fatigue	○ Support Brackets Fatigue, O.K.	○ Monitoring neutron fluence and spectrum ○ Chemical property eval. for embrittlement analysis ○ SCC of bolts and nuts
PZR Nozzle	Fatigue, Thermal Embrittlement	○ Fatigue and crack growth, O.K.	○ LBB analysis ○ Fatigue analysis for thermal stratification and striping
RCS Nozzle	Fatigue	○ Fatigue, O.K.	○ Monitor valve internal leakage ○ ISI method for thermal fatigue crack ○ Monitoring sys. for thermal stratification
Turbine	SCC, Fatigue	○ LP rotor disk SCC life exhausted	○ LP rotor replacement ○ Review SCC growth rate calculation formula
Cable	Thermal Aging, Radiation Aging, Moisture Aging	○ Low voltage power cable thermal aging evaluation, arrhenius method, cal'd residual life=98 yrs	○ Monitor containment inside T & humidity ○ Synergistic effect of thermal and irradiation aging
Containment	Corrosion, SCC, Fatigue	○ About 43 yrs residual life by corrosion life evaluation	○ Periodic program of checking shell thickness ○ Detail fatigue evaluation ○ Detail evaluation of design requirement change and re-design

CONCLUSIONS

This paper showed the procedure and methodology for selecting major components, identifying degradation mechanisms, calculating remained lifetime, and finally drawing recommendations for PLIM Phase II program. The results of PLIM Phase I major components lifetime evaluations suggested that plant-specific PTS analysis[9] should be performed to ensure the RPV integrity beyond 34 operating years.

Based upon the results and suggestions of this study we are planning to carry out the PLIM Phase II program known as "Detailed Lifetime Evaluation and Aging Management Program". Through the Phase II program, the detailed evaluation and diagnosis of Kori Unit 1 will be performed, license renewal related documents will be prepared, and finally, the schedule of PLIM implementation will be determined.

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