18th International Conference on Structural Mechanics in Reactor Technology (SMiRT 18)
Beijing, China, August 7-12, 2005
SMiRT18-D01-5

INTEGRITY ASSESSMENT TECHNOLOGY FOR THE AGING MANAGEMENT

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ABSTRACT

As the number of nuclear plants rise, technology development for aging and integrity assessment for the safety of existing operating equipments has increased. Kori Unit 1, which is the oldest nuclear power plant (NPP) in Korea has been in operation since 1978. Also, ten other nuclear power plants have been operating more than 10 years. Therefore aging and/or integrity assessment become a necessity at these plants. In this regard, the Korean regulatory authority recently established an institutional process through revision to the atomic energy act to introduce PSR. This PSR considers, among other factors, the cumulative effects of plant aging and operating experience. Plant aging is one of the most important factors in the process of license renewal in the US and includes identification of the system, structure and components (SSCs) for aging management, assessment of aging effects and planning of aging management implementation program. Aging evaluation results is one of the procedural requirements that is utilized to approve life extension or renew an operating license of a nuclear power plant. This paper describes current aging assessment regulation and technology for the lifetime management in Korea.

Keywords: Periodic Safety Review (PSR), License Renewal, Aging Management, Integrity Assessment, Regulatory Strategy, Safety Assessment

1. INTRODUCTION

Since the first NPP started its commercial operation in 1978, Korea has built and currently operates twenty NPPs with some additional NPPs under construction. The current status of Korean NPPs is shown in table 1[1]. Ten NPPs have been operating for more than 10 years while Kori Unit 1 approaches the end of its design life in 2008. The first CANDU NPP, Wolsung Unit 1, is also approaching the end of its design life in 2013 based on a design life of 30 years.

As the number of aging plants rise, public concern over the safety of existing operating nuclear power plants (NPPs) has increased. Systematic and comprehensive operational safety assessment and plant life management are necessary to maintain a high level of safety, taking into account improvements in safety standards and practices, the cumulative effects of plant aging, operating experience, and the evolution of science and technology.

Operating license is issued without a fixed term in Korea. Consequently, design life in FSAR is understood as the basis for the duration of operation of a NPP. An institutional process for the life extension of a plant is not fixed yet and it should be established in the very near future, considering the remaining life span of Kori Unit 1 and the time required to perform the safety assessment.

PSR system was introduced and well established with sound legal basis for the comprehensive and systematic safety evaluation of operating plants. The results of PSR focused on aging can be utilized in the safety assessment for life extension of the plant. The Nuclear Safety Commission in Korea determined that the results of the PSR would be used for life extension of the NPPs.

This paper describes the current regulation and some requirements for operational safety and aging management, including the status in relation to PSR. This paper, also, introduces the aging management activities briefly. The strategy and method for the application of PSR results to the aging management of nuclear power plants and further plant life extension are also included.

2. CURRENT REGULATION

2.1 Legal Basis for Regulation

The laws and regulations that establish the basis for operating NPPs in Korea are [2]:

- Atomic Energy Act
- Enforcement Decree of Atomic Energy Act (Presidential Decree)
- Enforcement Regulation of Atomic Energy Act (Ministerial Ordinance)
- Regulation on Technical Standards of Nuclear Installations (Ministerial Ordinance)
- Regulation on Technical Standards of Radiation Protection (Ministerial Ordinance)
- Notice of the Minister of Science and Technology

2.2 Procedural Requirements

A construction permit is issued based on the radiological environmental report, the Preliminary Safety Analysis Report (PSAR), and the Quality Assurance (QA) program for design and construction. An early site approval for limited construction work on a proposed site is approved before the construction permit is issued. The operating license is issued based on the operational technical specifications (TS), the FSAR, the quality assurance program for operation, the radiological environmental report and the radiological emergency plan. It should be noted that prescriptive limit on license term is not given; however, the FSAR states the design life.

2.3 Design Requirements

- Quality standards: Design, testing, and inspection of SSCs are conducted to quality standards which commensurate with the importance of the safety functions.
- Environmental and dynamic effects design basis: SSCs are designed to accommodate the effects of, and to be compatible with the environmental conditions including the effects of aging.
- Equipment qualification: Equipment is installed after qualification of its functional capabilities by experience, analysis, test or a combination there of.
- Testability, monitorability, inspectability and maintainability: SSCs are designed to be tested, monitored, inspected, and maintained to ensure that their structural integrity, leak tightness, functional capability, and operability are maintained during the life of the nuclear power plant.

2.4 Inspection Requirements

- Pre-operational inspection: A document and field inspection of the installation and performance is conducted.
- Periodic inspection: The eleven facilities including the reactor are inspected at intervals of 20 months during the outage for refueling.
- Quality assurance inspection: Quality Assurance activities are conducted in accordance with the qualify assurance program.

2.5 Requirements on Safety Measures for Operation

- Conformance to Technical Specifications (TS): Monitor operations under limiting conditions to ensure compliance with the TS.
- Feedback from operating experience: To incorporate the lessons learned from the operating experience of plant facilities in accordance with the safety related criteria, procedures, and training programs.
- Testing, monitoring, inspection and maintenance of SSCs
  - ISI : Aging related degradation in material and performance of safety related SSCs shall be monitored, evaluated and managed through timely remedial actions.
  - IST: For major pumps and valves necessary for safe shutdown and reduction of accident situations, the performance and aging related degradation shall be monitored evaluated and managed.
  - RPV surveillance test: The degradation in material properties of reactor pressure vessel due to neutron irradiation shall be monitored, evaluated and managed.
  - Instruments calibration: Instruments and radiation detectors directly related to monitoring of states of nuclear facilities shall be periodically calibrated.

2.6 Corrective Actions and Enforcement

Nuclear facilities shall be placed in operation only after the integrity and performance are confirmed to be satisfactory through pre-operational inspections for each construction process. The reactor is allowed remain in operation if the performance of the NPP is confirmed to be satisfactory through periodic inspections. The regulatory body could order the operator to take corrective or complementary measures, such as suspension of use, repair, or modification of guidelines for operation to remedy inadequate performance of facilities and safety measures during operation. They could also request submittal of reports or documents on corrective activities, and order the operator to take corrective or complementary measures as a result of the inspections.

3. PERIODIC SAFETY REVIEW (PSR)

3.1 Current Progress

The Nuclear Safety Commission decided basic framework for the implementation of the PSR in December 1999. Public hearing was also held in December 1999. MOST issued “Implementing Guidelines for PSR” in May 2000 after deliberation with the Nuclear Safety Commission. KHNP (Korea Hydro and Nuclear Power Company) submitted the PSR Plan on 30 May 2000, which includes the plan for Kori Unit 1 to be completed by November 2002 and Wolsung Unit 1 by June 2003. The Atomic Energy Act was revised to adopt the PSR system in January 2001, including basic direction and framework for the implementation of the PSR. Detailed provisions including review scope, method, procedure, and technical standards are included in the Enforcement Decree (Presidential Decree) and the Enforcement Regulation (Ministerial Ordinance) of the Atomic Energy Act.

3.2 PSR Implementation

PSR is specified to be carried out every 10 years after issuance of an operating license. The operator (KHNP) of NPPs has the responsibility for performing the PSR. MOST specifies PSR requirements and reviews the PSR results. Review scope is based on 11 safety factors suggested by IAEA in Safety Series No. 50-SG-O12 [3], and detailed scope may vary depending on plant age. PSR for twin plants having a single FSAR is assembled together into a single report but separately considers the aging of SSCs and the physical status of each plant. Once the PSR report is submitted, MOST/KINS (Korea Institute of Nuclear Safety) reviews the PSR results and prepares safety evaluation report (SER) with identification of safety issues. The implementing process of PSR is shown in figure 1.

3.3 Technical Requirements on Aging Assessment

Aging review is focused on the following issues to ensure that plant aging is being effectively managed so that required safety margins are maintained and adequate aging management program is in place for the safe operation of the plant:

- For those SSCs within scope, identify structures and components requiring an aging evaluation
- Identify and assess aging effects
- Perform detailed aging analysis
- Demonstrate that the effects of aging are managed

- Construct a program using the 10 NRC AMP elements

4. AGING MANAGEMENT ACTIVITIES

4.1 Experience in Aging Management

Based on a committed effort over the last twenty years, the following activities are implemented or are being considered to cope with the aging problems.

- Kori Unit 1 : Plant Lifetime Management Study, Phase I and Phase II
- Kori Unit 1 : Periodic Safety Review
- Kori Unit 2 : Periodic Safety Review
- Kori Units 3, 4 : Periodic Safety Review
- Wolsung Unit 1 : Plant Lifetime Management Study, Phase I and Phase II
- Wolsung Unit 1 : Periodic Safety Review
- Younggwang Units 1, 2 : Periodic Safety Review

4.2 Aging Evaluation Methodology

Based on the technical guidelines on the aging in PSR scheme, the following steps are generally applied to assess the effects and to prepare the management plan for plant aging.

- SSCs Screening: Identify SSCs within the scope of each aging management activity (PSR or PLIM etc.). Generally, all the safety related SSCs are selected and grouped to be reasonably managed. SSCs screening criteria in PSR is shown in figure 2.
- Aging Mechanisms Identification: Identify all the possible aging mechanisms for the screened and selected SSCs. All of the aging mechanisms defined in ASME Section III, Appendix W are reviewed whether it is possible for the each SSC/commodity group or not.
- Aging Evaluation: Evaluate all the aging effects from possible mechanisms for the SSC/commodity group components. Then compare them to the available acceptance criteria. All the evaluation methodology is based on the latest technology. Aging evaluation method of passive components in PSR is shown in figure 3.
- Site Walkdown: Compare the results of the aging evaluation to the plant via walkdown process before arriving at the final conclusion and management plan.
- Aging Management Plan: Four types of aging management programs are implemented; prevention, mitigation, condition monitoring and performance monitoring

4.3 Integrity Assessment Technology

An integrated methodology for aging evaluation and management in PSR that is applicable to all operating nuclear plants in Korea has been developed. Technically, there are three steps; identification of aging mechanisms, aging evaluation, and establishment of optimal aging management program.

Aging Mechanism Identification

Among the 17 potential aging mechanisms specified in ASME Section III Appendix W, significant aging mechanisms were identified at the component level on the basis of review of material/environmental conditions and operating experience. For instance, significant aging mechanisms such as IASCC, thermal embrittlement, corrosion, and fatigue for major components have been identified.

Aging Evaluation

The aging evaluation consists of following activities; review of design documents, review of operating condition and maintenance history and determination of whether the evaluation results satisfy operating acceptance criteria. The results of aging evaluation are connected to activities of selecting aging management programs for continued operation of the major SSCs. The examples of aging evaluation results are as follows:

- IASCC of Reactor Internals:

The effects of IASCC are significant for reactor internals that are fabricated from susceptible materials and subjected to a fluence level above $5x10^{21}$ n/cm^2 (E>1MeV). Failure history in the other plants showed that IASCC was a major contributor to baffle former bolts cracking of the reactor internals. As a result of aging evaluations, it is expected that core barrel inside and baffle former assembly could exceed the IASCC threshold value during 40 operating years. Theses results are shown in figure 4. Therefore, enhanced visual and ultrasonic inspection has been recommended to be performed periodically.

- Thermal Embrittlement of Class 1 Piping:

Cast austenitic stainless steel (CASS) piping which contain ferrite content above 20% (volume) has been known to experience thermal embrittlement when exposed to reactor operating temperatures of 280-320^{o}C. Reactor coolant loop piping of some plants is made of CF8A CASS. There are four straight pipes and five elbows in the reactor coolant loops. The ferrite content was obtained from the chemical compositions of certified material test report using Aubrey equation. The J-R curve is predicted from saturated Charpy impact energy and ferrite content. Ferrite content is 16.23% in elbow of the hot leg that is manufactured by static cast method. The value of $J_{2.5}$=255kJ/m^2 (1450in-lb/in^2) could be used to differentiate between CASS materials that is non-susceptible and potentially susceptible to the thermal embrittlement[4]. In figure 5, $J_{2.5}$ is 274kJ/m^2 that is not smaller than 255kJ/m^2. Consequently, reactor coolant loop piping has adequate fracture toughness for thermal embrittlement.

- Corrosion of Pressure Vessels:

In order to evaluate general corrosion, present wall thickness was calculated based on the corrosion rate depending on specific service condition. Also it was directly measured by in-situ UT and compared with minimum required thickness. The results show that the pressure vessels can be operated up to 60 years with adequate safety margin.

Aging Management Program

In order to determine the condition of SSCs, the various programs of the other plants such as in-service inspection program and maintenance programs etc. should be reviewed to detect and mitigate the effects of degradation. Also, various aging management programs of other plants in Korea and overseas plants should be reviewed and compared. When aging management programs are not effective in predicting the aging effects of SSCs, they need to be modified or new aging management programs should be developed. From the above procedure, modified and new aging management programs (AMP) including existing AMPs were preliminarily derived as shown in table 2.

5. APPLICATION OF PSR RESULTS

Possible models to be considered for life extension are license renewal, official approval for continued operation beyond design life, and continued operation with PSR results without any procedural requirements. For any model, PSR results can provide useful information for the decision of continued operation.

License renewal rule, 10CFR54 [5] specify the requirements for renewal of operating licenses for nuclear power plant. According to the rule, the requirements are composed of four areas such as general information, technical information, technical specification and environmental Information. With respect to technical area, licensee should perform IPA (Integrated Plant Assessment) and thus identify SSCs subject to AMR (Aging Management Review) and demonstrate aging effects will be adequately managed during the extended operation.

This IPA is comparable to actual condition of SSCs and aging of total 11 PSR safety factors. The objective of review of actual condition of SSCs is to determine the actual condition of structures, system and components (SSCs) important to safety and whether it is adequate to meet its design requirements. In addition, the review should confirm that the condition of SSCs is properly documented. Also, the objective of the review of aging is to determine whether aging in a nuclear power plant is being effectively managed so that required safety functions are maintained, and whether an effective management program is in place for future plant operation. Consequently, these two safety factors of PSR can cover IPA of license renewal application. CLB change, evaluation of TLAAs, and FSAR supplement which are part of requirement of license renewal rule can also be incorporated in the PSR results. Correlations between format for license renewal application and PSR implementation are shown in table 3.

6. CONCLUSIONS

Periodic safety review system was introduced and well established with sound technical basis for the comprehensive and systematic evaluation of the safety of operating plants. Institutional processes for the plant life extension will be established in the very near future, considering the remaining design life span of Kori Unit 1 and the time required to perform the safety assessment. Aging assessment in PSR is considered to be used directly for further purposes. As a result, PSR could be utilized in determining the plant life extension by adding some other factors like environmental effects as a complement to the FSAR (TS) in safety-related documents.

REFERENCES

1. MOST, (2004), White Paper on Nuclear Safety
2. MOST, (2001), Enforcement Regulation of Atomic Energy Act
3. IAEA, (2001), Periodic Safety Review of Nuclear Power Plants, IAEA Safety Series No. 50-SG-O12, 2001
4. USNRC, (2001), "Generic Aging Lessons Learned (GALL)", NUREG-1801
5. DOE, (2004), Requirements for Renewal of Operating Licenses for Nuclear Power Plants (10CFR54)
6. IAEA, (1999), Safe Management of the Operating Lifetimes of Nuclear Power Plants, IAEA INSAG-14

Table 1 Status of operating nuclear power plants in Korea

Plants	Reactor Type	Thermal Power (MWt)	TBN Power (MWe)	Operation Permission	Commercial Operation
Kori-1	PWR	1724	587	1972. 5.31	1978. 4. 29
Kori-2	PWR	1876	650	1983. 8.10	1983. 7.25
Wolsong-1	PHWR	2064	688	1978. 2.15	1983. 4.22
Kori-3	PWR	2775	950	1984. 9.29	1985. 9.30
Kori-4	PWR	2775	950	1985. 8. 7	1986. 4.29
Yonggwang-1	PWR	2775	950	1985.12.23	1986. 8.25
Yonggwang-2	PWR	2775	950	1986. 9.12	1987. 6.10
Uljin-1	PWR	2775	950	1987.12.23	1988. 9.10
Uljin-2	PWR	2775	950	1988.12.29	1989. 9.30
Yonggwang-3	PWR	2815	1000	1994. 9. 9	1995. 3.31
Yonggwang-4	PWR	2815	1000	1995. 6. 2	1996. 1. 1
Wolsong-2	PHWR	2061	700	1996.11. 2	1997. 7. 1
Uljin-3	PWR	2815	1000	1997.11. 8	1998. 8.11
Wolsong-3	PHWR	2061	700	1997.12.30	1998. 7. 1
Wolsong-4	PHWR	2061	700	1999. 2. 8	1999.10. 1
Uljin-4	PWR	2815	1000	1998.10.29	1999.12.31
Yonggwang-5	PWR	2815	1000	2001.10.24	2002. 5.21
Yonggwang-6	PWR	2815	1000	2002. 7.31.	2002.12.24
Uljin-5	PWR	2815	1000	2003. 10	2004. 7.29
Uljin-6	PWR	2815	1000	2004. 11	2005.4

(1) Issue date of construction permit

Table 2 Example of AMPs for some components

Components	Existing AMPs	Modified/New AMPs
Reactor Internals	ISI Program	Augmented ISI
	Chemistry Control	Fatigue Monitoring
Class 1 Piping	Chemistry Control	BACI Program
	ISI Program	Augmented ISI
Pressure Vessels	ISI Program	BACI* Program
	Chemistry Control	

* BACI : Boric Acid Corrosion Inspection

Table 3 Correlations between LR and PSR

<table>
<tr><th colspan="3">Requirements for Renewal of Operating Licenses for NPPs(10CFR54)</th><th>Requirements for PSR
(Atomic Energy Act, 23.3)</th></tr>
<tr><td colspan="3">Generation Information(§54.17 & 54.19)</td><td>Plant design</td></tr>
<tr><td rowspan="5">Technical Information (§54.21)</td><td rowspan="2">IPA (§54.21(a))</td><td>Identify SSCs subject to AMR</td><td>-SSCs screening
-Actual condition of SSCs</td></tr>
<tr><td>Demonstrate aging effects will be adequately managed</td><td>-Aging mechanism identification
-Aging evaluation
-Aging management program</td></tr>
<tr><td colspan="2">CLB changes (§54.21(b))</td><td>-Review of Technical basis</td></tr>
<tr><td colspan="2">Evaluation of TLAAs (§54.21(c))</td><td>-Aging evaluation
-Enhanced PSR</td></tr>
<tr><td colspan="2">FSAR supplement (§54.21(d))</td><td>-FSAR changes</td></tr>
<tr><td colspan="3">Technical Specifications(§54.22)</td><td>-Review of Technical basis</td></tr>
<tr><td colspan="3">Environmental Information(§54.23)</td><td>-Environment
(Radiological impact on environment)</td></tr>
</table>

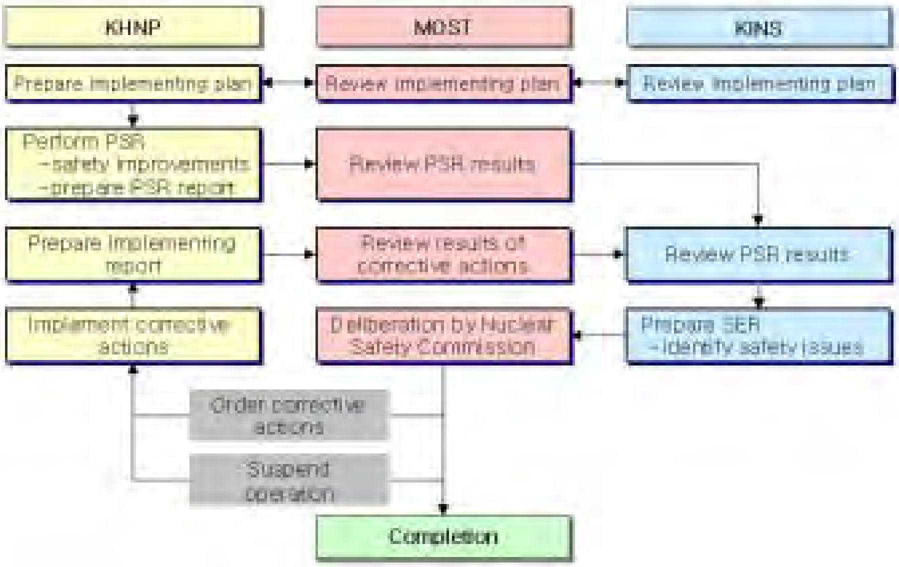

Fig. 1 PSR implementing process

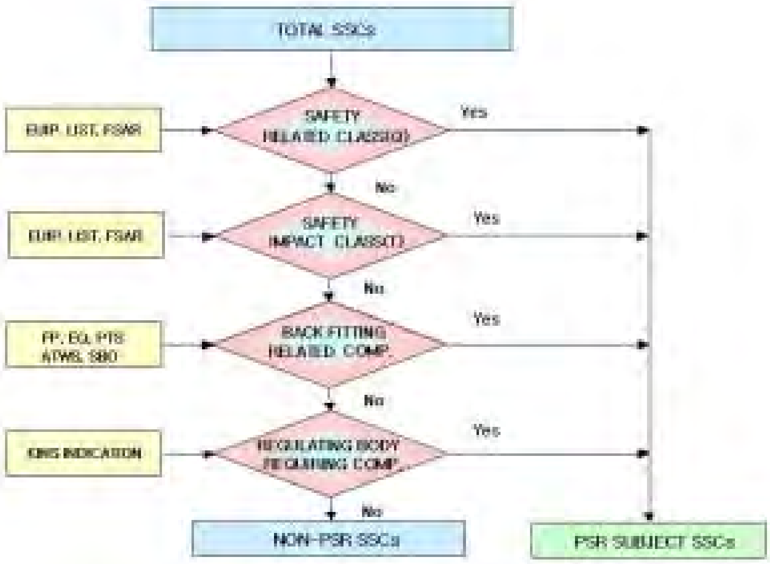

Fig. 2 SSCs screening criteria

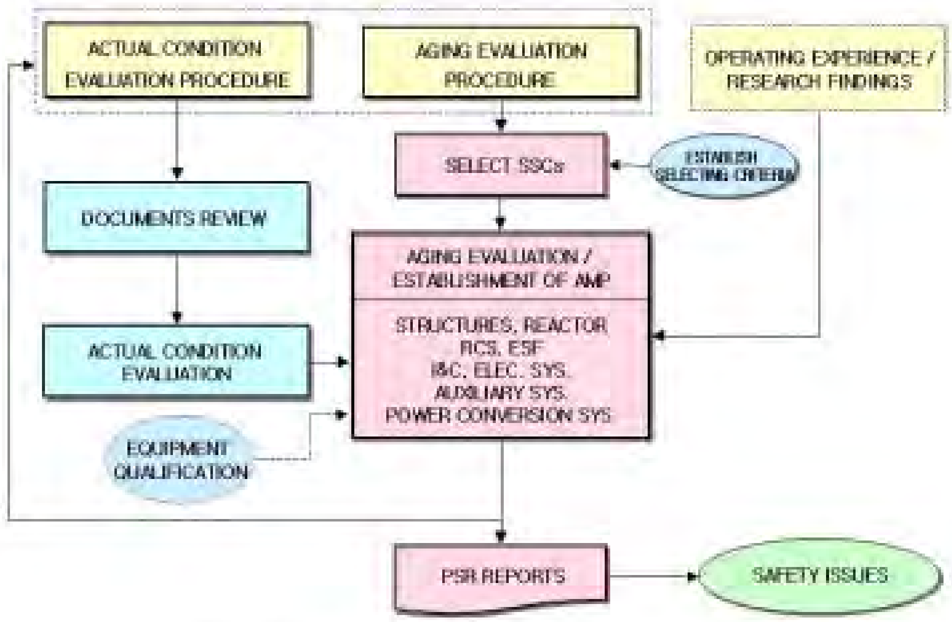

Fig. 3 Evaluation method of passive component in PSR

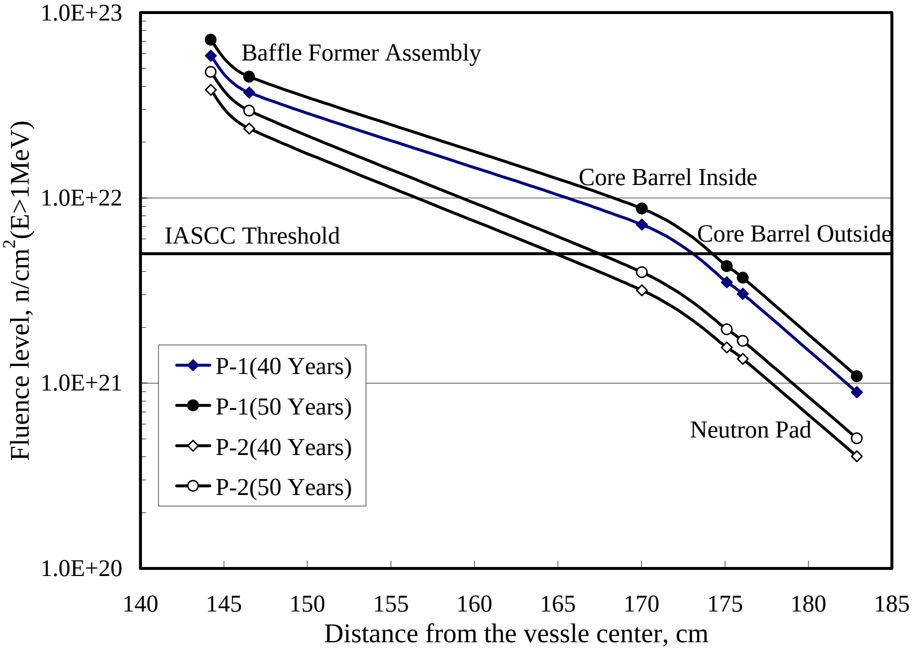

Fig. 4. Predicted fluence level of reactor core

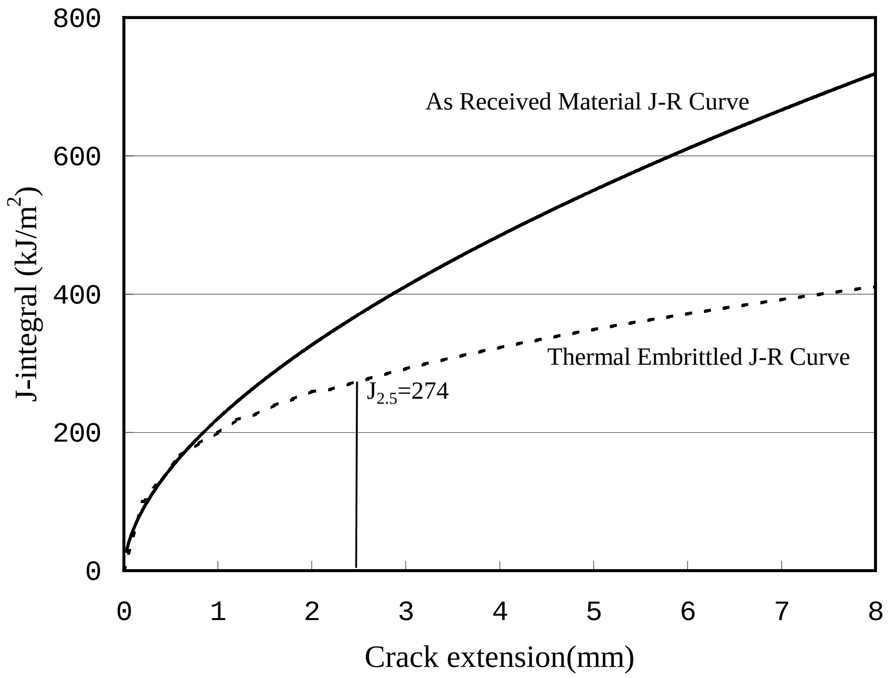

Fig. 5. As received and embrittled J-R curve