TECHNICAL EVALUATION OF THE CONTINUED OPERATION OF NPP

TAE-RYONG KIM*, TAE-EUN JIN[1]
*Korea Electric Power Research Institute, KEPCO
65 Munji-Ro, Yuseong-gu, Daejeon, Korea
[1]Korea Power Engineering Company
360-9 Mabuk-Ri, Guseong-eup, Yongin-si, Gyeonggi-do, Korea
*Corresponding author. E-mail : trkim@kepri.re.kr

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Recently, the long-term operation of a nuclear power plant beyond its licensed term has become a worldwide trend as long as the safety of the plant is maintained in the extended period. Kori Unit 1, the oldest PWR in Korea, is the foremost example of this type of long-term operation in Korea. Comprehensive technical evaluation of the long-term operation of this plant was completed to confirm the overall safety of the plant. The technical evaluation included a review of PSR results, an assessment on aging management programs and time limited aging analyses, and a statement of radiological impact on the environment. Based on all of the results of the technical evaluation activities, Kori Unit 1 was approved to operate for an additional 10 years beyond its original design life of 30 years.

KEYWORDS : Periodic Safety Review, Continued Operation, Safety Factors, Aging Management Program, Time Limited Aging Analyse

1. INTRODUCTION

Recently, the long-term operation of a nuclear power plant (NPP) has become a worldwide trend, partly because of the high oil prices and the greenhouse effect. This trend is viable as long as safety of the plant is maintained in its extended operation period. Long-term operation of a NPP is defined as operation beyond its established timeframe (e.g. licensed term, design life) according to the relevant International Atomic Energy Agency (IAEA) document [1]. It is sometimes expressed in other terminology such as; 'plant life extension' in some conferences, 'license renewal' in USA, 'continued operation' in Korea [2]. To acquire licensed approval of continued operation, prescribed regulatory requirements must be satisfied. Notice 2005-31 of the Ministry of Science and Technology (MOST), the nuclear regulatory body in Korea, prescribes the licensee with the items that they must submit for continued operation [3]. These include the reinforced periodic safety review (PSR) report, aging management programs with time-limited aging analysis and an assessment report of the radiological impact on the affected environments. In addition, a detailed life assessment of the SSC and the implementation plan of the operational experience of other plants and R&D findings should be included.

With all the requirements satisfied and plant safety during the continued operation period confirmed, Kori Unit 1 has acquired licensed approval of its continued operation for an additional 10 years beyond its original design life. This paper presents a summary of the technical evaluation on the safety assessment and aging management for the continued operation of Kori Unit 1.

2. TECHNICAL EVALUATION OF CONTINUED OPERATION

Typically, research regarding technical evaluations related to continued operation of a NPP includes aging and degradation assessments. These have been done by industries, research institutes and universities from the late 1980's in Korea. In the earlier stages, this research focused aging management to reduce unplanned outages rather than on aging assessments. Since then, various types of aging phenomena of critical components and sub-components such as neutron irradiated embrittlement and fatigue, corrosion have been investigating by many universities, research institutes and industries including the Korea Electric Power Research Institute, the Korea Atomic Energy Research Institute and Korea Power Engineering Company.

Based on the investigation of the major aging mechanisms, a project for PLiM (Phase I) was initiated in 1993 for comprehensive and systematic study of the aging assessment and management of the overall systems, structures and components (SSC) of Kori Unit 1, the oldest PWR in Korea. In the study [4], the feasibility and the licensibility of continued operation beyond the original lifetime of 30 years was evaluated in terms of technological and economical concerns. From a technological standpoint, SSCs concerning life management were classified and critical components were selected for a detailed lifetime evaluation.

For instance, the reactor pressure vessel was selected as a critical component due to its aging mechanism of pressurized thermal shock (PTS). The detailed lifetime evaluation including the selection of the initial event, a thermal-hydraulic system analysis, a heat flow and mixing analysis, and a probabilistic fracture mechanics analysis was carried out to test the overall structural integrity according to Regulatory Guide 1.154 of USNRC [5].

With the results and the assessment technologies developed during Phase I, the PLiM project (Phase II) was started in 1998. The remaining life of all SSCs in Kori Unit 1 was evaluated [6], which led to the adoption of the entire process of PLiM for the Korean nuclear industry.

2.1 Periodic Safety Review (PSR)

PSR is defined as a systematic re-assessment of the overall safety of a NPP [7]. The objective of a PSR is to determine by means of a comprehensive assessment of an existing NPP the extent to which the plant confirms to current international safety standards and practices, the extent to which the licensing basis remains valid, and the adequacy of the arrangements that are in place to maintain plant safety until next PSR or until end of the plant lifetime. For the scope and contents of a PSR, 11 safety factors recommended by the IAEA were used, as listed in Table 1. Typical PSR procedure is described in Fig. 1. The starting point of a PSR is the agreement between a NPP owner/operator and the regulating body. Once they have agreed upon the scope and requirements, safety factors are reviewed in detail.

Since the PSR was adopted as a regulatory tool in 1999, the first PSR on Kori Unit 1 according to Article 23-3 of the amended Atomic Energy Act was completed in 2002. As a result of the PSR, it was confirmed that Kori Unit 1 would be safely operable for 10 years, until the time of the next PSR [8]. To satisfy the regulatory requirements for continued operation, however, a reinforced PSR began in 2005.

The main point in the assessment of the safety factors of aging management is to determine whether SSC aging is being effectively managed and whether an effective aging management program (AMP) is in place. A review of the aging management program was performed according to the procedures described in NUREG-1800 [9], NUREG - 1801 [10] and NEI 95-10 [11]. The implementation status or plan for the 87 safety improvement items identified in the previous PSR was also reviewed.

Table 1. Safety factors in a PSR

No.	Safety Factor
1	Actual condition of the NPP
2	Safety analysis
3	Equipment qualification
4	Management of aging
5	Safety performance
6	Use of experience from other plants and research findings
7	Procedures
8	Organization and administration
9	Human factors
10	Emergency planning
11	Radiological impact on environment

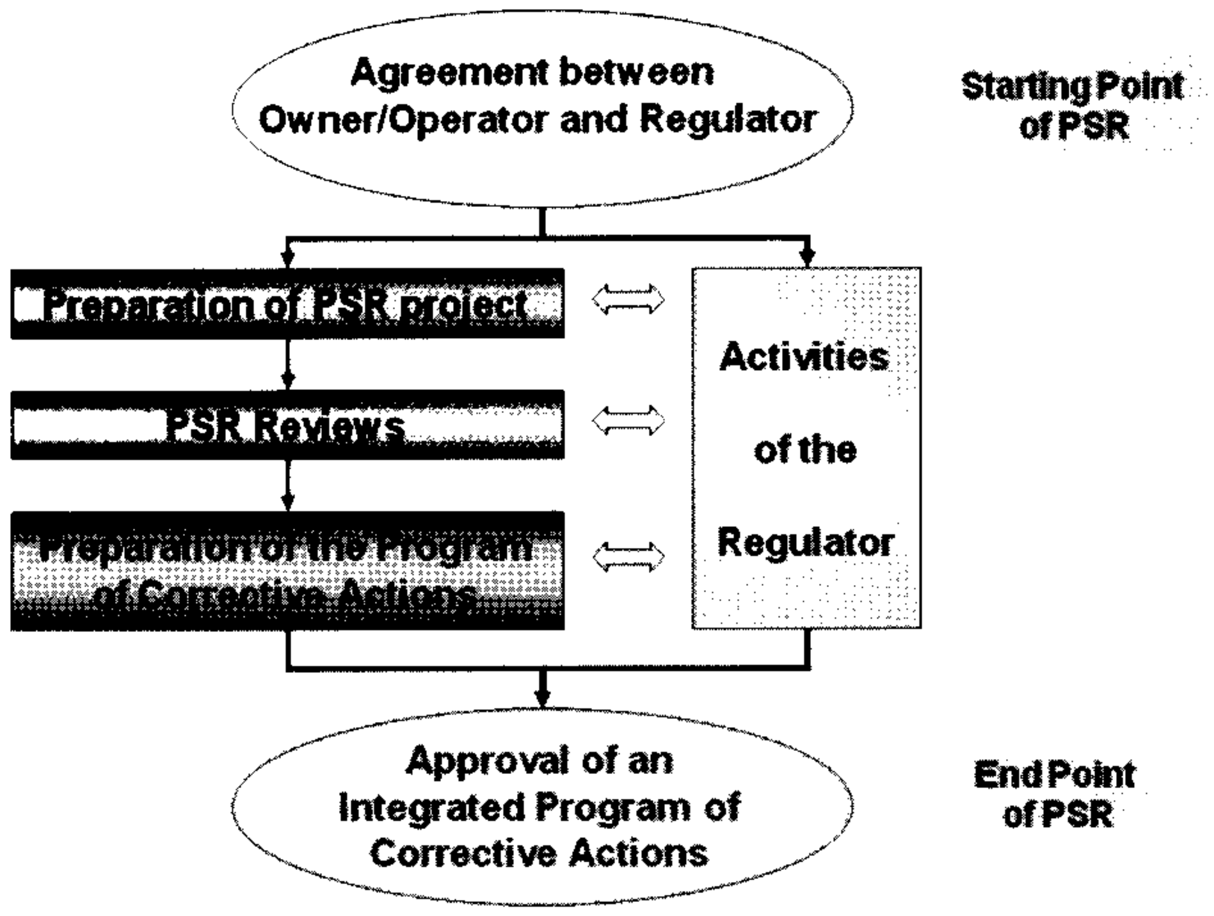

Fig. 1. Typical PSR Procedure

The appropriateness of the major aging mechanisms of critical components was also reviewed. For instance, aging mechanisms of irradiation-assisted stress corrosion cracking, embrittlement, swelling and stress relaxation were reviewed for reactor internals. These aging mechanisms are highly relevant to the total fluence that the reactor internals bear. The fluence level used for the review of the aging mechanism was calculated at the beltline region, where it was expected to have the maximum value. Fig. 2 shows a 1/8 model of the reactor core used for the fluence calculation.

For the assessment of neutron-irradiated embrittlement of the reactor pressure vessel, the Charpy upper shelf energy was reviewed per the requirement in Regulatory Guide 1.161 [12]. It was found to have a sufficient safety

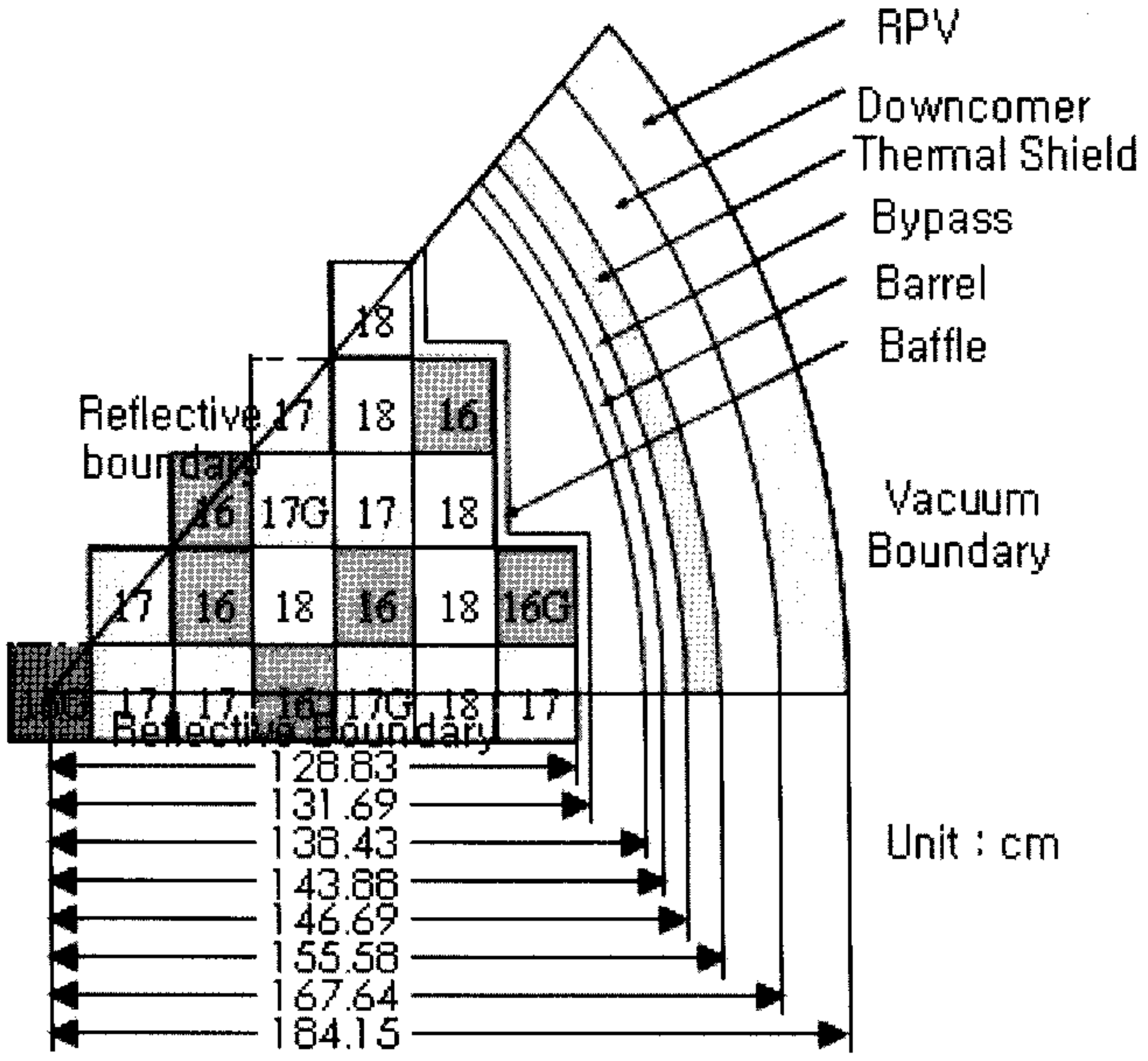

Fig. 2. Cross-Section of a 1/8 Model for Fluence Calculation

margin based on a fracture toughness test of a surveillance specimen that had been exposed to radiation for 32 effective full power years (equivalent to 40 calendar years). PTS was also assessed using specific data from Kori Unit 1 with the master curve methodology. It was found that the acceptance criteria (below 300°F) were satisfied, as shown in Fig. 3. A pressure-temperature limit operation curve for PTS was set up with the measurement error per ASME Sec. XI, App. G [13]. From the result of a detailed evaluation, the reactor pressure vessel of Kori Unit 1 was found to be operable in principle for more than 40 years. Core damage frequency due to PTS of the reactor pressure vessel was also found to satisfy the acceptance criteria of 5.0×10^{-6}/Rx-yr described in Regulatory Guide 1.154, based on an even 60 year operation of Kori Unit 1.

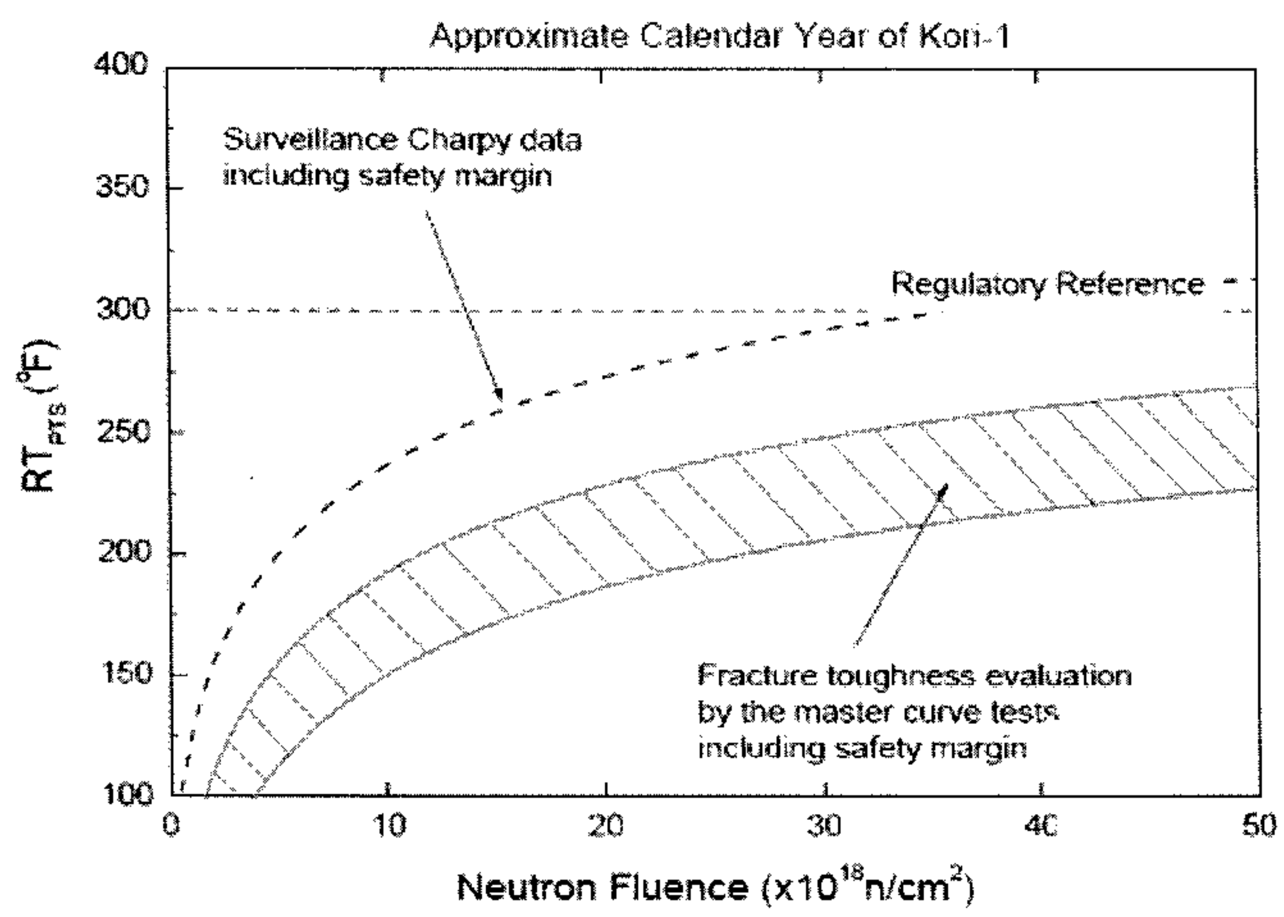

Fig. 3. RT_{PTS} Assessment Result of Welds in the Beltline Region (Kori Unit 1)

For the assessment of general corrosion or flow-accelerated corrosion in carbon steel pressure vessels and in secondary system piping, the wall thicknesses of the components were measured by ultrasonic testing, as shown in Fig. 4. The remaining lifetimes of the components were then assessed by comparing the measured thickness with the required minimum thickness per the ASME Code. It was found that the carbon steel components properly reflected their operation experience and research findings , and it was concluded that the safety issue related with aging degradation was properly managed.

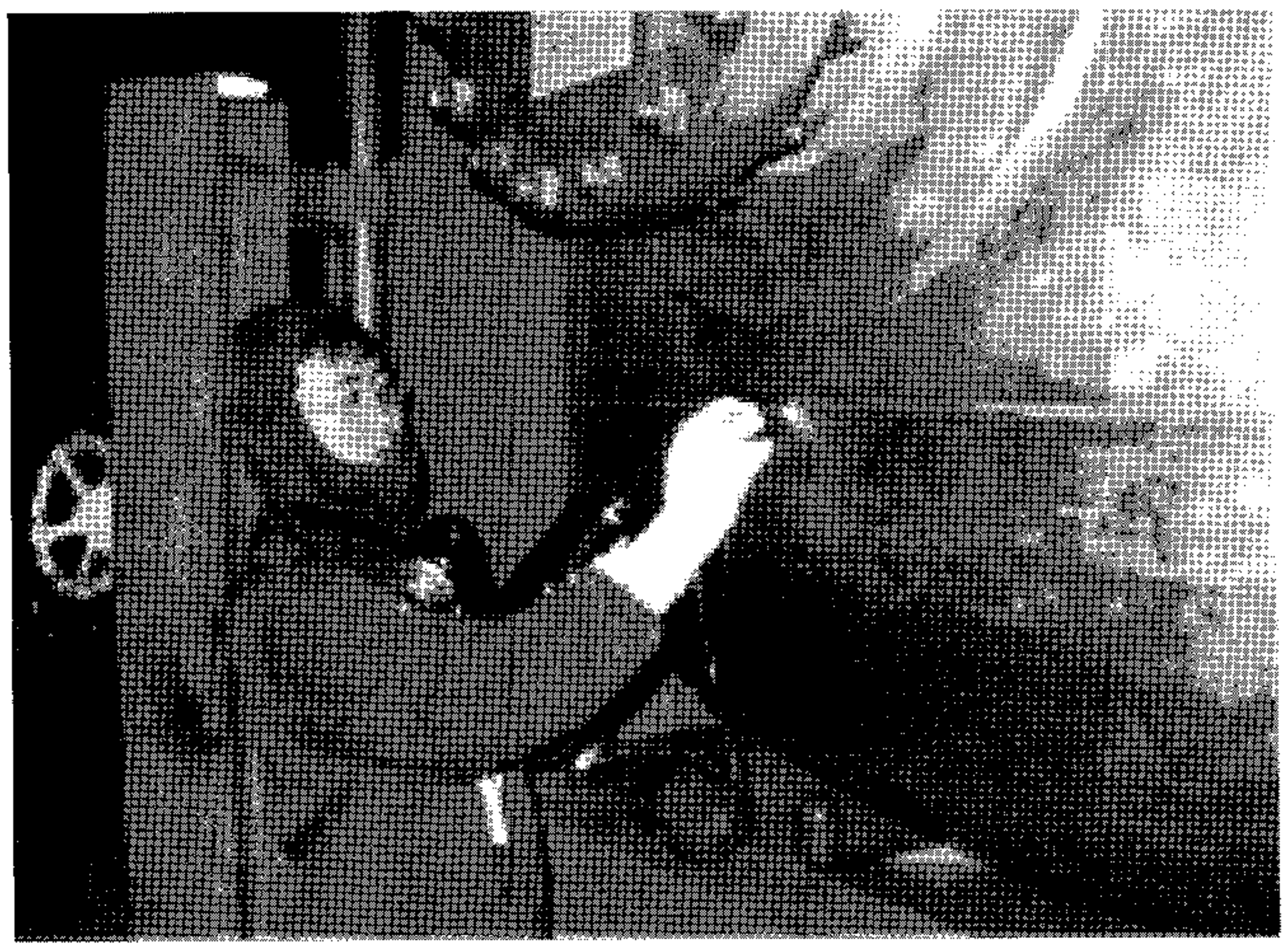

Fig. 4. Wall Thickness Measurement by Ultrasonic Testing

In the assessment of the environmental qualification of the equipment, Kori Unit 1 was found to be insufficient according to the recent standard IEEE323-1983[14], as it was designed according to an older version. The corrective actions recommended to enhance plant safety were completed before the approval of continued operation; work was done that included preparing a qualification plan and an equipment list subject to qualification, an accident analysis for cases of main steam line breaks and small loss-of-coolant accidents, and acquisition of the qualification test and evaluation documents. Furthermore, environmental temperature data surrounding the cables were acquired periodically for 18 months with a measuring device attached to the positions of concern, as shown in Fig. 5.

In addition to the document reviews and re-analyses, a substantial in-service inspection data review and site walkdown were simultaneously conducted for the assessment of the safety factors.

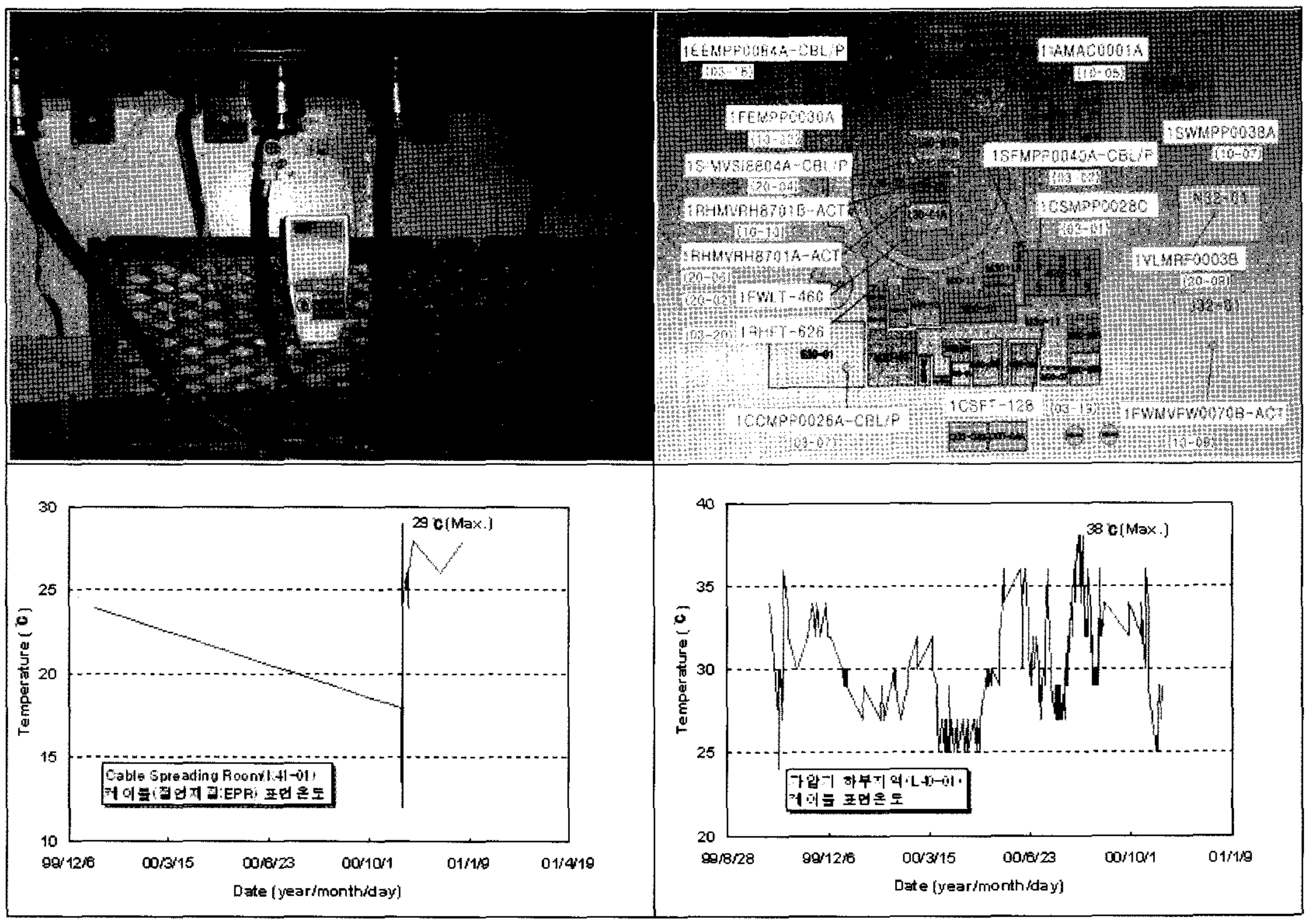

Fig. 5. Temperature Measuring Device Mounted and the Measured Data

From the results of the reinforced PSR, it was verified that Kori Unit 1 had been maintained in good physical and operational condition and that a sufficient safety margin would be ensured for the continued operation period.

2.2 Review of the Aging Management Program (AMP)

As mentioned earlier, another regulatory requirement for continued operation of a NPP is a review of the aging management program (AMP) that is used in the plant. For the effective management of SSC aging, an AMP should be in place. Review of the AMP includes an assessment of the present monitoring/management procedures and maintenance activities to confirm that the existing AMP simply and effectively manages the range of aging phenomena; otherwise, a new AMP should be prepared.

Each AMP consists of 10 attributes: the scope of the AMP, preventive actions, parameters monitored or inspected, the detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmatory procedures, administrative control, and operational experience.

Fig. 6 shows the typical procedure for the preparation of the AMP. The first step of the review of an AMP or preparation of a new AMP is the screening of critical SSC factors. Screening of critical SSCs for aging management was performed according to 10CFR54.21 [15], NUREG-1800 and NUREG-1801. Accordingly, 25 systems and 7 structures were selected as critical SSCs from the three categories of mechanical equipment, civil structures and electrical equipment.

A total of 39 AMPs were selected for review in Kori Unit 1 based on NUREG-1801. These included an in-service inspection of safety class 1,2,3, a one-time inspection, reactor vessel surveillance, loose part monitoring, neutron noise monitoring, bus ducts, fuse holders, electrical cables and connections not subject to environmental qualification requirements.

It was found that most of the existing AMPs can properly and effectively manage the aging phenomena shown in the critical SSC. However, for some of them, e.g., the one-time inspection and the selective leaching of materials, revision was necessary. Accordingly, 13 AMPs in total were revised or were newly prepared. Particularly, implementation of an AMP for one-time inspections and selective leaching of materials was successfully completed by the active cooperation of the plant personnel. These AMPs are the leading AMPs in the world, which implies that Korea acquired experience and proficiency regarding these AMPs prior to other NPPs overseas that remain at the implementation planning stage.

In addition, an AMP for nickel-alloy nozzles and

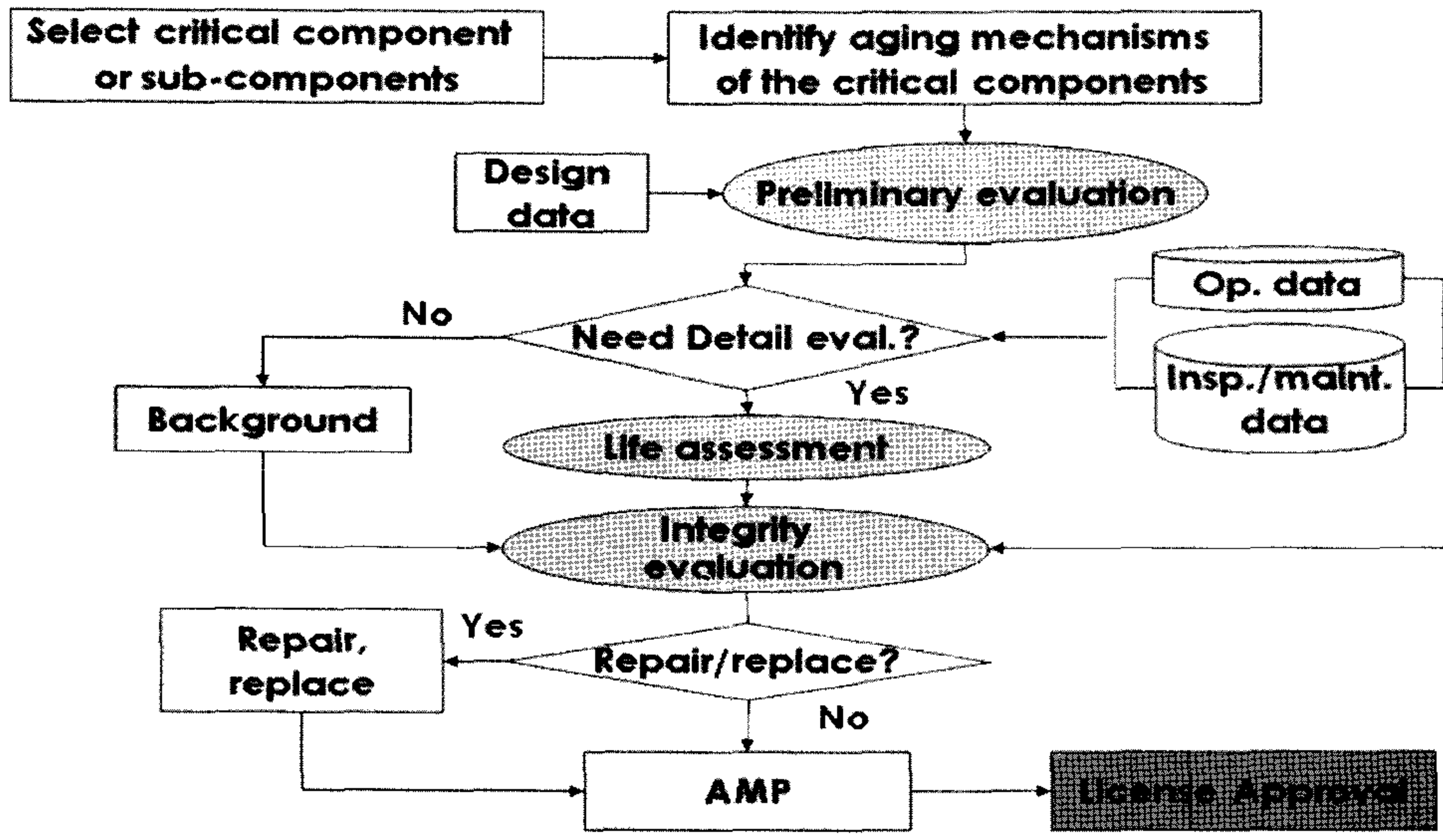

Fig. 6 Typical AMP Preparation Procedure

penetrations was established to ensure that integrity of the nickel-alloy nozzles and penetrations is maintained against primary water stress corrosion cracking (PWSCC), which recently has become an important issue worldwide. This AMP includes a survey of all Alloy 600 dissimilar metal welds in Kori Unit 1 and an assessment of the integrity of components highly susceptible to PWSCC. In the case of CRDM nozzle penetration in the reactor upper head, which is known to be highly susceptible to PWSCC, re-analysis of the stress distribution was done, as shown in Fig. 7, to ensure structural integrity.

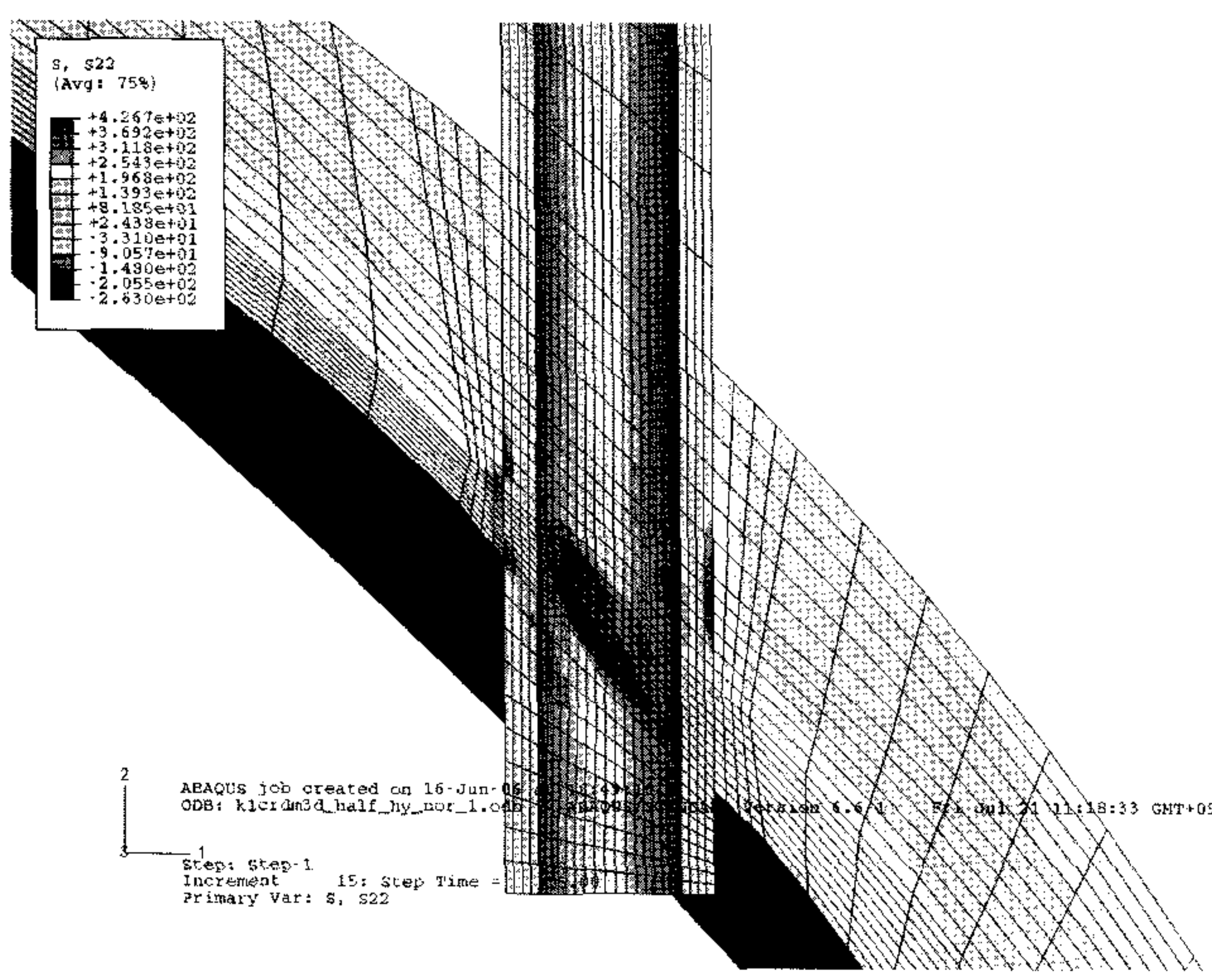

Fig. 7 Stress Distribution Due to Welding and Operating Conditions at the Point of CRDM Penetration in the Reactor Upper Head

From results of the review of AMPs and the preparation of the new AMP, it was confirmed that existing and/or enhanced aging management programs can effectively manage SSC aging during the continued operation period.

With the PSR and PLiM experience regarding Kori Unit 1, it can be said that safety assessment technology was established for the long-term operation of nuclear power plants.

2.3 Time-Limited Aging Analyses (TLAA)

MOST Notice 2005-31 requires the licensee to submit a time-limited aging analysis (TLAA) report of critical components related to safety for the continued operation of a NPP. TLAA can be defined as calculations and/or analyses done with time-limited assumptions defined by the operating term, for example, 50 or 60 years, to confirm the safety of the systems, structures, and components within the scope of continued operation. The confirmation can be achieved by showing that TLAA remains valid during the continued operation period, or that the integrity of the SSC is maintained through an analysis at the end of the extended period, or that aging effects on the inherent functions of SSC will be properly managed.

In case of Kori Unit 1, four (4) general TLAA items and six (6) specific TLAA items were identified and reviewed. The general TLAA items for Kori Unit 1 consisted of a reactor vessel neutron embrittlement analysis, metal fatigue analysis, environmental qualification of equipment, and containment liner plate, metal containment and penetration fatigue analyses. The specific TLAA items for Kori Unit 1 included wear of the neutron flux detector tube, the crane load cycle limit, the reactor coolant pump flywheel, the spent fuel pool liner, component and piping subsurface indications, and thermal aging embrittlement of the cast austenitic stainless steel.

In the TLAA of the metal fatigue of the main components (reactor vessel, control rod drive mechanism, reactor internals, reactor coolant pump, steam generator, reactor coolant system piping, safety injection tank), the cumulative usage factor was found to be less than 1.0 (acceptance criteria), even with the occurrence of transients conservatively predicted at the end of the extended operation of 40 years. It was subsequently confirmed that the integrity of the main components would be maintained until the end of continued operation of Kori Unit 1. Furthermore, a fatigue monitoring system was installed in Kori Unit 1 to ensure operational safety for another 10 years of operation, as shown in Fig. 8.

In the TLAA of metal containment and penetration fatigue analysis, it was found that the exemption requirements of ASME NE3221.5(d) were satisfied. Thus, the steel containment vessel was confirmed to maintain integrity in terms of fatigue for the period of continued operation. In the case of the main steam and feed water piping penetration bellows, it was also confirmed that the increase in transient occurrences expected during the continued operation term is less than that of the design transients; hence, they would maintain integrity regarding fatigue during the period of continued operation.

3. THIRD-PARTY PEER REVIEW OF THE TECHNICAL EVALUATION FOR CONTINUED OPERATION

To create an enhanced safety review report for Kori Unit 1 highly reliable and impartial, the plant operator, Korea Hydro/Nuclear Power Company (KHNP), and other organizations involved in the review engaged in activities such as managerial and progress check meetings, discussions with experts, seminars, and paper presentations. In the overall discussion of the safety review report, particularly, more than 50 experts from the domestic nuclear industry came together and twice conducted an in-depth review of the report. The safety review results of Kori Unit 1 were also presented in domestic and/or international conferences for the public verification.

With all activities completed, the enhanced safety review report for continued operation was submitted to MOST on June 16, 2006. During the investigation of the report for 18 months by the regulatory body, nearly 1,000 technical questions from the regulators were completely answered. In addition, a site walkdown of Kori Unit 1 was also performed by regulators to confirm not only that the plant has properly implemented the technical criteria for continued operation according to Article 9 of MOST Regulatory Guide 2005-31, but also to verify that the corrective actions specified in the safety review were properly implemented.

MOST decided to rely on IAEA for the third-party peer review to ensure that the enhanced safety review report

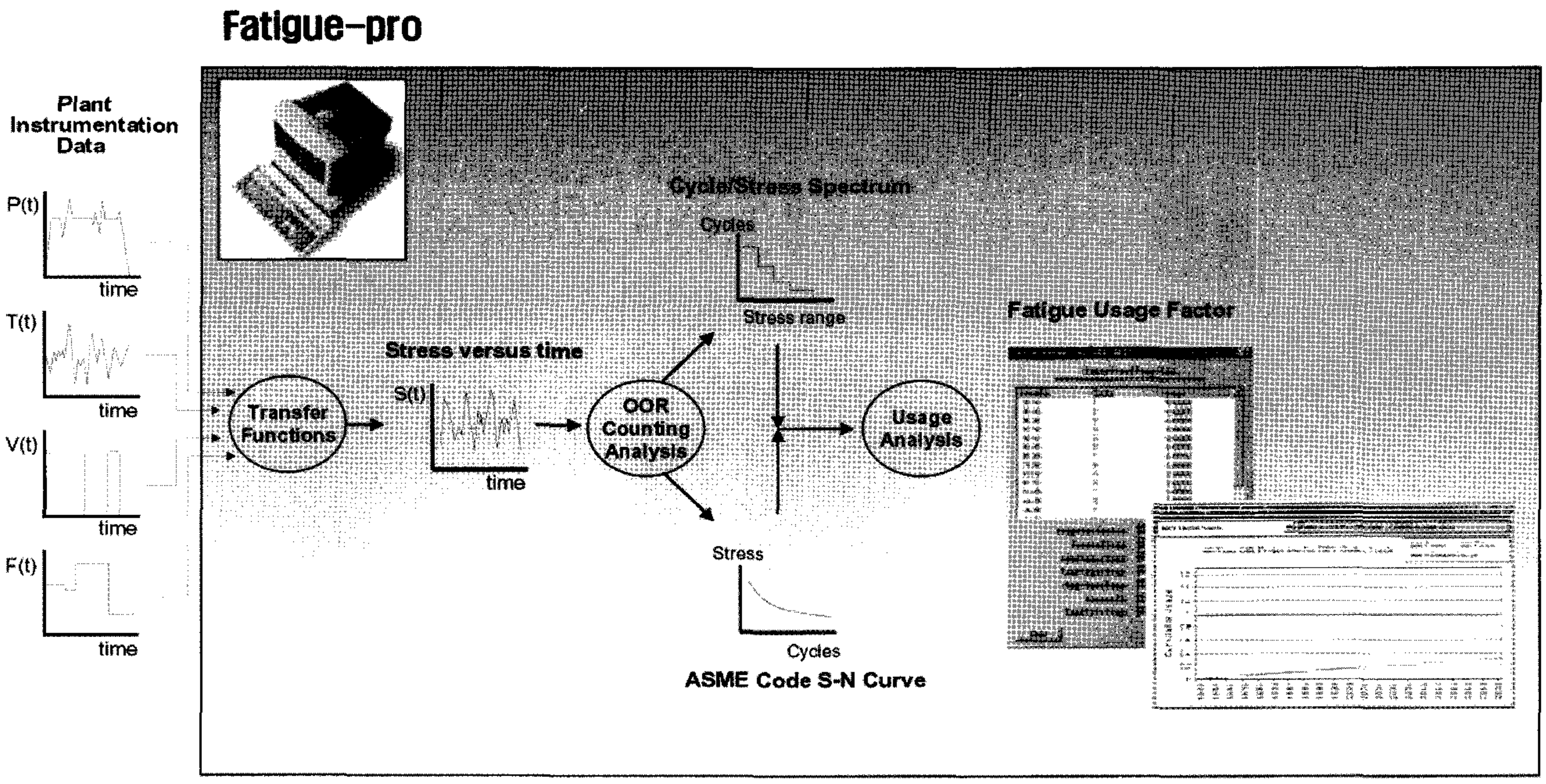

Fig. 8. Fatigue Monitoring System of Kori Unit 1

Fig. 9. Photo of the IAEA Peer review Meeting on Kori Unit 1

of Kori Unit 1 was trusted and completed. Accordingly, an IAEA Safety Aspect of Long-Term Operation (SALTO) team of seven specialists was dispatched to KHNP for two weeks. They reviewed the report and conducted a site walkdown in areas related to the basic principles of safety review, component classification, mechanical components, electric and I&C equipment, structures and in areas related to the radiological impact on the environment. Finishing the review, the IAEA team reported to MOST that Kori Unit 1 was expected to maintain a safe condition during its continued operation period. They also confirmed that it met the current international safety standards and practices [16, 17]. Fig. 9 shows a photo of the IAEA peer review meeting concerning Kori Unit 1.

4. APPROVAL OF CONTINUED OPERATION AND RE-START UP

Plant service life should be determined on a cost and benefit basis provided that the plant safety is maintained. As long as the aging phenomena of SSCs are confirmed to be properly managed during the period of continued operation, it is possible to operate the plant continuously. Kori Nuclear Unit 1 has been operated for its design lifetime of 30 years without any reportable trouble. A final safety analysis report was originally prepared based on the design code of ASME Sec. III, 1968 edition. Since the startup of the plant, the final safety analysis report has been revised to current codes and standards with the operation experience and research findings reflected.

With all the results of the technical evaluation activities for continued operation, the Korean Nuclear Safety Committee in MOST approved the continued operation of Kori Unit 1. The re-startup began on January 17, 2008 to last 10 years beyond the design life, which must be a landmark in the 30-year history of nuclear power generation in Korea.

It can be said that this is the result of efforts of all the individuals involved, including those of the plant operator, the regulatory body, research institutes, engineering companies, universities, and nuclear industries.

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