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OVERVIEW OF NPP's COMPONENT AGING ASSESSMENT RESEARCH PROGRAM IN TAIWAN

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ABSTRACT

A five-year pilot Component Life Assessment of Nuclear Power Plant (NPP) Program was initiated in 1987 to investigate the aging mechanisms and develop the evaluation methodology for the operating nuclear power plant in Taiwan. In this program the components were categorized into structural and functional components for easy implementation of life assessment methodology. The aging trends of the functional components were investigated by analyzing the work request records. The evaluation of the structural components has mainly focused on the failure mechanism studies and the integrity analysis.

Components reviewed in this study included part of the items where acute aging evaluations were required for 10-year operation license renewal application. The selected structural components for assessment were reactor pressure vessel, vessel internal components, and vessel support structure. The functional components were emphasized on mechanical components which included pumps, valves, and snubbers. In addition, some parallel research programs regarding to life assessment models for other components such as turbines, pipings and the heat exchangers were also briefly discussed in this paper.

INTRODUCTION

The life time of a component can be defined as the period during which the component can perform its intended function safely, reliably and economically. For a nuclear power plant, the service life time is currently determined by the 40-year license basis required by US-NRC. Taking into account the margins of design conservatism, conservative operation and maintenance conditions, there is a possibility that the plant components can continue to operate beyond its design life; on the other hand, the component might face early retirement, if it encounters the degradation conditions such as unanticipated stresses, operation conditions beyond design limit, corrosion or other environmental effects, thermal aging of materials, etc.

Intensive research programs related to nuclear power plant life extension or life assessment have been carried out in North America, Europe and Far Eastern. One of the noted endeavor is the EPRI/DOE utility-sponsored project on plant life extension for BWR (Monticello) and PWRs (Surry and Yankee Rov.). Industrial Reports (IRs)[1,2] for 40-year operation license renewal process were generated from that project, whose topics comprise systems, structures, and/or components that are vital to safety and continued plant operation, and address the issues generically rather than on a plant-specific basis. The IRs emphasized a comprehensive evaluation of the age-related degradation or any plausible mechanisms affecting the structure or component under consideration. Suggested approaches for few unresolved degradation mechanisms are also provided in IRs.

This paper will present the life assessment methods for nuclear power plant components in Taiwan. The key components such as reactor pressure vessel, nozzles, vessel internals, pipings, turbine rotors, condensers, heaters, and valves are emphasized. Each component is evaluated on the basis of its major aging mechanism.

APPROACH

Figure 1 shows a general process[1,2,3] for the component life assessment. In order to determine the priority of candidates for evaluation from about 5000 items in a nuclear power plant, a set of screening criteria has to be set up. A complete screening criteria for component life assessment usually include the general rules, feasibility of replacement, period and cost of replacement, effects on the surrounding components due to failure. In addition, the operation condition and history, code requirement, and the reliability of component also have to be taken into account.

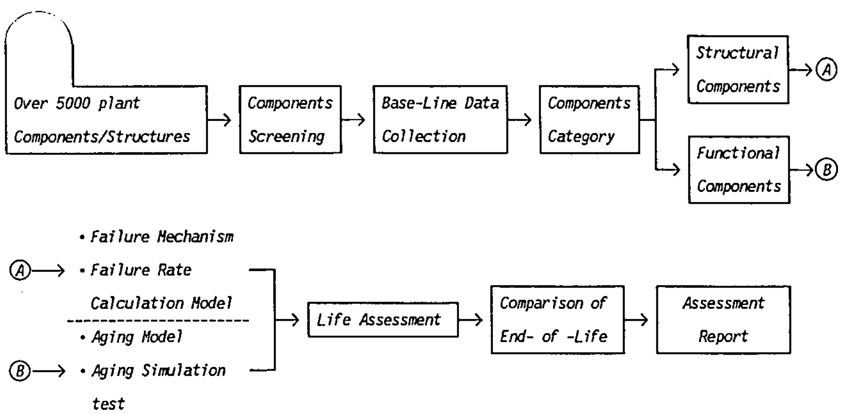

Figure 1. Schematics of component life assessment process.

When the component is chosen, the baseline data such as its function, constructing materials, operation and maintenance history, environmental condition, design specification, and stress reports have to be collected. However, the methods for life assessment are usually dependent on the categories of components. When considering the aging for structural components (e.g. reactor vessel, piping, etc.), emphasis is placed on their structural integrity. On the other hand, the emphasis for functional components (e.g. pump, valve, cable, etc.) is to assure that the function can still be satisfactorily maintained when the components continue in service. The definition of life limit also depends on the type of components. For example, critical crack length and material strength are used as life limit for structural components. Failure to operate and leakage are normally used for functional components.

Finally, a complete life assessment report will be issued, which normally include the following items: abstract, definition and function description of component, codes or regulatory requirements, evaluation records of component qualification, design basis and related testing, operation history, maintenance history, major degradation mechanism, life evaluation, renewal of operation and maintenance procedure, future corrective action, and further research and development.

RESULTS

The components were categorized into structural and functional components for easy implementation of life assessment process. The evaluation of the structural components were mainly focused on the failure mechanism and their integrity. The results are described as the following items.

(1) Reactor pressure vessels (RPV): The neutron irradiation embrittlement is the aging mechanism of most concern. The life time of RPV beltline materials limited by irradiation embrittlement were evaluated from upper shelf energy (USE), nil ductility transition temperature (RT_{NDT}), and pressure-temperature limit; whose requirements were issued by NRC (10CFR50 Appendix G). An example[3] of life assessment conducted for MNS-1(PWR) vessel shell is described in figure 2. The results show that the MNS-1 can satisfy the requirements of the hydro tests at 80 EFPY.

(2) RPV Nozzles: Fatigue due to system thermal transients is one of the most important degradation mechanisms for nozzles. The fatigue damage was evaluated by counting transient cycles occurred and the transient cycles were compared with the allowable cycles defined in the design stress report. The procedure for determining the cumulative usage factor as defined in ASME Sec. III is summarized in figure 3. In order to evaluate the fatigue usage accurately, the on-line fatigue monitoring system is being installed in local plant to keep track of the aging trend of the critical nozzles. Recently, intergranular stress corrosion cracking (IGSCC) of nickel alloy 600 penetrations has been observed in European PWRs. A local program to assess that fracture problem is also initiated.

(3) RPV internals: The BWR top-guide is considered potentially susceptible to irradiation assisted stress corrosion cracking (IASCC). A finite element beam model of the top-guide was developed based on the design drawing. The most critical locations for crack initiation and propagation were determined in figure 4[4]. Critical crack lengths of 4 crack types were also calculated for various loads in different directions.

(4) Pipings: IGSCC of 304 stainless steel is the degradation mechanism of most concern in BWRs. A PC version program[5] was developed to assess the safety life of the cracks on the pipe during plant operation. Comparison of the SCC growth rate data from the prediction and the field inspection for several cracks was also made in this study, which is shown in figure 5. Pipe rupture problem from erosion corrosion is another safety-related issue for carbon or low alloy steel pipings. The erosion corrosion rates were calculated from literature models and field inspection results. A procedure to assess the fracture criteria of uniform and local pipe wall thinning is also established.

(5) Turbine rotors: SCC of low pressure (LP) rotors is the major factor to affect their operation life. The life assessment method was focused on the determination of SCC growth rate and the calculation of critical crack length. The process of deterministic and probalistic life assessment for rotors with SCC cracks is presented in figure 6[6].

(6) Main condensers: The water side erosion corrosion, steam side impingement attack and fretting of copper-nickel alloy tubes were observed to be the most severe aging mechanisms. As no theoretical approach could be found to resolve the synergistic effects by these mechanisms, an empirical equation from inspection data was made to estimate the remaining

life before the condenser reaches its life limit of 10% tubes plugged. The residual life was assessed by the following equation[7],

$$N_{50} = n_1*N_{20} + n_2*N_{30} + n_3*N_{40} + n_4*(52000-N_P)$$

where N_{50}: the number of tubes which will exceed 50% wall thinning at next EOC

N_{20}: the estimated number of tubes with existing defect from 20 to 29% wall thinning at next EOC

N_{30}: the estimated number of tubes with existing defect from 30 to 39% wall thinning at next EOC

N_{40}: the estimated number of tubes with existing defect from 40 to 49% wall thinning at next EOC

N_P: the number of plugged tubes at the present EOC

n_1: is the probability of 20-29% defects growing to above 50% defects

n_2: is the probability of 30-39% defects growing to above 50% defects

n_3: is the probability of 40-49% defects growing to above 50% defects

n_4: is the probability of under 20% defects growing to above 50% defects

(7) Feed water heaters (HWHs): The major failure mechanism of 304 stainless steel tubes was identified as pitting corrosion by sea water intrusion. A pitting incubation model of 304 S.S. was derived, which described a linear relationship between pitting initiation potential (E_{np}) and logarithm of incubation time (t_i), such as[8]:

$$E_{np} = -0.467 + 0.267\log(t_i)$$

This equation is independent on temperature, chloride content in solution, and the degree of sensitization of 304 S.S.. From this model, an appropriate action plan could be set in the plant when an off-normal chloride solution is found in the heaters.

The functional components were emphasized on mechanical components which included pumps, valves, and snubbers. The aging trends were investigated by analyzing the work request records in the plants. Table 1[9] shows the trend of motor-operated valve failure modes in nine years for a local BWR. There was no specific trend in this table, but it was found that leakage is the most frequently occurred failure mode. Suggestions of corrective action for the aging components were issued after the analysis. However further efforts on the statistic model and simulation test for the functional components are needed to assess the remaining life accurately.

CONCLUSION

The importance of understanding and controlling aging effects on nuclear components are gradually being recognized by both the utility and regulatory body. The goal of life assessment methods described in this paper is to manage the aging problems for each component by suitable/in-time replacement, monitoring or refurbishment. To further extend the concept of aging management, it is necessary to carry out a detailed aging survey program on systems of interest or the whole plant. Through a precise assessment on the plant components, the aging problems can be adequately controlled and higher margins can be further achieved for plant life extension.

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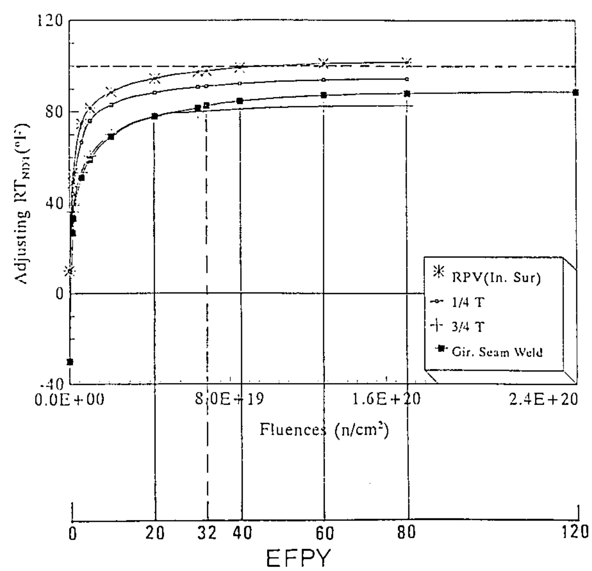

Figure 2.
The predicted Adj.RT_{NDT} for a PWR's reactor shell.

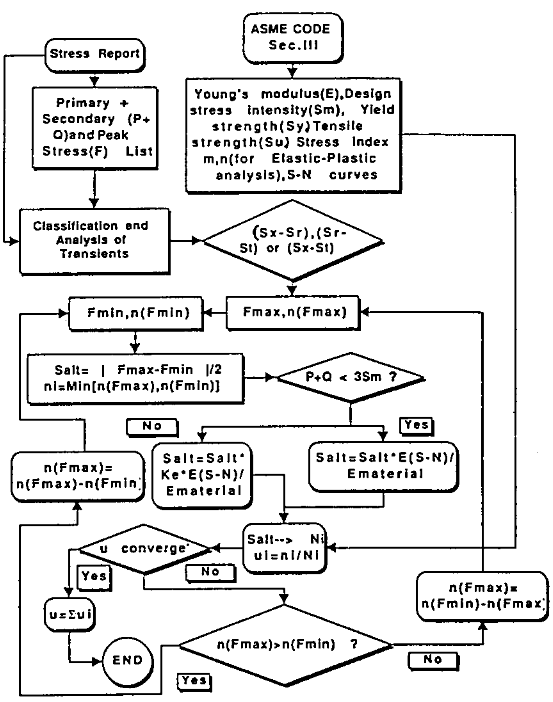

Figure 3.
Calculation procedures for fatigue usage factor.

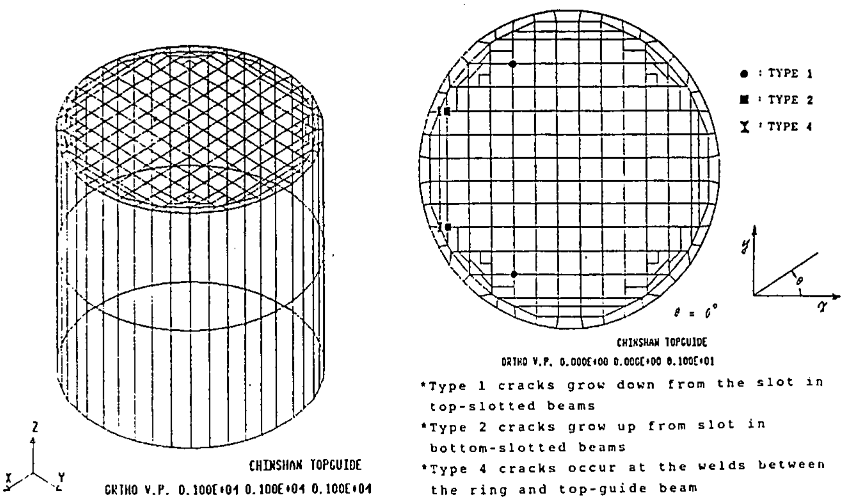

Figure 4. The FE model of top-guide and its critical locations of various crack type.

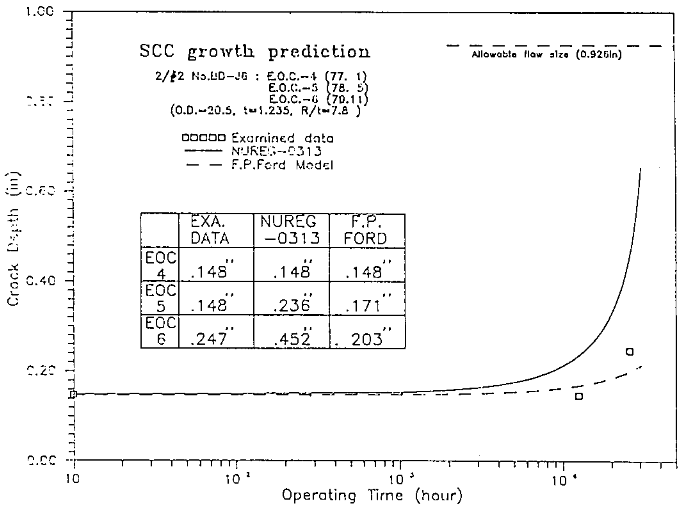

Figure 5.
Comparison of SCC growth rate data by prediction models and the field inspection data.

Table 1.
Trend of MOV failure modes in a local BWR.

	leakage	fail to open	fail to close	fail to operate	total
1980	8		1	2	11
1981	7		1	6	14
1982	16	1	6	5	28
1983	9		2	1	12
1984	3	5	1	3	12
1985	11	1	3	3	18
1986	21	2	1	4	28
1987	3			3	6
1988	4		2	2	8
total	82	9	17	29	137

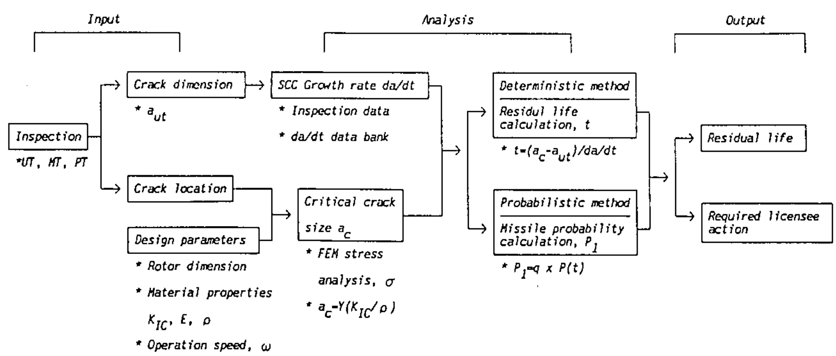

Figure 6. Life assessment process for nuclear LP turbine rotor.