Proceedings Series

Material Degradation and Related Managerial Issues at Nuclear Power Plants

Proceedings of a Technical Meeting
Vienna, 15–18 February 2005

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MATERIAL DEGRADATION AND RELATED MANAGERIAL ISSUES AT NUCLEAR POWER PLANTS

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PROCEEDINGS SERIES

MATERIAL DEGRADATION AND RELATED MANAGERIAL ISSUES AT NUCLEAR POWER PLANTS

PROCEEDINGS OF A TECHNICAL MEETING
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
HELD IN VIENNA, 15–18 FEBRUARY 2005

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FOREWORD

The world's fleet of nuclear power plants (NPPs) is, on average, more than 20 years old. Even though the design life of an NPP is typically 30 to 40 years, it is quite feasible that many NPPs will be able to operate in excess of their design lives, provided that nuclear power plant engineers demonstrate by analysis, trending, equipment and system upgrades, increased vigilance, testing, ageing management that the nuclear power plant will operate safely. In the operation of nuclear power plants, safety should be always the prime consideration. Plant operators and regulators must always ensure that plant safety is maintained, and where possible enhanced, during a plant's operating lifetime.

After the secondary system pipe rupture at Unit 3 of the Mihama nuclear power plant in Japan, which occurred on August 9, 2004, the IAEA recognized the increasing importance of material degradation and related managerial aspects, and held a technical meeting from 15 to 8 February 2005 in Vienna, Austria.

The purpose of the technical meeting was to provide an international forum to share recent knowledge and experience, and to share lessons learned. The discussion included lessons from recent events, results from research programmes, experiences and practices, ownership and responsibility, managing operational experience and regulatory aspects. The quality of the papers and presentations in the technical meeting was high, and they provided good technical and managerial insights to materials degradation issues. The main results and lessons learned from the technical meeting were compiled in this publication.

By evaluating common aspects of the issues presented, one can conclude that the initiating technical issue was rarely the root cause of the resulting problem. That is, each technical issue was relatively well understood at least in some segments of the industry and frequently had been experienced to some degree at other nuclear power plants. What complicated each issue and was at the root of most of the problems, was some shortcoming in the facilities' management process combining a gradual decrease in safety culture.

The IAEA wishes to thank the participants for their contributions, especially the meeting chairman, P. Tipping of the Swiss Federal Nuclear Safety Inspectorate (HSK) of Switzerland. Thanks also to M. Bakirov of VNIIAES of the Russia Federation, and R. Schrauder of the First Energy Nuclear Operating Company (FENOC) in the United States of America. The IAEA officers responsible for the organization of the meeting and the compilation of this publication were K. Kang of the Division of Nuclear Power and T. Inagaki of the Division of Nuclear Installation Safety.

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SUMMARY

This publication presents the main results and lessons learned from the IAEA Technical Meeting on Material Degradation and Related Managerial Aspects held on 15–18 February 2005. Managerial aspects and influences are seen to be just as important to the safe and reliable operation of nuclear power plants (NPPs) as technical issues and are frequently the root cause of incidents that occur.

The technical meeting consisted of four sessions in addition to the opening and closing sessions. The report of incidents session was to share information on recent events involving material and managerial issues affecting safe operation of NPP and to share lessons learned from those issues. The technical issues session was to share technical updates on material issues and to identify outstanding technical problems in the field. The managerial session shared lessons learned on the aspects of the management system coping with technical issues presented in the previous session, and to identify outstanding managerial issues in the field. The regulatory aspects session discussed the sharing of roles and responsibilities among parties involved in NPP operation and regulation.

A total of 40 experts from 23 Member States (MS) participated and 35 papers were presented during 4 days. A main result from the presentations was that NPP management performance, including questioning attitudes by personnel and a good safety culture were critical factors and thus key elements in determining how safely NPPs will operate. Also, constant vigilance is needed, even when a NPP has had a very reliable operating history.

During the course of the sessions, based on the papers presented, it became obvious that rarely does a purely technical issue lead to an incident or accident. In each case the technical issues were significantly complicated by the underlying management issues that became apparent during the investigations which followed the events. It is clear that management must establish and enforce a safety culture in order to safely, reliably and cost effectively operate today's NPPs. Clearly, close collaboration and communication between operators, safety experts, regulators and NPP management is very essential for safe operation of NPPs.

1. BACKGROUND

NPPs worldwide are showing continuous improvements in their performance and availability. This applies to overall plant safety as well. However, problems associated with materials (in more general terms, system, structure and component (SSC) degradation followed by spontaneous failure) have been observed. These material problems may, however, not be of a purely technical nature but may have factors present in the management system of the NPP concerned. The use of existing knowledge could have prevented some recent accidents or incidents, showing that knowledge management and application of lessons learned has not been done sufficiently. Such aspects lie within the management processes of a NPP. The recurrence of problems, despite overall, global knowledge of what can happen, is an indication of an insufficient corporate culture to acquire, process and apply the state of arts science and technology.

In order to reduce the likelihood of repeated occurrence of failures, despite already well-known and documented incidents, the combination of a wide spectrum of activities, ranging from the effective use of operational experiences, continued research into the mechanisms of material failure, transparency in sharing information, avoidance of complacency, and optimised knowledge management is needed. This is, in principle, already done, through the owner's groups (e.g. Boiling Water Owners Group (BWOG), Westinghouse Owners Group

(WOG) and World Association of Nuclear Operator (WANO) and information is also shared between regulators through their groups (e.g. Western European Nuclear Regulator Association (WENRA). The importance of such activities will be further emphasized in the light of ageing of the world's nuclear fleet, which is currently approximately 20 years old. As more NPPs age, the importance of the issue of ageing of SSCs and managerial issues, including knowledge retention due to retirement of experienced personnel, will be further increased.

Degradation and ageing are terms used to describe both the deterioration of components, but it is useful to distinguish between them.

- Degradation is immediate or gradual deterioration of characteristic of an SSC that could impair its ability to function within acceptance criteria.
- Ageing is general process in which characteristics of an SSC gradually change with time or use.

When ageing processes are known, they can be allowed to monitor through an appropriate ageing management programme (AMP) and plant life management (PLiM) programme and potentially mitigated. New or unexpected degradations, if left undetected on the other hand, can lead to accidents.

TABLE 1. MAIN DEGRADATION MECHANISMS IN PRESSURIZED WATER REACTOR COMPONENTS

Component	Irr. Emb.	Fatigue	Corrosion Fatigue	SCC	Corrosion	Thermal Ageing	Wear
Reactor pressure vessel	❍	❍	❍	❍ *	**	***	
Control rod drive mechanisms		❍		❍			❍
Internals structures	❍	❍		❍		❍	
Reactor coolant pump casing		❍	❍		❍	❍	
Piping and safe ends		❍		❍ +	❍ ++	❍	
Pressurizer		❍	❍				
Surge and Spray lines		❍					
Steam generator tubing		❍	❍	❍	❍		❍
Steam generator shell and nozzles		❍	❍	❍		❍	

Note : Irr. Emb : Irradiation Embrittlement
SCC : Stress Corrosion Cracking
* In closure head bolts
** When there is no cladding (VVER 230) or cladding defect corrosion is possible. RPV corrosion can occur on the outside, if leaking head penetrations allow borated water to contact the RPV vessel.
*** Depending on RPV material
\+ In instrumentation nozzles and heater sheaths
++ The combination of flow and corrosive fluid can lead to erosion-corrosion or flow-accelerated corrosion and thus reduce the wall thickness of piping.

It is of particular concern when well-known, avoidable degradation mechanisms, seen at other NPPs years before, recur and impact other NPPs. Age related component failure could occur by degradation processes which are common to all industrial plants. The components life times vary considerably as a factor of temperature, time, environment and stressing cycles. Table 1 shows an example for main degradation mechanism in pressurized water reactor components.

Degradation mechanisms include metallurgical phenomena such as irradiation embrittlement, fatigue, corrosion, interaction and a combination of mechanisms. It should be noted that Table 1 gives only main degradation mechanisms for different NPP components. There are many degradation mechanisms, some of which are shown in more detail in Figure 1.

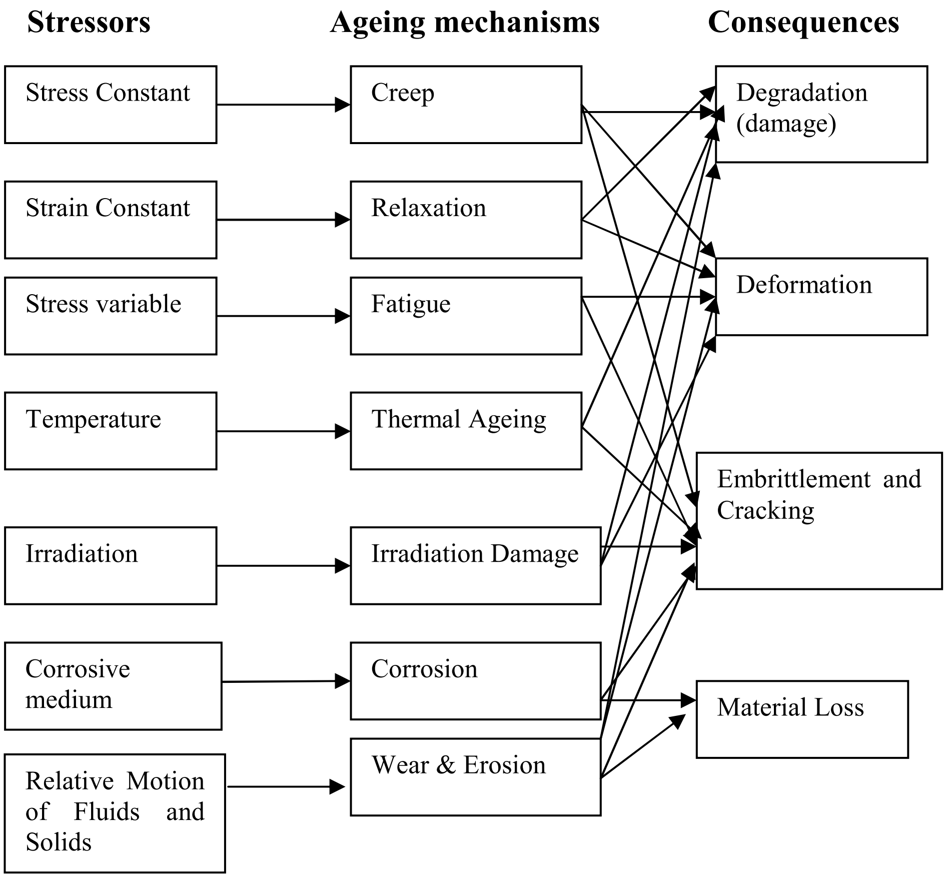

Fig. 1. Ageing factors, ageing mechanisms and possible consequences.

Ageing degradation mechanisms are usually classified into two main categories, which are those that:

- Affect the internal microstructure or chemical composition of the material and thereby change its intrinsic properties (thermal ageing, creep, irradiation damage, etc.).
- Impose physical damage on the component either by metal loss (corrosion, wear) or by cracking or deformation (stress-corrosion, deformation, cracking).

As can be seen above, the phenomenon of ageing degradation in NPPs is complex and thus requires sophisticated, state of science and technology procedures to effectively manage it and ensure safe, reliable operation. It will be observed that not only technology is involved, but also an effective management system is needed in order to correctly implement mitigative or monitoring actions.

Recognizing the importance of these aspects, the IAEA Divisions of Nuclear Power and Nuclear Installation Safety held a joint technical meeting from 15 to 18 February 2005 in Vienna, Austria. The purpose of the meeting was to provide an international forum to share recent technical knowledge and experience relating to material degradation issues, and to share lessons learned related to managerial issues. The meeting consisted of 4 sessions, not including the opening and closing sessions, as follows:

- Session 1: Report of incidents — This session was to share information on recent events involving material and managerial issues affecting safe operation of NPPs, which had implications for technical as well as safety management aspects;
- Session 2: Technical issues — This session was to share technical updates on material issues and to identify outstanding technical issues in the field;
- Session 3: Managerial issues — This session shared lessons learned on the aspect of the management system coping with technical issues presented in the previous session, and to identify outstanding managerial issues;
- Session 4: Regulatory aspects — This session discussed the sharing of roles and responsibilities among NPP parties involved, and showed regulatory aspects.

This report summarizes the main results obtained, and provides insight on how the issues can be managed or avoided. A list of good practices is also provided in Appendices 6.

2. REPORT OF INCIDENTS

This session provided information on recent events involving material and managerial issues affecting operation of NPPs having implications for technical aspects as well as safety management aspects. This information gave the basis for discussion in the latter sessions. The events presented by the Member States (MSs) include incidents concerning the RPV and piping systems.

2.1. Mihama Unit 3 pipe rupture accident in Japan

The Nuclear and Industrial Safety Agency (NISA) of Japan, through the Accident Investigation Committee has been conducting an investigation into the secondary system pipe rupture at Unit 3 of the Mihama Nuclear Power Plant, which occurred on August 9, 2004. The interim report was published on 27th September 2004. A summary of the report follows:

2.1.1. Pipe rupture mechanism

A secondary side carbon steel pipe ruptured (see figures 2 and 3). Thinning had developed at the downstream of an orifice due to the combination of erosion caused by the mechanical action of the internal water flow and corrosion caused by chemical reaction. The pipe had gradually lost its thickness, which led to a loss of strength and eventually to a rupture from internal pressure (about 1 MPa). The investigation has revealed that the dimple pattern typical in this kind of phenomenon was found on the inner surface of the pipe.

2.1.2. Pipe thickness management for the ruptured portion

The ruptured portion of pipe should have been subjected to Kansai Electric Power Company's (KEPCO) pipe thickness management process requiring the measurements of pipe wall thickness, according to their document entitled "The thickness management guideline for secondary system pipes," adopted in 1990. However, the ruptured portion of pipe was missing in the inspection list that Mitsubishi Heavy Industries, Inc. (MHI) prepared based on the Guideline. In the past, there had been opportunities to correct the list to include the ruptured portion. Specifically, when the transfer of the inspection activities from MHI to Nihon Arm Co. Ltd., took place, the correction of the inspection list at other NPPs by MHI, to include the same location as the ruptured one at Mihama Unit 3, was not done. The facts revealed, however, that the thickness measurement of the ruptured portion had not been conducted because those responsible had never checked whether there was any possible omission in the list.

The immediate cause of the pipe rupture at Mihama Unit 3 was "the omission of the failed portion from the inspection list, and the fact that this had not been corrected before the accident occurred" due to "the failure in pipe thickness management for the secondary system involving three parties, KEPCO, the MHI, and Nihon Arm Co. Ltd.". This means that quality assurance and maintenance management had failed to function properly in KEPCO, which was responsible for the safety of the NPP as its Licensee.

2.1.3. Pipe thickness management at NPPs other than Mihama Unit 3

NISA conducted a survey on pipe thickness management adequacy of all NPPs, and confirmed that inspections had been properly carried out for all NPPs except for the ones operated by KEPCO. As for KEPCO, in their ten NPPs other than Mihama-3, a total of 14 portions had never been checked for pipe thinning or they had been inspected using unauthorized procedures. For the portions KEPCO had inspected, including the 14 portions and the other portions NISA obliged them to inspect, NISA confirmed the integrity of all those portions, except for three portions, which needed pipe replacements.

2.1.4. Immediate actions by the government and licensees

The following are the actions to be taken by the Government, Licensees and other relevant parties:

(1) Actions taken by the Government:

- Clarification and dissemination of the licensces' inspection methods concerning pipe thickness management;
- Verification of the implementation system for pipe thickness management by the Licensees including their subcontractors during the periodic safety management review;
- Adoption of the standards being prepared by the Japan Society of Mechanical Engineers (JSME) concerning the pipe thickness management methods and use of these JSME standards as the criteria;
- Direction for, and supervision of, the Licensees (including their subcontractors) to conduct more careful subcontracting management.

(2) Actions taken by the Licensees:

- Review of the "PWR Thickness Management Guideline" based on the latest knowledge and actual data, including the handling of the "main systems to be inspected" and "other systems";
- Compiling guidelines for pipe thinning management for PWR;
- Construction of management system for systematic "inspection lists," which link the computerized piping system diagram to the management charts to conduct effective maintenance of all the equipment subject to the periodic licencees' inspections;

- Development and observance of rules specifying the methods and responsibility-sharing concerning subcontracting management for the maintenance management activities;
- Identification of matters to be specified in contracts and order documents, to facilitate classification of the rights and duties concerning subcontracting parties;
- Provide feedback on the lessons learned from the pipe rupture accident at Mihama Unit 3 so that they are reflected in the safety-ensuring activities of the Licensees;
- Implementation of training to ensure safety of workers and clear notification of risk information in critical areas.

2.1.5. Other matters for consideration

It was essential to implement the actions proposed without delay. Other preventive measures may be added, depending on the final findings and conclusions of the investigation. Some attribute the accident to the ageing of nuclear power plants, but its direct cause was the absence of the inspection, even under proper thickness management programmes. However, more comprehensive inspection management is required for the ageing of NPP SSCs. In view of this, even more important are the periodic safety reviews (PSRs) conducted every 10 years and the comprehensive evaluation of the ageing of NPPs over 30 years.

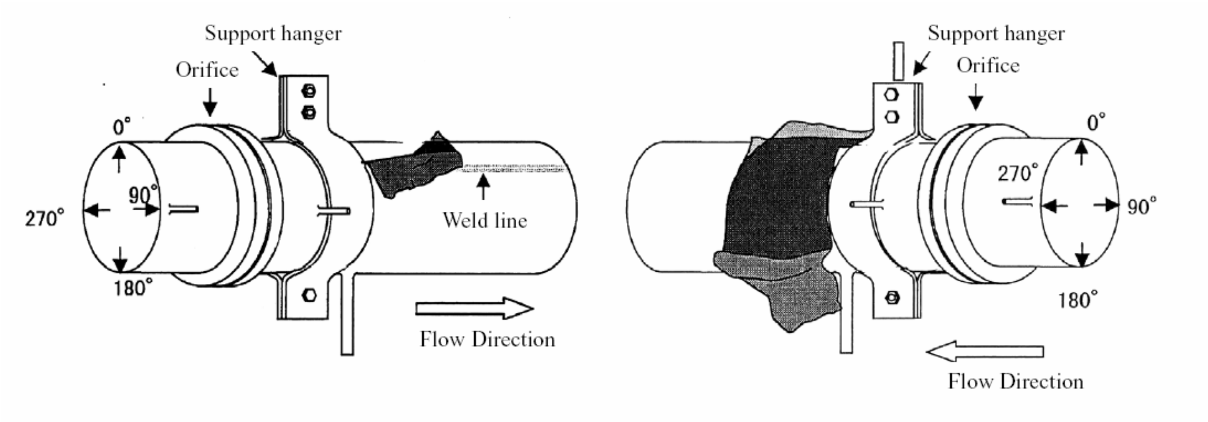

Fig. 2. and 3. Rupture of pipe downstream to orifice.

2.2. Lessons learned from Davis Besse NPP – reactor pressure vessel degradation

This section discusses the discovery of the near through-wall corrosion of the RPV closure head at the Davis-Besse Nuclear Power Station in March 2002. Although no loss of coolant occurred and the reactor core always remained fully covered and cooled, this incident represented a significant degradation of the nuclear safety margin at the facility. This section describes what caused the corrosion, why it occurred, what actions were taken to correct the root cause of the degradation and finally some lessons-learned from the incident.

2.2.1. Background

Davis-Besse is a pressurized water reactor (PWR), manufactured by Babcock and Wilcox with a licensed thermal power output of 2772 Megawatts. The owner/operator of the facility is the First Energy Nuclear Operating Company (FENOC). The facility is located in Oak Harbor, Ohio on the western shores of Lake Erie.

The plant began commercial operation in August 1978 and is currently licensed to operate until April 2017. The RPV has an operating pressure of 2155 psig (151.50 kg/square cm) and a design pressure of 2500 psig (175.75 kg/square cm). Davis-Besse had accumulated 15.8 effective full power years (EFPY) of operation when the plant shut down for its thirteenth refuelling outage on February 16, 2002. During that refuelling outage, while performing RPV vessel closure head inspections required by the United States Nuclear Regulatory Commission, workers discovered a large cavity in the 6 inch (15.24 cm) thick low-alloy carbon steel RPV head material. The cavity was about 6.6 inches (16.76 cm) long and 4 to 5inches (10.16 to 12.70 cm) at the widest point extending down to the 0.25-inch (0.635 cm) thick Type 308 stainless steel cladding. See Fig. 4 and 5.

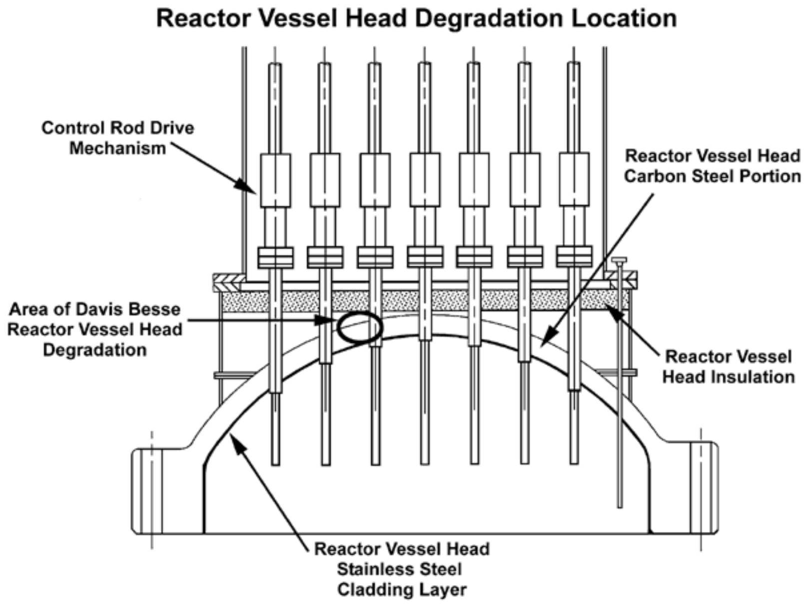

Fig. 4. PWR vessel head penetration cracking of Alloy 600 allowed leakage of the borated coolant to occur. This corroded the RPV head externally down to the stainless steel cladding.

FENOC promptly commissioned a root cause team to evaluate what had caused the corrosion. After initially considering a repair of the RPV head, FENOC purchased an unused head of the same design. The facility was out of service for over 2 years while the head was replaced and other wide-scale evaluations and improvements were made to the physical plant, programmes, and staffing organization.

The following summarizes lessons-learned from this incident:

- Past plant operating successes can lead to complacency and failure to maintain and advance standards of excellence;
- A strong Corrective Action Program (CAP) was essential for future safe plant operation;
- Minimum compliance with regulatory standards is not enough;
- Having a heavy production focus rather than a safety focus is a costly and flawed choice;
- Equipment and material problems and anomalies must be rigorously addressed in a thorough and timely manner;
- Oversight organizations must be involved and unbiased;
- Ignoring many small indications (fluid balance, increasing frequency filter change etc.) can lead to serious consequences.

Fig. 5. Davis-Besse Reactor Pressure Vessel Head Degradation.

2.3. Erosion-corrosion in the secondary system piping and components in Finland

The Loviisa NPP owned by Fortum (by IVO until 1998) consists of two VVER 440 units dating back to the late '70s and the early '80s. Special features of the secondary circuit systems, such as neutral water chemistry and the use of carbon steel, had made piping and components susceptible to erosion-corrosion (EC). A control programme was developed in 1982-1983 to manage EC in the secondary system piping and components.

The scope of the original control programme was enlarged after the NPP Surry/USA accident, caused by EC in 1986, but still the control programme was based on operating experience and two phase-flow conditions. Therefore, the flow control orifices were not included in the control programme. Due to shortcomings of the control programme, a guillotine pipe break of the feed water system piping occurred in 1990 at Unit 1. EC adjacent to the flow control orifice in the feed water discharge line was incorrectly assumed to be lower than in areas, which were found during previous outages when inspecting other fittings (Tees, 90 degree elbows, throttles, etc.).

After the guillotine pipe break, the extent of the control programme was further increased. Computer programmes for weak point analysis and the inspection data management were implemented, as well as a review of the extent of the annual inspection programme by the group of experts of the company. However, in spite of the large efforts put into improving the inspection programme, a second guillotine pipe break of the feed water system occurred in 1993 at Unit 2. See Fig. 6.

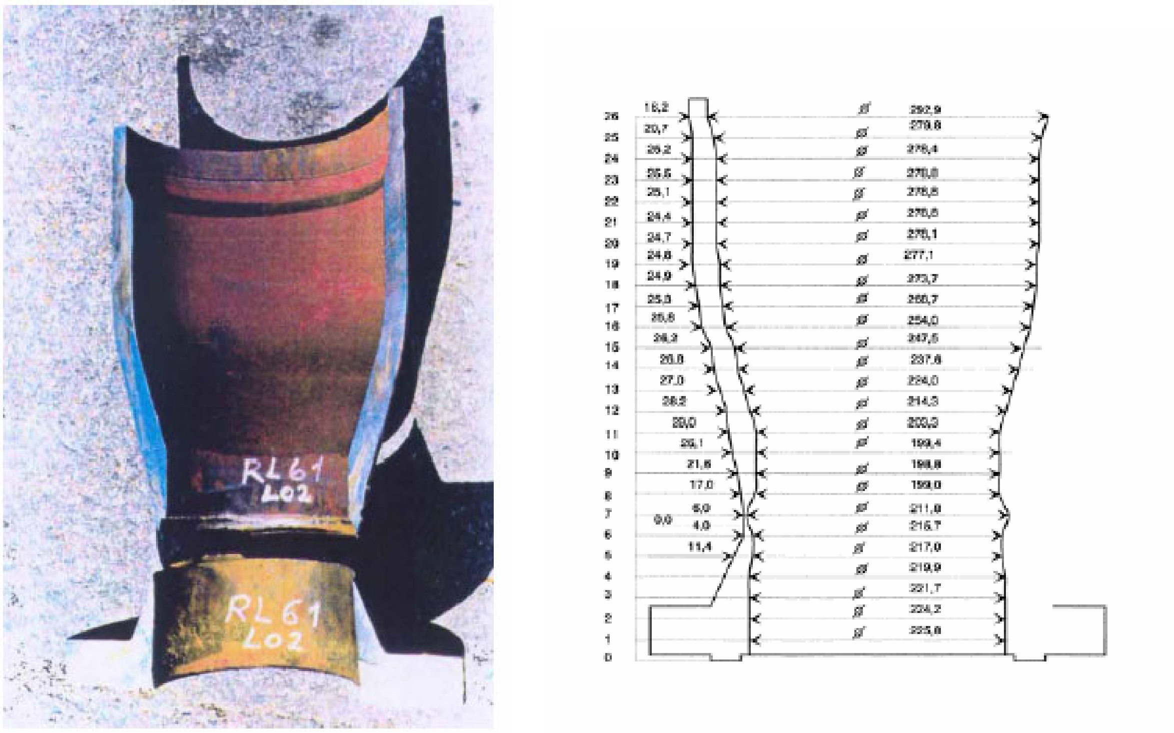

Fig. 6. Wall thinning of parallel FW-discharge lines (RL61).

A self-assessment performed revealed continuing shortcomings in inspection procedures and in the interpretation of previous inspection results. In order to diminish EC of the feed water system piping, the secondary water chemistry was changed from neutral to alkaline water chemistry in 1994 at Unit 2 and in 1995 at Unit 1. Also, a change from the carbon steel piping to low alloy or stainless steel piping was carried out. In order to improve the management of inspections, implementation of new computer software is underway.

Management of EC of the secondary system piping and components has turned out to be an important factor for the safety, plant availability and for the long-term operation of the plant. The following summarizes lessons-learned from this incident:

- It is not sufficient to assume that EC will be less in other areas: the exact conditions of flow (local turbulence and multi-phase conditions) may impact such assumptions;
- Evaluating accidents in other NPPs will give indications to inspect similar situations in other NPPs. Additional inspections may not lead to improved safety, if the inspection is not done in the suspected danger areas.

Changing piping materials and water chemistry, coupled with on-line monitoring will improve the level of safety.

It is important to note that in the Loviisa accidents there were no fatalities. This was due to the fact that no personnel were in the area at the time of the incident. In the Mihama Unit 3 accident, a final total of 5 people died since they were working in the immediate vicinity of the ruptured pipe. Workers routinely work on the secondary side of PWRs. Therefore, it is essential to their safety that effective processes are in place to prevent pipe ruptures.

2.4. Vandellós II – essential service water system pipe degradation

On August 25, 2004 at 5:25 a.m. with Vandellós II operating at full rated power, a circumferential break occurred on the neck of manhole EF-18-I when starting essential service water pump EFP01C. This took out of service one of the two trains of essential service water system (EF) of the plant. See Fig. 7.

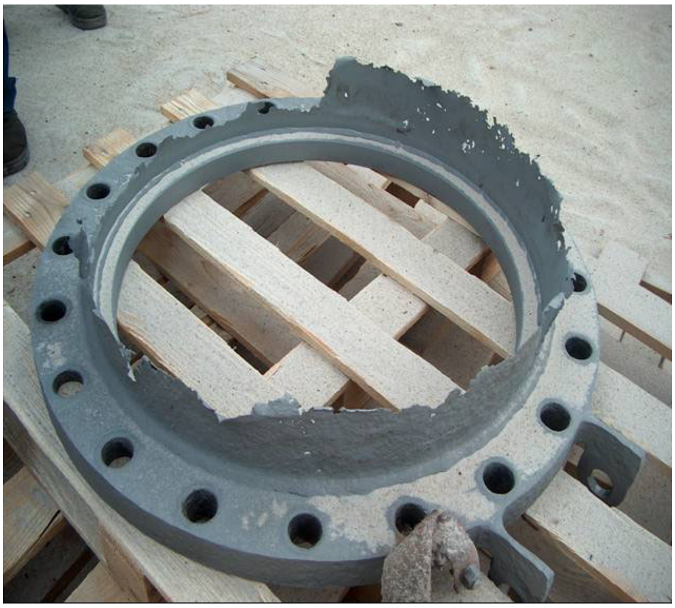

Fig. 7. Broken manhole neck of essential service water system.

The manhole EF-18-I is located in one of the collection boxes of the system (EF-28-Z3). The break occurred as a consequence of weakening of the carbon steel plate by external corrosion. After the incident, on inspection of the affected neck, it was seen that corrosion had progressed in a homogeneous way all around the neck, without a previous clear indication of a leak, although some slight weepage had been previously detected.

2.4.1. Essential service water system piping degradation

The necks on the manhole, as well as the transition between aerial (exposed) and buried piping, include a transition neck of carbon steel without any concrete protection. These local points had suffered corrosion with accumulated service years. This was produced by the aggressive external ambient conditions due to condensation and also the presence of seawater in some collection boxes coming from activities during outages when the pipe was drained for maintenance or inspection purposes. In this situation, degradation was increased significantly due to the presence of a high concentration of chlorides. Dating from the original construction, a layer of paint was the only protection against external corrosion.

Concerning the present case, the piping had been inspected (both EF trains) along the years during every outage until 2000 (refuelling outage 11). Since the initial operation of the NPP from March 1988, the same contractor had carried out the inspections. All inspection activities had been focused on detecting internal damage by seawater, and the state of the cathodic protection system, since internal corrosion by seawater had always been assumed to be the main potential mechanism of the pipe degradation. Throughout the successive inspections, the execution and data collection systems had been improving such that observations had become progressively standardised and a matter of routine.

There was no attempt to use systematic diagnostic techniques to identify potential problems with the neck of the pipe (measures of wall thickness). During the refuelling outage on September 2000, the inspection contractor generated an immediate report dated September 19, 2000 for Train A and September 25, 2000 for Train B. The report recommended some actions to be taken before returning the system to operation. The recommendations were:

- Clean and paint indicated surfaces;
- Clean the neck to the bare metal and measure the thickness.

The following summarizes two main lessons-learned from this incident:

- Avoid assumptions, based on routine.
- Be aware that the siting of a NPP may cause increased rates of corrosion (sea water equals chlorides).

3. TECHNICAL ISSUES

This session shared technical updates on material issues and identified outstanding technical issues in the field. Presentations covered issues related to the topic of the meeting such as:

- Overview of world material issues of NPPs;
- Behaviour and mechanism of pipe thinning and rupture;
- Inspection and monitoring of pipe thinning and rupture;
- IGSCC and IASCC including RBMK;
- Mechanical, thermal and corrosion fatigue;
- Ageing management (Technical point of view).

Presentations were followed by panel discussions.

3.1. Status report on material degradation at Paks NPP, Hungary

This paper described recent ageing related problems, including steam generator tube degradation and deformation of Control Rod Drive Mechanism (CRDM) sleeve and liner at the Paks NPP.

Outside diameter stress corrosion cracking (ODSCC) of austenitic stainless steel (08Ch18N10T) tubing in steam generators (SG) of Paks NPP caused a significant number of indications on the tubes. Macroscopic and microscopic examination, scanning electron microscopic examination, X ray diffraction analysis, tensile test and chemical analysis have been carried out to investigate secondary oxide layers and tube materials on steam generator tubes pulled from four Units of Paks NPP. Destructive examination of removed tubes indicated cracks had arisen at the tube support plate (TSP). Tensile test results showed higher values of mechanical properties than expected.

To mitigate the pollution of the SGs, the condenser tubing was replaced with austenitic stainless steel (AISI 316 L). Anomalous outlet temperatures and decreased coolant flow rates were discovered in the core of Units 1-2-3 at Paks NPP after SG decontamination. Magnetite crystals had formed on the surface of the fuel assemblies. As these deposits progressively blocked the cooling channels, the flow rate of water coolant through the reactor core decreased. Comprehensive studies were performed on the heat exchanger tubes originating from the different SGs of the Paks NPP. In 2002-2003, deformation of the sleeve and liner of the nozzles were observed on four CRDMs. The defective nozzles were repaired and their liners and sleeves were replaced. In order to determine the root cause of the failure, analyses have been started.

3.2. The Czech Republic erosion-corrosion programme and lessons learned from Mihama NPP

Flow Accelerated Corrosion (FAC) or EC is a specific process influencing piping components, manufactured from plain carbon steels. The result of two processes, creation and dissolution of the surface oxide layer, gives rise to the overall wall-thinning phenomenon, which may ultimately cause the pipe's rupture.

Efforts to minimize damage to the piping through FAC in the NPPs combines two approaches. First, it is the combination of the operational parameters, which excludes or significantly reduces FAC. The second approach constitutes the system measures – the systematic monitoring, reliable damage prediction and implementation of all accessible operational experiences to the evaluation of selected piping lines. Both methods are often combined, although significant changes of the operational parameters are difficult to make.

Although current results show good quality of the prediction model for both NPP Dukovany and Temelin, with respect to events in NPP Mihama, it has been decided to review the last inspections and plan additional inspections on the condensate and feed water piping.

TABLE 2. COMPARISON OF OPERATION CONDITIONS IN CONDENSATE SYSTEM

Parameter	Critical Value	Mihama	Dukovany	Temelin
Material, Cr content	< 1%	carbon steel SB42, <= 0.3	carbon steel 12 022, <= 0.3	carbon steel 12 022, <= 0.3
PH	< 9.5	8.8 – 9.3	8.6 – 9.9	10
Temperature	135 – 165°C	140°C	145°C	163°C
Liquid/Steam	Liquid is more than 95 %	All water	All water	All water

Note: Between 4th (5th) point feedwater heater and deaerator, upstream of the main feed pumps.

Attention should be focused on the pipelines, where the average temperature lies between 135 to 165 °C and locations immediately behind measurement orifices and other fittings, e.g. valves. In the case of NPP Dukovany, it was recommended to perform inspection revisions and trace the history of replacements/repairs in localities behind measurement orifices on all pipelines susceptible to FAC or behind orifices, where erosion or cavitation could occur during normal or non-standard conditions.

If during measurement the thickness is found to be close to the criterion value that is based on simple analytical evaluations of thickness, a more detailed analysis is performed. The analysis will cover all regions where significant thickness reduction is detected.

For detailed analysis, the following additional methods will be used to evaluate stress limits according to the Czech Normative Codes. First, stress analysis, using piping programmes usually applied for evaluation of influence of piping forces and piping moments on thickness of reduced areas, will be performed provided that the computer model of the piping system is available. Secondly, detailed finite element (FE) analysis of thickness-reduced areas, using measured thickness data, will be performed.

Attention will be concentrated on uncertainties where significant thickness reduction is detected or expected, such as at locations where piping hangers are fixed on pipe, piping supports, wall penetrations, location of pipe whip restraints, T- joints with reinforcing pads and generally inaccessible parts of piping.

3.3. COMSY software to assist lifetime management activities

The COMSY program has been developed to provide a tool for the plant life management of systems and mechanical components. The program utilizes more than 25 years of experience resulting from research activities and operational experiences. It is designed to support a plant-wide strategy, providing lifetime predictions for mechanical elements, which are validated by a small number of examinations at priority locations. The objective is to establish economically optimized inspection and maintenance programmes, while maintaining high levels of plant safety and availability. This is accomplished by focusing on inspection activities on the actual degradation-relevant locations.

The capability to perform service life predictions is the key function of a software system for ageing and plant life management. The lifetime prediction capability requires:

- An understanding of relevant ageing and damage mechanisms;
- The availability of data characterizing parameters required for lifetime calculations.

Parameters required cover the fields: System design and component geometry, material properties, the history of thermal-hydraulic and water chemical operation conditions, stress conditions and if available, results from non-destructive testing. To manage these parameters, the software utilizes a virtual power plant model. Based on the plant data, the program conducts a condition-oriented lifetime analysis for various damage mechanisms, which may occur in power plants (e.g. flow-accelerated corrosion (FAC), cavitation erosion, droplet impingement erosion, strain-induced cracking, material fatigue, etc.). This capability was achieved by integrating advanced analysis tools with extensive databases.

The resulting service life prediction is validated and optimized through the performance of a small number of examinations at priority locations, which are indicated by the program. Trending functions support the comparison of the as-measured condition with the predicted progress of degradation while making allowance for measurement tolerances. The results of this comparison are used to improve the accuracy of future life expectancy predictions. This systematic closed loop process ensures the generation of a quantifiable database, which is continually kept up to date with information related to the technical “as-is” status of the plant. On the basis of reliable and damage-relevant predictions, maintenance management and plant availability can be optimized. This capability is particularly useful for the service life extension of systems and components.

Within the last years the program has been successfully applied to various nuclear power plants, and the benefit of this software-based strategy is currently confirmed by field experience.

3.4. Guideline for management of piping wall thinning in PWR secondary systems in Japan

3.4.1. History of development

In the early 1980s, when some PWR plants in Japan experienced steam leakage due to piping erosion/corrosion, PWR-operating utilities in Japan responded to the problem of secondary system pipe wall thinning by launching a systematic investigation of pipe wall thickness at their PWR plants, paying due attention to fluid flow conditions and pipe configurations.

In 1986, a pipe rupture accident at the USA Surry NPP prompted Japanese utilities to work urgently toward the development of a guideline for the management of pipe wall thinning. In 1987, Japanese utilities started the analysis of wall thinning phenomenon using the available large amount of measurement data from operating plants. The evaluation covered a wide variety of data pertaining not only to two-phase flows, but also to feedwater and steam systems. As one of those utilities, Kansai Electric reviewed data taken at about 30,000 components in its nine PWR plants. Upon completing a statistical evaluation of the data produced by such review activities, the utilities established, in 1990, the “Guideline for Management of Pipe Wall Thinning in PWR Secondary Systems.”

3.4.2. Outline of guideline

The Guideline sets rules for the wall thinning management for carbon steel pipes in PWR secondary systems by stipulating the inspection method, inspection targets, inspection frequency, acceptance criteria, corrective actions (pipe replacement) and so on. Wall thickness is measured using ultrasonic test (UT) measuring equipment.

Regarding inspection targets, the guideline identifies those systems and components judged to be prone to major thinning based on the available data. For those components found not to be prone to thinning, the Guideline requires the locations of flow turbulence be inspected at a rate of 25% in 10 years (i.e., all such locations must be inspected over a forty year term).

With regard to inspection frequency, the guideline stipulates that the remaining service life (period of time remaining until a pipe is thinned to the calculated minimum thickness requirement) should be determined and that inspection activities should be continued until the remaining service life decreases to two years. Pipes found to have a remaining service life of two years or less require corrective action.

3.4.3. Further Initiatives for revision of guideline

PWR utilities in Japan checked the piping thickness of the components required to be inspected in a well-planned manner based on the guideline. Even after the recent secondary system pipe rupture accident at the Mihama Nuclear Power Station Unit 3, the guideline is still regarded to be suitable as a management programme in general.

However, it is pointed out that the utilities should establish a new private-sector guideline for pipe thickness management, based on accumulated measurement data and latest knowledge from overseas, in order to ensure more strict control over pipe wall thinning. The Japan

Society of Mechanical Engineers (JSME) is thus currently developing a private-sector guideline for pipe thickness management.

3.5. PERFECT Program on "Multi-Scale Model of Irradiation of Materials"

The key objectives of PERFECT can be summarized as follows:

- To build RPV-2, complemented with a toughness module for the modelling of irradiation-induced evolution of fracture toughness of in RPV steels;
- To build INTERN-1, complemented with an IASCC module for the simulation of irradiation effects in stainless steel, including irradiation assisted stress corrosion cracking;
- To 'validate' these tools and to carry out an European Collective Exercise on material behaviour using numerical tools;
- To disseminate knowledge and experience beyond PERFECT: Euratom fusion, other international organizations;
- To carryout training activities, seminars and workshops in international environment (can be EU funded) and make use of the Marie Curie grant system.

The project is run by a Consortium of 30 partners led by EdF (18 Universities), with a budget of 18 million Euros (7.5 million Euros is the budget provided by EURATOM FP6 via European Commission Research DG). The official kick-off was held in January 2004, and the foreseen duration of the project will be 4 years.

3.6. Erosion-corrosion monitoring, forecast and control of processes of second circuit of WWER NPPs' damage due to operating conditions

Safety depends on, to an important degree, on the erosion-corrosion durability of the equipment and especially pipelines, which work in single — phase and double — phase flows of coolant. A variety of destruction failure mechanisms of SSCs are relatable to differences in phase condition, thermodynamic, hydrodynamic, water chemistry and other parameters.

Analysis of statistic data about the damage of main feed water pipe and of steam-water pipe and investigations in this field shows that the most activities are done without taking into full account the prevailing conditions. This makes it difficult to find correlations and to apply results to avoid future incidents. Classification of failure mechanisms of SSCs gave the possibility to select two groups of mechanisms of destruction (failure) of the metals: the first — which damages surface (reduction of thickness), the second — which breaks down the inner structure of metals.

The following examples are from broken piping in NPPs from the Russian Federation (RF) and the Ukraine.

- Yuzhnoukrainskaya NPP, Unit 2, August 26, 2005: Unit was in operation and subjected to an emergency shutdown as a result of rupture of the condensate pipeline with diameter 219 x 8 mm (after the condensate gathering tanks of the separator-overheater of the first level towards deaerator).
- Yuzhnoukrainskaya NPP, Unit 2, May 19, 2005: Damaged of the piping of the heating steam condensate of the high pressure heater and a ruptured the drainage piping with diameter 500 mm.

- Zaporozhskaja NPP, Unit 4, September 24, 2004: Leakage in the steam pipeline of the first heating steam extraction towards the separator-overheater of the first level.
- Balakovo NPP, Unit 2, November, 2004: There was an emergency shutdown according to the signal of one reactor coolant pump (of two pumps under operation) switch-off as a result of leaking of the nominal diameter 100 (108 x 6 mm, material - steel 20) feed water piping before the water valve on the un-switched section related to SG-4 piping system. The cause was an erosion-corrosion wear of the piping - there was observed wall thinning close to the weld joint up to 3-3.5 mm with consequent crack formation having a length 185 mm and maximum crack opening dimension of 1.5 mm. An operational inspection work programme was insufficient that it did not allow detection of the damaged region efficiently. (See Fig 9).

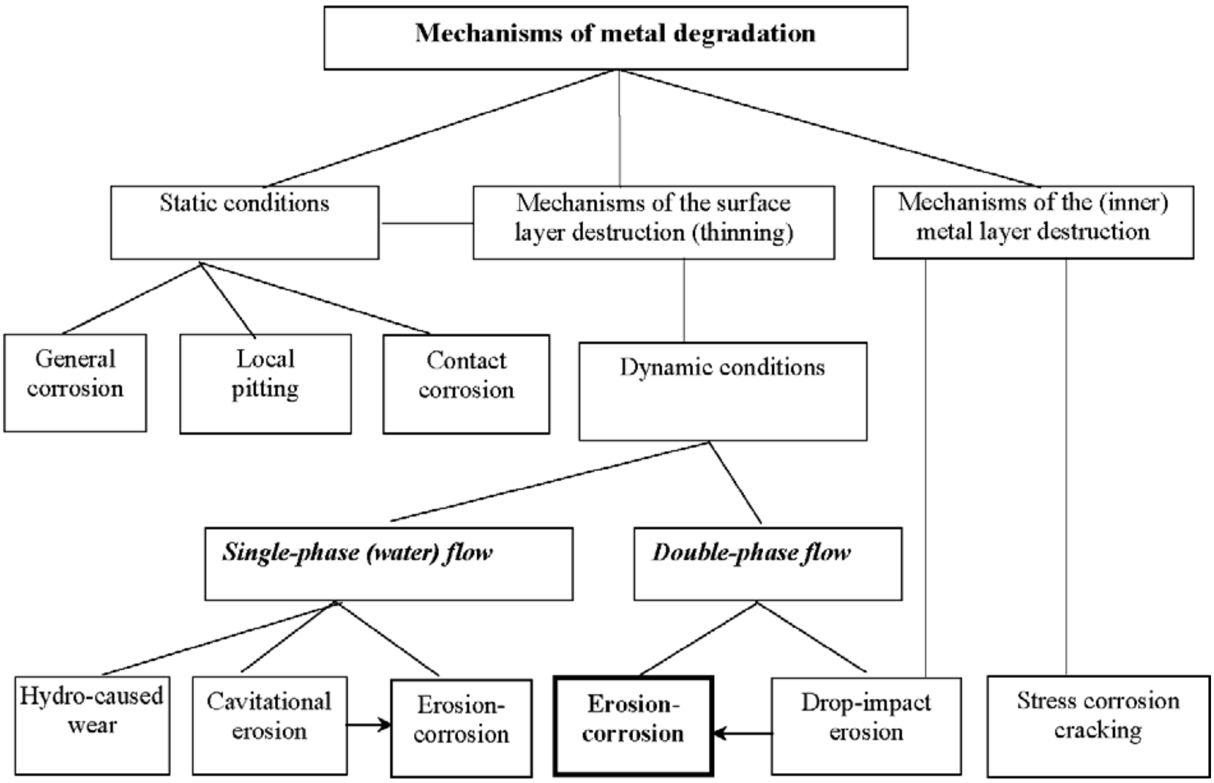

Fig. 8. Classification of the mechanisms of metal degradation in single- and double-phase flows in NPP operational circuits.

During a period between 1986-2002 in the RF and the Ukraine NPPs there were 45 cases registered of NPP's operational failures as result of pipelines' FAC:

- WWER-440 NPPs - 19 failures,
- WWER-100 NPPs - 17 failures,
- RBMK NPPs - 7 failures.

Failures in NPP operation as a result of FAC are event, which occur one or two times per year in Russian Federation and Ukraine NPPs.

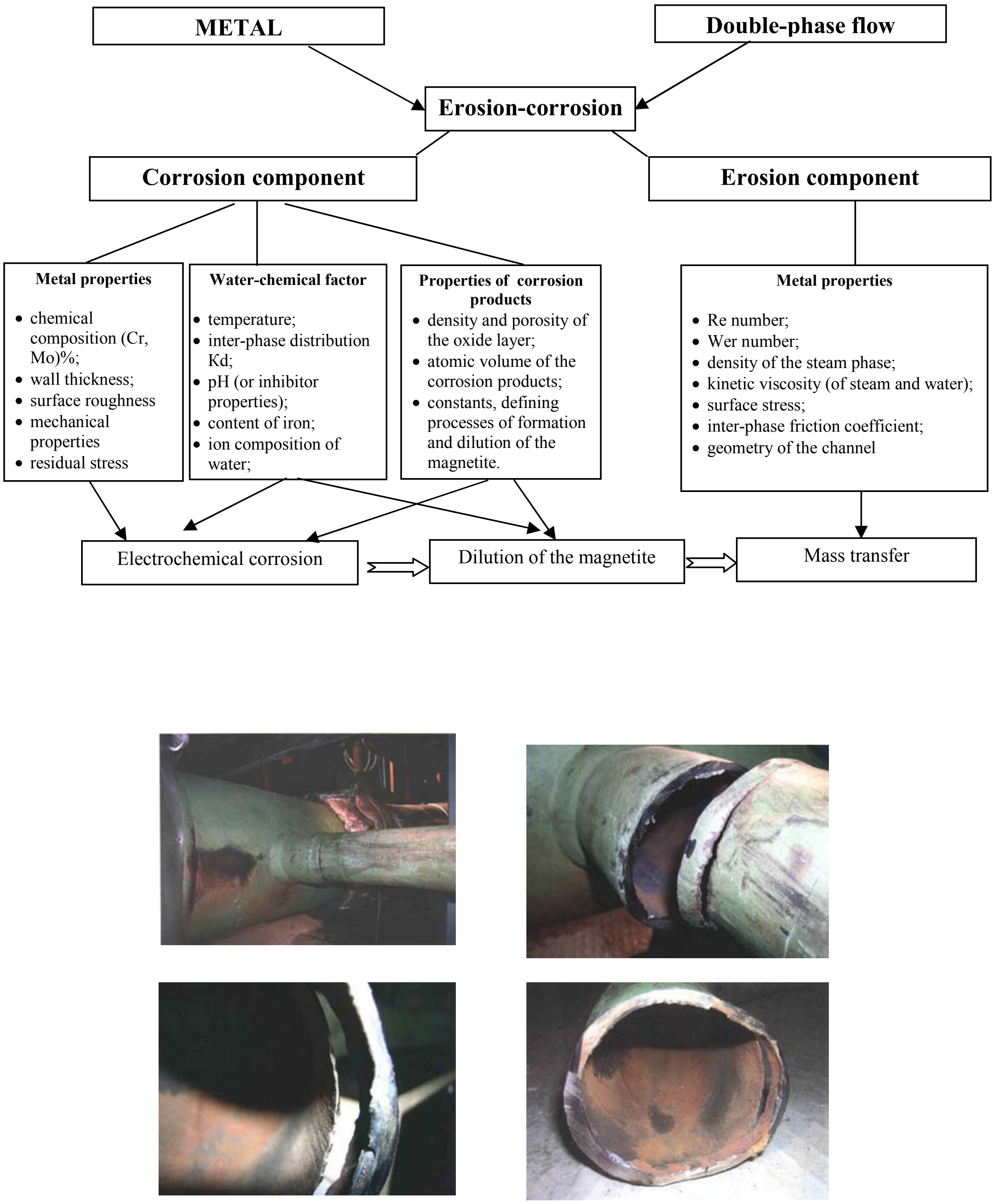

Fig. 9. Main factors and final results of EC in secondary piping.

4. MANAGERIAL ISSUES

This session shared and addressed lessons learned on the aspects of management systems. This was focused on the technical issues presented in the previous sessions and aimed to identify outstanding related managerial issues in the field. Presentations covered management system issues such as:

- Ageing Management (managerial point of view);
- Managing operational experience;
- Qualification of contractor's competence and relationship with contractors;
- Management system/ decision making process;
- Concern over healthy and safe working environment.

4.1. Leibstadt (Switzerland) NPP programme for the prevention of piping degradation due to flow-accelerated corrosion phenomena

FAC also known as erosion-corrosion has always been an operational and safety problem in nuclear and fossil-fuelled power plants. The tragic accidents at the US NPP SURRY-2 in 1986 and recently in the Japanese NPP Mihama-3, as well as other failures in power plants elsewhere, prompted utilities and authorities to consider actions to detect, evaluate and monitor such degradation mechanisms. FAC is a degradation process resulting in wall thinning of piping, vessels and equipment made of carbon or low-alloy steel. Areas suffering from FAC are often difficult to locate, as this degradation process occurs only locally under specific conditions of flow, medium phases present, water chemistry, temperature and material response.

For the Swiss NPP Leibstadt (KKL) a systematic screening procedure was applied with the intention to reliably identify system areas, which may be subject to FAC. During the screening process, the heat balance diagramme of the power plant was modelled using the functionality of the COMSY software tool, including system parameters for each relevant system area. Based on this model, an analysis of the water chemistry cycle was performed, considering associated thermo-hydraulic parameters. Taking into consideration the materials used in each case, the system areas were studied with respect to the degradation potential imposed by FAC

For system areas liable to degradation, a detailed lifetime analysis was performed for individual piping elements. Based on the predicted service life, elements were prioritized for inspection programmes. The result of element examinations were fed back to further improve lifetime predictions. On this basis, the FAC inspection programme was optimized. (see Fig. 10)

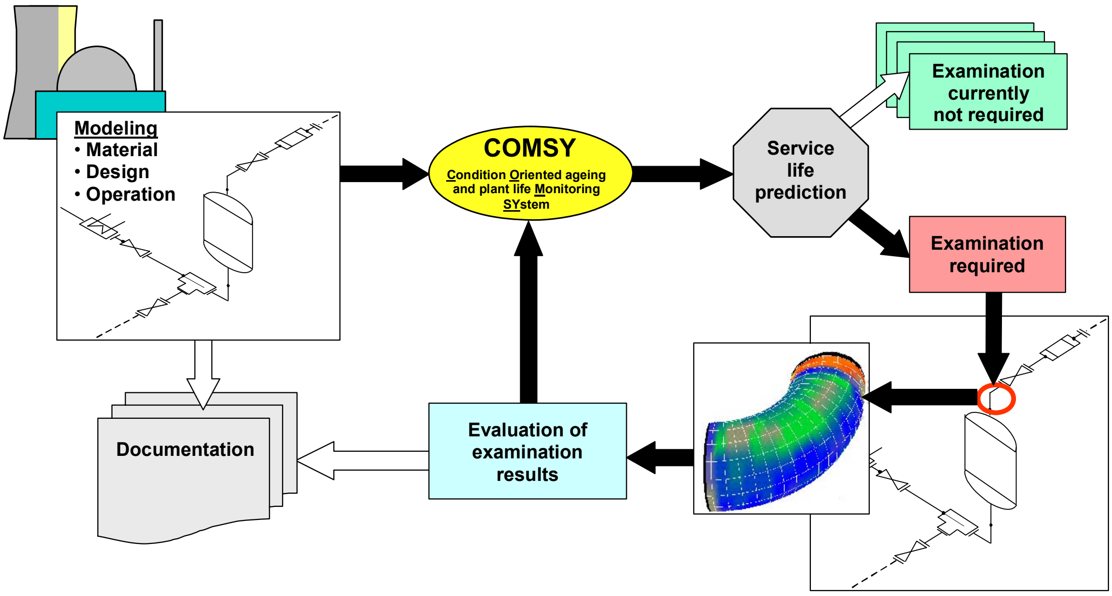

Fig. 10. The closed loop process for PLiM.

4.2. Managerial improvement efforts after finding unreported cracks in reactor components

In 2002 Tokyo electric Power Company (TEPCO), Japan found that there were unreported cracks in reactor components, of which inspection records had been falsified. Stress corrosion cracking (SCC) indications found in core shrouds and primary loop recirculation pipes at some plants were removed from the inspection records, and not reported to the regulators. Cultural and organizational weaknesses were discovered during the following investigations. A production-oriented culture prevailed, and continuous power generation was pursued based on one-sided judgments without sound independent verification and control. Subjective reporting criteria were used, and the crack indications were not reported and only disclosed after they had been judged to be significant. Complacency concerning operational performances weakened the organizational learning attitude and continuous improvement. Quality management was not fully functional because work processes were complicated and accountability was not clear.

The top management of TEPCO took full responsibility and resigned, and recovery was started under the leadership of a new management team. First of all, behavioural standards were reconstituted to strongly support the safety-first value. Ethics education was introduced and a corporate ethics committee was organized with participation of external experts. An independent assessment organization was established to enhance quality assurance. Information became more transparent through a Non-Conformance Control Programme. As for the material management, prevention and mitigation programmes for SCC of reactor components were re-established. (See Fig. 11)

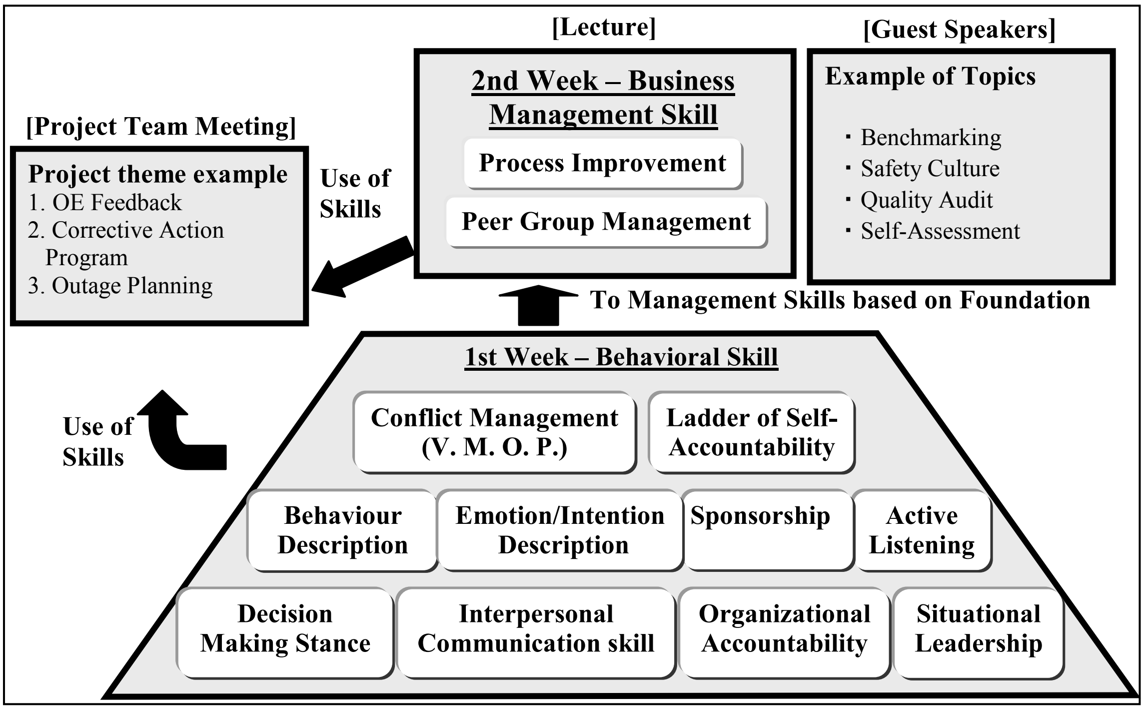

Fig. 11. Leadership training program model.

In addition to the above immediate recovery actions, long term improvement initiatives have also been launched and driven by an aspiration to excellence in safe operation of nuclear power plants. Vision and core values were set to re-align the personnel. Organizational learning was enhanced by benchmark studies, better systematic use of operational experience, self-assessment and external assessment. Leadership training was introduced to develop people to lead the change in various levels of the organization of TEPCO and contractor companies. Based on these foundation blocks, and with strong sponsorship from the top management, work processes were analyzed and improved by peer groups. In addition, peer groups improved commonality of the processes so that best practices could be shared more easily among different plants. Also, a measurement system is now introduced to monitor the progress of improvement.

4.3. Aspects of unexpected events in nuclear power plants

Unexpected events in nuclear power plants (NPP) may lead to upset conditions or even accidents. Events such as these affect not only safety, but also the economic viability of NPP operation. Another facet of such events, virtually irrespective of their degree of severity, is the generally negative impact on public acceptance of nuclear power, such as was seen as a direct result of the Three Mile Island (USA - 1979) and Chernobyl (Ukraine - 1986) accidents. The operators of NPPs are responsible for their safe operation, whilst regulators ensure that NPP operating practices (e.g. start-up, shut-down procedures, inspections, monitoring, and compliance with technical specifications (TS)) are such that the highest possible levels of safety are a priority present at all times. As a matter of engineering principles, designs of NPPs feature safety margins and they are based on conservative assumptions, mostly to allow for material response to the operating conditions and environment (e.g. neutron embrittlement, fatigue usage). Inspections and monitoring have the purpose to check whether systems structures and components (SSC) are behaving according to the design with regard to compliance with safety requirements even when “aged”.

The paper examined aspects concerning events or accidents in NPPs, despite generally high levels of SSC monitoring and inspection and regulatory oversight. The importance of materials selection at the design stage, and the need for vigilance and questioning attitudes was stressed. The necessity to learn from accidents or events that have occurred in other NPPs is shown to be an important tool and source of information for NPP designers, manufacturers, operators and regulators.

4.4. Overview of the French Atomic Energy Commission R&D Programme for reactor material ageing phenomena understanding and modelling

In co-operation with French Nuclear Industry, a major part of Research and Development programme performed at the French Atomic Energy Commission (CEA) is dedicated to the ageing management of NPPs, mainly PWRs. The paper described the R&D projects performed under the industrial and utility partnership. The different R&D projects dealt with the main following topics of importance for the component ageing of PWRs.

The first topic dealt with the RPV materials irradiation embrittlement investigation to ensure 50 or 60 years for the life time of the vessel based on projects on a large experimental programme mainly performed in the OSIRIS reactor and LECI hot cells labs, material modelling and improvement in new methods for structural mechanics margin determination. The second topic dealt with the issue of Irradiated Assisted Stress Corrosion (IASCC) and Irradiation Swelling phenomena for internal structure materials under irradiation, mainly in the framework of international programme as the CIR2 and GONDOLE Projects.

5. REGULATORY ASPECTS

This session discussed the sharing of roles and responsibilities among parties involved, and polices on regulatory aspects. Following topics were covered by presentations:

- Regulatory vs. non-compulsory inspection;
- Regulatory control of sub-contractors;
- Risk and reliability evaluation of components and piping;
- Incident Report System (IRS) topical study on corrosion cracking events.

5.1. Overview of regulatory aspects of ageing issues of nuclear power plants in Japan

As of January 2005, approximately half of 53 units of BWRs and PWRs have been in commercial service more than 20 years, and 7 units over 30 years in Japan. The Nuclear Safety Commission and the Nuclear and Industrial Safety Agency are concerned with various potential ageing effects threatening safety of the NPPs, and are proactively working on the ageing issues from the regulatory standpoints.

The overview paper of regulatory aspects of the ageing issues covered the following topics: Ageing effects observed in some recent cases; roles and responsibilities shared among relevant organizations; topical regulatory activities such as revision of laws and regulations; development of technical codes and standards and incorporation in the prioritized nuclear safety research programme.

Japan is aware that (1) Ageing issues are of importance for safety assurance of NPPs on the background of increasingly aged plants and ageing effects observed in the recent events and cases, (2) Upgrading of regulatory activities is in progress in establishing legislation and conducting reviews and inspections to ensure operator's appropriate ageing countermeasures, and (3) One of the most and effective measures responding to ageing issues is to share experiences and knowledge on ageing problems among generations, organizations and countries. International engagement on the issues should continue to be encouraged.

5.2. Canadian regulatory approach towards ageing management programmes and critical component condition monitoring and evaluation

Effective ageing management programmes and condition monitoring of key safety-related SSCs are an essential aspect for ensuring the long-term safety and reliability of NPPs. The paper presented the Canadian Nuclear Safety Commission's (CNSC) approach towards ensuring that licensees operate and maintain their NPPs in a safe condition. It described the processes and requirements in place that ensure prompt notification is given to the regulator following the discovery of previously unconsidered ageing phenomena through in-service and periodic inspections or through in-service failures.

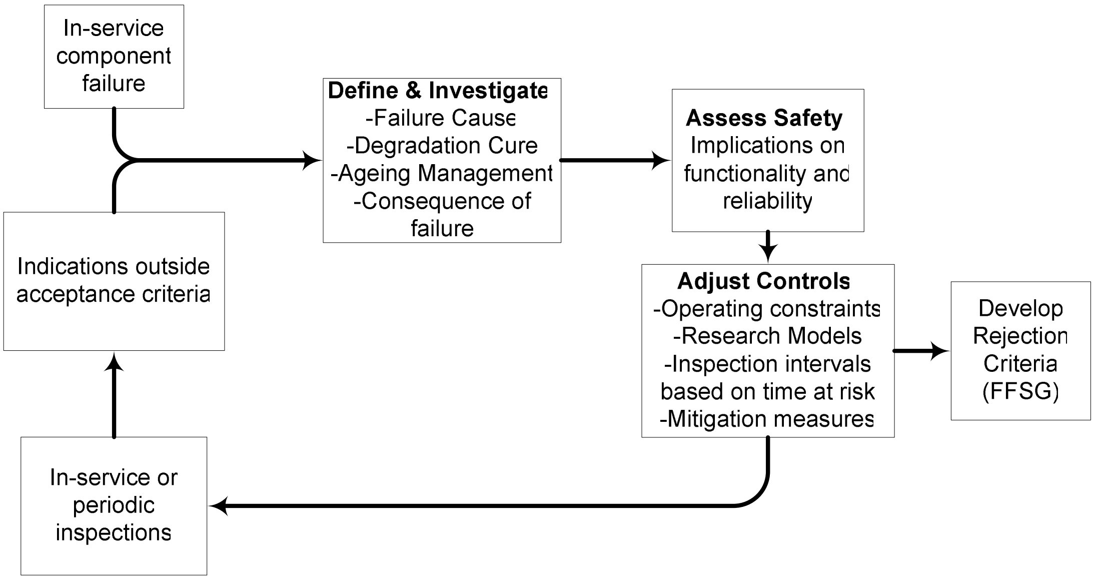

Fig. 12. Regulation by feedback process for managing component degradation.

The paper went on to describe the regulatory response towards these events, which involved requiring the licensee to investigate the cause of the failure, re-assess the safety of the facility, adjust the controls on plant operation and surveillance, and implement measures to monitor the condition of the SSC (See Fig. 12).

The paper also briefly discussed the known degradation mechanisms of key CANDU SSCs, and described the requirements in place to ensure licensees sufficiently monitor the condition of these SSCs and appropriately disposition the results of these inspections. Finally, the paper described the current and planned initiatives to improve the Canadian regulatory requirements for ageing management programmes, as well as the oversight for the surveillance of critical NPP SSCs, including the use of probabilistic tools for condition evaluations and in-service inspections, and discussed the need to increase these efforts in order to account for the increasing effects of degradation mechanisms as Canada's power reactors approach their end of design life.

6. CONCLUSIONS AND RECOMMENDATIONS

6.1. Administrative issues

The technical meeting was realised in a very short time from notification. The response to the call for papers was quick, reflecting the importance of the subject matter of the WS to the MS. The IAEA demonstrated its ability to react to issues as they arise, and showed its goal to provide MS with actual information, which is necessary in order to assure the safe operation of NPPs. The organization of the technical meeting was good as reflected by the answers to the questionnaires distributed at the end.

6.2. Technical/managerial issues

The quality of the papers and presentations was high, and they provided good technical and managerial insights to materials degradation issues. An overview of lessons learned are as follows:

- A strong and comprehensive corrective action programme is necessary to avoid an accident or a "near miss;
- Recovery of a NPP after an accident or "near miss" is both lengthy and costly. In fact more costly than to effectively manage the issues in the first place;
- Equipment and material problems and anomalies must be rigorously addressed in a thorough and timely manner. Awareness of similar problems in other NPPs should guide corrective and mitigative actions;
- Minimum compliance with regulatory standards is insufficient, as this "minimalist" attitude may erode safety culture standards. NPPs must always strive for excellence in operation;
- Organizational learning through benchmark studies and continual self-assessment should be part of daily operation;
- Active involvement in several international projects/groups on ageing management (IAEA, OECD-NEA, EC) is advantageous to all. Knowledge sharing and implementation will improve awareness to potential problems;
- Administrative "ageing documentation" is not an optimum way to transfer knowledge concerning ageing management. Implicit knowledge (from documentations) is not as good as explicit knowledge gained by "hands-on" experience through appropriate training schemes;
- Past plant operating successes can lead to complacency and failure to maintain and advance standards of excellence;
- Having a high production goal rather than a safety focus is a costly and flawed choice;
- Involving independent quality assurance organizations is a way improve the overall attitudes and approaches to plant operational processes.
- Analyse and consider the impact of cost reductions on safety and reliability;
- Ethics programmes, including behavioural standards and organizational transparency are an important aspect to nurture a sense of pride of "ownership" of the NPP within the operating personnel.

6.3. Underlying causes

By evaluating of common aspects of the issues presented in this material, one can conclude that the initiating technical issue was rarely the root cause of the resulting problem. That is, each technical issue was relatively well understood (at least in some segments of the industry) and frequently had been experienced to some degree at other NPPs. What complicated each issue and was at the root of most, was some shortcoming in the facilities' management process.

The table 3 shows some common shortcomings in management processes and indicates what can occur if these shortcomings are evident to a significant degree.

TABLE 3. COMMON SHORTCOMINGS IN MANAGEMENT PROCESSES

Cause		Description
1	Management becomes over confident or complacent.	This can occur when plants run reliably for long periods of time. Programmes and processes may not be continuously improved to incorporate best industry practices. Benchmarking other good facilities may not occur. Minor problems may be overlooked or not evaluated for adverse trends.
2	Ineffective corrective action programme.	Workers are not encouraged to find and fix problems. Based on managements' reaction to problems identified, workers may stop bringing problems to management's attention. Minor problems are not critically assessed for possible consequences and therefore lessons learned are not developed and incorporated into similar programmes and processes.
3	Ineffective evaluation and use of operating experience.	Issues or events at other NPPs are not adequately evaluated and lessons learned are not effectively incorporated into the facility's processes and programmes.
4	Findings of external auditors, inspectors, or regulators are not implemented or even accepted.	Management may believe that external sources' findings are not as significant as the evaluator's depict. There may be a sense that the evaluators are biased in their findings.
5.	Management becomes over focussed on production, sometimes at the expense of safety	Incremental short cuts for efficiency gains may be taken to improve production without an adequate review of the increase in risk. The synergistic effects of these incremental changes in risk may not be well evaluated, understood, or fully appreciated.

Not all of these causes may be present in each case. Moreover, these causes are not always evident or plentiful. At times it is a gradual loss of safety culture that when combined, can lead to unexpected and intolerable consequences.

6.4. Main recommendations

- It was seen that NPP management performance was a critical aspect in determining whether NPPs are operated safely;
- Questioning attitudes and a good safety culture are key elements in NPP operation;
- Vigilance is always needed, even when a NPP has had a very reliable operating history in the past;
- Anticipate material degradation areas by effective use of experience and available literature to avoid unexpected events;
- Close collaboration between operators, safety experts, regulators and NPP management must be nurtured at all times to create an open and transparent approach to operation and appropriate regulation of NPPs;
- There is a significant and basic difference between maintaining safety and improving safety;
- Plan for obsolescence and procurement of replacement SSCs to facilitate the business case;
- Keep all processes as simple and transparent as possible (inspection, management, monitoring).

PAPERS PRESENTED AT THE TECHNICAL MEETING

REPORT OF INCIDENTS

(Session 1)

SUMMARY OF THE INTERIM REPORT ON THE SECONDARY SYSTEM PIPE RUPTURE AT UNIT 3, MIHAMA NUCLEAR POWER PLANT

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Abstract. The Nuclear and Industrial Safety Agency of Japan, through the Accident Investigation Committee, has been conducting an investigation into the secondary system pipe rupture at Unit 3 of the Mihama Nuclear Power Plant, which occurred on 9 August 2004. The interim report was publicized on 27 September 2004, while further detailed analytical investigation is expected. A secondary side carbon steel pipe ruptured. Thinning had developed at the downstream of an orifice due to the combination of erosion caused by the mechanical action of the internal water flow and corrosion caused by chemical reaction. The pipe had gradually lost its thickness, which led to a loss of strength and eventually to a rupture from internal pressure. The investigation has revealed that the dimple pattern typical in this kind of phenomenon was found on the inner surface of the pipe.

1. PIPE RUPTURE MECHANISM

In the ruptured pipe (carbon steel), thinning had developed at the downstream of an orifice (see Figures 1 and 2) due to the combination of erosion caused by the mechanical reaction of the internal water flow and corrosion caused by chemical reaction. It is estimated that the pipe had gradually lost its thickness and that the loss of strength led to a rupture from internal pressure (about 1 MPa). The investigation so far has revealed that the dimple pattern typical in this kind of phenomenon was found on the inner surface of the pipe.

2. PIPE THICKNESS MANAGEMENT FOR THE RUPTURED PORTION

The ruptured portion should have been subjected to KEPCO's pipe thickness management scheme requiring the measurements of pipe wall thickness, according to their document entitled "The thickness management guideline for PWR secondary system pipes," adopted in 1990. However, the ruptured portion was missing in the inspection list that Mitsubishi Heavy Industries, Inc. (MHI) prepared based on the Guideline. In the past, there had been some occasions when the list might have been corrected to include the ruptured portion, e.g., the transfer of the inspection activities from MHI to Nihon Arm Co. Ltd., the correction of the inspection list at other NPPs by MHI to include the same location as the ruptured one at Mihama Unit 3, revision so as to add the ruptured portion to the inspection list by Nihon Arm Co. Ltd., etc. The facts revealed, however, that thickness measurement of the ruptured portion had not been conducted because KEPCO and the others had never checked whether there was any omission in the list.

The immediate cause of the pipe rupture at Mihama Unit 3 was "the omission of the failed portion from the inspection list, and the fact that this had not been corrected before the accident occurred" due to "the failure in pipe thickness management for the secondary system involving three parties, KEPCO, the MHI, and Nihon Arm Co. Ltd.". This means that quality assurance and maintenance management had failed to function properly in KEPCO, who is responsible for the safety of NPP as Licensee.

3. PIPE THICKNESS MANAGEMENT AT NPPS OTHER THAN MIHAMA UNIT 3

NISA conducted survey on pipe thickness management adequacy of all NPPs, and confirmed that inspections have been properly carried out for all NPPs except for the ones operated by KEPCO. As for KEPCO, in their ten NPPs other than Mihama-3, a total of 14 portions had never been checked for thinning or had been inspected using unauthorized criteria. For the portions KEPCO had inspected, including the 14 portions and the other portions NISA obliged to inspect, NISA confirmed the integrity of all those portions, except for three portions which needed pipe replacements.

4. IMMEDIATE ACTIONS BY THE GOVERNMENT AND LICENSEES

The following are the actions to be taken by the Government, Licensees and other relevant parties:

4.1. ACTIONS TO BE TAKEN BY THE GOVERNMENT

- Clarification and dissemination of the licensees' inspection methods concerning pipe thickness management
- Verification of implementation of pipe thickness management and observance of internal rules during the nuclear safety inspection; verification of the implementation system for pipe thickness management by the Licensees (including their subcontractors) during the periodic safety management review
- Adoption of the standards being prepared by the Japan Society of Mechanical Engineers (JSME) concerning the pipe thickness management methods and use of these JSME standards as the criteria
- Direction for and supervision of the Licensees (including the subcontractors) to conduct more careful subcontracting management

4.2. ACTIONS TO BE TAKEN BY THE LICENSEES

- Review of the "PWR Thickness Management Guideline" based on the latest knowledge and actual data, including the handling of the "main systems to be inspected" and "other systems"
- Compiling guidelines for pipe thinning management for BWR
- Construction of management system for systematic "inspection lists," which link the computerized piping system diagram to the management charts to conduct effective maintenance of all the equipment subject to the periodic Licensees' inspections
- Development and observance of rules specifying the methods and responsibility-sharing concerning subcontracting management for the maintenance management activities
- Identification of matters to be specified in contracts and order documents, in order to clarify the rights and duties concerning subcontracting parties
- Provide feedback on the lessons learned from the pipe rupture accident at Mihama Unit 3 so that they are reflected in the safety-ensuring activities of the licensees
- Implementation of training to ensure safety of workers and clear notification of risk information in critical areas

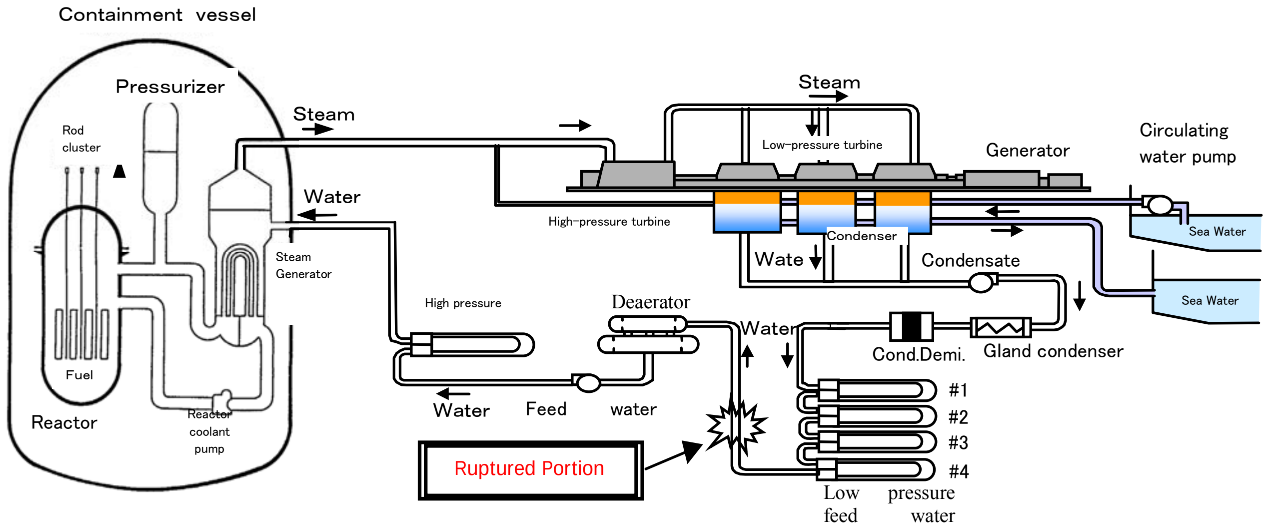

Fig. 1. Mihama unit 3 system diagram.

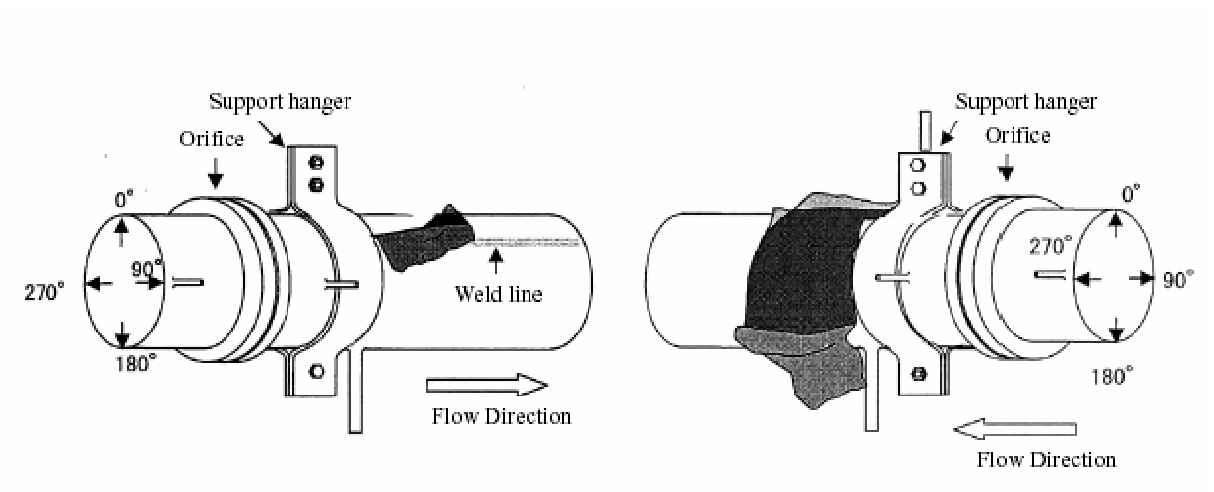

Fig. 2. Configuration of rupture.

5. OTHER MATTERS FOR CONSIDERATION

It is essential to implement the actions proposed by this report without delay.Other preventive measures may be added, depending on the findings in the investigation toward the final conclusion. Some attribute the accident to the aging of nuclear power plants, but its direct cause was the absence of the inspection even under the proper thickness management. However, needless to say, more comprehensive inspection management is required for aging nuclear power plants. In view of this, even more important are the periodic safety review conducted every 10 years and the comprehensive evaluation of the aging of power plants over 30 years.

LESSONS LEARNED — REACTOR PRESSURE VESSEL DEGRADATION

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Abstract. This paper discusses the discovery of the near through-wall corrosion of the reactor pressure vessel closure head at the Davis-Besse Nuclear Power Station in March 2002. Although no loss of coolant occurred and the reactor core always remained fully covered and cooled, this incident represented a significant degradation of the nuclear safety margin at the facility. This paper describes what caused the corrosion, why it occurred, what actions were taken to correct the root cause of the degradation and finally some lessons-learned from the incident.

1. INTRODUCTION

This paper discusses the discovery of the near-through corrosion of the reactor pressure vessel closure head at the Davis-Besse Nuclear Power Station in March 2002. Although no loss of coolant occurred and the reactor core always remained fully covered and cooled, this incident represented a significant degradation of the nuclear safety margin at the facility.

This paper describes what caused the corrosion, why it occurred, what actions were taken to correct the root cause of the degradation and finally some lessons-learned from the incident. The FirstEnergy Nuclear Operating Company, owner operator of Davis-Besse, is committed to sharing its experiences and lessons-learned throughout the world in the hopes of helping assure that such a lapse in safety never occurs again.

2. BACKGROUND

Davis-Besse is a raised loop pressurized water reactor (PWR), manufactured by Babcock and Wilcox. The reactor licensed thermal power output is 2772 megawatts. The architectural engineer and constructor was Bechtel Power Corporation. The owner operator of the facility is the FirstEnergy Nuclear Operating Company (FENOC). The facility is located in Oak Harbor, Ohio on the western shores of one of the Great Lakes — Lake Erie.

The plant began commercial operation in August, 1978 and is currently licensed to operate until April 2017. The reactor pressure vessel has an operating pressure of 2155 psig (151.50 kg/square cm) and a design pressure of 2500 psig (175.75 kg/square cm). Davis-Besse had accumulated 15.8 effective full power years (EFPY) of operation when the plant shut down for its thirteenth refueling outage on February 16, 2002. During that refueling outage, while performing reactor pressure (RPV) vessel closure head inspections required by the United States Nuclear Regulatory Commission, workers discovered a large cavity in the 6 inch (15.24 cm) thick low-alloy carbon steel RPV head material. The cavity was about 6.6 inches (16.76 cm) long and 4 to 5inches (10.16 to 12.70 cm) at the widest point extending down to the 0.25 inch (0.635 cm) thick Type 308 stainless steel cladding.

FENOC promptly commissioned a root cause team to evaluate what had caused the corrosion. After initially considering a repair of the RPV head, FENOC purchased an unused head of the same design. The facility was out of service for over 2 years while the head was replaced and other wide-scale evaluations and improvements were made to the physical plant, programs, and staffing organization.

3. HISTORICAL PERSPECTIVE

The Davis-Besse plant had an extended shut down in 1985 following a loss of feedwater incident. Since that time and throughout the 1990's the plant had been a top industry performer. The plant's capacity factor continuously improved during this period and by 2001 had reached 99.5%. During this same period the plant experienced few forced outages. The Institute of Nuclear Power Operations (INPO) consistently rated Davis-Besse as a superior performer. From 1991 until it changed its rating process in 1997 the Nuclear Regulatory Commission gave Davis-Besse a rating of 1, its best rating.

Although the plant continued to operate well and by outward appearances continued to be a strong performer things were occurring in the plant and more importantly throughout the organization that would soon become evident and have significant ramifications. The impact would affect not only Davis-Besse but all other FENOC plants, all U.S. reactors and the Nuclear Regulatory Commission. These impacts are discussed in more detail in another paper to be presented at this workshop.

4. CHRONOLOGY

The plant was exhibiting signs that indicated conditions inside containment were changing. During earlier refueling outages boron was observed on the RPV head (see Figure 1). This was believed to be from control rod drive mechanism flange leakage above the RPV head as the plant had a history of leakage at this location. Unidentified leakage had historically been very low (~0.05 gpm) (~0.189 l/min) in the early 1990's. In the late 90's leakage increased to ~0.8 gpm (~3.03 l/min) which was attributed to relief valve leakage. The relief valves were repaired but leakage remained at 0.2 gpm (~0.757 l/min). Radiation monitors inside containment were requiring filter replacement with significantly greater frequency. Additionally, the containment air coolers were accumulating boron and required more and more frequent cleaning while the plant was in service.

While station personnel addressed each of these symptoms on an individual basis, a systematic assessment was not done which may have linked all the symptoms and identified the true cause earlier. During this same period the organizational infrastructure was also degrading. Due to strong production performance the plant received less Nuclear Regulatory Commission inspections than other plants in its region. The organization began shifting from a strongly led Operations-oriented organization to more of an Engineering-led organization. As the plant focused more on containing operating costs, it became isolated from the rest of the industry and did not keep up with increasing industry standards and operating experience. Management allowed a minimum compliance standard to evolve within the organization instead of a standard of excellence.

Fig. 1. Condition of Reactor Vessel Head in Refuel Outage 12 (2000).

In the early 2000's INPO ratings began to decline. A decline in plant performance began to become apparent although the plant continued to experience long operating periods. On February 16, 2002, Davis-Besse shut down to refuel and perform mandatory inspections of the RPV head. In early March, significant degradation of the reactor vessel head was discovered. Corrosion of the 6-inch (15.24 cm) carbon steel head material to the underlying 0.25 inch (0.635 cm) stainless steel cladding existed in an area approximately 6 inches (15.24 cm) long by 4 to 5 inches (10.16 to 12.70 cm) wide with approximately 16.5 square inches (106.5 square cm) surface area of the cladding exposed. This exposed cladding had been subjected to an operating pressure of 2155 psig (175.75 kg/square cm) (see Figure 2). Upon discovery of the degradation FENOC initiated two root cause evaluations. One evaluation was to determine the technical root cause of the degradation while the other concentrated on the management and organizational issues that led to this incident.

Fig. 2. Cavity with CRDM nozzle removed.

5. ROOT CAUSE FINDINGS

The technical root cause determined that the corrosion was the result of boric acid interaction with the carbon steel on the RPV head. The source of the boric acid was via a through wall crack in a control rod drive mechanism nozzle (Figure 3). This crack was initiated as a result of Primary Water Stress Corrosion Cracking.

The Davis-Besse organization believed that the boron accumulated on the RPV head was due to leaking control rod drive mechanism flanges above the RPV head and that such accumulation would not cause corrosion due to the elevated temperatures at that location. This boron accumulation on the top of the RPV head was not fully removed during refueling outages and it masked the typical "popcorn" boron indication from a nozzle containing a crack (see Figure 4). Accordingly, the boric acid emitted through the nozzle crack was allowed to corrode the carbon steel head creating a cavity.

Following the discovery of the cavity and the removal of the head, a disk containing the cavity was cut out and sent to a laboratory for examination of the degradation in detail. A complete copy of the technical root cause report and the degradation examination report are available on the Nuclear Regulatory Commission's web-based Agency-wide Documents Access and Management System (ADAMS) at http://www.nrc.gov.

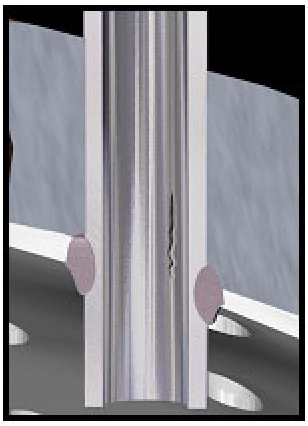

Fig. 3. Depiction of Through Cracking Leakage.

Fig. 4. Typical Appearance of Through Wall Nozzle Crack.

The "Management" root cause concluded that the failure to identify and prevent the degradation of the reactor vessel head was attributable to:

(1) There was less than adequate nuclear safety focus (a production focus combined with taking minimum actions to meet regulatory requirements).

(2) Implementation of the Corrective Action Program was less than adequate as indicated by:

- Addressing symptoms rather than causes
- Low categorization of conditions
- Inadequate cause determinations
- Inadequate corrective actions
- Inadequate trending

(3) The organization failed to integrate and apply key industry information and site knowledge and to compare new information on plant conditions to baseline knowledge.

(4) Personnel did not comply with the Boric Acid Corrosion Control Procedure and In-service Inspection Program, including failure to remove boric acid from the RPV head and to inspect the affected areas for corrosion and leakage from nozzles.

Additional observations from the root cause included:

- Quality Assurance's role was inconsistent or minimal
- Operations had minimal involvement
- Written policies on safety were weak
- Condition reports remained unresolved
- Training weaknesses existed
- Management presence in the field was minimal
- Proper balance of long-term production and safety was lost.

A complete copy of this root cause report is also available at http://www.nrc.gov.

6. ACTIONS TAKEN

Following substantial management changes. FENOC developed the Davis-Besse Return to Service Plan. This was an integrated plan to address all the causes identified in the root cause reports and to satisfy any additional concerns of the Nuclear Regulatory Commission. Figure 5 shows a pictorial of the integration and various building blocks of the Return to Service Plan.

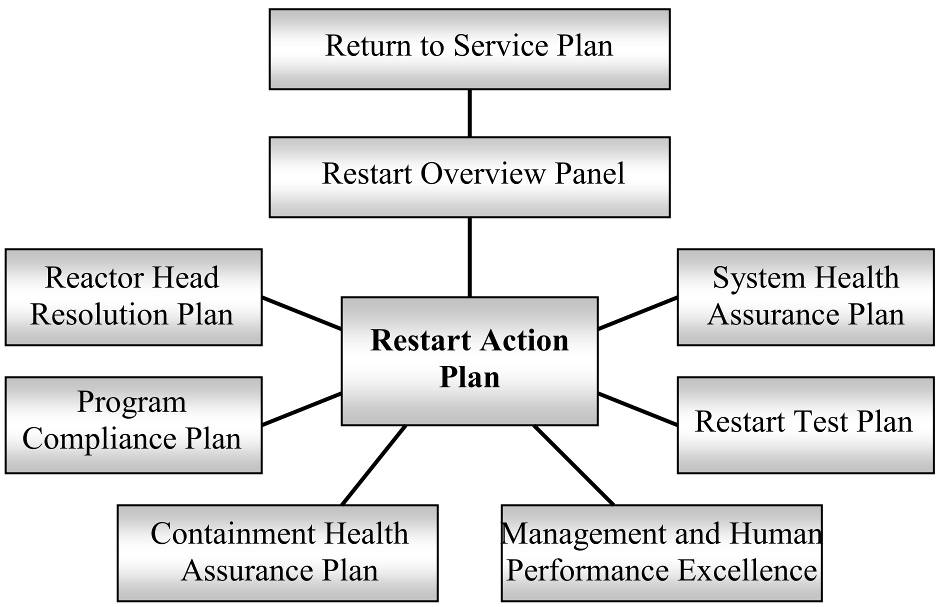

Fig. 5. Return to Service Plan.

To resolve the degradation of the reactor vessel head FENOC initially considered repairing the damaged area but ultimately decided to replace the head. An unused head from an uncompleted facility was procured and installed on the Davis-Besse reactor vessel. The service structure for the head was modified to enable enhanced inspections in the future. A new head, with nozzles constructed of Alloy 690, a material less susceptible to PWSCC, has also been ordered and will be installed in a future outage.

During the comprehensive inspections and reviews that followed, several design deficiencies were identified and corrected. Most notable among these are a significantly enhanced design for the containment re-circulation sump and improvements in the design of the high-pressure injection pumps. As a result of past primary system leakage substantial amounts of boron were deposited on equipment and surfaces throughout the containment. These deposits were all removed and evaluations performed to determine any adverse effects.

In addition to the physical enhancements made to the plant, programmatic improvements were also made. Implementation improvements in the Corrective Action Program received significant attention. Organizational changes included an expanded corporate nuclear structure. Additional management expertise was recruited. A new position of Vice-President Quality Assurance was created, reporting directly to the President of FENOC.

7. LESSONS-LEARNED

The following summarizes lessons-learned from this incident:

- Past plant operating successes can lead to complacency and failure to maintain and advance standards of excellence
- A strong Corrective Action Program is essential for safe plant operation
- Minimal compliance with regulatory standards is unacceptable
- Having a heavy production focus rather than a safety focus is a costly and flawed choice
- Equipment and materiel problems and anomalies must be rigorously addressed in a thorough and timely manner
- Oversight organizations must be involved and unbiased

GUILLOTINE BREAKS OF FW LINES AT LOVIISA 440

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Abstract. The Loviisa Nuclear Power Plant owned by Fortum (by IVO until 1998) consists of two VVER 440 units dating back to the late '70s and the early '80s. A control programme was developed in 1982–1983 to manage erosion corrosion in the secondary system piping and components. The scope of the control programme was enlarged after the Surry accident in 1986 but still the control programme was based on the operation experience and two-phase flow conditions. Due to shortcomings of the control programme, the guillotine pipe break of the feed water system piping occurred in 1990 at Unit 1. Erosion corrosion adjacent to the flow control orifice in the feed water discharge line was incorrectly assumed to be lower than in other during the previous outages inspected fittings (tees, 90 degree elbows, throttles, etc.). However, in spite of the large efforts put on the improving the inspection programme the second guillotine pipe break of the feed water system occurred in 1993 at Unit 2. In order to diminish erosion corrosion of the feed water system piping secondary water chemistry was changed from neutral to alkaline water chemistry in 1994 at Unit 2 and in 1995 at Unit 1, respectively. Also change of larger extent of carbon steel piping to low alloy or stainless steel piping has been carried out.

1. INTRODUCTION

Loviisa 1 and 2 VVER 440 units owned by Fortum started their operation in 1977 and 1981, respectively. The nominal gross electrical power per unit was originally 440 MW, but this has been increased up to 510 MW. Power upgrading was performed 1996–97.

Special features for Loviisa, typical for VVER-440 plants, differing from the western PWR plant secondary system design, are secondary system parameters (e.g. neutral water chemistry until 1995), design and materials (e.g. horizontal steam generators with stainless steel heat exchanging tubes).

The underlying philosophy behind the neutral water chemistry was to protect the most important components in the secondary circuit, i.e. steam generators, by keeping the conductivity of feed water as low as possible and by avoiding to make any chemical addition to it. On the other hand this has increased erosion corrosion susceptibility of the secondary system piping and components. In the early '80s, it became obvious that erosion corrosion of the secondary system piping and components would be an important factor for the safety and availability of the plant [1, 2].

2. EXPERIENCE AND COUNTERMEASURES WITH EROSION CORROSION AT THE LOVIISA POWER PLANT

Quite soon after commissioning, several leaks were discovered in various fittings of the secondary system piping. Due to leaks, these fittings had to be replaced, and subsequently the first inspection programme, was developed in 1982–1983 to manage erosion corrosion in the secondary system piping and components [1, 2].

2.1. EROSION CORROSION IN WET STEAM CARRYING LINES AND COMPONENTS

First wall thinning was discovered in the wet steam systems, such as reducers of auxiliary steam system, cross under piping from HP turbine to MSRH (Moisture Separator Reheater), high and low pressure turbine bleed steam lines and steam generator blow down lines. Therefore also the inspections were at first concentrated on these systems.

The number of single items (T pieces, elbows, reducers, etc.) to be inspected was increased year by year, based on the experience gained from the inspections performed and operation experience from the plant and other plants. In HP-turbine bleed steam (extraction) lines to preheaters, auxiliary steam system and steam generator blow down system erosion corrosion has been very pronounced.

Not only pipe fittings, but also the main components of the secondary circuit were found to be susceptible to erosion corrosion. It was first discovered in wet steam systems in the casing of the HP-turbine at the beginning of the '80s. HP-preheaters and MSRHs have also suffered from erosion corrosion [1, 2]. It was discovered soon after the power upgrading that erosion corrosion of MSRHs is more pronounced than thinning of other components.

2.2. EROSION CORROSION IN WATER CARRYING LINES AND COMPONENTS

Erosion corrosion problems were, however, not concentrated on the systems operating in wet steam conditions only, but also on the components and fittings in single phase (water) conditions, such as feed water, condensate and drain systems.

First heavy erosion corrosion in single phase conditions was discovered at the inlet of the heat exchanger tubes of HP preheaters. The main reason, however, for the severe attack is believed to be not only the unfavourable operation conditions, but also inferior design of the HP preheaters.

After the Surry-2 accident in 1986, an updating of the inspection programme was performed in 1987. The number of individual items (pipe fittings such as tees, elbows, etc.) in the single phase (water) conditions to be inspected during the outages was increased, and the temperature range was increased to cover the whole secondary circuit.

2.3. GUILLOTINE BREAK OF THE FEED WATER LINE AT UNIT 1 IN MAY 1990

During operation at full power 28 May at 10.27 a.m., an electric failure caused stopping one of the four feed water pumps and switching on the reserve pump. This resulted in a pressure shock in the FW-line, which led to a break in one of the five MFWS pipe branches (diameter 325 mm, figure 1).

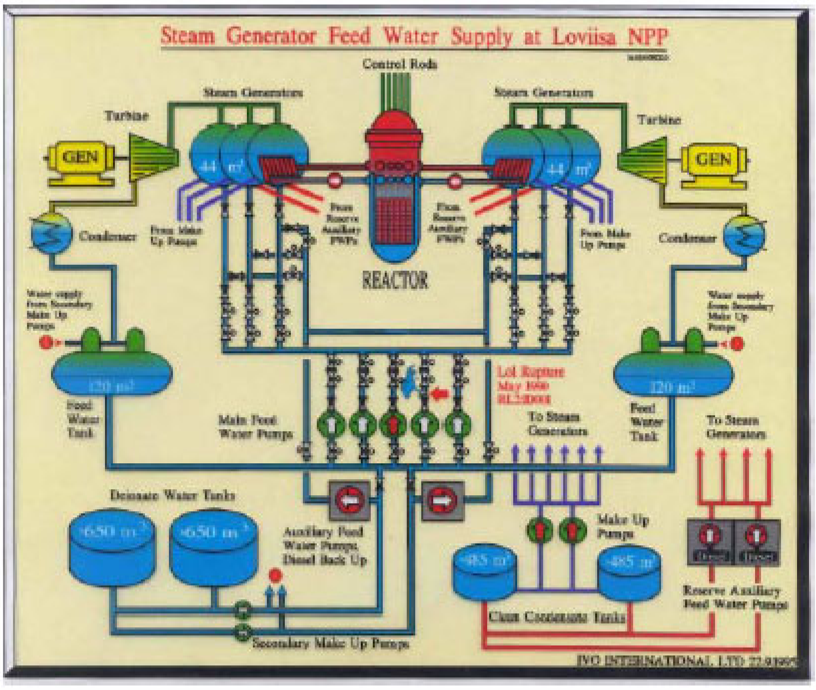

Fig. 1. Loviisa 1, feed water supply and location of guillotine break.

The reactor and turbines were tripped manually. The auxiliary feed water systems and other vital systems operated normally during the incident. The leak was isolated in 17 minutes and the unit was shut down according to the normal routines. There were no injuries of the personnel [3].

The exact location of the break was adjacent to the flow control orifice, Figure 2. The root cause for the failure was that erosion corrosion after the flow control orifice was incorrectly assumed to be lower than in other fittings (tees, 90 degree elbows, etc.). Other fittings in this particular FW-discharge line were inspected during the previous outages prior to the pipe break.

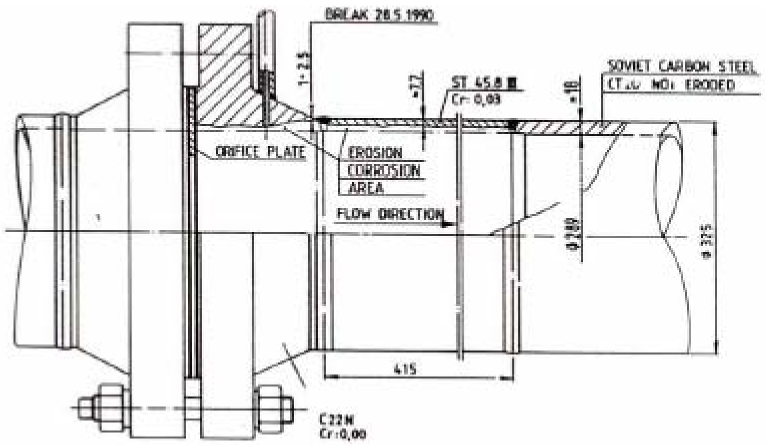

Fig. 2. Exact location of the break.

Chemical analysis of the eroded parts showed that the chromium content was very low (0.00–0.03%). This aspect is also believed to have increased erosion corrosion susceptibility adjacent to the flow control orifice. Wall thinning behind other similar flow control orifices was also detected.

As a countermeasure, flow control orifices and piping adjacent to them in the feed water lines and in other susceptible systems have been changed to more corrosion resistant ones (low alloyed or austenitic stainless steel) in 1990–1996. After the pipe break, special interest has also been paid to inspection of flow control orifices and other throttling devices and piping adjacent to them.

In 1990 an advisory group was nominated within IVO. The task of this group was to evaluate annual inspection programme. A classification of secondary circuit systems based on their safety functions and erosion corrosion susceptibility was made by this on the basis of the operational experience and company know-how.

To support the work of this group a more detailed weak point analysis of the chosen systems was performed by using the WATHEC-software designed for calculation of wall thinning caused by erosion corrosion. A new QC-engineer also was employed for planning and for execution of the annual inspection programme.

2.4. GUILLOTINE BREAK OF THE FEED WATER LINE AT UNIT 2 IN FEBRUARY 1993

On 25 February 1993 at 2.30 p.m., during operation at full power, the start up procedures of one of the feed water pumps were going on. This resulted to a guillotine break in one of the five MFWS pipe branches (figure 3). The reactor and turbines were tripped manually. The auxiliary feed water system and other major systems operated normally during the incident. The leak was isolated in 9 minutes, and the unit was shut down according to the normal routines. There were no injuries of the personnel [4].

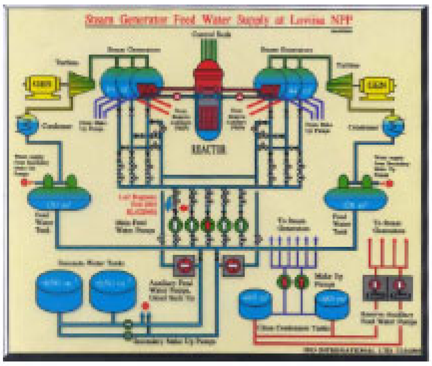

Fig. 3. Location of the broken FW-discharge line   Fig. 4. Exact location of the break.

The exact location of the break in the discharge feed water line was in the flange immediately after the non-return valve, figure 4. There were most probably several reasons which caused rapid wall thinning in the broken flange, such as high turbulence caused by the feed water pump and adjacent components (T piece, non-return valve), additional connection piece (stainless steel for few years) between the non-return valve and the flange and quite low chromium content of the flange. But it is still difficult to explain different erosion corrosion behaviour of similar flanges in parallel lines.

Weak point analysis performed by the WATHEC®-software in 1991–1992 demonstrated that the flange after the non-return valve was one of the weakest points of this part of the feed water system. Based on the wall thickness measurements and the above mentioned analysis, the discharge lines from the feed water pump to the feed water line collector were replaced in 1992 at Unit 1. At Unit 2, a similar replacement was scheduled for 1994. Unfortunately, the inspection technique used in the outage 1992 was unable to detect the wall thinning of the flange at Unit 2. Also misinterpretation of previous inspection results was one of the root causes for the failure [5].

Similar wall thinning was also detected in one of the parallel discharge lines, but it was more concentrated on the weld between the flange and the following expansion than in the flange, Figures 5 and 6. No significant wall thinning of similar flanges in the other three parallel lines was detected [6].

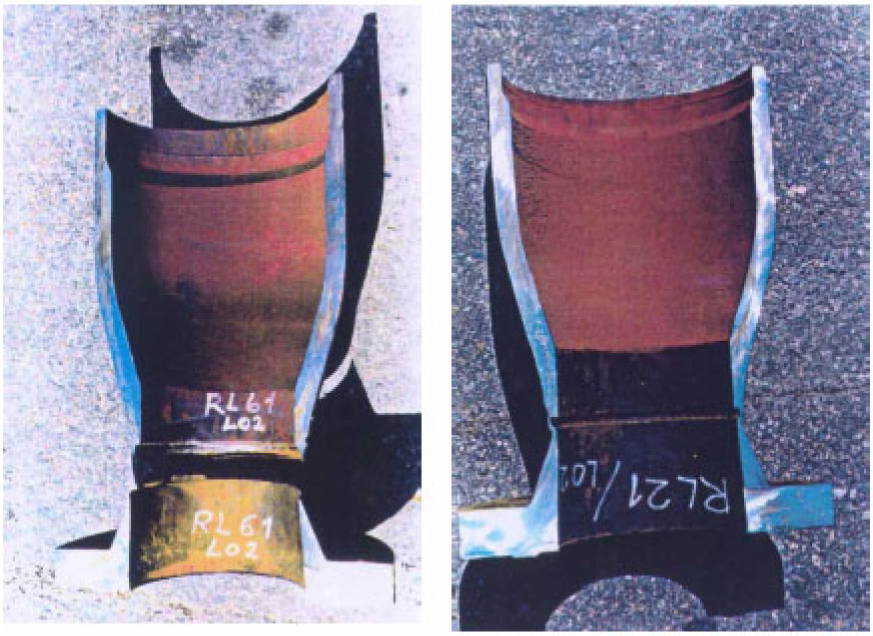

Fig. 5. Wall thinning of parallel FW-discharge lines (RL61/RL21).

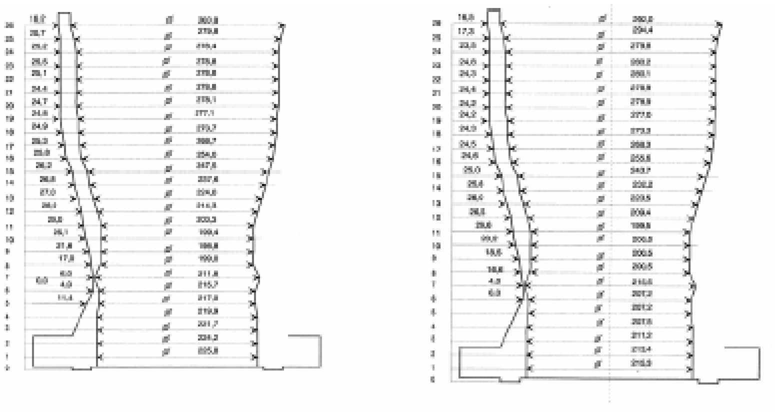

Fig. 6. Wall thickness measurement of flange connection and expansion cone of parallel FW-lines (RL61/RL21).

3. LESSONS LEARNED ON EROSION CORROSION OF THE SECONDARY SYSTEMS PIPING AND COMPONENTS

As a remedy, several options have been implemented both in wet steam and water carrying lines. Since wall thinning of similar piping and components in parallel lines might be completely different and in certain parts very local, great attention has been paid to the inspection programme, inspection techniques and handling of large amount of inspection data. Management of erosion corrosion of the most important secondary systems is part of the PLIM-programme of Loviisa NPP.

3.1. OPERATION EXPERIENCE FROM OTHER PLANTS

The first guillotine break showed the importance of the operation experience from the other plants. Similar break after the flow control had also been experienced at Santa Maria plant in Spain half a year earlier.

3.2. CONTROL PROGRAMME

Over the years, the number of single items (T pieces, elbows, flow control orifices, etc.) to be inspected has been raised. Inspection extent and grid of the inspected piping, fittings and components, which fulfilled the requirements of the ASME Code Case N-480 and ASME XI IWT (draft), have also been increased over the years particularly in the systems most susceptible to erosion corrosion.

In order to improve management of large amount of inspection data and e.g. to analyze the effect of power upgrade the old WATHEC-software has been replaced by new COMSY-software. Inspections performed after the power upgrade showed that location of the wall thinning might be changed or wall thinning was discovered in complete new places or it might be more pronounced than before.

Due to extensive replacement of the piping the number of single inspection items has been reduced during the recent years. Since 1990 the annual inspection programme has been reviewed and evaluated by the advisory group.

3.3. INSPECTION TECHNIQUE

Extensive inspections performed in outages since 1990 have also revealed that wall thinning in certain parts of the systems may concentrate on a very local area, e.g. in the weld and/or counter bore of a fitting, valve or even straight pipe. In order to find out this kind of wall thinning, the amount of TV inspections were increased, and special attention has been paid to inspection of the weld areas. Due to the local nature of wall thinning, inspection techniques have also been improved in the case of flanges and welds, for example. Efforts have also been taken to proper calibration as well as repeatability of successive inspections ("permanent" grids with litho chalk).

3.4. WEAK POINT ANALYSIS AND MANAGEMENT OF WALL THINNING INSPECTION RESULTS

As mentioned before after the first feed water pipe break, a classification of secondary circuit systems based on their safety functions and erosion corrosion susceptibility was made by the company personnel on the basis of the operational experience and company know-how.

A more detailed weak point analysis of the chosen systems was performed in 1991- 1992 by using the computer code WATHEC designed for calculation of wall thinning caused by erosion corrosion. This weak point analysis was updated yearly by taking into consideration the wall thickness measurements performed in the previous outages.

The management of large amount of inspection data (at maximum 400-600 items/unit/year) gathered during the years has been one of the most problematic issues. Extensive work has been done to handle and evaluate this inspection data. Recently the old software (IVO-NDT, WATHEC) used for management the data as well as to perform weak point analysis has been replaced by new COMSY-software.

3.5. CHANGE OF WATER CHEMISTRY

Instead of the neutral water chemistry applied at Loviisa, compatibility of different water chemistries was intensively studied in a feasibility study. Based on the literature study, three alternative water chemistries were evaluated to be implemented at Loviisa, i.e. N2H4 water chemistry, ODA (octadectylamine) water chemistry and morpholine water chemistry. Based on the evaluation, N2H4 water chemistry was chosen for the optional water chemistry (pH< 9). At Unit 2 the change of water chemistry to N2H4 water chemistry was implemented in June, 1994 and at Unit 1 in July 1995, respectively.

3.6. REPLACEMENT OF THE THINNED PIPING AND COMPONENTS

At the beginning, the most common option was the replacement of the thin walled fitting with a similar (material) fitting or with stainless steel fitting.

A step taken to diminish the root course of erosion corrosion, i.e. wet steam, was installation of HVSs (High Velocity Separators) in the 2nd extraction steam lines of HP turbine in 1989-1991 and cross under piping from HP turbine to MSRHs in 1990–2000. One of the options carried out was coating of a component (e.g. HP turbine casing) with spraying.

In the years 1993–2004 change of larger parts of the system has been adopted more often. In these cases also more corrosion resistant material has been used. As an example of the replaced piping systems it can be mentioned:

- Feed water discharge and suction lines
- Most of parallel feed water lines inside the containment
- Most of the HP-extraction lines
- Some of LP-extraction lines
- Some condensation lines
- Auxiliary steam lines

Primary justification for some of the piping replacements has been to increase the plant's safety level for other reasons than erosion corrosion. SME HP-preheaters and modification of MSRHs and FW-pumps are some examples of the replaced and modified components.

4. SUMMARY

Management of erosion corrosion of the secondary systems piping and components has turned out to be an important factor for the safety, plant availability and for the long term operation of the plant. Therefore, management of erosion corrosion of the most important secondary systems is part of the PLiM-programme of Loviisa NPP.

In spite of the extensive attention paid to the inspection programme of the secondary system piping, two guillotine breaks of the feed water system piping occurred in 1990 and in 1993 at Unit 1 and Unit 2, respectively.

After the first FW-guillotine break, a classification of the secondary systems based on their susceptibility to erosion corrosion and their safety functions was performed. A detailed weak point analysis of the chosen systems was also performed in the beginning of 90's. The second pipe break revealed, however, shortcomings in the inspection technique used and in the inspection data management.

Inspections performed have revealed that wall thinning may be very local in nature. Wall thinning of similar components in parallel lines might be completely different. Inspections in the case of the second pipe break revealed that no unambiguous correlation between the parameters evaluated and different wall thinning of similar components in the parallel feed water lines was found. It seems that, under the operation conditions prevailing (neutral water chemistry, T, flow conditions), even minor changes in these conditions and/or geometry and chemical composition may have a dramatic effect on wall thinning of similar components, fittings or piping.

Improvements in the inspection technique and data management have been implemented, and special attention has been paid to the inspection of welds and flange connections. The following options have also been implemented in order to reduce erosion corrosion and the scope of annual inspections:

- Change of the water chemistry from neutral to hydratsine water chemistry
- Change of large part of the piping of the susceptible systems
- Change or modification of the components of the susceptible systems

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VANDELLOS II - ESSENTIAL SERVICE WATER SYSTEM PIPE DEGRADATION

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Abstract. On 25 August 2004 at 5:25 AM with Vandellós II operating at rated power, a circumferential break occurs on the neck of manhole EF-18-I when starting EFP01C pump. This rendered out of service one of the two trains of essential service water system (EF) of the plant. The manhole EF-18-I is located in one of the collection boxes of the system (EF-28-Z3). The break occurred as a consequence of weakening of the carbon steel plate by external corrosion. After the incident, on inspection of the affected neck it was seen that corrosion had progressed in an homogeneous way all around the neck, without a previous clear indication of leak, although oozing had been detected before.

1. EF SYSTEM FUNCTION

The Essential Service Water System (EF) supplies cooling water to the heat exchangers of the component cooling water system (EG), condensers of the essential cooled water system (GJ), and heat exchangers of the emergency diesel generators (KJ).

EF system carries out the thermal load from their cooled systems to the ultimate heat sink, the Mediterranean Sea, so the system flow is seawater. There are two redundant trains, each one able of fulfill their safety function during and after an accident.
One EF train is always running in normal operation (at power and during shutdown of the plant). The two trains of EF system are alternate on operation every two weeks. In case of accident the system must carried out the same thermal loads from safety related system than in normal operation, as well as the additional load coming from mass and energy release to the containment in case of LOCA or HELB inside containment. To fulfill these functions, both of the systems start by SIS (Safety Injection) or PSE (Loose of outside power) signals.

2. EF SYSTEM PIPING DESCRIPTION

The original design of the essential service water system (EF) was made in 1984. The piping of the system is the so-called Bonna pipe mainly. The main features for the piping are:

- Outside the buildings: buried reinforced concrete pipes with a steel plate of 2.5 mm thickness for 800 mm pipes and 1.5 mm for 300 mm pipes. The pipe is internally lined with mortar of 16 mm thickness and outside reinforced concrete layer of 59 mm with reinforcement made by 33 circumferential bars of 10 mm diameter per meter, and

- Inside the buildings:
    - Carbon steel pipes, internally coated with rubber for diameter greater than 3”; and
    - Stainless steel or monel pipes for diameter smaller than 3”.

Based on the operating experience from Vandellós I, (internal corrosion of steel plate) the original design was modified to include in the buried piping a series of manholes located in collection boxes at a distance from each other providing access into the pipe for inside inspection and repair. Also a cathodic protection system by impressed current was installed based on a platinum-coated titanium anode. The control of the cathodic protection system includes reference electrodes installed in the manholes above mentioned. Concrete slabs without seals covered the inspection boxes. The manholes in the slabs also had no seals, which allowed for rainwater in-leakage.

The engineering hazard analysis of the initial design did not include the consideration of external corrosion on the EF system, as an important question to be followed.

4. EF SYSTEM PIPING DEGRADATION

The necks in the manhole as well as the transition between aerial and buried piping include a transition neck of carbon steel without concrete protection.

These singular points had suffer along the years a process of corrosion produced by the aggressive external ambient due to condensation and also presence of sea water in some collection boxes coming from activities on outage when pipe was drained for maintenance or inspection purposes. In this last situation degradation was increased significantly due to the presence of a high concentration of chlorides. Since the original construction, an adequate painting was the only protection against external corrosion on all this points.

One of the necks in the manhole (EF-18-I) on train B breaks on August 25 of 2004. The piping has been inspected (both EF trains) along the years every outage until the 2000 year (refueling outage 11) since the initial operation of the NPP on March 1988, by the same contractor. All the inspection activities were focused on inside damage by seawater, and the state of the cathodic protection system. Internal corrosion by seawater was always seen as the main potential mechanism of pipe degradation. Throughout the successive inspections, the execution and data collection systems have been improving such that observations have become progressively standardised.

Different contractor, according with the same inspection specification performed the inspections during following refueling outages (12 and 13).

First time corrosion has been reported was in 1989 and it was most severe in the collection boxes closest to the sea. From 1993 inspection reports indicated corrosion in elements inside boxes and recommended draining every 4 months. During the refueling outage of 1999, the recommendation was take measure of the neck thickness, but it was in the third collection box away from the sea in the series of 8 boxes. There was no attempt too use systematic diagnostic techniques to identify potential problems with the neck of the pipe (measures of wall thickness).

During the refueling outage on September 2000, the inspection contractor generated immediate report dated September 19, 2000 for Train A and September 25, 2000 for Train B. The report recommended some actions to be taken before return system to operation. The recommendations were:

- Clean and paint indicated surfaces; and
- Clean the neck to white metal and measure thickness.

Although a work request was initiated (work order) it was closed without performing the requested activities. There was no feedback mechanism to tell the work requester that activities were not carry out.

The original schedule for inspection was both trains each 12 months (every refueling outage). The frequency decreased to every 18 months (linked to the extended length of the cycle). The frequency decreased again to 36 months (one train each outage) with the 2002 outage. Inspection frequency was decreased due to the success on preventing internal corrosion, based on results of internal piping inspection.

In May 2004, the presence of dampness was detected in the manhole EF-18-I. The contractor taken care of cathodic protection reported it to the NPP. The maintenance staff proceeds to inspect the manhole with engineering people support. Water samples were taken, analysed, and proceeds to dry the partially flooded collection box. In was determined that water sample corresponds to rainwater, which seems to indicate there was not a leak from the inside of the pipe. After a complete drying of collection box, a slight "oozing", not affecting the system's pressure, was also noticed. The detected symptoms did not give rise to the assumption that the corrosion might be deep and circumferentially extended, lead to a no realistic assessment of the nature and progress of the degradation and consequently to not consider additional verifications to confirm or alter the initial assumptions. In fact, the internal mortar was acting as a leakage barrier.

As a compensatory action after May 10, 2004, surveillance was increased to weekly intervals. On June 8, the oozing detected was reported at the daily meeting as seawater coming from inside the neck. Still no measure of the wall thickness was performed. A decision was taken to perform daily inspections for monitoring increased leakage. This led the engineering and plant staff to believe that there was no need for an immediate response.

The design team started working on the temporary fix to be implemented in September 2001 using a 72-hour limiting condition for operation (LCO). The design was to be completed in early August and transmitted to maintenance for scheduling in September. The circumferential break in the neck of EF-18-I occurred on August 25, 2004 after the start-up of pump EP-P01-C aligned to Train B.

After that fact, the plant was shutdown in order to repair the neck break and the corresponding neck in the other train. Also thickness measurement was done in the rest of the necks in the manholes for 800 mm and 300 mm piping. Calculations determined that remaining thickness was enough for the loads on the necks.

In October 2004, the decision was taken to implement temporary repairs in all the necks, until next refueling outage in March 2005, when all elements with extended corrosion will be replaced by new ones.

On December 2004, another "oozing" was detected in a singular point: the transition from aerial piping to buried piping in the discharge of train B pump. This singular point is not included in a collection box, but open to the atmosphere. This point and his equivalent in the other train undergo also temporary repair.

Different types of temporary repairs have been performed until the next refuelling outage in March of 2005, depending on the piping configuration. In the mean time, the design of the permanent solution is in progress. The entire T-components in all collection boxes will be replaced, as well as other singular elements of the piping.

5. EF SYSTEM DIAGNOSIS OF THE DEGRADATION

The diagnosis made inside the organisation indicates external corrosion by seawater condensation and in some collection boxes by flooding with seawater. This diagnostic has been confirmed by two independent organisations: a specialised materials company and a public institute.

The first one analysed a sample of the break neck in his laboratory, and confirmed the pipe has suffered general external corrosion, excluding degradation due to internal corrosion.

The public institute has also undertaken different activities in order to assess the state of buried pipe. Samples have been taken from soil to determine the chloride and sulphate content, as well as soil resistivity measures, in order to determine aggressiveness of soil. First results indicate a non-aggressive soil, and that buried pipe is not affected by the same problems that necks and transitions from aerial to buried pipes. During next outage a more complete analysis would be made using replaced parts of the pipe in order to confirm first diagnostic, and particularly a laboratory determination of the effect of cathodic protection on the reinforcing bars.

SERVICE INDUCED DEGRADATIONS OF CANDU FEEDER PIPING -FAC WALL THINNING AND CRACKING

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Abstract. Since the mid-1990s, degradation has been identified in feeder piping at CANDU stations, such as excessive local pipe wall thinning and cracking. The local wall thinning is thought to be attributed to the phenomenon known as Flow Accelerated Corrosion (FAC). Although the cause of cracking is still under investigation, Intergranular Stress Corrosion Cracking (IGSCC) or hydrogen assisted creep cracking are proposed as candidates for the cracking mechanisms. This paper describes the experience of Canadian CANDU industry in the degradation of feeder piping and how it has been managed. The paper also briefly discusses Canadian regulatory actions taken to ensure a safe operation of degraded feeder piping and requirements.

1. BACKGROUND

Feeder piping, shown in Fig. 1, is an integral part of the CANDU Primary Heat Transporting System (PHTS), carrying pressurized heavy water (D2O) to and from the reactor fuel channels to remove heat produced by the fission of uranium fuel. All CANDU plants in Canada share a similar design. The material for feeder piping, carbon steel SA106 Grade B, and PHTS chemistry vary little between reactors. The primary difference affecting feeder degradation is the operating temperature of the outlet feeders, which can vary from ~295 degree C to ~312 degree C. This results in two-phase flow passing through the outlet feeders in some reactors that operate at higher temperatures. In addition, the number of feeders differs depending on the number of fuel channels. At Darlington there are 960 feeders (i.e., one inlet and one outlet feeder for each of the 480 fuel channels) per reactor unit. Pickering A has 390 inlet feeders and 390 outlet feeders, while CANDU-6 plants and Pickering B have 380 inlet feeders and 380 outlet feeders. The terminal points of all feeders are a nozzle connection to a header at one end and a feeder coupling assembly, known as a Grayloc connection, at the other end. The nominal diameters of feeder pipes vary from 1.5 inches to 3.5 inches with nominal pipe size of 80.

If FAC wall thinning and/or cracking are allowed to progress unchecked, there is a potential for pressure boundary failure resulting in a leakage of reactor primary coolant. Although the rupture of a single feeder pipe is within the parameters of the safety report, there is still the potential for damage to the reactor core and a release of radioactivity to the atmosphere. As a result, it has been a Canadian regulatory requirement and industry practice to prevent leakage from the feeder piping system.

2. DEGRADATION MECHANISMS OF THE CANDU FEEDER PIPING SYSTEM

The main known degradation mechanisms having the potential to affect the operating life of the CANDU feeder piping are excessive pipe wall thinning and cracking, both of which are not unknown to the nuclear industry. Both SCC and FAC accounted for 42 percent of the large bore (greater than 2 inch diameter) piping service failures in U.S. commercial nuclear power plants from 1961 through 1995 [1].

2.1. PIPE WALL THINNING DUE TO FLOW ACCELERATED CORROSION (FAC)

Since 1995, the carbon steel outlet feeder piping in most Canadian CANDU reactors has been observed to experience greater than anticipated wall losses at the feeder bends. This was originally uncovered through wall thickness measurements which were prompted by the discovery of considerable quantities of magnetite (Fe_3O_4) deposition originating from the inside surface of the feeders. Current understanding attributes excessive wall thinning primarily to Flow Accelerated Corrosion (FAC), with possible minimal contribution from other erosion-corrosion mechanisms. Although the mechanistic understanding of FAC is not yet complete, it is commonly taken to mean corrosion caused by the flow accelerated dissolution of protective corrosion product films, i.e. dissolution of the magnetite layer.

Although the general kinetics of wall loss are understood, in terms of the dissolution rate of the surface oxide and the mass transfer rate of dissolved iron from the surface layer to the bulk coolant, FAC involves a complex interaction of several variables such as water chemistry (pHa), hydrodynamic variables (velocity, mass flow rate, steam quality, flow geometry), material composition (chromium content) and operating temperature. As a result, wall-thinning rates vary significantly from feeder to feeder based on their differing hydrodynamic parameters. Coolant velocity, however, has been identified as the primary thermal-hydraulic variable and has been correlated with the loss of wall thickness, with greater wall loss being associated with a higher velocity [2]. Although the wall loss occurs along the entire length of the outlet feeder pipe, the rate appears to be highest in the lower feeders, particularly at bends closest to the end fitting. Inlet feeder pipes do not appear to be suffering from wall loss at a measurable rate.

2.2. CRACKING

Since 1997, seven outlet feeders have required corrective maintenance due to axial cracking in the feeder bends at one CANDU station. In addition, one outlet feeder was removed due to a circumferential crack in a repaired field weld at another CANDU station. Two of the axial cracks as well as the circumferential crack were through-wall cracks resulting in leaks, while the others were discovered by NDE as part-through-wall cracks. In all three cases where the cracks resulted in leaks, the reactor was shut down before the crack reached an unstable length.

All feeder pipe cracks found to date were entirely intergranular as shown in Fig. 2, and were accompanied by secondary cracking in locations of high residual tensile stresses. The cracks, however, differed as to their point of initiation, with some initiating at the inside surface of the bend, others initiating at the outside surface, and others possibly being sub-surface breaking. The understanding of the cracking mechanism is still in the stage of speculation, however the axial cracks initiating at the inside surface of bends is believed to be caused by Intergranular Stress Corrosion Cracking (IGSCC), while both the outside surface cracks and weld cracks may have been caused by hydrogen assisted creep cracking.

CANDU feeder pipe bends have been fabricated by a variety of methods in different plants, e.g., hot formed elbows with normalization, cold bent bends without stress relief, cold bent bends with stress relief, and warm formed bends without stress relief. Bends have also been formed with or without compression boost. Hence, depending on the fabrication processes used for each particular station, the amount of residual stress in the feeder bends differs.

High residual tensile stresses in feeder bends accompanied by a mildly oxidizing water environment produces favorable conditions for IGSCC and the probability of IGSCC occurrence increases when the residual tensile stress exceeds the yield strength of the material. Cyclic strain is also likely to increase this probability. The primary factor affecting creep cracking is the time spent under both high tensile stresses and elevated temperature. Creep cracking is believed to be associated with secondary material factors such as pipe ovality, hardness, high general creep strength, and susceptibility to strain ageing. It is also believed that atomic hydrogen from FAC may facilitate creep cracking under less aggressive conditions.

3. SIGNIFICANCES INHERENT TO CANDU FEEDER DEGRADATION

(a) Extent of degradation

One of the reasons that the degradation of CANDU feeder piping is significant is that there is a large amount of piping, making it very difficult to access for inspection as well as associated economic and radiation dose issues. There are several hundreds feeder bends and field welds in each CANDU station. Accordingly, the inspection and repair of such a large number of bends or welds will be expensive and would expose personnel to significant radiation doses.

(b) Lack of a database

IGSCC has been primarily discovered in austenitic stainless steel that becomes sensitized through the welding process and is subject to BWR operating environments. As a result, although the mechanistic understanding of FAC in carbon steel pipe has been improved throughout the nuclear industry, intergranular cracking of carbon steel under CANDU water chemistry is very rare. This makes it difficult to find references to rely on in terms of environmental parameters, threshold values for the parameters in the initiation or propagation of cracks and crack grow rate (CGR).

(c) Difficulties in flaw detection

Inspection of feeder piping for IGSCC is difficult due to poor accessibility, as discussed above, as well as due to crack characteristics such as a scalloped surface, secondary cracking, multiple surface cracks and discontinuities of cracks causing ultrasound reflections. As a result, defects are only detectable above a certain size. The effectiveness of ISI procedures has therefore been a primary concern, with key parameters being demonstrated through a qualification process. Human factors are also a concern regarding the quality of the inspection.

(d) Two different degradation mechanisms at the same location

Recently, several part-through-wall cracks have been discovered at the bend extrados, which had already thinned during the bending process. More significantly, FAC wall thinning is most active in this area. The stability of these extrados cracks is therefore a concern. The possibility of a coupling effect between FAC and cracking must also be considered, because it could accelerate the degradation. In addition, one hypothesis is that hydrogen generated by FAC could assist cracking, as higher service stresses due to a reduction in wall thickness could lead to an increased susceptibility to stress corrosion cracking.

4. MANAGEMENT AND MITIGATING EFFORTS

4.1. FAC WALL THINNING

Regular thickness measurements of susceptible locations, assessment of thinning rates, and accurate prediction of wall thickness at the next planned inspection compared with pre-established minimum required thicknesses are the most important aspects in managing FAC wall thinning.

(a) Thickness measurement

Most Canadian CANDU stations have already inspected 100% of their feeder pipes for thickness measurements during the last several years and will continue to inspect sufficient sample sizes of feeders during each outage. The selection of feeders for inspection is based on a number of criteria, but in general include feeders at the highest risk while providing a statistically valid sample size. The highest FAC rates are observed in high turbulence locations within the highest velocity channels. Therefore, tight radius bends, immediately downstream from the Grayloc coupling where there is no dissolved iron, should thin at the highest rates. Other locations that experience high flow and turbulent conditions are also believed to be susceptible to a high rate of FAC.

(b) Thinning rate assessment

Currently, the available measurement samples that have been obtained to date make it very difficult to provide accurate estimations of wall thinning rates. Until additional data is available, Canadian utilities predict wall thicknesses at the end of the evaluation period by means of linear regression analysis based on QV parameters, where Q is the volumetric flow rate and V is the flow velocity.

Thinning rate estimations are different at each plant due to different flow characteristics, with a range of approximately 0.05 mm/year to 0.2 mm/year. A typical example of a thinning rate assessment is shown in Fig. 3 of this paper.

(c) Minimum required thickness criteria

The plant owners must evaluate the acceptability of a feeder if the predicted thickness, tp, is found to be less than the higher of 87.5% of the nominal wall thickness (0.875tnom) or 60% of initial wall thickness. The CNSC has approved the use of a fitness for service guideline (FFSG), developed by utilities, which provides a methodology for the evaluation of thinned piping which satisfies ASME Boiler and Pressure Vessel Code, Section III and Code Case N-597[3]. The Canadian Standard CSA/CAN N285.4-94 [4] stipulates acceptance criteria and requirements for reporting and dispositioning thickness measurements of feeder pipes. Figure 4 shows the actions to be taken by utilities based on the predicted feeder thickness. If a disposition is required, it must be submitted to the CNSC for acceptance, prior to returning the component to service. An inspection report, including assessments of the results of the inspection, i.e., determination of thinning rates and predicted time to reach the minimum required wall thickness must be submitted to the CNSC within 90 days of the Unit's return to service

Since pipe wall thinning due to FAC is the most exclusive degradation mechanism from the leak before break point of view, thickness measurements on a regular basis are thought to be the only way of management.

Several feeders located in Canadian stations have been dispositioned due to wall loss of more than 40% of the initial thickness, by demonstrating that the predicted thickness is still greater than the minimum required thickness.

(d) Mitigation

FAC is assisted by a coolant chemistry that accelerates magnetite dissolution. The most important chemistry requirement appears to be the existence of a coolant unsaturated in dissolved iron. Coolant pHa is also considered a factor which minimizes magnetite solubility and FAC rate. The range of pHa of Canadian CANDU has been maintained at 10.1 to 10.4. It is expected that continuous monitoring of the pHa in the primary coolant and an on-line thickness measurement system installed in a station will provide a more accurate correlation between pHa and thinning rate.

The addition of a corrosion inhibitor, such as titanium dioxide, has also been shown to reduce FAC in laboratory simulations and is presently being considered at one Canadian CANDU station.

In addition, chromium presence in carbon steel is known to be highly beneficial for suppressing and even preventing FAC damage [2]. This is further supported by data obtained at one CANDU plant. Therefore, for future plant design or repairs, FAC wall thinning may be prevented by using carbon steel with a specified minimum chromium content.

4.2. CRACKING

The ongoing plan to manage the potential of feeders failure due to cracking is to perform 100% inspections of susceptible sites, such as at both first and second bends having tight radiuses. In addition, all the accessible repaired field welds in outlet feeders should be inspected. Although the probability of inlet feeder failure is much lower a sample of inlet feeder bends and inlet repaired welds must also be inspected because inlet feeders have higher consequences of failure compared to outlet feeders.

The inspection scope is expanded if any cracks are discovered during the inspection. At present, any indications that are confirmed to be cracks are not dispositioned because there is currently no methodology for determining the acceptability of feeder cracks has been developed or validated. Pipe sections with confirmed cracks must therefore be removed and replaced. Considering the current limited understanding of cracking, it is not reliable to prevent cracking by changing reactor operating condition or chemistry control. Expanded inspection programs will therefore be maintained in order to reduce the risk of a feeder failure.

A leakage rate-monitoring program has been recognized as an important element of managing the risk of feeder cracking. The Canadian CANDU industry has made an effort to improve the sensitivity of the leak detection system up to 5 kg/hr and 1-day response time.

5. REGULATORY POSITIONS

Due to safety concerns stemming from service-induced CANDU feeder degradation, licensees are required to maintain a low risk of multiple feeder ruptures as well as a low probability of an inlet feeder stagnation break. Feeder piping should be therefore maintained so that consequential feeder rupture and multiple feeder ruptures are not credible. A multiple feeder break case is not considered a credible event and therefore was not included in the safety report. Stagnation due to inlet feeder breaks, which are the largest contributor to severe core damage among the feeder breaks, should also be an incredible event. The CNSC has not accepted any procedure for assessing crack-like flaws or defects detected during in-service inspection. The current regulatory position is to replace the affected portion of the feeder and to augment the inspection scope if a crack-like defect is discovered. When feeder cracking cannot be precluded, the CNSC requires a shutdown leakage limit of 20 kg/hr for the feeder piping.

The CNSC considers that the effectiveness of ISI should be improved in order to ensure that safety of the plant is not adversely affected by degradation. ISI procedures, equipment, and personnel should be demonstrated to ensure that the ISI system is capable of meeting this objective. In addition, qualification criterion for NDE in detecting and sizing the feeder cracks needs to be developed. Recently, the Canadian nuclear industry has begun investigation the application of risk-informed in-service inspection methodologies. CNSC staff believe that considerable efforts should be made to evaluate the failure probability, including all key contributing factors for each degradation mechanism prior to making use of these methods.

In addition, the Canadian CANDU industry is currently exploring the possibility of the application of the LBB concept to CANDU feeder piping where IGSCC and wall thinning are the active degradation mechanisms. CNSC staff considers that this effort is to demonstrate that defense-in-depth will not decrease due to the degradation. Accordingly, no credit has been given to the application of LBB at this time.

6. PATH FORWARD

CNSC staff considers that the level of understanding of the degradation mechanisms should be increased for effective management of the degraded CANDU feeder piping, thereby ensuring safety of CANDU stations. In-service Inspection on a regular basis is considered to be vital for feeder piping wall thinning assessment and cracking management. However, CNSC staff recognizes that the capability of NDE needs to be improved in terms of detection sensitivity, detection probability, accessibility to difficult-to-access sites, automation to reduce dose rate, and so on.

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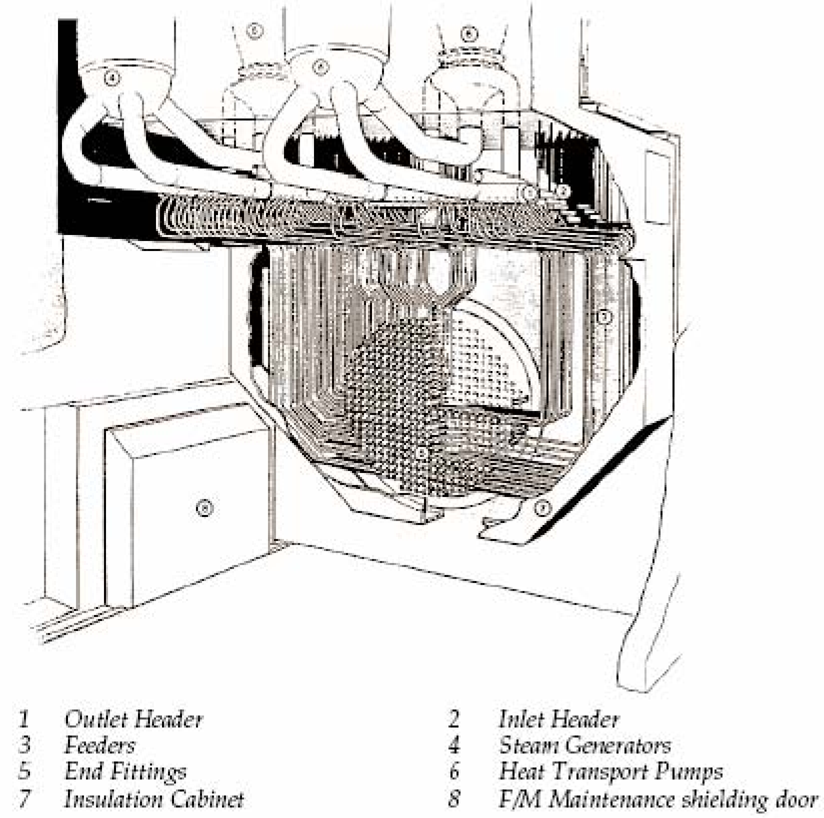

Fig. 1. Typical CANDU feeder piping layout.

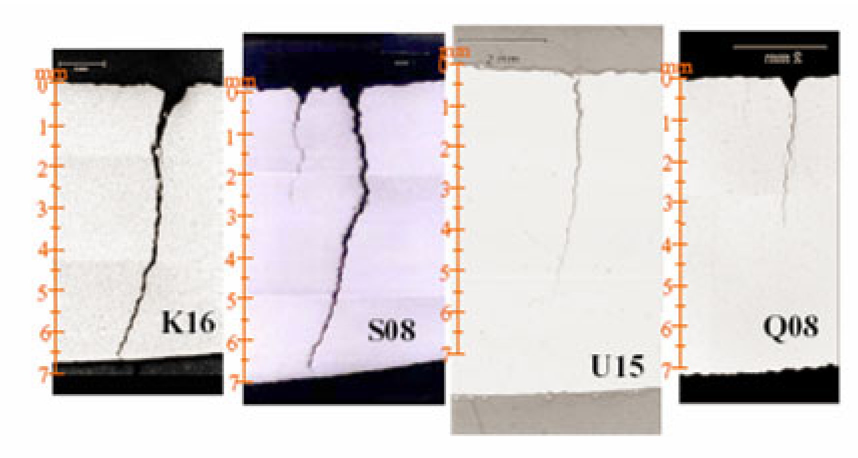

Fig. 2. Typical examples of CANDU feeder cracking.

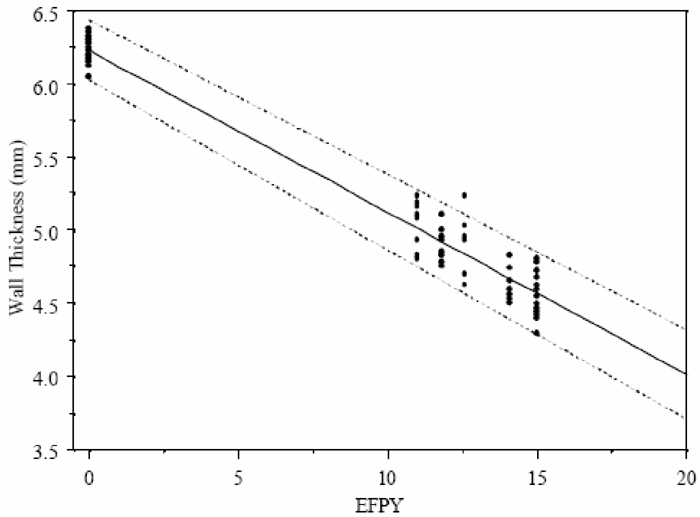

Fig. 3. Typical example of wall thinning rate assessment. The solid line is the mean wall thickness calculated using a weighted linear regression. The dashed lines enclose the 95% prediction interval.

Thickness	Action required by utilities
$t_{initial}$	
87.5% $t_{nominal}$	
80% $t_{initial}$	To be reported to the CNSC
60% $t_{initial}$	Disposition to be accepted by the CNSC
$t_{ASME\ III\text{-}3600}$*	Disposition to be accepted by the CNSC

$t_{initial}$ = Actual thickness at the time of installation of feeder pipes
t_{nom} = Nominal thickness of feeder pipes
$t_{ASME\_III\text{-}3600}$ = A pre-determined thickness that satisfies NB-3600 requirements

Fig. 4. A Schematic Illustration of Pre-Established Thickness (es) for Comparison.

TECHNICAL ISSUES

(Session 2)

STATUS REPORT ON MATERIAL DEGRADATION AT PAKS NPP

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Abstract. This paper describes the recent ageing related problems include steam generator tube degradation and deformation of CRDM (Control Rod Drive Mechanism) sleeve and liner at Paks NPP. Outer stress corrosion cracking (ODSCC) of austenitic stainless steel (08Ch18N10T) tubing in steam generators (SG) of Paks NPP caused lot of indication on the tubes. Macroscopic and microscopic examination, scanning electron microscopic examination, X-ray diffraction analysis, tensile test, chemical analysis have been carried out to investigate secondary oxide layers and tube materials on steam generator tubes pulled from four Units of Paks NPP. Anomalous outlet temperatures and decreased coolant flow rates were discovered in the core of Unit 1-2-3 at Paks NPP after steam generators decontamination. Comprehensive studies have been performed on the heat exchanger tubes originating from different steam generators (SGs) of the Paks NPP. In 2002-2003 deformation of sleeve and liner of the nozzle were observed on four CRDM (Control Rod Drive Mechanism). The defective nozzles were repaired and their liners and sleeves were replaced.

1. CURRENT STATUS OF STEAM GENERATOR HEAT EXCHANGER TUBE NON-DESTRUCTIVE EXAMINATION RESULTS

At Paks NPP the operational utility anometer the degradation monitoring of the tubes. In order to assure the integrity of this safety barrier, periodic non-destructive testing using eddy current method is carried out. The In-service-inspection program, prescribing as a minimum 10% scope determined the scope of the inspections before 1997. This inspection scope well responded to the international practice, according to which inspections also covered such percentage. According to the eddy current inspections requirements for the heat exchange tubes of the SG the registration limit is the 20% deep eddy current indication gained from circumferential flaw of the outer diameter. An indication exceeding 50% is considered to be defect limit, being at the same time criteria for plugging. In August 1997 there were found much more damaged tubes inspected by eddy current testing. Due to this matter a medium term heat exchange tube inspection program was developed. Till now all tubes have been examined.

The percentages of the plugged tubes are shown in Fig1. It is clear from the figure, that in SGs of Unit 2 and Unit3 much more indications were found then in others. In most cases the indications were initiated on outer surfaces of the tubes at the tube support plates. In few cases the defects were located on the free tube section too, mainly on the free span (FS). The schedules of inspections were so tight that at one outage only 50% of SG tubes were examined and next time another 50%. In following operational time the frequency of defect practically not changed Fig. 2. After testing of 100% of tubes a new inspection period started and new results show us that only few defects have been found mainly in case of Unit 2 (Fig. 3.). Analysing statistically the appearance of defect it can be stated that the frequency of the indication haven’t risen since 1998.

2. MACROSCOPIC AND MICROSCOPIC EVALUATION

Contact between two surfaces cause abrasion, which can interfere whit subsequent examinations. Some of the factors to consider are: the distortion associated with fracture, dislocation of the fracture surfaces, corrosion products, the number, size, and location of fragments, the roughness or smoothness of the surface, and any relation of the fracture to external damage, such as nicks. The chemical communication between bulk water and the TSP was good allowing for chemical exchange with bulk water through the porosity and capillary network during power operation this is why the corrosive substances concentrated in crevices.

The surface covered by magnetite crystals, which filled up other heterogeneous oxides. In case of FS (free span) the surface covered by iron oxide and small amount of corrosion product were observed. The thickness of the magnetite depends on the elapsed time since chemical cleaning. Stress corrosion mechanism generally exhibits little ductility and has the macroscopic appearance of brittle fracture. The susceptibility of the material is then determined using metallographic and fractographic techniques. Of course, materials can display varying degrees of susceptibility, which would fall between these two extremes. There may be multiple cracks originating from the surface, but failure usually results from the progression of single cracks on a plane normal to the main tensile stress. In austenitic stainless steels, cracks are usually transgranular and are frequently associated with specific crystallographic plane.

Energy-dispersive spectroscopy (EDS) chemical analysis was performed to identify the red, brown-coloured phase covered the surfaces. Deposits of TSP regions contained preferentially Al, Si, Mg, Ca, Mn and most regions at the support plate presence of Cu, S, Cl, Zn was also established.

The magnetite deposits on the free span are consisted low contents of Mg, Si and Ca. Analysis of the corrosion by-product from the surface is revealed significant peaks for chloride and sulphur. The most important impurities (Cl, S, and Cu) are anometers in Figs.4-5. It is clear then concentrations of the polluting components have been mitigated since the main condensers changed and chemical cleaning completed.

3. DETERMINATION OF THE CHEMICAL COMPOSITION OF TUBE MATERIAL

The corrosion behaviour of austenitic alloys is depending on the nickel and chromium content. The influence of the nickel content on the stress corrosion cracking processes in 18% chromium austenitic alloys is well known when stressed chloride ions. The 08Ch18N10T-type steel is susceptible to stress corrosion cracking. As we indicated before the worst SGs are in Unit II and Unit III. The nickel content of the removed tubes mainly lower then 10% but only few of them damaged. This is why nickel content hasn't significant effect damaging of tubes.

3.1. ANALYSIS OF THE TRACE POLLUTING MATERIALS IN SG FEEDWATER SYSTEMS

From 1997 to 2000 significant efforts were made at Paks NPP in order to decrease the secondary circuit risk sources, by replacing the condensers followed introduction of the modified water regime. The new leak-tight condensers prevent the anion polluters (chloride, sulphate) acting as stress corrosion activators from entering into the secondary circuit. By these concentrations of the polluting components in the steam generators decreased, lowering the probability of the incorporation and local increase of the concentrations of the ions causing corrosion.

The polluting materials are present in a concentration lower approximately of a factor one or two, then they used to be in case of the condensers earlier. Below the results of some of the several systematically performed operational measurements are given. On the Figures 6–7 from the anion polluters investigated in the condense water of turbines Unit I, Unit II the concentration of the two most critical from corrosion point of view components (chloride and sulphate) are given. It is well traceable on the basis of the changes when were the cooling water side leakage, as well as leaks of the raw water in the period between 1988 and 1996 followed by significant decrease of the volume of leaks in 1996 and 1997. Most noticeable is the effect of the condenser refurbishment from copper-based alloy to austenitic steel.

3.2. IN-SERVICE INSPECTION AND MONITORING METHODS

In next eight years 100% of SGs tube inspections have to perform at Paks NPP. If additional inspected tubes are found more defective or degraded, the sample size increases and, ultimately, inspection of all a tubes in one more or more steam generators and notification to HAEA have required.

3.3. EFFECT OF CHEMICAL DECONTAMINATION PROCEDURE

Anomalous outlet temperatures and decreased coolant flow rates were discovered in the core of Unit 1-2-3 at Paks NPP after steam generators decontamination. (Fig. 8). Magnetite crystals have grown from solution onto the surface of the fuel assemblies. As deposits blocked the cooling channels, the flow rate of water coolant through the reactor core decreased. Before outages some steam generators were decontaminated. Comprehensive studies have been performed on the heat exchanger tubes originating from different steam generators (SGs) of the Paks NPP [3].

3.4. STRUCTURE OF THE SURFACE

Austenitic alloys owe their high resistance to corrosion in water and aqueous solutions at ambient temperature to the formation of a thin oxide layer, the passive film. The structure, composition and the thickness of the passive film depend on the chemical composition of the alloy and the exposure conditions and can be identified for the situation in question by means of surface analysis techniques. Typical features of the passive film on austenitic stainless steels are [2]:

The chromium content is usually higher than fifty percent. The thickness of the film is normally in the range of several anometers. A normally appearance and chemical composition of the oxide layer on tube are given in Fig. 9. The surface of the inner SG tube is covered with oxide crystals. The sizes of the crystals are up to 10mm and their shape becomes clearly octahedral.

3.5. CHANGING OF THE SURFACE STRUCTURE

After decontamination the structure didn’t changed. We can see same crystals and the chemical composition is the same than was before. Drastic changing was observed when the decontaminated and one year or more time operated tube surface is analysed. The so-called octahedral crystals are disappeared and there is formed an amorphous layer. The chemical composition is significant because it has an enrichment of Cr and reduces of Fe content. The formation of the layer can be related to a passivation reaction i.e. to a solid state growth mechanism. It is assumed for simplicity that only Fe and Ni ions are released from the metal. After then new crystals have been grown on the hybrid structure and nowadays the flow resistance and flow rate volume are in normal condition. General considerations of the layer structure and mobility are under discussion.

Continue the comprehensive program to qualify the general corrosion state of heat exchanger tubes study by electrochemical (voltammetry) and surface analyzing (SEM-EDX, CEMS, XRD) methods.

3.6. INVESTIGATIONS ON DEFORMATION OF CRDM NOZZLE SLEEVES

In 2002 during CRDM dropping test performed by the Safety Systems Section in shutdown state of unit 2, personnel experienced that emergency dropping speed of CR drive 21-34 was reduced within one period (204 mm/s), however, this minimum value and the average speed for full path length still remained within the tolerance band specified in TS.

Special group of the Reactor Section, Heat Exchanger and Vessel Maintenance Section made a video recording of the gap between sleeve and outer liner of the CRDM nozzles in core cell 21-34. In the gap of core cell deformation was well visible also on the liner. The gap between the sleeve and the liner of 37 CRDM nozzles in unit 2 was measured with a cylinder gauge (Record No. 11063/Rf/2/02). During inspection personnel determined that no anomalies additional to the detected failures affecting core cell 21-34 existed and other CRDM nozzles showed no deviations in dimension of the gap between the sleeve and the liner.

The control rod nozzles base material is carbon steel covered with stainless liner from the inside in order to prevent the coolant contact with carbon material. The stainless liner is welded on the carbon nozzle cladding in the bottom and top. Special group of the Reactor, Heat Exchanger and Vessel Maintenance Section performed visual inspection of the top dome nozzle. The video film showed a significant deformation towards the interior on lower part of the nozzle with major scratch indications. The Material Control Section's visual inspection of the same scope was completed with the same result. After cleaning dye-penetrate testing were performed on upper part and the lower section of 150–200 mm of the nozzle (including welds).

The tests revealed no defect indications on inner surface of the nozzle. The liner of CRDM nozzle 21-34 was removed by using the procedure GYT-403. When the limiter contour appeared during removal of lower weld joint of the liner, personnel detected some liquid leaking from the area between the guide tube and the carbon steel nozzle. The liquid flowed away initially with significant, later with a gradually reducing pressure. The same was experienced in Unit 1 too. In 2003 same failure was appear in Unit 2 when the replaced tube damaged. In this case there was a borehole drilling into the house and a special heating device was mounted onto the penetration tube. After heating to 150°C a leak test was performed using helium noble gas. This test was successful and the indication was identified.

4. MACROSCOPIC AND MICROSCOPIC EVALUATION

After leak test an onsite microscopic examination was performed using a stereo microscope.

The defect is a small flaw located in the weld 21/h. The liner tube was pulled out and comprehensive metallographic examination was performed on the remained weld .The flaw is a horizontal direction grown crack.

A simple tensile and fatigue test was performed on corner weld specimens. The failed welds areas were mounted for a metallographic examination. In case of tensile test the cracks are propagated in minimum cross-section. But in case of fatigue test the crack arise in two directions.

Microscopic cracks were created during operation due to thermal fluctuation on the control rod liners welds. Cooling and heating period are induced tensile and compressive stresses in the structure, which is lower than yield stress. In case of weld No 21/h is much more and caused low cyclic fatigue. Due to this, cracks initiated into two directions and the circumferential crack seems to be a diode crack [4].

When the small crack reached the inside surface the coolant penetrated through the weld cracks to the small area between stainless liner and carbon steel nozzle. All area filled within power operation. The coolant in the area reduced its volume in course of cool down and additional water leak to area. The water volume increased relatively quickly during next heat up based on specific water volume increased. High pressure is created in the area in order to run out the water volume corresponding to specific volume increment. The high pressure deformed the nozzle liner towards inside and the bulge is created. This progress repeated within every coolant temperature variation. The bulge grew, deformed the nozzle sleeve and clamped the control rod drive shaft or limiting nut. Till now, four sleeve and liner have been replaced at Paks NPP.

ACKNOWLEDGEMENT

The author wishes to thanks to the staff of the CORWELD Ltd and Department of Radiochemistry, University of Veszprém for assistance and support.

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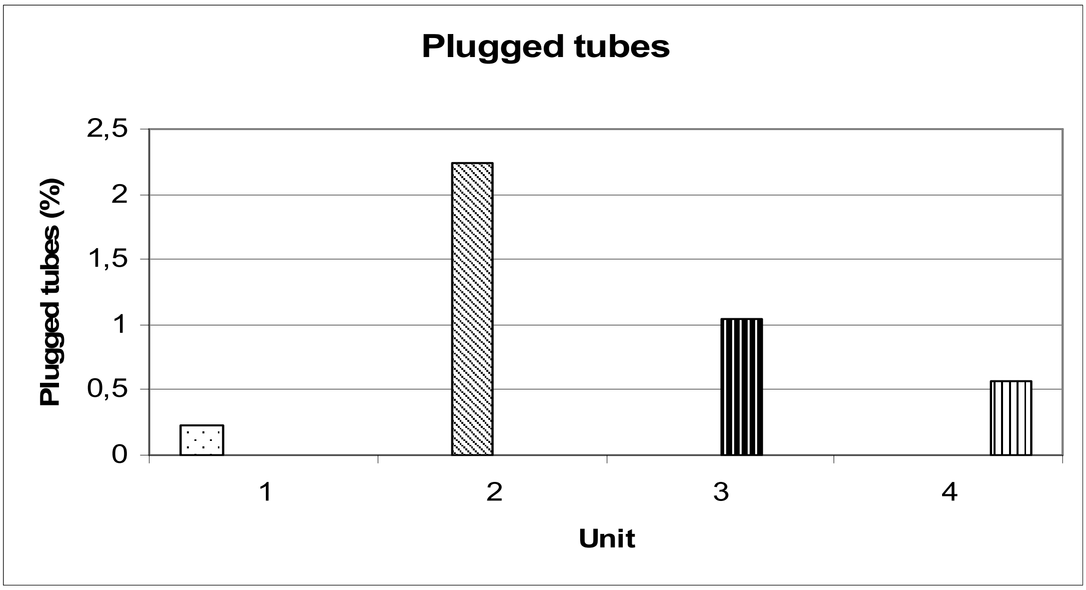

Fig. 1. Percentage of plugged tubes on SGs.

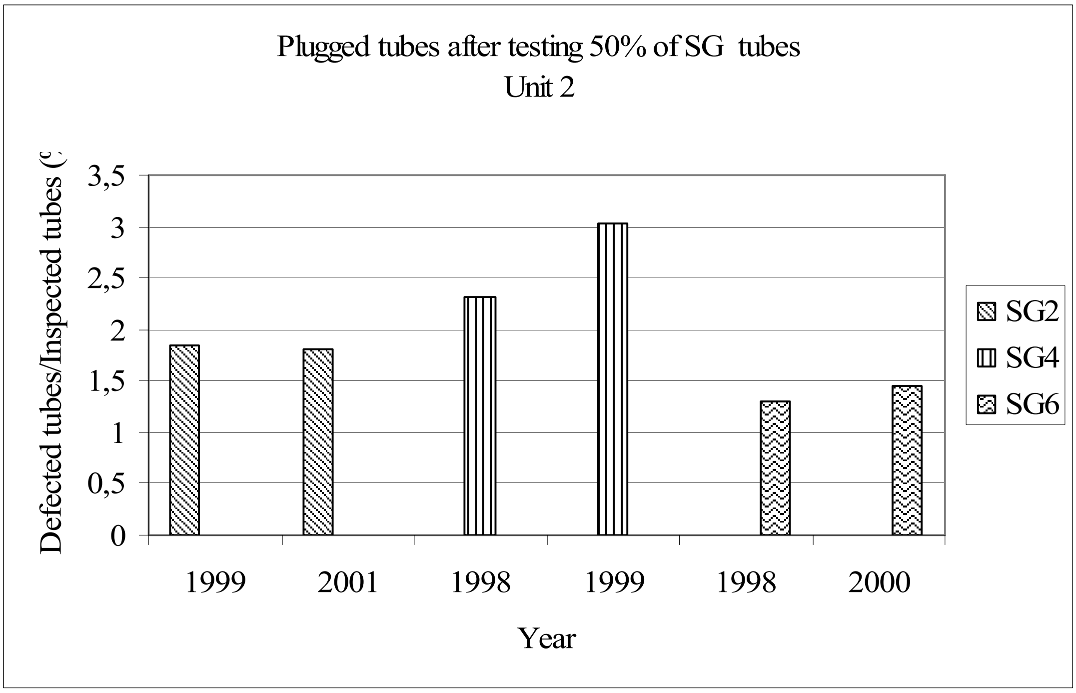

Fig. 2. The frequency of defect in Unit 2.

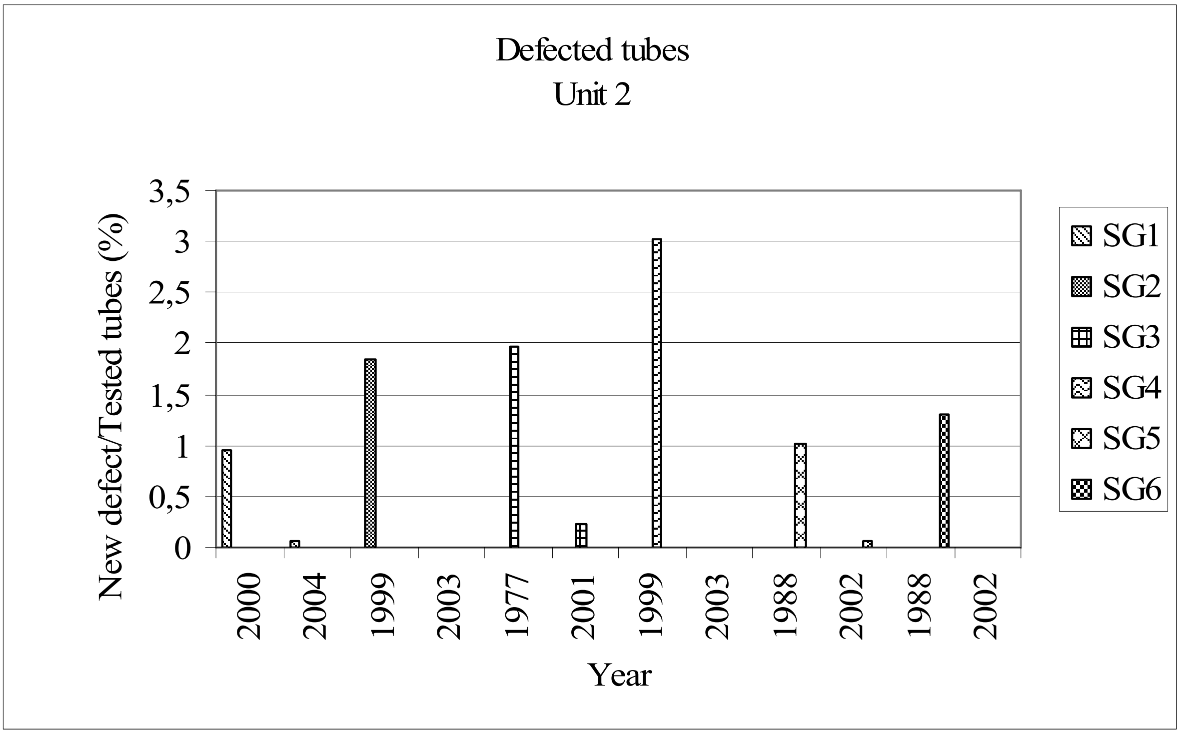

Fig. 3. New testing results.

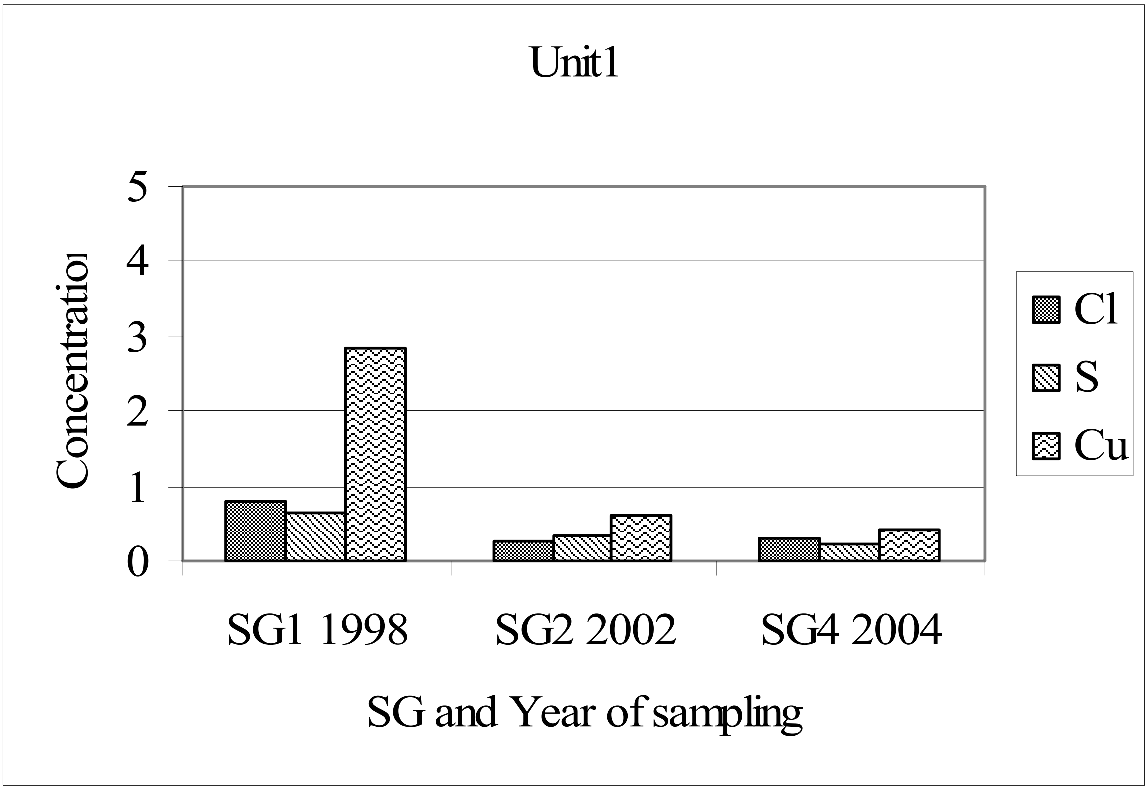

Fig. 4. Concentration of the polluters under tube support plate Unit 1.

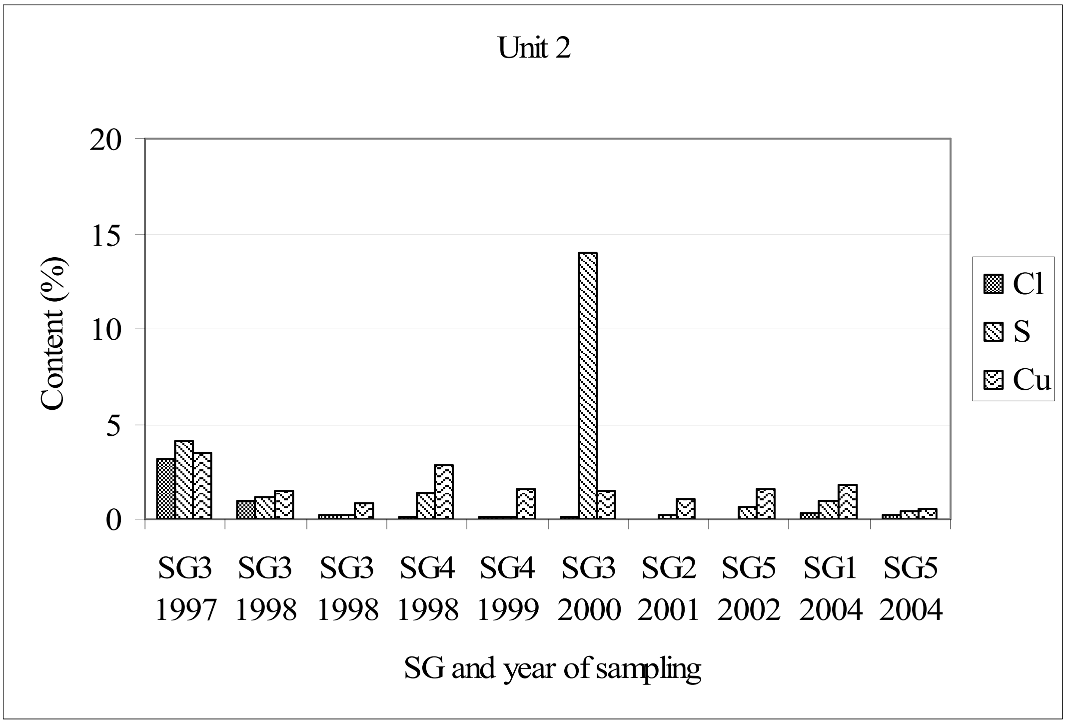

Fig. 5. Concentration of the polluters under tube support plate Unit 1.

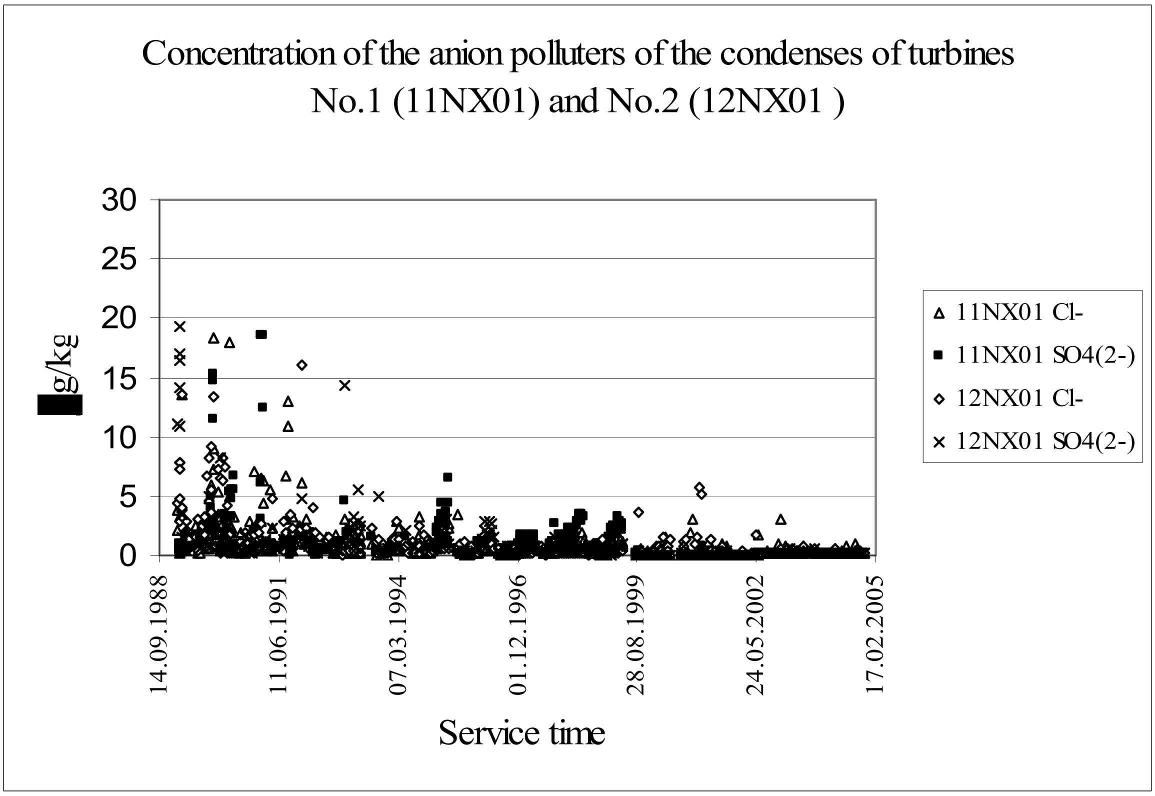

Fig. 6. Concentration of the anion polluters of the condenses Unit 1.

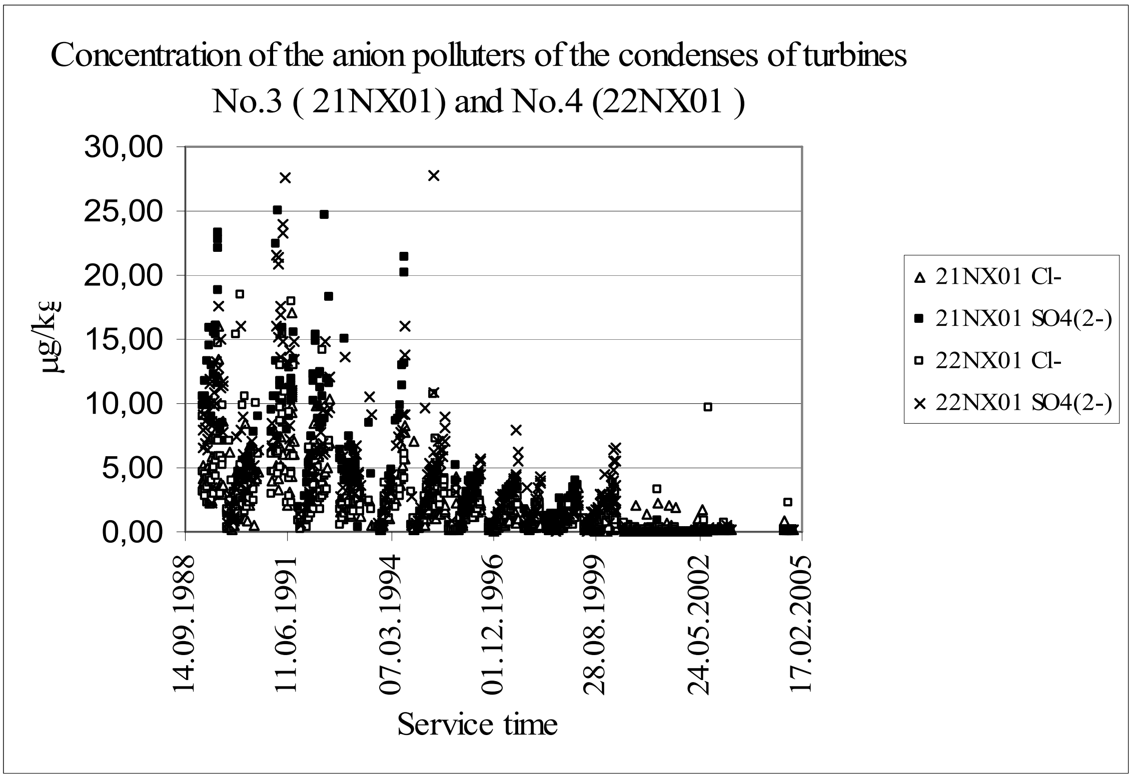

Fig. 7. Concentration of the anion polluters of the condenses Unit 1.

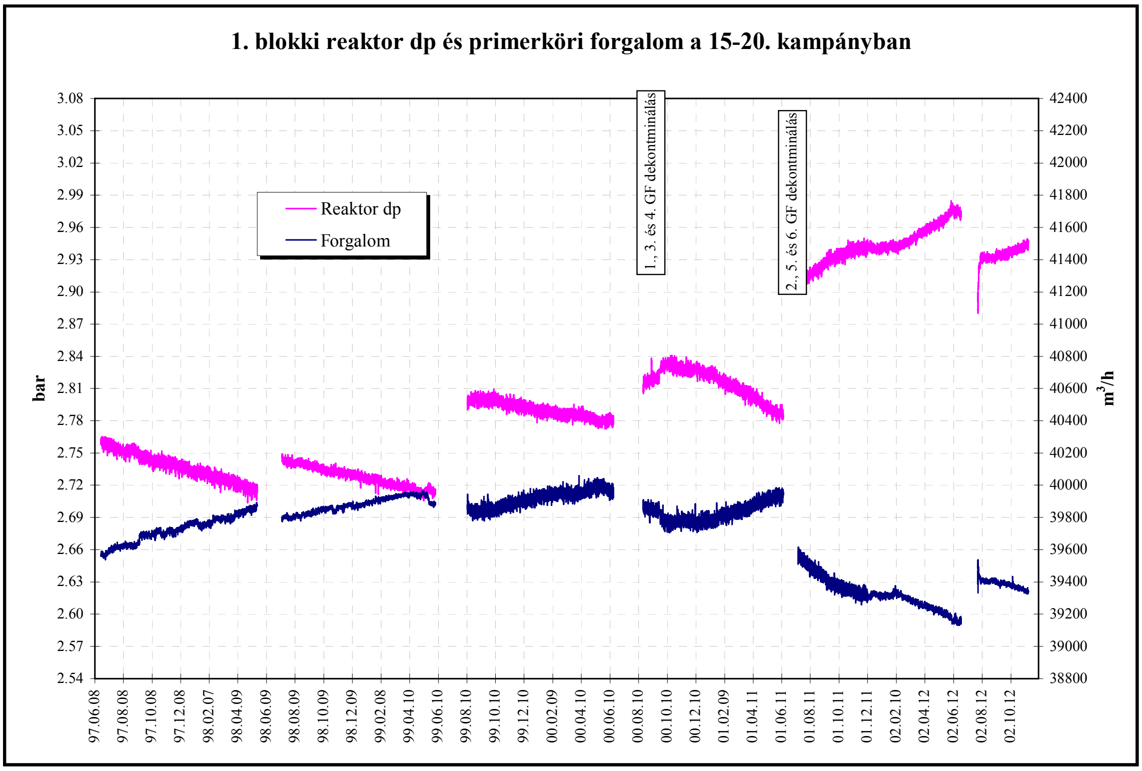

Fig. 8. Flow resistance and flow rate volume Unit 1

THE CZECH REPUBLIC EROSION-CORROSION PROGRAMME AND LESSONS LEARNED FROM MIHAMA NPP

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Abstract. Flow Accelerated Corrosion (FAC) is a specific process influencing piping components, manufactured from less noble materials. The result of two processes, creation and dissolving of the surface oxide layer, is the wall thickness thinning, which may cause its rupture. Effort to minimize damage to the piping by FAC on the nuclear power plants combines two approaches. At first it is combination of the operational parameters, which excludes or significantly reduces FAC. The second approach constitutes the system measures – the systematic monitoring, reliable damage prediction and implementation of all accessible operational experiences to the evaluation of selected piping lines. Both methods are often combined, because significant changes of the operational parameters are difficult to make. The combined approach has been chosen in case of NPP Dukovany, where the monitoring and prediction system was introduced in 1993. Similar program was introduced in NPP Temelin in 1996.

1. INTRODUCTION

Flow Accelerated Corrosion (FAC), often called Erosion Corrosion (EC), is a specific process influencing piping components, manufactured from less noble materials. The result of two processes, creation and dissolving of the surface oxide layer, is the wall thickness thinning, which may cause its rupture.

The main parameters influencing FAC are temperature, flow rate, oxygen contents and environmental pH, chemical composition of the steel, component geometry and thermodynamic state of the environment (water, mixture steam-water, live steam). The secondary circuit's parameters of the nuclear power stations are in many cases close to the critical limits for influence of the individual parameters. Temperatures between 130–260°C, pH25 below 9.5, carbon steels and flow rates above 1 m/s form favorable conditions for FAC development.

Effort to minimize damage to the piping by FAC on the nuclear power plants combines two approaches. At first it is combination of the operational parameters, which excludes or significantly reduces FAC. This may be for example replacement of the materials for heat exchange surfaces or condensers and following change of the coolant chemical regime or replacement of the whole piping line with the material more resistant to FAC.

The second approach constitutes the system measures – the systematic monitoring, reliable damage prediction and implementation of all accessible operational experiences to the evaluation of selected piping lines. The target of the second approach is to eliminate in advance possibility of the defect, which may cause damage to the piping system. Both methods are often combined, because significant changes of the operational parameters are difficult to make.

2. CZECH REPUBLIC EROSION-CORROSION PROGRAMME

The combined approach has been chosen in case of NPP Dukovany, where the monitoring and prediction system was introduced in 1993. At first to the feed water and live steam piping and gradually extended to other secondary circuit systems, where potential damage would cause significant safety or operational and economical problems – condensate, turbine extraction lines, moisture separator, low-pressure regeneration and several auxiliary lines.

Similar program was introduced in NPP Temelin in 1996. Pre-operational thickness of certain components has been measured in Temelin. Currently, first operational inspections and wall thickness measurements are under way.

The evaluation process has three main parts: damage prediction, built on the computer code CHECWORKS, operational experiences and supplementary evaluation of non-standard situation, wall thickness measurements of the selected components and procedures for qualification of components with defects for future operation.

The prediction computer code CHECWORKS, used for modeling, is based on evaluation of effects of the individual operational parameters on the given component, with respect to influence of the neighboring components. The resulting parameter is speed of the wall thinning and time to reach limit wall thickness. These values are referenced to the individual components and create general indication of the speed of the system aging. The components for the wall thickness measurement are selected according to the accepted criteria. The objective of the measurement on the components with significant prediction of the wall thinning is to evaluate size and location of the defect. Other components are measured in order to verify the prediction process.

3. LESSONS LEARNED

Although current results show good quality of the prediction model for both NPP Dukovany and Temelin, with respect to events in NPP Mihama it has been decided to review the last inspections and plan another inspections on the condensate and feed water piping.

Table1. Comparison of operation conditions in condensate system (between 4th (5th) point feedwater heater and deaerator, upstream of the main feed pumps)

Parameter	Critical Value	Mihama	Dukovany	Temelin
Material, Cr content	< 1%	carbon steel SB42, <= 0.3	carbon steel 12 022, <= 0.3	carbon steel 12 022, <= 0.3
PH	< 9.5	8.8 – 9.3	8.6 – 9.9	10
Temperature	135 – 165°C	140°C	145°C	163°C
Liquid/Steam	< 0.95	all water	all water	all water

Attention should be focused on the pipelines, where the medium temperature is between 135–165°C and locations immediately behind measurement orifices and other fittings, e.g. valves. In case of NPP Dukovany it was recommended to perform inspection revisions and trace history of replacements/repairs in localities behind measurement orifices on all pipelines susceptible to FAC or behind orifices, where erosion or cavitation could occur during normal or nonstandard conditions.

If we find during measurement the thickness to be close to criterion value that is based on simple analytical evaluations of thickness, a more detailed analysis will be performed. The analysis will covered all regions where significant thickness reducing is detected.

For detailed analysis the following sophisticated methods will be used to evaluate stress limits according The Czech Normative Codes. First stress analysis using piping programs usually used for evaluation of influence of piping forces and piping moments on thickness reduced areas will be performed provided that computing model of piping system is disposal. Secondly detailed FE analysis of thickness reduced areas using measured thickness data will be performed.

Attention will be concentrated on uncertainties where significant thickness reducing is detected or expected such as locations where piping hangers are fixed on pipe, piping supports, wall penetrations, location of pipe whip restraints, T- joints with reinforcing pads and generally inaccessible parts of piping.

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COMSY SOFTWARE ASSISTS LIFETIME MANAGEMENT ACTIVITIES

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Abstract. The COMSY program has been developed to provide an effective software tool for the plant life management of systems and mechanical components. The program utilizes more than 25 years of experience resulting from research activities and operational experiences. It is designed to support a plant-wide strategy providing lifetime predictions for mechanical elements, which are validated by a small number of examinations at priority locations. The objective is to establish economically optimized inspection and maintenance programs, while maintaining high levels of plant safety and availability. This is accomplished by focusing inspection activities on the actually degradation relevant locations based on reliable service life predictions. This capability is particularly useful for the service life extension of systems and components. Within the last years the program has been successfully applied to various nuclear power plants and the benefit of this software-based strategy could be confirmed by field experience.

1 INTRODUCTION

The long-term and safe operation of thermal power plants calls for a methodology, which takes precautionary measures against problems typically experienced in power plants after operating periods of time. These problems cover ageing effects on the fields of mechanical components, electrical components and civil structures.

For mechanical components efficient lifetime management strategies are based on reliable lifetime predictions to indicate when a system or component has reached the end of its lifetime. The most frequently degradation mechanism which occur in thermal power plants is flow-accelerated corrosion (FAC). An example for the extension of FAC shows the current large pipe rupture at the Mihama-3 nuclear power station in Japan, which caused four fatalities and the pipe break at the American Surry-2 NPP in 1986. The intentions of a systematic lifetime management for thermal power plants require a detailed understanding of the degradation mechanisms and functional interactions of the relevant parameters.

Research activities and operational experiences have been conducted in this field (e.g. FAC) for some 25 years in the Framatome ANP laboratories in Erlangen, Germany. Based on these activities a software tool for ageing and plant life management (PLIM) and plant life extension (PLEX) for passive mechanical components was developed. It is designed to support a plant-wide strategy providing lifetime assessment for piping and vessels in respect to degradation mechanisms typically experienced. The lifetime assessment process requires detailed information on design and operating conditions as well as the components' as-is state. This information is kept in a user-friendly database application, which is continuously updated via the integrated inspection management functionality. Hence, the software tool serves to address the following PLIM related topics:

- Prevention of failures by identifying degradation sensitive systems and components;
- Streamlining of inspections by focusing inspections activities on priority locations utilizing a combination of condition-oriented and risk-informed methods;
- Know-how conservation by providing compiled information on the service history of relevant components; and
- Up-to-date documentation including memos and technical reports, which are directly associated with the system and component level.

The purpose of a systematic ageing and plant life management program is to allow the lifetime of plant components to be planned, and to indicate when a component has reached the end of its effective lifetime before it fails. Another important function of such a strategy is to increase the availability of power plants and to enable implementation of a targeted maintenance strategy in terms of its economic and technical effect.

2. THE COMSY CONCEPT

The COMSY software system (Condition-Oriented ageing and plant life Monitoring System) was developed by Framatome ANP GmbH as a tool for ageing and plant life management of mechanical components. This knowledge-based program system allows the overall lifetime of the pressure boundary of mechanical components to be tracked. The concept is based on a comprehensive combination of deterministic and probabilistic approaches, including the option for risk-informed prioritization of activities.

The application of the software system for the purpose of tracking the overall service life promises the following economic advantages to the plant operator:

- Concentration of inspection activities on safety or availability relevant system areas where a degradation potential exists,
- Comprehensive documentation continuously visualizing the current as-is condition of the plant,
- Assessment of the effects of refitting work prior to its performance through the use of simulation calculations.

The COMSY software system acquires, manages and evaluates component and operating parameters relevant to service life. Plant data pertaining to individual vessel elements, piping elements and systems are stored in a "virtual power plant data model". Based on these plant data, the program performs a condition-oriented lifetime analysis for various degradation mechanisms which are commonly experienced in power plants. Those are e.g. erosion corrosion (FAC), cavitation erosion, droplet impingement erosion, material fatigue (transient, stratification and cycling), strain-induced cracking, stress corrosion cracking (IGSCC, TGSCC, PWSCC), pitting, crevice corrosion and microbiological corrosion (MIC).

This process is supported via an intelligent user interface, effective analysis functions (stress analysis, thermal-hydraulic and flow analysis functions, water chemistry cycle analysis), comprehensive material libraries (e.g. material data catalog, database of material acceptance values), a module for management and evaluation of examination results and a risk-informed function, which allows to prioritize and to optimize the inspection date by means of risks and costs criteria.

The concept is based on comprehensive experience gained in the use of the predecessor software tools WATHEC & DASY in conducting analyses for weak points due to flow-accelerated corrosion over a period of more than ten years. In the following the ageing management strategy implemented in COMSY is described in more detail including the relevant sub-items:

- Plant wide diagnosis,
- Closed loop process,
- Lifetime prediction for elements,
- Optimization of inspection activities and
- Software tools supporting the analysis.

3. PLANT WIDE DIAGNOSIS – SCREENING

In order to efficiently screen a plant unit for systems potentially affected by specific degradation mechanisms, a so-called rough analysis is performed (Fig. 1). During this screening process the heat balance diagram of the water/steam cycle in the power plant is modeled using graphical tools. In a second step the system parameters (pressure, temperature, mass flow and enthalpy) are specified for each system area. The resulting model establishes the basic data structure of the virtual power plant, and allows for an analysis of the water chemistry cycle to be conducted based on e.g. the known injection rate of an alkalizing agent for the associated system. The water chemistry cycle calculation subsequently provides the distribution of local water chemical conditions for each system and sub-system of the BOP.

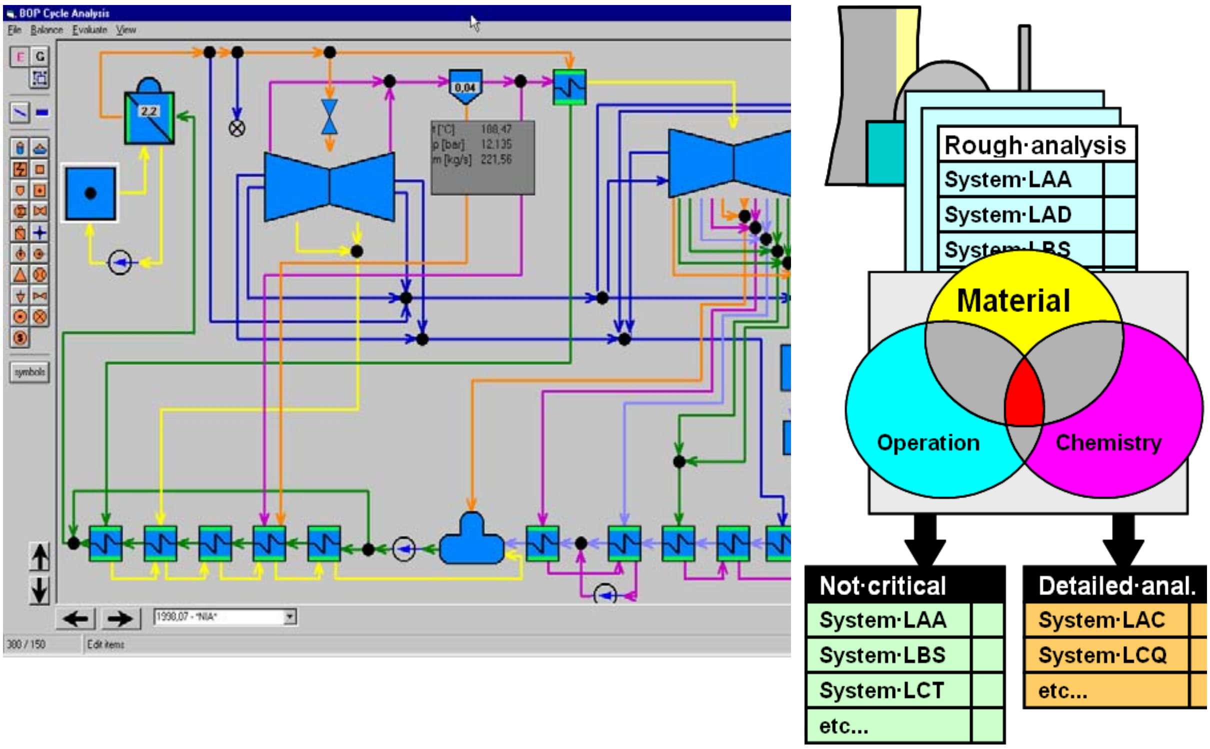

Fig. 1. Rough analysis using COMSY.

In the next step representative materials brands are specified for each system area. As the sensitivity in respect to degradation mechanisms is primarily controlled by the combination of operating parameters, chemical conditions and materials applied, the system areas can now be evaluated by the program to determine the local degradation potential. The resulting table indicates which degradation mechanism may be relevant for the individual system or sub-system. It also indicates, which degradation type can be disregarded due to the systems design and operating parameters.

The screening process guarantees an economically and technically useful application strategy. Based on the priorities determined, the program provides the option to analyze selected systems areas in detail, in order to further localize and quantify an existing degradation potential.

4. CLOSED LOOP PROCESS

Service life assessment is the key function of a software system for ageing and plant life management. On this basis inspection management and plant availability can be optimized and the service life of components can be extended. An efficient service life management program builds on these degradation predictions, which are used as a basis for inspection optimization. Inspection results are used to validate and to individually calibrate lifetime predictions. By establishing a closed loop process, the program is capable to adapt a condition-oriented approach on a step-by-step basis. This makes lifetime predictions more and more accurate with every year or program application.

Based on the predicted service life, components can then be prioritized for examination programs. Probabilistic tools are provided to set-up software guided inspection plans, as shown in Fig. 2. The results of component examinations are fed back into the program system, and are used for further optimization of service life predictions over the life cycle of the component. Overall, this systematic, closed-loop process enables up-to-date inspection strategies, utilizing quantifiable data characterizing the technical as-is status of the plant.

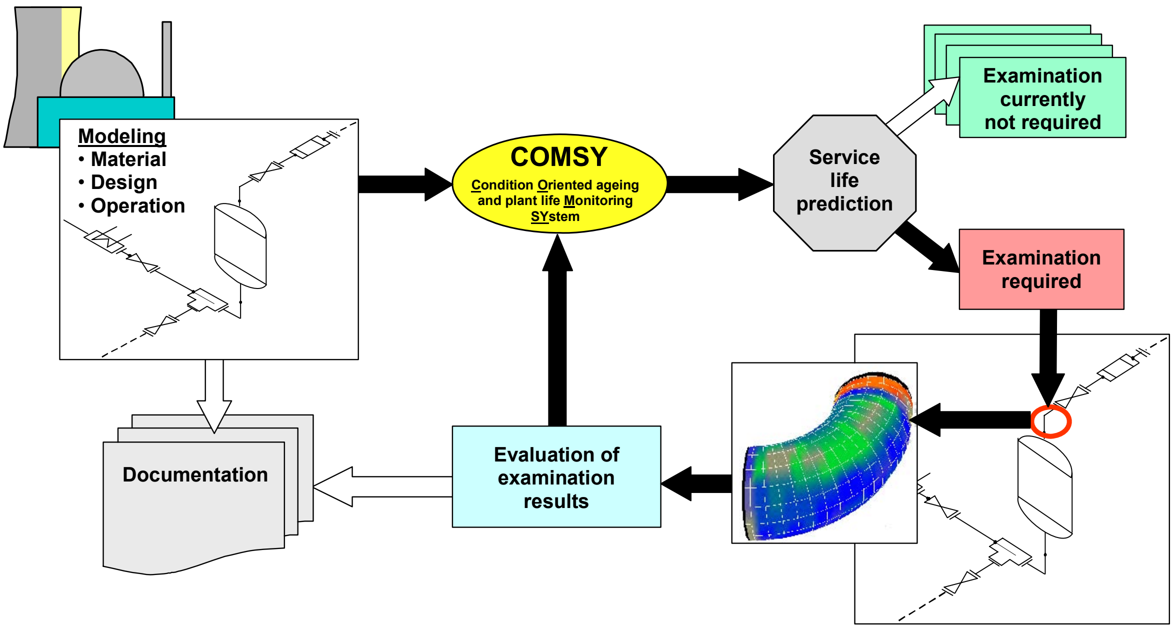

Fig. 2. The closed loop process for PLIM.

Based on a known type of degradation and a validated rate of degradation progression, suitable remedies and preventive measures can be implemented in order to extend the service life of components. Experience over many years has shown that an inspection management program based on reliable service life predictions enables costs to be minimized and plant availability to be increased.

5. LIFETIME PREDICTION FOR ELEMENTS

Ageing and wear mechanisms – in particular corrosion and fatigue – are the cause for a limited service life of mechanical components. In order to assess the service life of a component, the following questions have to be answered:

- Which degradation mechanisms are relevant to the material under its intended conditions of use?

- What rate of component degradation progression is to be expected under those conditions?
- Which limiting condition caused by the progression of the degradation restricts the service life of the component?

The properties of the material, the ambient water chemistry and thermal-hydraulic conditions and the mechanical load on the component must be evaluated in order to assess the type of corrosion to be expected as well as the rate of degradation progression.
The limit on the service life of the component is reached, for example, when

- the maximum allowable stress in the pressure-retaining boundary is reached,
- the maximum allowable utilization factor is reached with respect to material fatigue,
- the toughness of the material drops below the required values.

5.1. LIFETIME PREDICTION MODELS

The preparation of degradation models presumes a detailed understanding of the type of degradation concerned as well as the functional interactions of the relevant parameters which influence the rate of degradation progression. Studies and degradation analyses have been conducted in this area for some 25 years in the Framatome ANP laboratories in Erlangen, Germany. The experience gained from these activities has been compiled in analytical and semi-empirical corrosion models for each degradation mechanism. To date, degradation models have been elaborated for the following types of corrosion: erosion corrosion (FAC), cavitation erosion, droplet impingement erosion, material fatigue (transient, stratification and cycling), strain-induced cracking, stress induced corrosion cracking (IGSCC, TGSCC, PWSCC), pitting, crevice corrosion and microbiological corrosion (MIC).

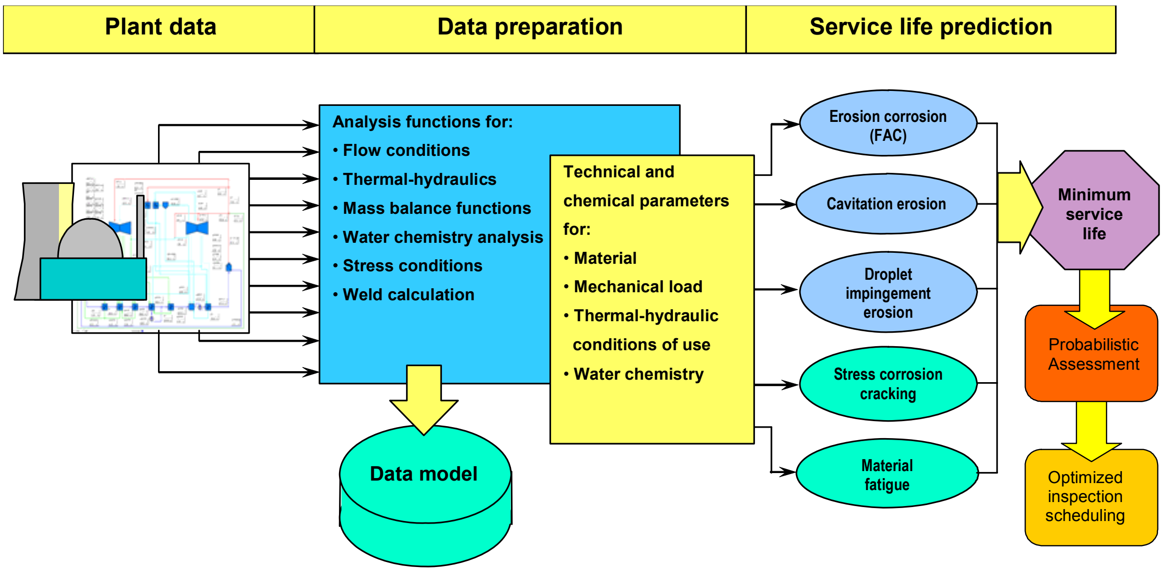

Fig. 3. Data conditioning for lifetime prediction.

The corrosion models used to make deterministic service life predictions require the use of a number of physical and chemical parameters, which cannot always be taken directly from the plant documentation. In order to enable economical application of service life predictions, COMSY includes appropriate analysis functions and engineering for pre-processing corrosion-relevant parameters based on the available documentation, see Fig. 3.

The rate of degradation progression is determined for the relevant degradation mechanism in each case using these degradation models, whereby a corresponding safety factor is used to allow for expected uncertainties. The calculated rate of degradation progression and the strength boundary conditions calculated for the component are used by COMSY to determine the minimum service life.

The probabilistic assessment provides information on the likelihood of failure for individual components based on analysis of existing component data. In combination with detailed analysis of existing examination data, this method allows to focus inspection and maintenance efforts on those components with the highest failure probability. This process utilizes e.g. a crack growth model and considers the failure consequences for risk-informed approaches.

5.2. CALIBRATION OF LIFETIME PREDICTIONS

The COMSY software system acquires and assesses measurement results from non-destructive component examinations and visual inspections, see Fig. 4. The examination and inspection results are linked to the examined component for documentation of the as-is condition at that specific time in the operating history of the plant, and are integrated into the virtual power plant data model.

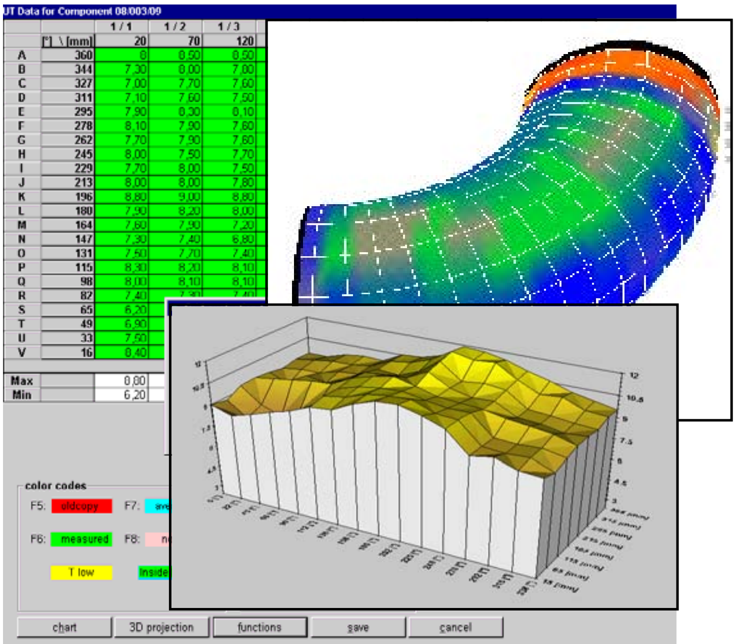

Fig. 4. Evaluation of component examinations.

The evaluation of component examinations is sup-ported by e.g. interactive analysis functions, which greatly simplify the geometry-dependent evaluation of measurement results, among other things. A calibration function supports the comparison of the as-measured condition with the predicted progression of the degradation, while making allowance for measurement tolerances. The results of this comparison are used in order to improve the accuracy of future service life predictions.

This process ensures that experience gained from evaluation of examination data will be fed back into the performance of analytical service life predictions. Examination data resulting from in-service inspections are thus consistently used in the preparation of a reliable database, which is kept continually up to date. This process increases the informational value of the examination results, and makes it possible to provide a legible description of the as-is condition of components and systems. The quality of the condition documentation will continue to increase with every year the process is in use.

6. SOFTWARE TOOLS

6.1. POWER PLANT DATA MODEL

A systematic program for ageing and plant life management requires an overall view of the power plant, because individual components cannot be studied in isolation when attempting to predict the remaining service life.

The information used in making such a determination must on the one hand be stored in a component-specific manner (e.g. geometry, material, examination results, material acceptance data, etc.), but on the other hand must be valid across components (design criteria, thermal-hydraulic and water chemistry conditions of use) and across systems (flow rates, availability times). This is achieved by using a "virtual power plant data model" which enables the assignment of parameters to various plant element relations. A significant characteristic of the data model is the specification of environmental operation conditions as a function of time in service.

6.2. COMPONENT MODEL

The program processes components such as piping or vessel elements individually. If a piping element is to be generated in COMSY for a specific piping run, for example, the user selects the corresponding system area and picks the component type from a list of predefined component symbols. The operator selects the desired diameter, wall thickness and material from the integrated standard libraries. Subsequently, the program generates a component data sheet. Using integrated analysis tools, it calculates the related operating conditions which apply to the component in question based on existing thermal-hydraulic and water chemistry data (flow condition, degree of turbulence, pH and oxygen concentration of the fluid). In a next step, the program computes the strength conditions of the component for the given design criteria – in accordance with GOST, ASME, DIN, or JIS – and displays this information on the component data sheet.

A plausibility routine checks the design of the component for compliance with applicable standards, and indicates possible input errors. Subsequently, the automatic generation of service life predictions for each component is provided. The completed component data sheet is used as the basis for further evaluations. If required, additional information can be supplemented over the life cycle of the component.

In addition, details concerning examination results such as examination records and pictures taken during visual inspections can be accessed for the component in question. The component data sheet also makes it possible to access static documents – drawings, parts lists, reports and acceptance certificates which exist as files – also in power-plant-specific documentation systems. Moreover, the software system enables the use and/or integration of information extracted from existing databases.

6.3. RISK BASED ASSESSMENT

The probabilistic assessment tool provides the option to utilize existing data from detailed analysis and previous inspection results to evaluate the degradation potential within a given time period. The damage probability is associated with the expected damage consequence based on PSA data. Additional criteria are the cost for resulting repair and/or replacement activities. The degradation potential for individual components is based on a deterministic evaluation of operating and design conditions in combination with examination data, which are administered in the database.

The damage consequence is evaluated considering the key results from the PSA, e.g. the conditional core damage probability (CCDP) or the conditional large early release probability (CLERP). Finally, the risk is determined by consideration of damage potential and damage consequence. Only, if the risk does not represent a danger for environment and personnel, equipment financial effects are also assessed. In order to find the measure with the optimum risk/cost ratio the following options can be taken into account:

- Repair or replacement associated with unplanned shutdown necessary
- Redundant system available
- Damage figure is difficult of access during normal operation
- Arising expenses for repair or replacement
- Increasing costs if repair or replacement is postponed.

The result is illustrated by a four to three risk matrix (Fig. 5). It shows a classification for judging the necessity of measures to be taken.

This method allows focusing inspection and maintenance efforts on the components with highest failure probability. The results of component examinations are fed back into the program system and are used to update probabilistic assessment criteria for further optimization of service intervals.

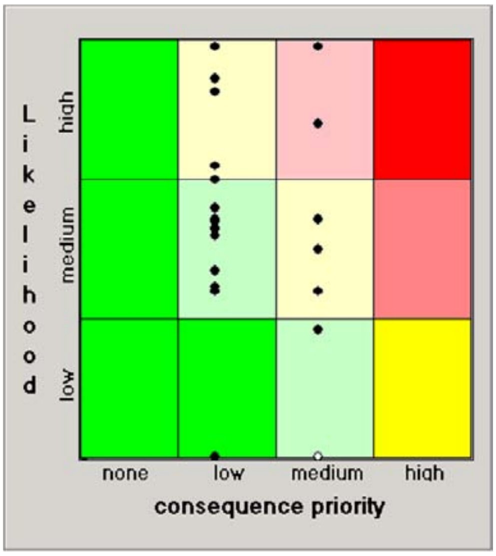

Fig. 5. Risk matrix

7. CONCLUSIONS

For many utilities a systematic and efficient ageing and plant life management system is becoming more and more important to ensure an economical power plant operation in spite of continuous facility ageing. In this regard the COMSY software system makes a knowledge-based program system available, which integrates advanced analysis tools and comprehensive material libraries with a “virtual power plant data model”. It enables the condition-oriented of evaluation service life of vessels, piping systems and entire plant units with respect to relevant degradation mechanisms. The results of component examinations are fed back into the program system, and are used for further status evaluation over the life cycle of power plant systems. Overall, this systematic, closed-loop process enables up-to-date maintenance utilizing quantifiable data characterizing the technical as-is status of the plant. On the basis of reliable predictions and of relevant degradation, maintenance management and plant availability can be optimized and the service life of costly systems and components extended.

The implementation of COMSY in various power plants within and outside Germany (e.g. Japan, Spain, Finland, Bulgaria and Hungary) has confirmed that systematic plant life management makes good economic sense, and that the process can be greatly streamlined through software support.

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GUIDELINE FOR MANAGEMENT OF PIPING WALL THINNING IN PWR SECONDARY SYSTEMS IN JAPAN

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Abstract. In the early 1980s, when some PWR plants in Japan experienced steam leakage due to piping erosion/corrosion, PWR-operating utilities in Japan responded to the problem of secondary system pipe wall thinning by launching a systematic investigation of pipe wall thickness at their PWR plants paying due attention to fluid flow conditions and pipe configurations. The Guideline sets rules for the wall thinning management for carbon steel pipes in PWR secondary systems by stipulating the inspection method, inspection targets, inspection frequency, acceptance criteria, corrective actions (pipe replacement) and so on. With regard to inspection frequency, the Guideline stipulates that the remaining service life (period of time remaining until a pipe is thinned to the calculated minimum thickness requirement) should be determined and that inspection activities should be continued until the remaining service life decreases to two years. Pipes found to have a remaining service life of two years or less require corrective action. PWR utilities in Japan checked the piping thickness of the components that are required to be inspected in a well-planned manner based on the Guideline. Even after the recent secondary system pipe rupture accident at the Mihama Nuclear Power Station Unit 3, the Guideline is regarded by the Japanese regulatory authorities to be "suitable as a management program in general." However, it is pointed out that the utilities should establish a new private-sector guideline for pipe thickness management based on accumulated measurement data and latest knowledge from overseas in order to ensure more strict control over pipe wall thinning. Now, the Japan Society of Mechanical Engineers (JSME) is currently developing of a new private-sector guideline for pipe thickness management.

1. INTRODUCTION

In the early 1980s, when some PWR plants in Japan experienced steam leakage due to piping erosion/corrosion, PWR-operating utilities in Japan responded to the problem of secondary system pipe wall thinning by launching a systematic investigation of pipe wall thickness at their plants paying due attention to their own fluid flow conditions and pipe configurations.

Later on, alerted by the secondary system pipe rupture accident at the Surry Nuclear Power Station (USA) in 1986, these utilities in Japan, analyzed the wall thinning measurement data accumulated up to that point in time, to develop a guideline applicable to the secondary system pipe inspection planning. This was finally established in 1990 as the "Guideline for Management of Piping Wall Thinning in PWR Secondary Systems". Since then, the PWR-operating utilities in Japan have followed the Guideline when performing systematic monitoring of pipe wall thickness, covering all components that were identified as being prone to thinning.

This paper describes the Guideline adopted by the utilities in Japan and our approach in addressing the issues that came into our focus following the pipe rupture accident at the Mihama Nuclear Power Station Unit 3.

2. HISTORY OF PIPE WALL THINNING MANAGEMENT BY PWR-OPERATING UTILITIES IN JAPAN

The Kansai Electric Power Co., Inc. (hereinafter "Kansai Electric") has been conducting wall thickness monitoring activities for secondary system piping since the 1970s. Later on, when

Takahama Unit 2 (commissioned in 1975) and Mihama Unit 3 (commissioned in1976) experienced steam leakage due to pipe wall thinning, the company extended the wall thickness monitoring activities to the entire secondary system piping system including the feedwater and condensate piping, paying due attention to fluid flow conditions and pipe configurations. In 1986, a pipe rupture accident at the Surry Nuclear Power Station (USA) prompted the company to work urgently toward the development of a guideline for management of pipe wall thinning.

In 1987, aiming to establish a guideline for management of pipe wall thinning in PWR secondary systems, Kansai Electric, with Mitsubishi Heavy Industries, started the analysis of wall thinning phenomenon using the available mass of measurement data from the operating plants. The analysis covered a wide variety of data from about 30,000 components inspected. It included data not only of those systems where a two-phase flow existed, but also data from those systems where water existed in either only the liquid or gaseous phase. In 1990, the statistical analysis of such data resulted in the establishment of the "Guideline for Management of Pipe Wall Thinning in PWR Secondary System." Since then, the five PWR-operating utilities in Japan have followed the Guideline as they managed pipe wall thinning at their plants (see Table 1).

Table 1. History of pipe wall thinning management

1970's	Kansai Electric started the measurement of wall thickness.
1983	Steam leakage at Takahama Unit 2
1984	Steam leakage at Takahama Unit 2 and Mihama Unit 3
1985	Kansai Electric started systematic investigation on wall thinning.
1986	Pipe rupture accident at Surry Unit 2
1987	Kansai Electric and Mitsubishi Heavy Industries started the development of the Guideline for Management of Pipe Wall Thinning.
	The Guideline was issued. Inspection according to the Guideline (5 PWR utilities)
2004	Pipe rupture accident at Mihama Unit 3

3. OUTLINE OF THE GUIDELINE FOR MANAGEMENT OF PIPE WALL THINNING

The Guideline established in 1990 sets rules about the wall thinning management for carbon steel piping in PWR secondary systems by stipulating the inspection targets, inspection method, inspection frequency, acceptance criteria, corrective actions (pipe replacement) and so on. The following subsections describe the approaches and requirements set forth in each section of the Guideline.

3.1. GUIDELINE REQUIREMENTS ON THE SELECTION OF INSPECTED COMPONENTS

3.1.1. Selection of the systems to be inspected

The criteria for selecting the systems to be inspected were established through the assessment of the primary factors influencing on pipe wall thinning, which we conducted using the available mass of inspection and measurement data. The figure below shows the workflow concerning the selection of the systems to be inspected (see Fig. 1):

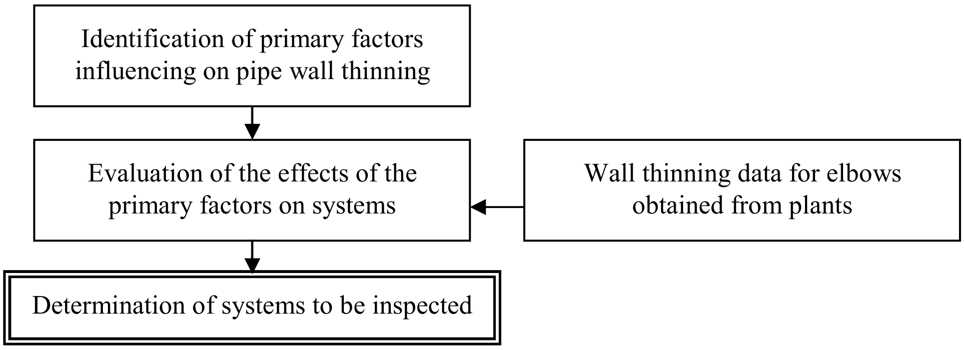

Fig. 1. Selection flow.

3.1.2. Identification of the primary factors influencing pipe wall thinning

When determining the criteria for selecting the systems to be inspected, we examined the fluid temperature, fluid flow velocity and steam quality (wetness), which are listed as primary factors affecting wall thinning in Table 2, below. Different factors contribute in differing amounts to the wall thinning for the various systems.

The chromium (Cr) content is another primary factor and contributes to the inhibition of wall thinning; however, we decided not to consider it during the selection of inspected systems because the Guideline applies to carbon steel pipes only, which should all have a similar Cr content.

Table 2 Selection of Primary Factors Influencing Pipe Wall Thinning

Influential Factors	Conditions	Adopt or Not
Flow velocity	Fluid flow velocity is different among systems.	A
Temperature	Fluid temperature is different among systems.	A
Steam quality	Steam quality is different among systems. (i.e., single-phase, two-phase, superheat)	A
pH	Values are equivalent throughout all systems, being controlled at pH 8.8 to 9.2.	N
Dissolved oxygen	In PWR secondary systems dissolved oxygen levels are kept as low as possible, and low level of dissolved oxygen have dramatic effects on the wall thinning rates. However, they are equivalent throughout all systems, being controlled at 5 ppb or below.	N
Chemical composition of material	Cr is the most important alloying element for improving resistance. Mo and Cu may also have beneficial effect. However carbon steel pipes should have a similar content.	N

With regards the pH value and dissolved oxygen density, it is known that a higher pH value and a higher oxygen density are likely to inhibit pipe wall thinning. In PWRs, the water quality is controlled in such a way that the pH value of around 9 is maintained, which should inhibit wall thinning. On the other hand, the dissolved oxygen density is maintained at a low level in order to limit the transportation of iron into the steam generator, and this low oxygen density may contribute to wall thinning. The pH value and dissolved oxygen density, however, are uniform within the entire secondary system and therefore we excluded them from the list of primary factors that require evaluation when selecting the systems to be inspected (see Table 2).

3.1.3 Evaluation of the effects of the influencing factors on different systems

We classified systems with reference to the chosen factors influencing pipe wall thinning (steam quality, temperature and flow velocity). With each category of system, we checked whether the systems were prone to wall thinning or not prone to wall thinning by examining the available mass of inspection data from Kansai Electric's nine PWR plants. The figure below gives an example of our summary of investigation over different system categories: (see Fig. 2).

Steam Quality

Two-phase
Wetness of 15% or more

《Group A》 Temperature

Flow Velocity

Temp / Velocity	Less than 100deg C	100deg C - 150deg C	150deg C - 200deg C	200deg C - 250deg C	250deg C or more
Less than 30 m/sec	**N**	**P**	**I**	**N**	**N**
30 m/sec - 50 m/sec	-	-	-	-	-
50 m/sec or more	-	-	**S**	-	-

Two-phase
Wetness of 15 % to 5 %

《Group B》 Temperature

Temp / Velocity	Less than 100deg C	100deg C - 150deg C	150deg C - 200deg C	200deg C - 250deg C	250deg C or more
Less than 30 m/sec	-	-	-	**N**	-
30 m/sec - 50 m/sec	-	-	**I**	-	-
50 m/sec or more	-	-	**I**	**I**	-

I : indication of thinning, **N** : no indication of thinning, **-** : no applicable piping
P : no indication of thinning excluding particular sections (downstream of control valves and of glove check valves)
S : no available data because measures have been already taken (modification to SUS).

Fig. 2. Effect evaluation of influential factors among systems

3.1.4. Evaluation of the influence for piping geometry

The comparison of wall thinning for each piping geometry was done. The evaluation targeted the extracting system, condensate system and heater drain system, which were prone to wall thinning. The result of the evaluation was the establishment of a relative value for the wall thinning rate as compared to other part shapes. A standard value of (1.0) was assumed for the Elbow as is shown in the figure below. The difference in wall thinning rates as a function of the difference of the piping geometry is as follows: Bend pipes < Reducers < Straight pipes downstream of swing check valves < Elbows, Tees < Straight pipes downstream of ball check valves < Reducers downstream of control valves (see Fig. 3).

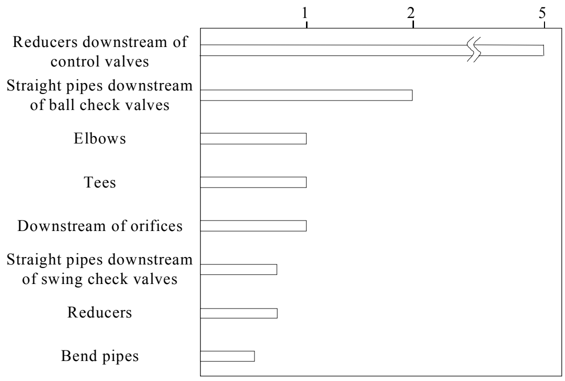

Fig. 3. Evaluation of relative tendency of wall thinning rate among different piping geometry.

3.1.5. Determination of systems to be inspected

In accordance with the system categories mentioned in Section 3.1.3 for evaluating the susceptibility to pipe wall thinning, we classified the criteria for selecting inspected systems in terms of steam quality, temperature and flow velocity. With the systems that were selected according to the criteria, we decided to require inspection for those pipe components where flow turbulence tends to happen, which included (see Table 3):

- Components downstream of a control valve;
- Components downstream of a ball check valve;
- Components downstream of an elbow, tee, or orifice;
- Components downstream of a swing check valve;
- Reducers and bends.

Table 3. Systems and components requiring inspection

Criteria for the systems to be inspected:
Carbon steel piping systems as follows (a) Two Phase Flow Lines: • pipes with wetness ≥ 5% & temperature 150–250°C • pipes with wetness < 5%, temperature ≥150°C & drain entrainment (b) Single Phase Water Flow Lines: • pipes with the temperature of 100–200°C (c) Downstream of Control Valve, Downstream of Ball Check Valve: • pipes with the temperature of 100–250°C
Components to be inspected:
Components that could cause flow turbulence: • Downstream of Control Valve, Downstream of Ball Check valve, • Elbow, Tee, Downstream of Orifice, Swing Check Valve, • Reducer, Bent Pipe

For those systems that are judged to be not being prone to significant wall thinning according to the selection criteria, we still decided that the components that could cause flow turbulence should be inspected at such a rate that the 25% of these components are inspected in the course of ten years. The Guideline is applicable to carbon steel piping systems. Stainless and low alloy steel piping systems are excluded from the scope.

3.2 GUIDELINE REQUIREMENTS ON THE INSPECTION METHOD

The Guideline requires that measurements be done using an ultrasonic thickness-measuring instrument. Thickness measuring points are specified differently for different components. In principle, a pipe size of 6B or larger requires inspection at eight points around the circumference and a pipe size of 5B or smaller requires inspection at four points around the circumference (see Fig. 4).

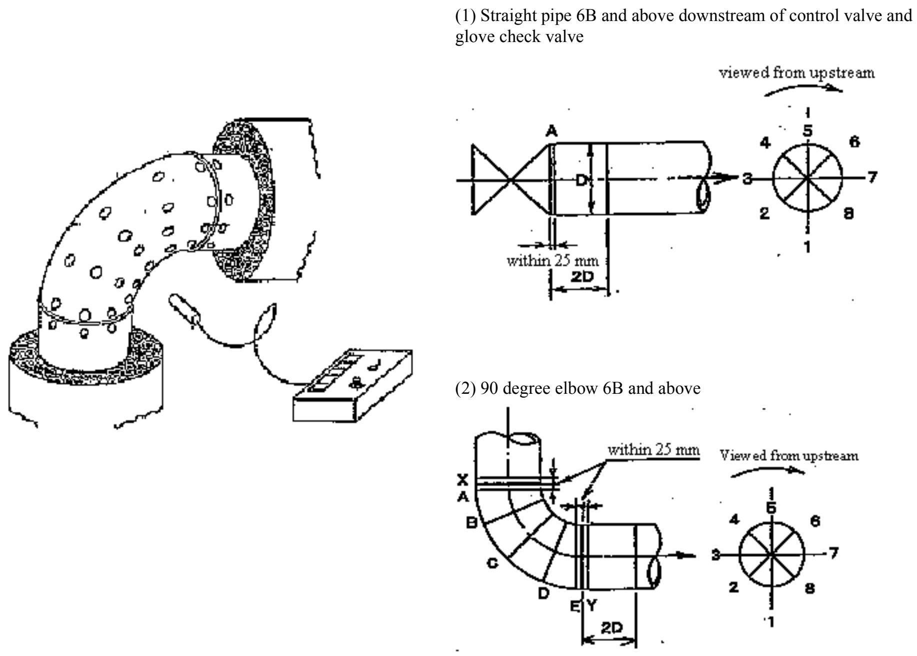

Fig. 4. Inspection Points.

According to the inspection manual of Kansai Electric, if these measurements reveal that a part has been thinned by one third or more of the normal thickness, the pipe is inspected more precisely with measurements performed in the manner of forming a 20 mm pitched grid. Thus, any pipe component that shows a significant tendency toward thinning is subjected to a detailed observation of progress of thinning (see Figs 5, 6).

3.3. GUIDELINE REQUIREMENTS ON THE INSPECTION FREQUENCY

The inspection frequency of each component is determined based on the remaining service life calculated from the thinning rate.

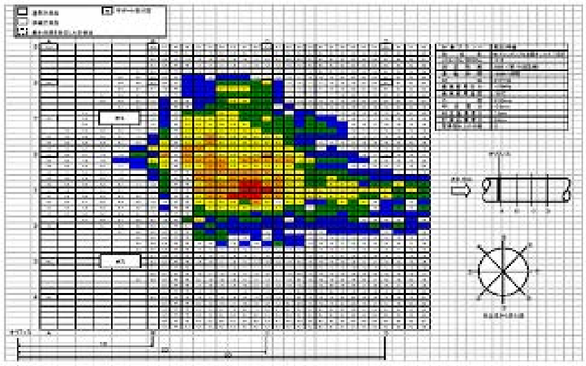

Fig. 5. Detailed measurement at Mihama Unit 3 (sample).

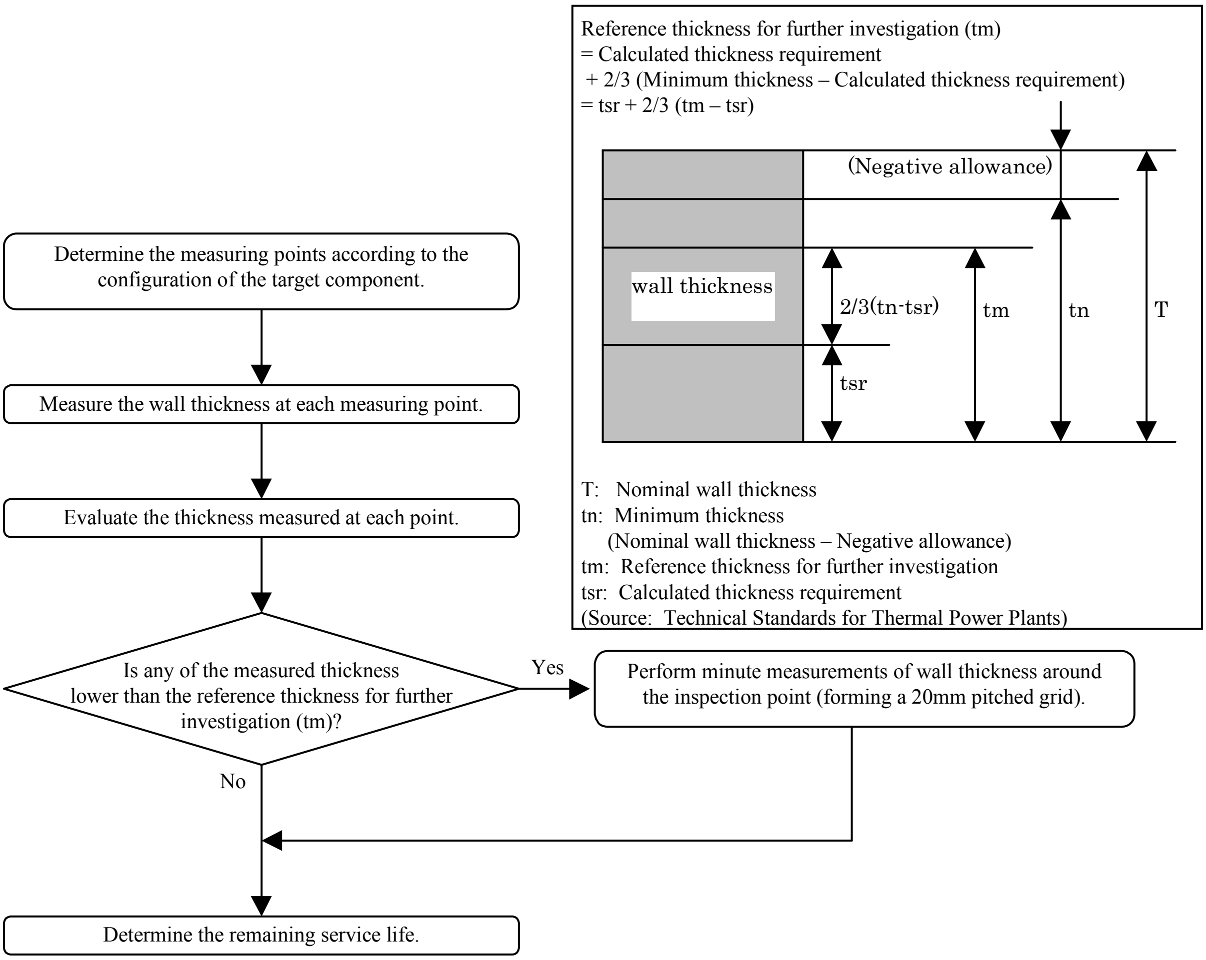

Fig. 6. Measurement procedure for "Major systems requirig inspection".

The remaining service life is calculated from the minimum thickness (tmin), minimum thickness requirement (tsr) and thinning rate (Wr). The next inspection for each component shall be scheduled for a date such that at least two years remained until the calculated end of lifetime (see Fig. 7).

The procedure to be followed when determining the thinning rate (Wr) depends on the number of inspections that have been performed so far, as described in the following subsections.

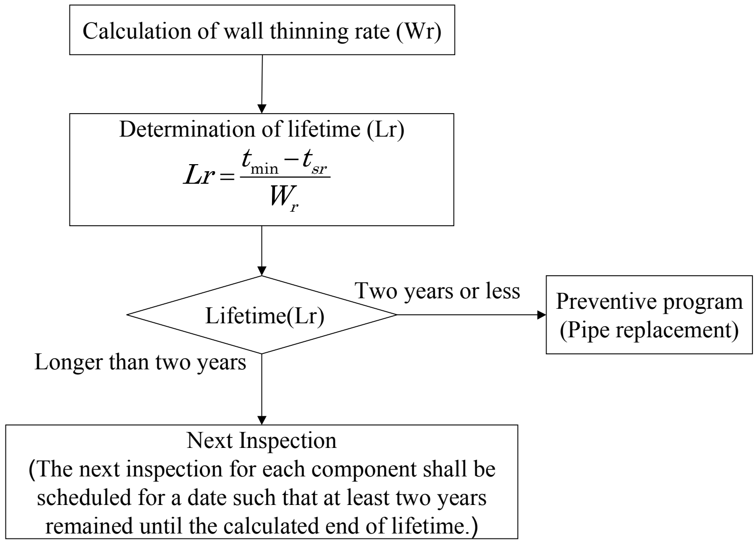

Fig. 7. Determination of inspection frequency.

3.3.1. Thinning rate of uninspected components

Since the thinning rate Wr is unknown for uninspected components, the initial thinning rate Wr(0) for these components was estimated from the thinning rate of elbows, determined by the statistical analysis of available inspection data. However, special provisions are made for components downstream of a control valve or a ball check valve because the analysis of available inspection data indicated that these components tended to suffer a high thinning rate. The initial thinning rate Wr(0) in components downstream of a control valve was estimated to be five times larger than the that of elbows. The initial thinning rate Wr(0) in sections downstream of a ball check valve was estimated to be twice as large as the that of elbows (see Fig. 8).

Wr for components which have not been inspected:

<table>
<tr><th rowspan="2"></th><th rowspan="2">Flow velocity</th><th colspan="5">Temperature</th></tr>
<tr><th>< 100 deg C</th><th>100-150</th><th>150-200</th><th>200-250</th><th>≥ 250 deg C</th></tr>
<tr><td rowspan="3">Two-phase flow with the wetness of 15%</td><td>< 30m/s</td><td></td><td rowspan="3">0.30</td><td colspan="2">0.35</td><td></td></tr>
<tr><td>30 - 50m/s</td><td></td><td colspan="2" rowspan="2">1.15</td><td></td></tr>
<tr><td>≥ 50m/s</td><td></td><td></td></tr>
<tr><td rowspan="3">Two-phase flow with the wetness of 5-15%</td><td>< 30m/s</td><td></td><td></td><td colspan="2">0.35</td><td></td></tr>
<tr><td>30 - 50m/s</td><td></td><td></td><td colspan="2" rowspan="2">1.15</td><td></td></tr>
<tr><td>≥ 50m/s</td><td></td><td></td><td></td></tr>
<tr><td rowspan="3">Two-phase flow with the wetness of less than 5%</td><td>< 30m/s</td><td></td><td></td><td colspan="3">0.35</td></tr>
<tr><td>30 - 50m/s</td><td></td><td></td><td colspan="3" rowspan="2">1.15</td></tr>
<tr><td>≥ 50m/s</td><td></td><td></td></tr>
<tr><td rowspan="3">Single-phase water flow</td><td>< 3m/s</td><td></td><td colspan="2" rowspan="3">0.45</td><td rowspan="3">0.30</td><td></td></tr>
<tr><td>3 - 6m/s</td><td></td><td></td></tr>
<tr><td>≥ 6m/s</td><td></td><td></td></tr>
</table>

Notes:
1. The unit for the initial thinning rate is 10^{-9}mm/hr.
2. Instructions about the meshed area in the table:
(1) Sections downstream of a control valve shall be assigned an initial thinning rate that is five times as large as that of elbows.
(2) Sections downstream of a ball check valve shall be assigned an initial thinning rate that is twice as large as that of elbows.

Applicable only to components downstream of a control valve

Applicable only to components downstream of a control valve or a ball check valve

Fig. 8. Initial thinning rate.

3.3.2. Thinning rate of components that have been inspected once or twice

With regards the components that have been inspected once or twice, we assumed that the amount of thinning equaled the difference between the maximum measured thickness and the minimum measured thickness. We divided this difference by the operating hours to determine the thinning rate. Each component has an uneven distribution of wall thickness already at the time of commissioning and the pattern of this distribution differs among components. With elbows and other components that should have the longitudinal evenness of wall thickness, we assumed that the amount of thinning equaled the difference between the longitudinally-measured maximum and minimum thickness values. With reducers and other components that should have the circumferential evenness of wall thickness, we assumed that the amount of thinning equaled the difference between the circumferentially-measured maximum and minimum thickness values (see Table 4).

Table 4. Thinning rate for components that have been inspected once or twice

Elbows and other components with the longitudinal evenness of wall thickness	Longitudinal thickness difference method: $Wr = \frac{\Delta t_{\text{Longitudinal}}}{T}$
Straight pipes, reducers and other components with the circumferential evenness of wall thickness	Circumferential thickness difference method: $Wr = \frac{\Delta t_{\text{Circumferential}}}{T}$

$\Delta t_{\text{Circumferential}}$ difference between the circumferentially-measured maximum and minimum thickness values

$\Delta t_{\text{Longitudinal}}$ difference between the longitudinally-measured maximum and minimum thickness values

T operating time

3.3.3 Thinning rate for components that have been inspected three times or more

With regards the components that have been inspected three times or more, the thinning rate was calculated for each measuring point using the least square method. The largest of the thinning rates from different measuring points on a component was taken as the thinning rate (Wr) of the given component.

The least square method:

$$Wr = -\frac{n\sum_{i=1}^{n} t_i T_i - \sum_{i=1}^{n} t_i \sum_{i=1}^{n} T_i}{n\sum_{i=1}^{n} T_i^2 - \left\{\sum_{i=1}^{n} T_i\right\}^2}$$

t_i Thickness values at i times inspection

T_i operating time to i times inspection

3.4. GUIDELINE REQUIREMENTS ON CORRECTIVE ACTIONS

The Guideline requires the utilities to re-calculate the remaining lifetime from the inspection data. In case that the remaining lifetime is less than 2 years, the utilities must develop a replacement plan for each component, and replace such component with anticorrosive materials or equivalent material.

4. CHALLENGES CONCERNING THE APPLICATION OF THE GUIDELINE

The PWR-operating utilities in Japan have followed the Guideline described in Section 3 as they systematically conducted the measurement of pipe wall thickness, covering all components that have been identified as being prone to thinning.

To our regret, however, a pipe rupture accident took place at Mihama Unit 3 in August 2004, causing the death of five workers and the injury of six workers. After the accident, it was revealed that the affected component was not registered in the pipe lists subject to the management. As a result, the affected component has not been inspected for wall thickness at all, since the beginning of plant operation. At the governmental Accident Investigation Committee meeting, the regulatory authority of Japan commented that the Guideline was "generally suitable as a management method."

However, in order to ensure complete management of pipe wall thinning, we shall develop a new private sector guideline using the mass of inspection data accumulated so far and latest findings from overseas. Discussions are required on the following matters:

(a) Redefinition of the scope

The new guideline should address not only the flow-accelerated corrosions but also other mechanisms that can contribute to pipe wall thinning. At the same time, discussions are required for the possible inclusion of other pipe materials such as stainless steel.

(b) Thinning rates and inspection targets based on measurement results

The initiation and progress of wall thinning have been observed in some systems that have been exempted from inspection under the existing Guideline. The inspection requirements must therefore be updated to include such systems (see Fig. 9).

(c) Inspection frequency determined by the remaining lifetime assessment

While the existing Guideline requires the utilities to schedule the next inspection on a date that leaves at least two years until the end of lifetime, this requirement may need to be made more conservative in view of preparation of the components in adequate time.

(d) Measuring method

The existing Guideline does not describe the minute details of the measurement method for thinned components, which should be included in the new guideline.

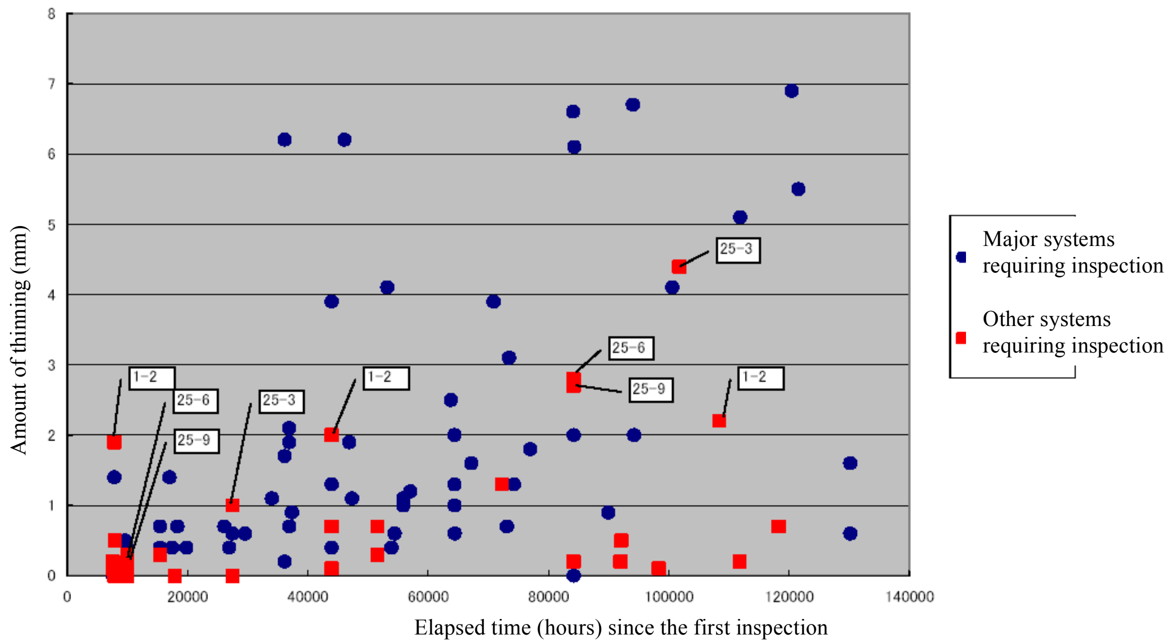

Fig. 9. Measurement results.

5. CURRENT STATUS OF DEVELOPMENT OF THE PIPE WALL THINNING MANAGEMENT CODES

The Japan Society of Mechanical Engineers (JSME) is leading an initiative to establish an open private-sector guideline through a transparent development process in order to address the issues described in Section 4. Kansai Electric and other utilities are actively involved in the process.

In response to a social pressure concerning this issue, JSME formed a task group with the membership of experts qualified by their experience and knowledge on pipe wall thinning and started to develop pipe wall thinning management codes. JSME intends to divide the code development activities into the two following phases:

5.1. PHASE 1: DEVELOPMENT OF THE PERFORMANCE REQUIREMENT CODE

Phase 1 is dedicated to the development of a performance requirement code. The code mainly deals with the requirements concerning the management of pipe wall thinning. Work on this code started in September 2004 and the code will be issued in March 2005 following a period of public comment.

This code, applicable to the management of pipe wall thinning not only at nuclear power stations but also at fossil-fired power stations, will specify the following:

- Technical matters that need to be addressed by each utility's management guideline;
- The procedure to be followed by JSME when giving qualification to these guidelines;
- General requirements concerning the management of pipe wall thinning.

Thus, this code is intended to promote the responsible management of pipe wall thinning by utilities.

5.2. PHASE 2: DEVELOPMENT OF TECHNICAL REQUIREMENT CODES

Phase 2 is dedicated to the development of a series of technical requirement codes. The codes will specify technical requirements such as the following:

- the method for selecting the pipe sections to be inspected;

  - Life prediction method;
  - Method for selecting the inspection timing;
  - Measurement methods and measurement ranges.

A series of technical requirement codes will be developed for different types of installations: PWR plants, BWR plants, large fossil-fired power stations and small fossil-fired power stations. The figure below shows the two phases of development activity of JSME codes. A table below shows the development schedule.

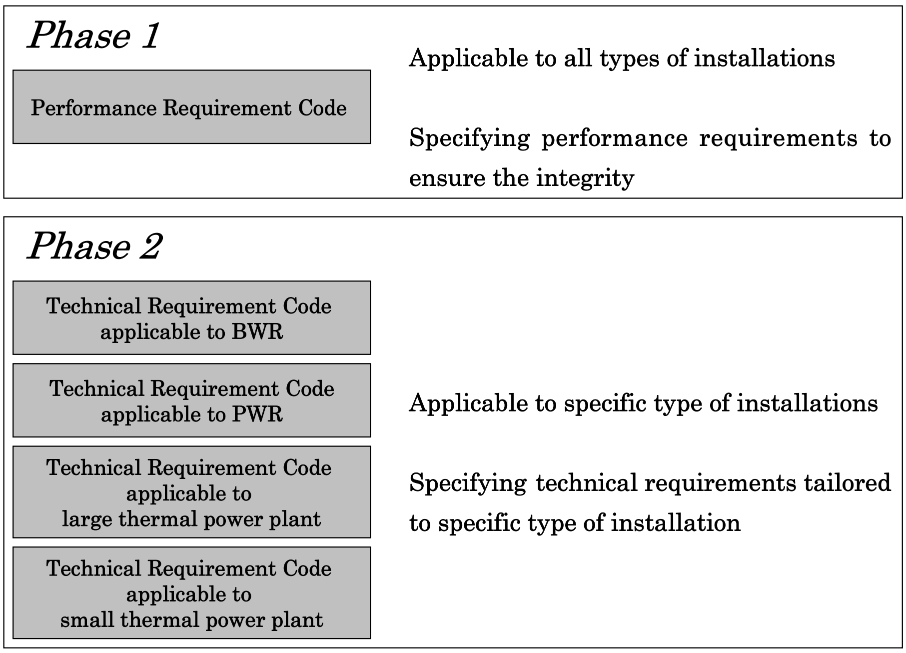

Fig. 10. Two phases of development of JSME codes.

Table 5 Schedule for development of JSME codes

Schedule	Detailed activities
November 2004:	JSME established the Task Force on Pipe Thinning.
March 2005:	JSME will publish the Performance Requirement Code for Pipe Thinning Management, applicable to both nuclear and fossil-fired power stations.
2005–2006:	JSME plans to publish the Technical Requirement Codes for Pipe Thinning Management (separately for PWRs and BWRs).

6. CONCLUSION

The PWR-operating utilities in Japan have addressed the problem of pipe wall thinning by developing a guideline using the available mass of inspection data from their plants and finally establishing it as “Guideline for Management of Piping Wall Thinning in PWR Secondary Systems” in 1990. Since then, the utilities have followed the Guideline as they systematically conducted the measurement of pipe wall thickness.

To our regret, however, a pipe rupture accident happened at Mihama Unit 3 in August 2004. In response to this accident, we decided to work toward the development of a new private-sector guideline to ensure complete management of pipe wall thinning. The new guideline will reflect the experience that we have accumulated so far through the monitoring of piping wall thickness.

LAGUNA VERDE U2 NPP / REACTOR VESSEL SHROUD IGSCC SUSCEPTIBILITY ASSESSMENT AND ITS INSPECTIONS RESULTS

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Abstract. This paper shows the IGSCC behavior in Laguna Verde Unit 1 & 2. LVNPP 1&2 have identical material and structural designs as well as a similar operational water chemistry and neutron fluency conditions. However, IGSCC behavior in core shroud welds are strongly depending on fabrication and construction residual stresses conditions for each of them.

1. REACTOR VESSEL SHROUD (RVS) FABRICATION

Laguna Verde Reactors Unit 1 and 2 are GE /BWR-5, 2027 Mwt each.

Laguna Verde RVS is built with 304L material plates, carbon content 0.014–0.030% and Solubilized Heat Treatment (SHT), and it is divided into three main sections: Shroud Upper Half, Shroud Mid Section and Shroud Lower Half. Figures 1 and 2 show plate components and welding lines, respectively.

Although Laguna Verde Unit 1 & 2 has identical shroud designs, manufacturers were different, therefore some assembly and built process in between are assumed different.

2. SHROUD IGSCC ASSESSMENT

According to operational experience gathered by CFE assessment in 1999, for category “B” Plants with 8 operational years and above, the cracks initiation occurs on “beltline” welding (maximum fluency zone), more on H4 than H5. Some plants with “ring-plate” type welding as H3 and H6 have also more incidences.

In 2000, Laguna Verde Unit 1 NPP, with an identical design as Unit 2, was categorized as a “A” Plant according to BWRVIP-76 Guideline. In May 2001, during 8th RO, Unit 1 UT examination results confirmed flaws on horizontal welds H3 and H4, and found additional flaws on H1, H5 and H7. In 2002, a Unit 2 Reactor Vessel Shroud Susceptibility Assessment was prepared during 6th operational cycle due to results of a Root Cause Analysis of same component flaws experienced in Unit 1. Although Units 1 & 2 shroud base material is the same (304L), different manufacturers were contracted so that differences in fabrication process were estimated. Unfortunately there are not enough Unit 2 fabrication records to be considered, but data from Laguna Verde Unit 1 and another plant with same manufacturer were used in the analysis.

Unit 2 Shroud IGSCC Susceptibility Assessment concluded that: weld H3 was the mayor based on surface flaws, highest ECP and construction welding type; welds H4 and H5 as medium due to maximum Neutron Fluency and high residual stresses; weld H7 as medium due to an extra welding material used; the remaining horizontal and all vertical welds were classified as a low susceptibility component. Special inspection requirement was given to weld H6A welding due to welding process and fabrication steps sequence used.

Short Term Actions were taken so that welds H3, H4 and H5 were scheduled to be visually inspected on that next 6th RO; welds H3, H4, H5, V3, V4, V5 and V6 were scheduled to perform an Accessibility Analysis for UT with more coverage before 7th RO; and perform UT inspection with a minimum 50% coverage on 7th RO for welds H3, H4, H5, H6A and H7.

As a Medium Term Action an aggressive HWC and NMCA injection program was recommended.

3. INSPECTIONS AND EXAMINATION RESULTS

During 6th RO Unit 2, visual inspections (EVT1) was performed on ID and OD in weld H3 without flaws, meanwhile H4 and H5 were partially visually inspected without flaws. During 7th RO Unit 2, the inspection results were as follows (see also figure 3):

H1, ≈ 62% of weld coverage (UT), NO indications

H2, ≈ 62% of weld coverage (UT), NO indications

H3, ≈ 90% of weld coverage (53% UT, 27% EVT1 [ID, OD]), NO indications

H4, ≈ 90% of weld coverage (53% UT, 27% EVT1 [ID, OD])

INDICATIONS ≈ 36% cracking of weld (27% UT, 9% EVT1 on OD)

H5, ≈ 90% of weld coverage (53% UT, 27% EVT1 [ID, OD])

INDICATIONS ≈ 42 % cracking of weld (28% UT, 14% EVT1 on OD)

H6A/B, ≈ 53% of weld coverage (UT), NO indications

H7, ≈ 53% of weld coverage (UT),

INDICATIONS ≈ 27% cracking of weld on ID. UT by OD Tracker

As a additional data, at EOC 7, Laguna Verde Unit 2 was » 8.6 years of operation with average Conductivity < 0.15 m S/cm.

4. CONCLUSIONS

Even that Laguna Verde Unit 1 & 2 have identical material and structural designs as well as a similar operational water chemistry and neutron fluency conditions, IGSCC behavior results in both typical core shroud welds are strongly depending on fabrication and construction residual stresses conditions for each of them.

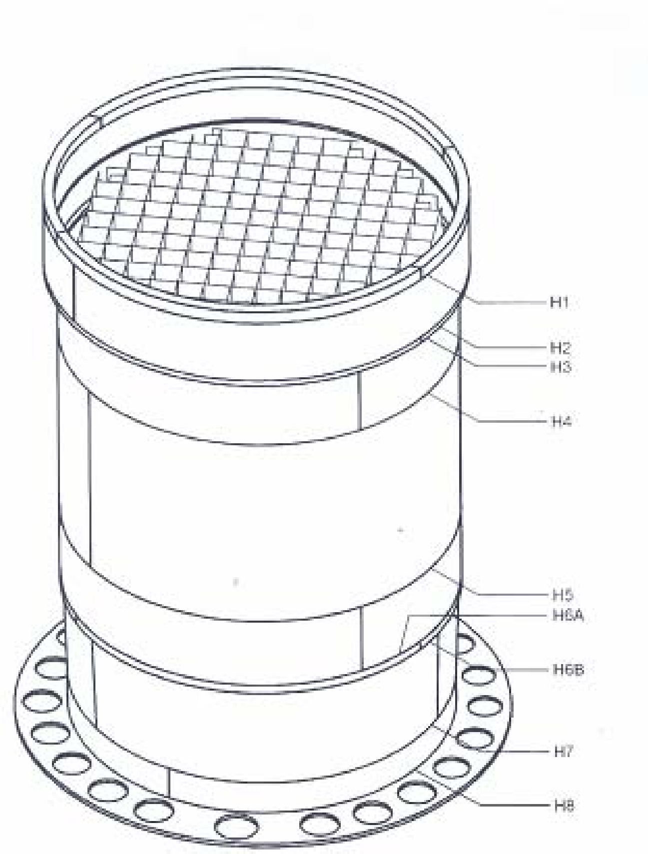

Fig. 1. Laguna Verde Core Shroud.

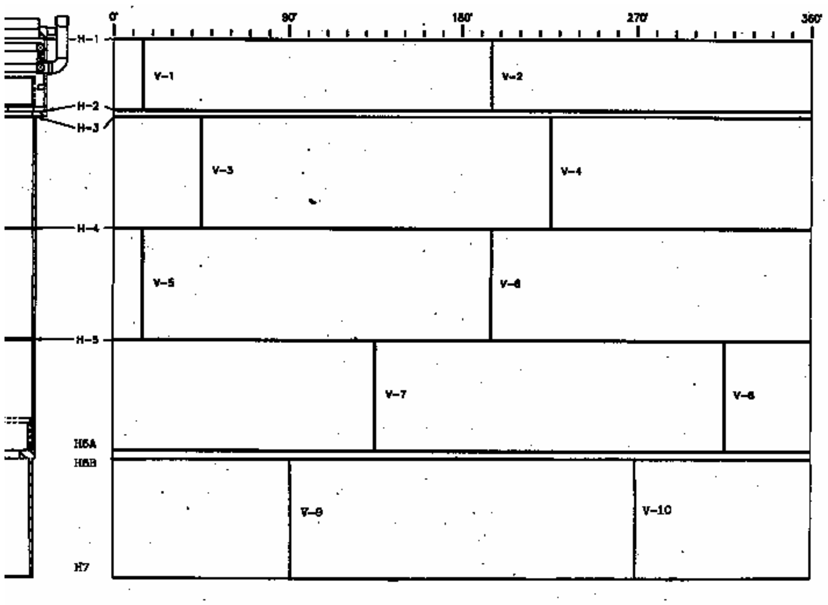

Fig. 2. Welds Arrangement for Laguna Verde Core Shroud.

MANAGERIAL ISSUES

(Session 3)

THE KKL PROGRAM FOR THE PREVENTION OF PIPING DEGRADATION DUE TO FLOW-ACCELERATED CORROSION PHENOMENA

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Abstract. Flow-accelerated corrosion is a degradation process resulting in wall thinning of piping, vessels and equipment made of carbon or low-alloy steel. Areas suffering from flow-accelerated corrosion are difficult to locate, as this degradation process occurs only locally under specific conditions of flow, water chemistry, temperature and material behavior. For the Swiss NPP Leibstadt (KKL) a systematic screening procedure was applied with the intension to reliably identify system areas, which may be subject to a flow-accelerated corrosion attack. During the screening process the heat balance diagram of the power plant was modeled using the functionality of the COMSY software tool, including system parameters for each relevant system area. Based on this model an analysis of the water chemistry cycle was performed, considering associated thermo-hydraulic parameters. Taking into consideration the materials used in each case, the system areas were studied with respect to the degradation potential imposed by flow-accelerated corrosion. This paper presents the approach to counteract flow-accelerated wall thinning with the help of computerized tools developed by Framatome ANP, which the operator of Leibstadt NPP (KKL) in Switzerland adopted in 1989 in response to regulatory requirements and inquiries. The results of the methodology applied since then with commendable success will be discussed.

1. INTRODUCTION

Flow-assisted corrosion (FAC), better known as but incorrectly named erosion-corrosion, has always been an operational and safety problem in nuclear and fossil-fueled power plants. The tragic accident at the American NPP SURRY-2 in 1986 and other failures in power plants elsewhere, prompted utilities and authorities to consider actions to detect, evaluate and monitor such causes of degradation/damages. Because of the frequently parallel occurrence of erosion-corrosion and cavitation erosion in single-phase water flow and erosion-corrosion and droplet impingement erosion in wet steam flow, it is expedient to extend such actions to all flow-induced metal loss phenomena and their consequences.

The main characteristics of the degradation processes, their definitions and associated explanations demonstrate that cavitation erosion and droplet impingement erosion are primarily mechanical defect mechanisms assisted by chemical corrosion processes whereas erosion-corrosion is mainly a chemical corrosion process (assisted by fluid dynamic mechanisms) and is caused by intense mass transfer due to highly turbulent flow keeping the metal surface in a permanent state of elevated reactivity [1, 2]. As the key parameters affecting this process can generally only be modified in the context of major pipe replacements or construction of new plants, methods for early detection of degradation, safety assessment and monitoring of degraded components are of major importance.

The approach to counteract flow-induced wall thinning, which the operator of KKL adopted in 1989 in response to regulatory requirements and inquiries, is presented in the following. The results of this approach after its application on the KKL plant will be discussed. But first the flow-induced material degradation phenomena will be defined and the tools used for the KKL approach will be described.

2. DETECTION OF DEGRADED COMPONENTS AND THEIR SAFETY ASSESSMENT

Framatome ANP (former Siemens/KWU) had developed procedures and computer programs as response to this task, see Fig. 1. The PC program WATHEC (Wall Thinning due to Erosion Corrosion) determined the susceptibility of piping systems to flow-induced material degradation. The PC program DASY (Data Management System) was applied for recording, managing, evaluating and documenting the data obtained from nondestructive examination (NDE) of individual piping components. Important features of both programs can be seen from [2].

In 1999 the COMSY software system (**C**ondition-**O**riented ageing and plant life **M**onitoring **SY**stem) was introduced as an integrated tool for ageing and plant life management of mechanical components. This knowledge-based program system allows the overall lifetime of the pressure boundary of mechanical components to be tracked. The concept is based on the comprehensive experience gained in conducting analyses for weak points due to flow-accelerated corrosion over a period of more than ten years.

As a first step a rough analysis (screening) is performed in a simple and cost-effective way taking only into consideration design and operating parameters as well as worst geometric and flow conditions. This investigation identifies those piping systems, which will not suffer a reduction in service life due to material degradation and can be disregarded during future inspection campaigns, see Fig. 1.

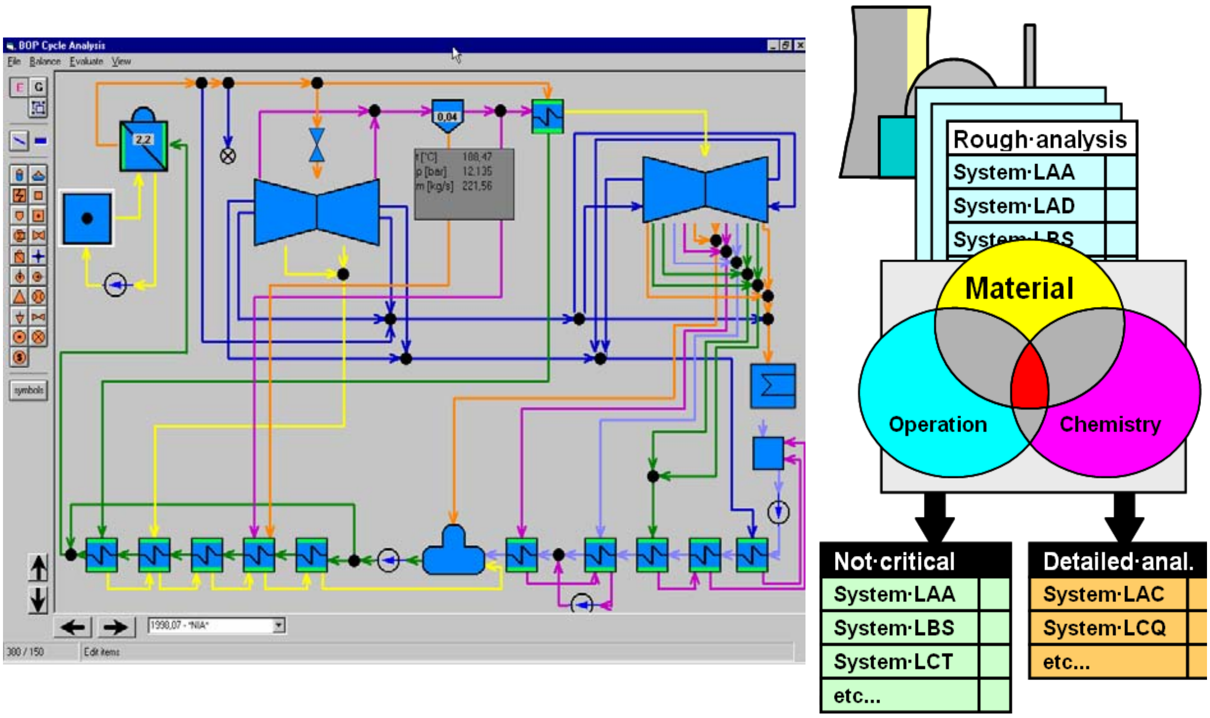

Fig. 1. Rough analysis using COMSY.

A detailed analysis as the second step is thus only carried out for piping systems for which it cannot be ruled out that material degradation may jeopardize operating safety. As also local stress conditions are modeled and analyzed the results provide besides information on the

- wall thinning distribution for the piping selected showing in which component the highest wall thinning is to be expected also the

- structural condition of each component selected to make a conservative estimate of residual life expectancy and to allow preparation of wall thickness measurements or inspections programs as well as implementation of repair/replacement activities at an early stage.

The reliability of the calculation results is dictated by the accuracy of the input data. One possibility of obtaining more reliable results on deadlines for wall thickness inspections is the use of a limited number of quality assured wall thickness readings. With the aid of such readings the results can be "calibrated" by performing a repeat calculation as this eliminates uncertainties due to inaccurate knowledge of input parameters, see Fig. 2. The wall thickness of components potentially at risk is usually checked by means of NDE methods, the most common of which presumably is ultrasonic examination. The amount of data gathered varies according to the number of components examined, their size and the grid density considered necessary. The often very large amount of data was managed with DASY and automatically used in WATHEC for optimizing prediction accuracy (the COMSY software introduced in 1999 now integrates both functionalities).

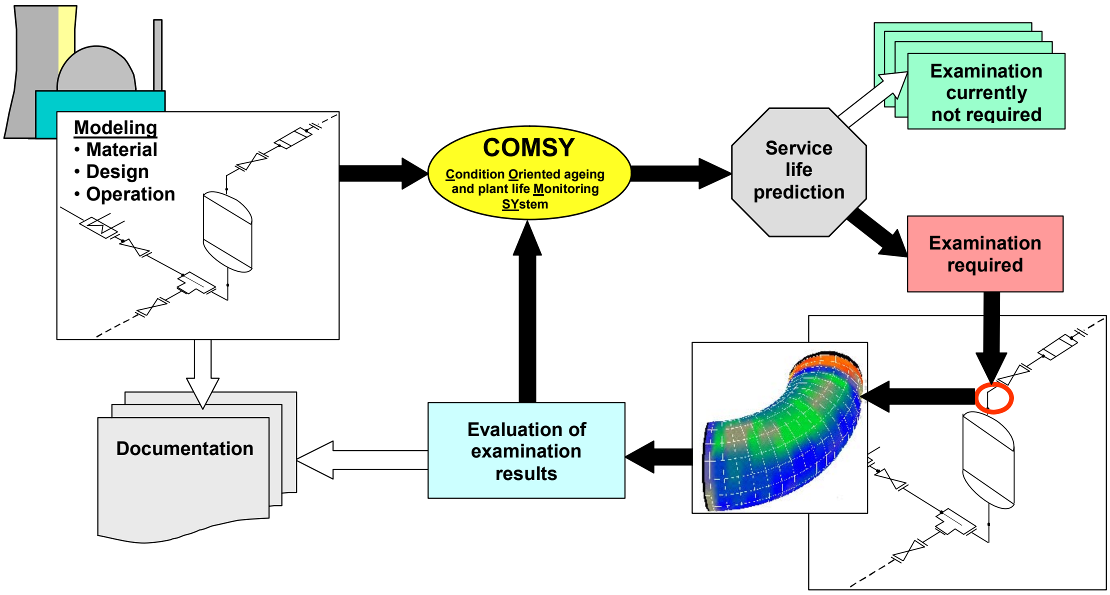

Fig. 2. The closed loop process for PLIM.

3. APPROACH FOR LEIBSTADT NUCLEAR POWER STATION (KKL)

Safety-related aspects and a targeted high availability of the power plant as well as the requirements stipulated by Swiss regulatory authorities as the result of the accident at SURRY-2 prompted the operator of KKL to introduce a procedure for the early identification of susceptible piping sections and components and the prevention of damage due to flow-induced material degradation mechanisms, see also [3]. This included the following steps and measures:

- Identification of pipes and piping components (potentially) susceptible flow-induced wall thinning

- Inspection of components by measuring their wall thicknesses once or periodically

- Walk down inspections of large-diameter piping or visual examination with optical aids

- Evaluation and documentation of wall thickness measurements and inspections, and result assessment (verification of integrity)

- Initiation of repair and replacement measures, if necessary.

The scope of the mentioned steps is not only largely identical to measures taken in other countries but in most cases even more extensive. An essential feature of this procedure, which was initiated by KKL in 1987 as a so-called EROSKO program (see Fig. 3), is its systematic approach. This was made possible by computer program WATHEC which was selected for this purpose following thorough research by the operator of KKL and then customized to suit the specific conditions prevailing at Leibstadt NPP and to meet Swiss regulatory requirements. The EROSKO program is characterized by the following features:

- The computer program WATHEC considered all known parameters influencing metal loss mechanisms and allows the integration of experience and data from laboratory tests performed by Siemens/KWU (now: Framatome ANP GmbH), from the literature as well as from other power plants also using this code.

- It was identified at a very early stage that the results of the predictive analyses should be examined by wall thickness measurements on a small number of components as spot checks. The predictive analyses already suggested a limited number of components where wall thickness measurements should be performed. These measurements carried out allowed the reliable assessment of the actual conditions of a component or a piping section.

- Additionally, the susceptibility calculated by the predictive code was assessed by appropriately trained and experienced power plant personnel.

This interaction of different activities ensured the optimum use of all resources required to prevent damage due to flow-induced material degradations.

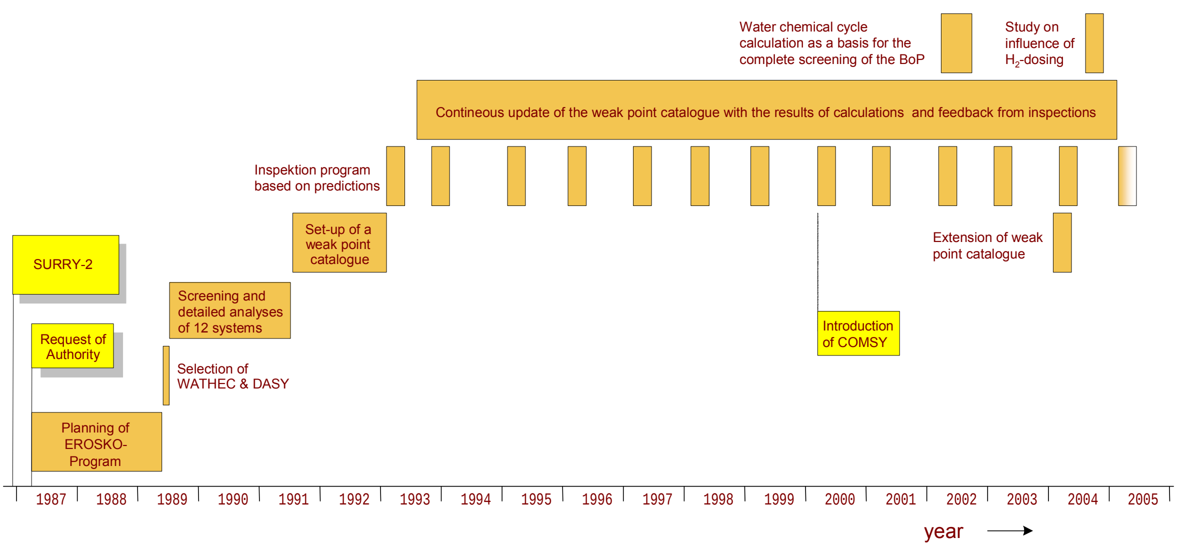

Fig. 3. The closed loop process for PLIM.

The approach with an initial rough analysis enabled a quick categorization of the piping systems into those, which were (potentially) susceptible to flow-induced wall thinning and those, which were not. This limiting step was performed using the experience available at KKL as well and covered safety-related and non-safety-related systems as well as small-bore piping systems. On the basis of this rough analysis, detailed analyses were performed for potentially susceptible piping systems and components only. These analyses yield a list of recommendations for wall thickness measurements to be performed during the next refueling outages (see Fig. 3). The results of the measurements are obtained by appropriately instructed inspection personnel in line with the approach adopted by KKL and the requirements derived there from. These data enables the "calibration" of the predicted susceptibility for all components considered. The "calibrated" calculation of the remaining service life of each component allows individual inspection intervals to be scheduled and can thus be used for refueling outage planning.

The interactive approach described above improves year to year the ability to predict (potential) susceptibility of piping components and particularly the effectiveness of the inspection program for the next refueling outage. The combination of predictions, the measured data and other inspection results (e.g. from visual examinations made during plant walk downs) available to the personnel of KKL allows an extensive catalog of weak points to be prepared (see Fig. 3). This enables the NPP personnel to access a comprehensive summary of information about component histories as well as data from previous measurements and walk down inspections, and thus to optimize the assessment of component conditions. The weak-point catalog is thus a tool for information and inspection of piping systems and piping components in order to optimize the condition-oriented planning of future measurements and inspections. To this end, the catalog is revised and updated on an annual basis.

After the introduction of the COMSY program with its new ability of performing a complete water chemical cycle calculation the oxygen concentration in the balance of plant (BOP) systems were re-evaluated which led to an extension of the number of systems considered in the weak point catalogue. Also special questions were solved using this new tool, e.g. the influence of dosing hydrogen into the feedwater for protection of RPV internals against stress corrosion cracking.

4. RESULTS FROM THE WEAK POINT ANALYSES OF KKL PIPING SYSTEMS

A sample of the piping systems at KKL for which a rough analysis has been performed is given in Fig. 4. The table shows an approximate assessment according to flow-accelerated material degradation. The piping systems selected for a detailed analysis on the basis of this evaluation are shown in the heat flow diagram in Fig. 5. The results of the detailed analysis lead to recommendations for wall thickness measurements.

A comparison of wall thinning based on quality-assured wall thickness measurement data of four selected systems with the wall thinning calculated by the predictive software is shown in Fig. 6. The chart shows a relatively good agreement between the data and also that the prediction made tends to be conservative. The scatter of data is caused by inaccuracies included with input parameters, e.g. true material composition, i.e. content of Chromium, Copper, Molybdenum of piping elements, etc.

This conservative approach with regard to the predicted remaining wall thickness and the remaining service life of the component is intentional in order to ensure that countermeasures can be initiated prior to any damage. The computed wall thinning is considered the maximum probable loss of material for piping components under specified operating conditions. The tolerance band included will only be reduced with reliable information regarding component integrity being available, i.e. difference between smallest quality-assured measured wall thickness and minimum allowable wall thickness.

System	Temperature [°C]	Steam Quality	02 water [ppb]	Material
ND-Turbine 3 zu ND-VW2a	104.8	.942	0	X6CrNiMoTi17-12-2
ND-Turbine 2 zu ND-VW2b	104.9	.925	0	X6CrNiMoTi17-12-2
ND-Turbine zu ND-VW3 bis Abzweig	152.6	.98	2	X6CrNiMoTi17-12-2
ND-Turbine zu ND-VW3a ab Abzweig	152.6	.98	2	X6CrNiMoTi17-12-2
ND-Turbine zu ND-VW3b ab Abzweig	152.6	.98	2	X6CrNiMoTi17-12-2
HD-Turbine zu Sp.W.Beh.	184.6	.86	5	10CrMo9-10
HD-Turbine zu HD-VW5 bis Abzweig	207.8	.883	10	10CrMo9-10
HD-Turbine zu HD-VW5a ab Abzweig	207.8	.883	10	10CrMo9-10
HD-Turbine zu HD-VW5b ab Abzweig	207.8	.883	10	10CrMo9-10
HD-Turbine zu HD-VW6 bis Abzweig	229.3	.912	18	13CrMo4-4
HD-Turbine zu HD-VW6a ab Abzweig	229.6	.912	18	13CrMo4-4
HD-Turbine zu HD-VW6b ab Abzweig	228.9	.912	18	13CrMo4-4
VA Drainage zum Sp.W.Beh. bis Sammelbehälter	185.2	.002	5	St35.8
HGA Drainage zum Sp.W.Beh. bis Sammelbehälter	185.	.002	5	St35.8
WA Drainage zum Sp.W.Beh. bis Sammelbehälter	184.6	.002	5	St35.8
Abscheider-Kondensate ab Sammelbehälter bis Pumpe	184.6	.001	23	St35.8

Fig. 4. Rough analysis on risk potential due to FAC.

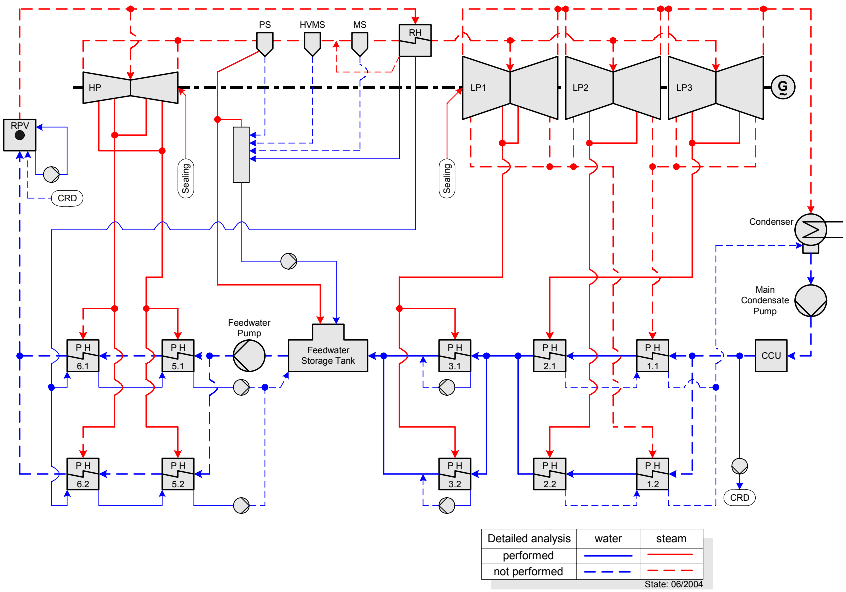

Fig. 5. Piping systems for detailed analysis.

A more detailed representation of this fact is shown in Fig. 7, which gives the initial wall thickness, the smallest measured wall thickness, the conservative prediction made and the minimum allowable wall thickness for various components of the extraction line #5. The results of this diagram can be summarized as follows:

- The piping system is affected by FAC because all components inspected show measurable wall thinning (lowest wall thicknesses measured are smaller than original piping component wall thicknesses).
- Lowest wall thicknesses measured are higher than conservatively forecasted by the program.
- Degradation detected does not require short term actions (repair/replacement) in most cases because in spite of 20 years of operation the lowest wall thicknesses measured are at least twice as thick as wall thicknesses required according to the allowable stress.

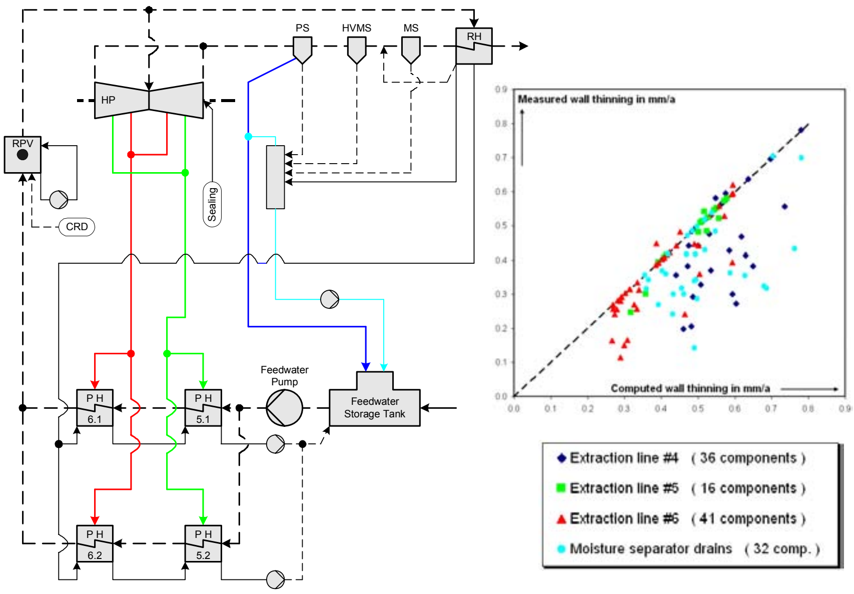

Fig. 6. Comparison of wall thinning measurements vs. predictions.

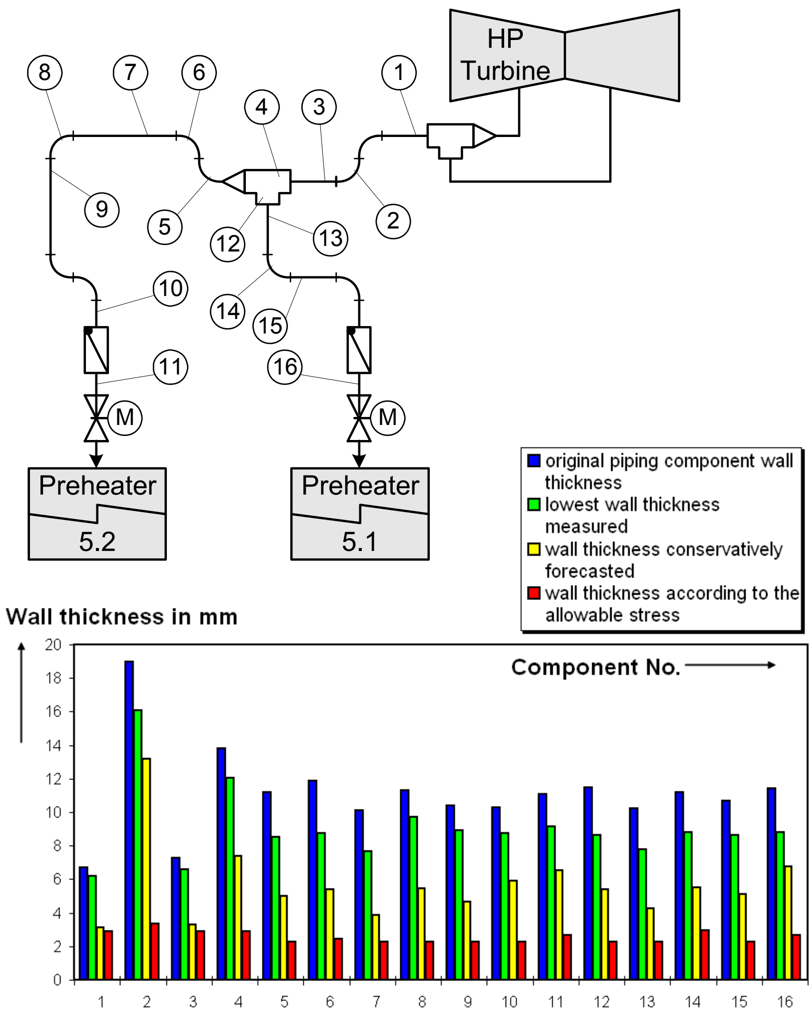

Fig. 7. Comparison of measured and computed wall thicknesses for extraction line #5.

In order to be able to assess the result of optimization of the measuring and inspection program for the weak point analyses of the KKL piping systems it is necessary to refer to the number of piping components which have to be subjected to a wall thickness measurement/inspection during the annual refueling outages. Thanks to the systematic approach this number has been continually reduced during the first years (see Fig. 8) and kept on this level over the last years of application by simultaneously fulfilling the safety requirements. This reduction in the monitoring activities plus the benefits associated with the reduced radiation exposure of personnel and the minimization of cost, time and effort during the annual refueling outages means that inspections activities are now required for only 35% of the number of components originally examined in 1993.

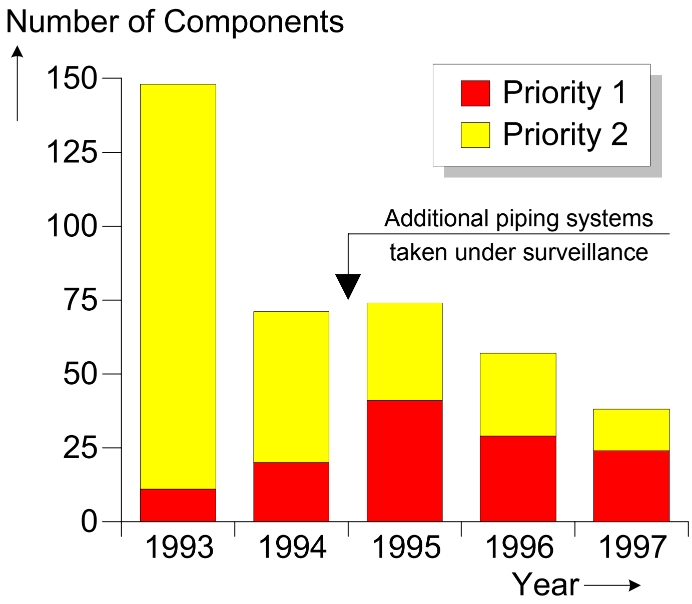

Fig. 8. Reduction of Examination Effort after Systematic Weak Point Analyses over 4 Years.

5. CONCLUSIONS

Framatome's computer program COMSY (and its predecessor programs WATHEC and DASY) focuses on the reliable identification of piping and piping components susceptible to flow-induced corrosion and on the reduction of the inspection scope and NDE efforts by optimizing the actions to be carried out during plant outages.

By customizing these measures to suit the individual conditions of Leibstadt Nuclear Power Station and to meet Swiss regulatory requirements by using experience and data obtained from plants in other countries and the experience of the operating personnel of KKL, it has been possible to establish an approach which is specifically developed to deal with flow-induced material degradation mechanisms and resulting risk imposed on the affected piping systems. The approach even allows predictions to be made with regard to a potentially increased susceptibility and/or an increase in the iron content in the steam, condensate and feedwater cycle due to an actual or planned upgrading measure.

The approach was applied with commendable success and is/will be essential part of the present/future-monitoring program EROSKO at KKL. The results gained up to now enable reliable conclusions to be drawn with regard to the frequency of component inspections and corrective actions.

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MANAGERIAL IMPROVEMENT EFFORTS AFTER FINDING UNREPORTED CRACKS IN REACTOR COMPONENTS

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Abstract. In 2002 TEPCO found that there were unreported cracks in reactor components, of which inspection records had been falsified. Stress Corrosion Cracking indications found in Core Shrouds and Primary Loop Re-circulation pipes at some plants were removed from the inspection records and not reported to the regulators. Top management of TEPCO took the responsibility and resigned, and recovery was started under the leadership of new management team. First of all, behavioral standards were reconstituted to strongly support safety-first value. Ethics education was introduced and corporate ethics committee was organized with participation of external experts. Independent assessment organization was established to enhance quality assurance. Information became more transparent through Non-conformance Control Program. As for the material management, prevention and mitigation programs for the Stress Corrosion Cracking of reactor components were re-established. In addition to the above immediate recovery actions, long term improvement initiatives have also been launched and driven by our aspiration to excellence in safe operation of nuclear power plants. Vision and core values were set to align the people. Organizational learning was enhanced by benchmark studies, better systematic use of operational experience, self-assessment and external assessment. Based on these foundation blocks and with strong sponsorship from the top management, work processes were analyzed and improved by Peer Groups.

1. FALSIFICATION OF MAINTENANCE RECORDS AT TEPCO PLANTS

In 2002 TEPCO found that there were unreported cracks in reactor components, of which inspection records had been falsified. Investigations by a TEPCO internal organization independent from Nuclear Power Division and external investigations by the regulators revealed that the inspection records were covered up in 29 cases over a period of 1988 to 1998. In a case, crack indications were found on the reactor core shroud of the Fukushima Daini Unit #3 in 1994, but no report was made to the regulators and the inspection findings were removed from the record. The cracks were characterized as Stress Corrosion Cracks, and the inspections were continued to be conducted periodically. In 2001 the cracks were reported to the regulators as newly found information, and repair was conducted.

Another falsification issue was found later in the same year. Stress Corrosion Cracks were found at Fukushima Daiichi Unit #1, 2, 3, 4 and 5 over a period of 1992 to 1995 on the Primary Loop Recirculation pipes which were made of Type 304SS material with Induction Heat Stress Improvement technique applied. The pipes were replaced without reporting the existence of the cracks to the regulators. From 1997 to 2002 Stress Corrosion Cracks were also found but in different material, Type 316LC-SS, of the Primary Loop Recirculation pipes at Fukushima Daini Unit #3, Kashiwazaki-Kariwa Unit #1 and 2. But these were also not reported to the regulators.

2. RECOGNITION OF WEAKNESS

There were cultural and organizational weaknesses found in the investigations. Production oriented culture was prevailed and continuous generation was pursued based on one-sided judgments without sound independent verification and oversight. Electricity demand grew rapidly until early 1990s and there was a chronic risk of supply shortfalls that might cause a massive blackout in Tokyo metropolitan area. Under the pressure to continue stable supply of the electricity, opinions justifying continuous generation were praised, and different views and second thoughts did not tend to be welcomed.

Overconfidence was another cause of the problems. The reporting criteria were described in a subjective way, requiring reports only when inspection results were judged as abnormal. But in many cases when inspection results were out of normal conditions such as finding of minor crack indications, unless judged as significant, the results were described as “no abnormality” and were not reported to the regulators. At that time there were no regulatory standards in Japan for the evaluation and repair of the cracks on reactor components, and it tended to take a long time to get approval from the regulators. In addition, negative public sensation was expected. Under such circumstances, overconfidence that “we know technology best” fostered over a long period a mistaken idea that failure to report could be permitted as long as there were no safety concerns over the physical plant conditions.

Complacency on the operational performances weakened organizational learning attitude and continuous improvement. Operating performance of Japanese nuclear plants were good and stable in 1980s and people lost interests in learning the operating experience and technical information from other countries. Stress Corrosion Cracking of reactor components were reported by US utilities in early 90s, but overconfidence in the selection of material such as Type 316LC-SS material caused lack of questioning attitudes, which allowed excessive cold work on those components while US utilities continued to pay attentions. Good performance also made an impact on public communications. It made people hesitate to communicate bad news with the public.

Strong vertical chains of commands and little cross-functional rotation of the human resources caused weak capability of systematic independent verification and oversight of engineering decisions for maintenance. Quality management was not fully functional because work processes were complicated and accountability was not clear.

3. RECOVERY ACTION

Top management of TEPCO took the responsibility and resigned, and recovery was started under the leadership of new management team. The first set of the recovery actions were focused on short-term sweeping changes in the areas needed to be improved quickly.

First of all, TEPCO reinforced corporate ethics programs for stringent compliance with laws and codes of conducts. Corporate Ethics Committee was established for this purpose. It is chaired by the president of TEPCO, and the members are independent experts such as lawyers and academics, executives of other companies, as well as top labor union officials. The committee put together a guideline for the behavioral standards for ethics as a specified measure in Charter for Corporate Behavior. There are three pillars in the guideline: compliance with rules, reliable behaviors and open communications. And the guideline clearly states that safety must always be given the first priority in our thoughts for any actions. All the executives and employees are required to take trainings based on this guideline.

Organizational transparency was improved. Non-conformance Control Program was totally changed by introducing a model of corrective action program from the United States. Employee Concerns Program was established, and a contact point was set up for consultation on compliance with laws and corporate ethics. The program is available to any sections of TEPCO, its subsidiaries and business partners. Operation records become more open to the public. Liaison committees are set up in the vicinity of TEPCO nuclear power stations to ensure that all information is disclosed. The members are representatives from local communities including local government officials, assembly members and residents, and they are provided with full information relating to the plants, including operational status, problems discovered during operations, inspection reports which include problematic findings, and reports on TEPCO internal assessments.

At each nuclear power station, TEPCO set up a Site Nuclear Quality Management Department that conducts independent assessment of nuclear safety and quality. This is under the direct control of the head office and is completely independent of the power stations. The department assesses the status of safety and quality and reports the results to the head office on a daily basis. If it identifies any problem with safety and quality at the power station, it will instruct the top management of the plant to make improvements. At the head office, a Nuclear Quality Management Department was set up. It collects information from the above departments stationed at nuclear power stations, and reports to the president at any time.

TEPCO also set up a Nuclear Safety and Quality Assurance Committee. The members of the committee consist of external experts such as nuclear experts, lawyers and academics. The committee reports to the president of TEPCO, and it ensures safety and quality by selecting the subject matter of independent internal assessments, examining the assessment reports, and offering suggestions for improvements.

As for the material management, prevention and mitigation programs for the Stress Corrosion Cracking of reactor components were re-established. For each reactor internal component at each plant, susceptibility to IGSCC was evaluated based on the base material, welding material, water chemical condition, construction work records and past inspection records. Then based on the degree of the susceptibility, necessary preventive actions such as residual stress improvement by under-water peening or increased monitoring were decided. A more comprehensive material management program for a wider variety of degradation mechanisms of various structure and components is now on agenda for future developments.

4. CONTINUOUS IMPROVEMENT IN PURSUIT OF EXCELLENCE

The above are the immediate remedial actions, and TEPCO continued to embark upon long term, continuous improvement activities in 2003. Rather than driven by minimum requirements of regulatory compliance, the long term, continuous improvement activities are driven by our aspiration to excellence in safe operation of nuclear power plants. By applying a change management method, the activities include the followings for continuous improvement:

- Setting an aspiring vision and core values;
- Organizational learning through benchmark studies and self-assessment;
- Leadership development;
- Process improvement and enhanced commonality of the processes throughout the fleet; and
- Measurement of the progress.

4.1. VISION AND CORE VALUE

First, vision and core values were set to align the people. The vision must inspire the people to drive the change. It needs to clearly represent the desire of people at all levels of TEPCO organizations and contractors to achieve higher levels of safety and quality. It must be also related to the values that make people feel proud of working in their workplaces. After some discussion, TEPCO decided the following statement as the vision of continuous improvement: "We will aim to realize trustworthy nuclear power stations that have the world's highest levels of safety and quality. We will create workplaces full of energy and confidence by making good use of the ability of all the personnel working at power stations and building innovative work processes."

This vision is supported by a core value of "Safe operation first". The following basic elements are recognized as values to support continuous improvement: Firstly effective communication, clear responsibility, and authority as important mechanisms; secondly integrity, confidence and pride, and faith in changes as important human factors; thirdly transparency, promotion of security and reliability, and respect of local communities as important factors for good partnership. Also self-accountability is emphasized as underlying value of individual behaviors for change.

4.2. ORGANIZATIONAL LEARNING

Organizational learning was enhanced by domestic and international benchmark studies. TEPCO have been making benchmarking trips to nuclear power stations in the United States to learn best practices in operational standards, equipment reliability process, work management, quality assurance, and systematic development of human resources.

We have also learned the best practices from Japanese companies. For example, TOYOTA is strong at "soft skills" to cultivate good behaviors and ways of thinking as well as their famous management techniques such as lean management. They value the behaviors of listening to people carefully, thinking what essential issues are, encouraging and proposing, providing ideas and wisdoms to win, asking colleagues' opinions, respecting facts and realities, and rising to the challenge. It was found that successful companies are based on continuous improvement processes enabled by both western rational way of process management and Japanese way of conventional human oriented management.

Gap analysis and self-assessment become more effective in identifying specific needs for improvements concretely after benchmark studies. The self-assessment is formally integrated in quality management systems and its results are subject to the periodic management review.

TEPCO recognizes that external assessment is also an important part of our continuous improvement process. Just before shifting into the long-term continuous improvement from the immediate recovery actions, TEPCO invited a group of nuclear industry experts who have hands-on experience in successful turnaround of plant safety performance in the United States. Their findings and recommendations were incorporated into the plans for the continuous improvement. Since then, we have been receiving the same experts in every quarter for continuous monitoring of our improvement status and expert opinions for change management. TEPCO also received an assessment by IAEA and a WANO peer review recently.

Systematic use of operating experience is also important. But it was difficult for many people to learn other countries' operating experience because of limited background knowledge. Therefore TEPCO created a new dedicated group at its head office to analyze the information and distribute it to relevant organizations within the company.

4.3. LEADERSHIP DEVELOPMENT

A leadership training program was introduced to develop people to lead the change in various levels of the organization of TEPCO and contractor companies. The program was designed and performed by the experts who have successful experiences in the PECO turnaround in the United States.

This is a two-week program providing a combination of behavioral skills such as communication skills, decision making stance, conflict management and accountability model, and business management skills such as process of process improvement and peer group management. These two different types of skills are important for the people to lead changes at any level of the organizations, and we believe that there is a synergic effect of learning both of them in an intensive training course. Guest speakers are invited from US nuclear industry organizations and nuclear operators so that participants can learn from their first-hand experience and knowledge.

TEPCO plans to achieve 500 graduates by the end of FY 2005. This accounts for about 20% of the personnel in TEPCO Nuclear Power Division. Targets are selected for the participation from TEPCO and contractor companies based on their potential for leading changes enthusiastically in their organizations. They are expected to participate in process improvement activities after the training.

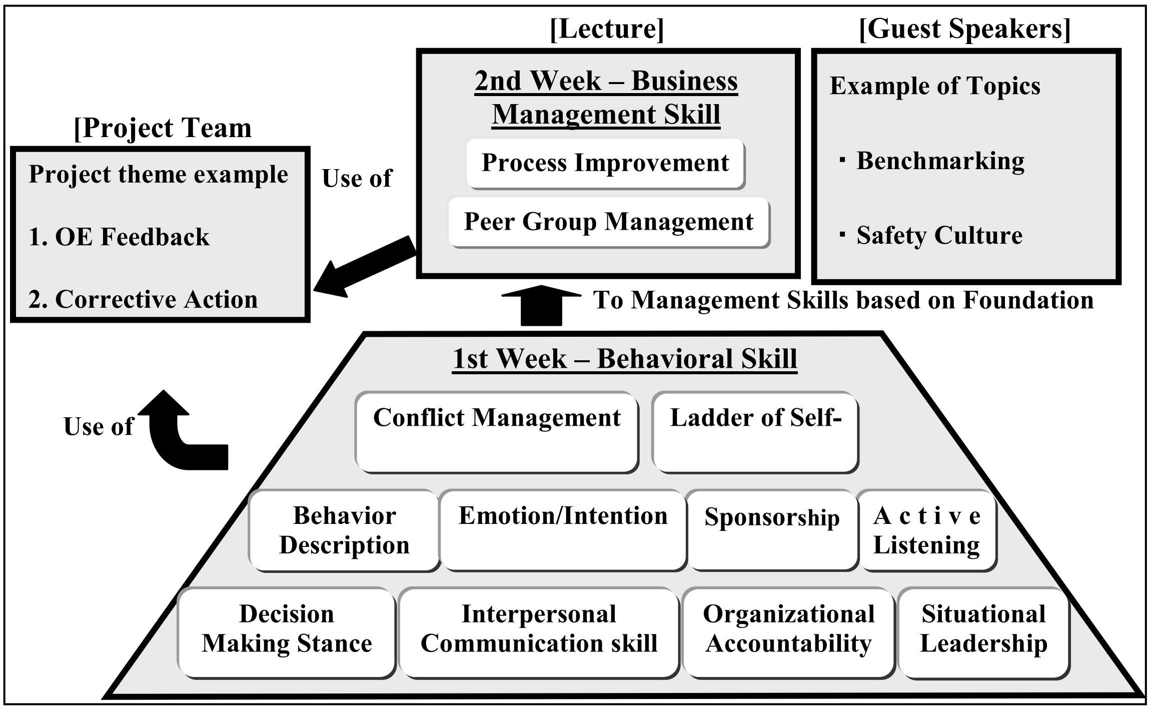

Fig. 1. Leadership Training Program Model.

4.4. PROCESS IMPROVEMENT AND SPONSORSHIP FOR CHANGE

Based on the above activities as foundation blocks and with strong sponsorship from the top management, work processes are analyzed and improved by Peer Groups so that best practices can be shared more easily among the plants.

The process improvement is designed and implemented by the following ten steps:

- Step 1: Determine the boundaries of the process that requires improvement, Organize, Capture “As-Is” SIPOC
- Step 2: Benchmark the process against industry “Best Practices” - Set Goals
- Step 3: Capture the current “As-Is” Process
- Step 4: Map any Sub-Processes
- Step 5: Capture issues with the current process, “Brainstorm”
- Step 6: Prioritize the issues
- Step 7: Create metrics/measures
- Step 8: Create schedule and assign responsibility for each issue. Determine the Root Cause & potential correction plans
- Step 9: Create the “To-Be” Process Diagram & SIPOC
- Step 10:Implement improvements and measure effectiveness

Peer Groups are set up for the following areas: Operation, Maintenance, Radiation Control, Quality and Safety, Internal Communications, External Communications, and Human Resources Development. They create common implementation of improved processes, develop and manage Performance Indicators, redesign functional area activities, and drive changes.

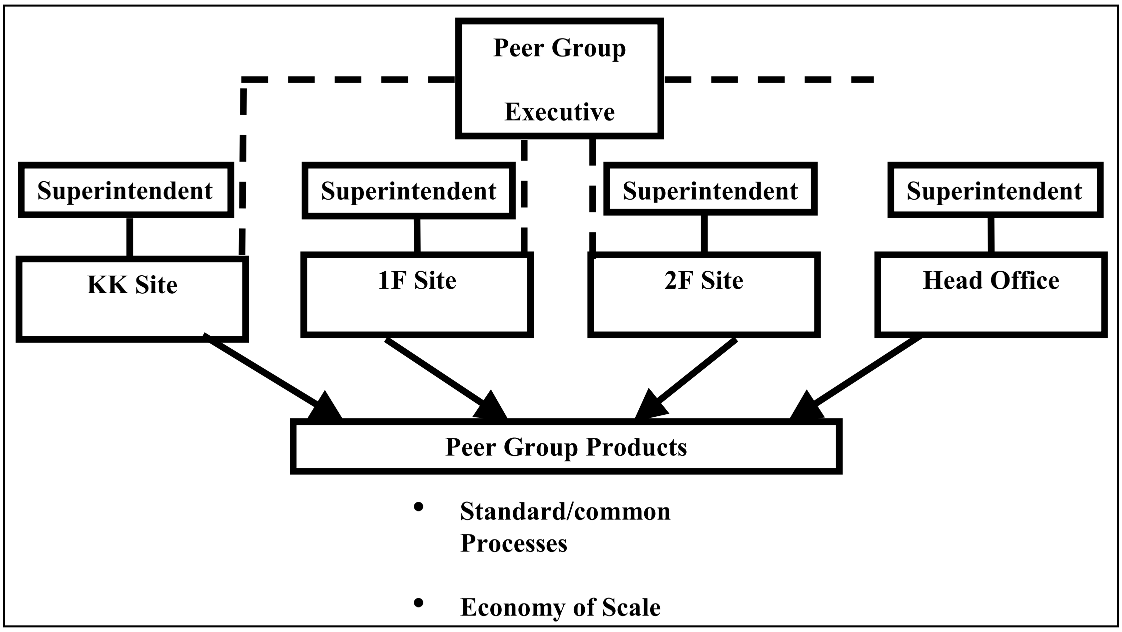

Fig. 2. Typical Configuration of a Peer Group.

Peer Groups are the most effective means to involve the people most knowledgeable of the current processes. They are also effective in promoting better commonality in future processes. We believe in the value of commonality because it makes sharing of best practices among different plants easier. It also improves transparency, enables effective oversight, and makes personnel rotation easier.

The Peer Groups have the following responsibilities:

- Standardizing and optimizing programs, processes, and procedures across the organization within their respective process areas as appropriate,
- Process ownership for respective area processes,
- Driving change initiatives and ensuring implementation & organizational compliance,
- Creating functional area goals and Performance Indicators,
- Monitoring functional area performance and Performance Indicators, striving for continuous improvement,
- Fostering identification and transfer/sharing of experience and company best practices as well as industry Operating Experience and best practices,
- Managing the sharing of resources across the organization,
- Continuous improvement of respective processes and sub processes (self-assessment, new technology applications, etc), and
- Developing business plans and monitoring progress.

In addition to the Peer Groups, a project team was set up for cross-functional improvements in Plant Operational Management. Strong sponsorship from the management is essentially important for the above activities. For each Peer Group and project team, a senior manager is designated as fleet-wide change sponsor. The sponsor expresses clear expectations for change, gives guidance for issue prioritization, and considers resource allocations for improvement activities. And progress of the improvement is reported to the Chief Nuclear Officer of TEPCO by those sponsors.

4.5. MEASUREMENT OF PROGRESS

TEPCO is now developing a new system of Performance Indicators (PIs). The PIs are designed to measure the progress toward improvement goals that are set up from the view points of internal and external customer value including safety and quality, business value, and human and technological development value. There are three levels of PIs for each kind of the value. The first level PIs measure station performance. The second level PIs measure the progress of improvement in selected key process areas. And the third level PIs measures some key activities for process improvement. A monthly review meeting is designed to be held at nuclear power stations with participation of senior managers from the head office.

5. CONCLUSIONS

In 2002 TEPCO found that there were unreported cracks in reactor components, of which inspection records had been falsified. Production oriented culture, overconfidence, complacency on operational performances, strong vertical chain of command, and dysfunctional quality management system caused this problem.

The immediate recovery actions focused short term sweeping changes by reinforcing corporate ethics program including behavioral standards, improving organizational transparency, independent quality assurance organization, better material management programs for SCC.

In 2003 TEPCO launched long-term improvement initiatives driven by the aspiration to excellence in safe operation of nuclear power plants. By applying a change management method, the activities include the followings for continuous improvement: setting an aspiring vision and core values, organizational learning through benchmark studies and self-assessment, leadership development, process improvement and enhanced commonality of the processes throughout the fleet, and measurement of the progress of change. Although these long-term improvement initiatives are still in an infant phase, we will continue the efforts in a step-by-step manner.

ASPECTS OF UNEXPECTED EVENTS IN NUCLEAR POWER PLANTS

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Abstract. Unexpected events in nuclear power plants (NPP) may lead to upset conditions or even accidents. Events such as these affect not only safety, but also the economic viability of NPP operation. Another facet of such events, virtually irrespective of their degree of severity, is the generally negative impact on public acceptance of nuclear power, such as was seen as a direct result of the Three Mile Island (USA - 1979) and Chernobyl (Ukraine - 1986) accidents. The operators of NPPs are responsible for their safe operation, whilst regulators ensure that NPP operating practices (e.g. start-up, shut-down procedures, inspections, monitoring, and compliance with technical specifications (TS)) are such that the highest possible levels of safety are a priori present at all times. As a matter of engineering principles, designs of NPPs feature safety margins and they are based on conservative assumptions, mostly to allow for material response to the operating conditions and environment (e.g. neutron embrittlement, fatigue usage). Inspections and monitoring have the purpose to check whether systems structures and components (SSC) are behaving according to the design with regard to compliance with safety requirements even when “aged”. The paper examines aspects concerning events or accidents in NPPs, despite generally high levels of SSC monitoring and inspection and regulatory oversight. The importance of materials selection at the design stage, and the need for vigilance and questioning attitudes is stressed. The necessity to learn from accidents or events that have occurred in other NPPs is shown to be an important tool and source of information for NPP designers, manufacturers, operators and regulators.

1. INTRODUCTION

Commercial nuclear power generation goes back nearly 50 years, whilst commercial aviation is almost twice as old. Although totally different in nature, both activities have close similarities. Technical safety and security are priority items and are correspondingly regulated. For both activities, it is also a goal to remain economically competitive in open markets whilst still ensuring safe operation.

Accidents can occur due to materials failure, design inadequacies, insufficient monitoring or surveillance and human performance factors. In fact, there are many industries or human activities that are subject to reliability-safety-regulations-economics aspects. Examples of non-nuclear activities/accidents are Bhopal (India - 1984)/chemical, the Titanic disaster (1912)/marine transport, Flixborough (UK - 1974)/chemical and the DC-10 aircraft accident (Paris – 1974).

The following paper first discusses the roles of regulators and operators of NPPs, and goes on to show the importance of human factors. Two examples concerning unexpected events that have occurred in NPPs are presented to underline why the event or accident occurred, and how it may have been avoided.

2. ROLES AND RESPONSIBILITIES OF OPERATORS AND REGULATORS

Those who exploit nuclear power are responsible for the operational safety of the NPPs. This includes not only the “nuclear side” but also the “secondary side” (non-nuclear) and even final treatment and disposal of radioactive waste arising from the NPP. Given the experiences gained thus far in the nuclear power industry and its regulation, it appears reasonable to state that, although high levels of inspection, control and vigilance are in place, there have still been events, accidents, “near-misses” and unforeseen impacts on safety and availability. This is also true for other industries (e.g. chemical, transport).

As NPPs approach their originally planned operating lives, and utilities and operators apply for license renewal for long-term operation, they must do so using arguments and facts based on evidence that the SSCs can still reliably perform within design requirements and TS. Periodic safety reviews (PSR), repairs, replacements, inspection and monitoring and improvements to operating conditions are therefore essential elements in the overall process.

As NPP SSCs accrue time under their specific operating conditions (temperature, load cycling, pressure, irradiation, chemical environment) it may be necessary, or make economic sense, to modify or re-evaluate their inspection schedules. In some cases, operating experience may have shown that a particular SSC was not so affected by a given ageing mechanism as was previously assumed in the design specifications. This may justify lessened inspection, or the depth thereof, thus allowing resources to be focused on more critical SSCs that could be showing degradation at rates in excess to those assumed at the design level. Risk-informed methods facilitate this approach. Credit for implementing improvements, such as water chemistry adjustment, neutron shielding and low-leakage core configuration and for substituting better-designed components featuring optimised materials (e.g. better alloys) and production methods (e.g. surface and heat treatments), must be given. Actions such as these must be scientifically validated (state of science and technology) to obtain regulatory approval.

3. HUMAN FACTORS AND THE IMPORTANCE OF LEARNING FROM EVENTS IN NPPs

There have been several instances where existing knowledge regarding the potential for a materials failure was not used adequately enough to check whether the same problems could occur in other NPPs. Human error may have aspects concerned with inadequate information, insufficient training, poor communications, and inadequate design, lapses of attention, wrong actions, wilfulness, and misperceptions of the signals and correspondingly wrong actions taken and mistaken priorities in attempting to gain control of a given situation.

Cognisance of weaknesses in a SSC and ways to avoid or mitigate their effects are essential to safety. Clearly, a priori, any known weaknesses will anyway be accounted for, and corresponding strategies will be in place to handle failures. The caveat lies in the fact that potential weaknesses or precursors for accidents may not be readily identified, and corresponding contingency plans are then not available. The following two examples are given to emphasize the importance of implementing actions based on available knowledge and lessons learned.

3.1. THINNING OF LOW ALLOY CARBON STEEL PIPING (FLOW ASSISTED CORROSION (FAC)/EROSION CORROSION (EC))

An accident with fatalities occurred in the USA Surry NPP, in Virginia, in 1986 due to low alloy carbon steel pipe wall thinning; a similar accident occurred in the Japanese Mihama Unit 3 NPP in August 2004. In the latter case, a turbine return pipe suddenly burst, scalding workers who were in the turbine hall, four of which died instantly. The pipe, some 27 years in service, was scheduled for inspection in November 2004, so the accident occurred just 3 months before the inspection. When new, the pipe walls measured 10 millimeters in thickness, but were found to be 1.4 millimeters thick when the accident occurred. This is an 86% loss in thickness and the pressure-temperature conditions correspondingly became too demanding for the remaining pipe material.

Problems associated with FAC-EC in this class of steel have been known for many years. Carbon steel piping in feed water, recirculation, extraction and heater drains is liable to be affected by FAC-EC (thinning of the alloy through loss of protective magnetite oxide layers due to flow of water or wet steam). It appears that the lessons learned from the Surry accident had been forgotten. More important, the difficulty is in where to monitor and measure. It is noted here that various proprietary tools and computer software packages are available to predict the wall-thinning rate, average rate of thinning since start-up, total loss in thickness to date and remaining life. Such on-line tools have been installed in many NPPs.

3.2. VESSEL CLOSURE HEAD DEGRADATION

Another case is the external corrosion of RPV closure heads from boric acid attack (wastage) due to leaking control rod drive mechanism (CRDM) nozzles caused by primary water stress corrosion cracking (PWSCC) of Alloy 600. The problem of wastage was discovered already in 1971 in a Swiss reactor, after a short time in service. In 2002, (31 years after) significant wastage was discovered in a USA reactor. In the latter case, the RPV's relatively thin (about 7 millimetres) stainless steel cladding was exposed locally, thus making it a pressure boundary. If this cladding had been breached, then a loss-of-coolant-accident (LOCA) would likely have occurred, although the emergency cooling and other measures were deemed adequate to limit the consequences. A lengthy outage of the Davis-Besse NPP resulted.

The problems due to wastage are being addressed through improved inspections and intervals thereof, monitoring, and in many case, through total replacement of the RPV closure head, featuring materials for the penetrations that are expected to be more resistant to PWSCC. The presence of insulating material is a hindrance to RPV head inspection, and this feature has thus created conditions where a developing problem has been hidden from view.

The unsuitability of Alloy 600 in some applications in PWRs, due to PWSCC, has been known for many years (steam generator tubing degradation), so it was to be expected that where it is present (e.g. CRDM penetrations) cracking may become a problem. Subtle factors such as product form and surface treatment (surface tensile stress) and heat treatment applied make the alloy variable in its resistance to PWSCC. In this case, the leaking CRDM penetrations were responsible for a far more serious secondary degradation of another component (RPV head).

4. DISCUSSION

Accidents do not really occur by accident, but rather they happen due to unforeseen and unfavourable situations and factors, and combinations thereof, that create conditions sufficient to overwhelm or by-pass safety features or to breach directly the integrity of critical SSCs. The precursors (errors) for accidents may be present in buying, commissioning, design and specifications, management (human), production, operation, inspection and repair actions. Avoidable and unavoidable errors are directly rooted in human actions.

Much research and operational experience has been accrued since the first commercial NPPs were connected to the supply grid. These first NPPs were, in essence, pioneer installations, built and operated to the best possible specifications and designs available at the time. Materials that had performed well in other industries (e.g. chemical) were natural candidates for NPP applications. Nevertheless, unexpected degradations have occurred under the prevailing NPP conditions. The interaction of SSC materials with the working environments and the actions (experience gained) of operators are time-dependent aspects.

Databanks have been set up and updated over the years as information on root causes of accidents, events or impacted safety has become available. Information has been exchanged between vendors, operators, regulators and research facilities. This, per se, is a necessary and good practice, but, nevertheless, new occurrences of old problems have still taken place. Excluding the case where a previously unknown degradation mechanism develops, it can be seen that the failure of humans to use existing knowledge correctly or to implement appropriate and timely strategies (inspections, monitoring, replacements etc.) can be an important factor where events or accidents are involved.

5. CONCLUSIONS

Safe operation is the sole responsibility of the NPP operators. Regulators ensure that operation rules and conditions (e.g. TS are complied with) and that state of technology principles are applied in all areas important to safety.

Design codes address as-new conditions, and they are not intended to cover assessment of NPPs that are in service. To address this, design codes feature conservative assumptions, based on engineering judgement and/or scientific knowledge and service experiences. Periodic assessment, inspection and monitoring of (aged) SSCs is therefore required to provide assurance that they are still capable of performing their foreseen design functions. Comprehensive ageing management and surveillance and inspection programmes are essential strategic tools to achieve reliable and safe operation.

Risk-informed approaches have the potential to better focus resources on those SSC requiring increased attention.

Human errors (insufficient number and depth of inspections, false responses to situations, failure to implement “lessons learned”, etc.) are often involved where accidents have occurred in NPPs. Levels of training, questioning attitudes and vigilance have to be maintained to the highest levels.

Safety levels can a priori be increased in designing SSCs that are intrinsically tolerant to operator and/or maintenance errors. Passive safety features should be implemented wherever they are technically possible and real gains in safety levels can be demonstrated/achieved.

Critical areas in SSCs should be designed to allow for ease of inspection/monitoring and, if required, easy replacement. These attributes will not only improve the reliability of inspection results, but will also favour radiological dose planning (i.e. as low as reasonably achievable (ALARA)) for involved personnel.

Various networks exist for sharing of experiences and lessons-learned (e.g. NPP owners groups, regulatory information exchange groups and the IAEA/OECD-NEA advanced incident reporting system (AIRS)). Nevertheless, such information sources must be fully analysed and knowledge regularly applied to raise questions on the possibility of the same events occurring in other plants. If information is available, it must be used correctly and efficiently. Accident and event databanks are only useful if their contents, especially the analyses of root causes, are intelligently applied to assess operating NPPs.

The “negative equity” impact of accidents involving human life, health and the environment, as well as the attendant public/political rejection of nuclear power are dominant factors surrounding nuclear power generation. The costs of accidents therefore cannot be measured solely in terms of money (NPP repairs, decontaminations, loss of energy production etc.).

The total monetary costs of an accident, depending on the severity, may be many orders of magnitude greater than the projected cost of a major SSC replacement, for example.

REGULATORY ASPECTS

(Session 4)

OVERVIEW OF REGULATORY ASPECTS OF AGING ISSUES OF NUCLEAR POWER PLANTS IN JAPAN

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Abstract. As of January 2005, approximately half of 53 units of BWRs and PWRs have been in commercial service more than 20 years, and 7 units over 30 years in Japan. The Nuclear Safety Commission and the Nuclear and Industrial Safety Agency are deeply concerned with various aging effects threatening safety assurance of the nuclear power plants, and are proactively working on the aging issues from the regulatory standpoints. This overview paper of regulatory aspects of the aging issues covers the following topics: Aging effects observed in some recent cases; Roles and responsibilities shared among relevant organizations; Topical regulatory activities as for revision of laws and regulations; Development of technical codes and standards; and Incorporation in the prioritized nuclear safety research program. Japan is aware that (1) Aging issues are of importance for safety assurance of NPPs on the background of increasing aged plants and aging effects observed in the recent events and cases, (2) Upgrading of regulatory activities is in progress in establishing legislation and conducting reviews and inspections to ensure operator’s appropriate aging countermeasures, and (3) One of the most and effective measures responding to aging issues is to share experiences and knowledge on aging problems among generations, organizations and countries. International engagement on the issues should be most encouraged.

1. INTRODUCTION

In 1970, the first BWR and the first PWR have started operation in Japan. As of January 2005, approximately half of 53 units of the nuclear power plants (NPPs) have been in commercial service more than 20 years, and seven units over 30 years. This time trend is shown in Figure 1, which indicates that Japan is due to have increasing number of aged reactors in the coming years.

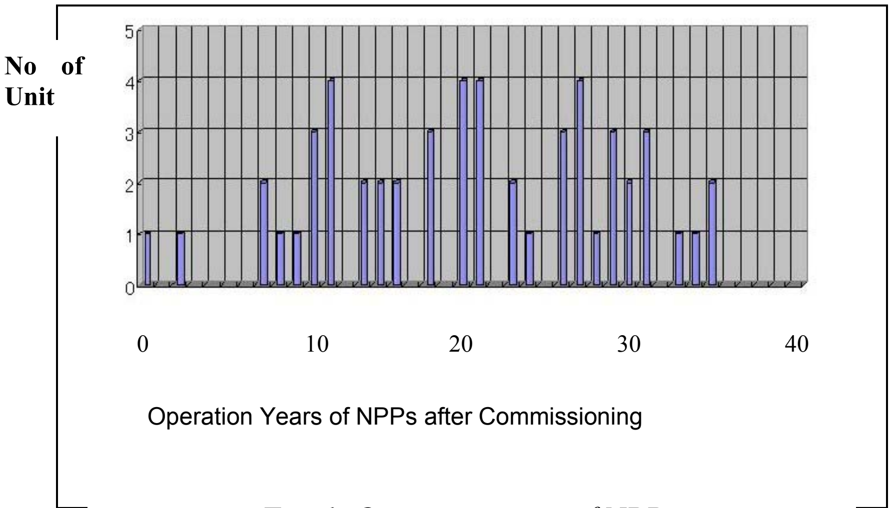

Fig. 1. Operation years of NPPs.

Several countermeasures have already been taken to cope with the aging situation by nuclear plant operators since the middle of 1990s, and generally speaking, the average performance of NPPs is stable. In the past several years, however, it was observed that various aging effects had formed basic and common reasons for some incidents and malfunctions threatening safety assurance. The Nuclear Safety Commission (NSC) and the Nuclear and Industrial Safety Agency (NISA) are deeply concerned with the aging issues from the regulatory perspective for ensuring nuclear safety and restoring public trust and confidence.

Mr. Matsuura, the chairperson of NSC, made a presentation on key safety issues related with NPP aging in the IAEA topical issue meeting in October 2004 (1). Incorporating the essentials in his presentation, this paper describes an overview of regulatory aspects of the aging issues of NPPs in Japan. In what follows, described are the aging effects observed in some recent cases, roles and responsibilities shared among relevant organizations, and topical regulatory activities regarding revision of laws and regulations, development of technical codes and standards, and incorporation in the nuclear safety research program prioritized by NSC.

2. AGING EFFECTS IN RECENT CASES

In the past several years, various aging effects have been observed in checkouts and inspections during maintenance outage and in incidents and malfunctions occurring during operation.

Examples of the aging events during 2001 to 2004 are:

(A) Due to Stress Corrosion

- Cracks in a lower ring of the core shroud (Fukushima No.2–3, BWR, 2001)
- Leakage from a CRDM housing at the bottom of the reactor pressure vessel (Hamoka-1, BWR, 2002)
- Cracks in piping of the reactor recirculation system (Onagawa-1, BWR, 2003)
- Leakage from a nozzle for a relief valve of the pressurizer (Tsuruga-2, PWR, 2003)
- Cracks in a nozzle for a CRDM housing at the head of the reactor pressure vessel (Ohi-3, PWR, 2004)

(B) Due to Fatigue

- Leakage from outlet piping of the regenerative heat exchanger shell (Tomari-2, PWR, 2003)

It could be found that the aging effects have formed basic and common reasons for the incidents and malfunctions threatening safety assurance from the view points of material/mechanical degradation, failures in sharing past technical experiences and knowledge, and lack of appropriate evolution of management and inspection systems. Hereafter discussed are the aging effects in the following three particular cases that made great impact on public trust and confidence in nuclear safety assurance:

- Case 1: Hamaoka-1 (BWR), “Pipe rupture in the residual heat removal system” (2001) [2]
- Case 2: Falsified recording of operator’s self-imposed inspections at TEPCO’s BWRs (disclosed in 2002) [3-4]
- Case 3: Mihama -3 (PWR), “Pipe rupture of the feed water line” (2004.8) [5-6]

Case 1: Hamaoka-1 (BWR), "Pipe rupture in the residual heat removal system" (2001) [2]

On November 7, 2001, a part of pipe in the steam condensation line of the residual heat removal system was ruptured during rated power operation at the Hamaoka Nuclear Power Station Unit-1 (BWR) of the Chubu Electric Power Company. The reactor was manually shut down immediately after the pipe rupture and there was no radioactive release into the environment. This incident was evaluated to be the level 1 of INES. The investigation concluded that the rupture had been caused by detonation of hydrogen gas accumulated in the ruptured piping section. No notable material degradation was observed in the piping. A quite amount of hydrogen gas was generated in the reactor core due to radiolysis of water.

While this incident is considered to have no implication of mechanical aging effects, NSC pointed out an organizational aging effect as its root cause: In the early stage of BWR development during 1960s to 70s, hydrogen explosion in discharging pipeline was a technical matter of careful concern. The past experiences and knowledge on hydrogen explosion, however, have faded away and the importance of the problem has not been continuously transferred and shared among reactor operators. Hence, the operator has paid no attention to possible hydrogen accumulation in that pipeline during operation.

Case 2: Falsified recording of operator's self-imposed inspections at TEPCO's BWRs (disclosed in 2002) [3–4]

At the end of August 2002, it was disclosed by a whistle blower that the Tokyo Electric Power Company (TEPCO) had concealed and falsified records related to the cracks in the core shrouds of BWRs detected during the operator's self-imposed inspections. The relevant regulatory authority, the Ministry of Economy, Trade and Industry (METI), carried out a very severe and detailed investigation, and identified that there were the following operator's problems at 9 reactors involving 16 cases as follows:

- Failed to observe the appropriate duties for technical standards. (6 cases)
- Reported inaccurate information or neglected to report to the government based on official notifications. (5 cases)
- Relied on inappropriate operator's self-imposed inspection procedure. (5 cases)

The cause of the cracks in the core shrouds was concluded to be the stress corrosion of stainless steel, which is a technical aging effect. In addition, NSC stressed that the root-cause of this case was connected to problems that reactor operator's complacency had increased during long periods of operations. Sense of compliance had gradually decreased, and also technical requirements by regulation had been neither clear nor practical enough for a long time.

Case 3: Mihama -3 (PWR), "Pipe rupture of feed water line" (2004.8) [5–6]

A part of the main feed water pipe in a turbine system was ruptured during rated power operation at the Mihama NPP-3 (PWR) of Kansai Electric Power Company (KEPCO). A quite amount of high-energy water flushed out from the pipe in the turbine building. This accident claimed five lives and injured other six workers on duty in a vicinity of the broken piping. The timing of the accident, just before a scheduled refueling and maintenance shutdown, was very unfortunate since many workers were in the turbine hall. Intensive investigations have been carried out by the regulatory authorities, both NISA and NSC.

It was revealed that the pipe rupture had been caused by erosion/corrosion on carbon-steel material at the downstream of an orifice-type flow meter in high energy water for a long time of operation. Decrease in the pipe wall thickness at the broken portion had never been inspected for 28 years after commissioning, because that portion was not listed in the inspection procedure document prepared by operator's subcontractors.

The operator had well recognized the aging effect on pipe wall thickness, but had continuously overlooked to check the thickness at the broken portion. The root cause of this misbehavior is a defect in operator's quality assurance activities including quality management of subcontractors.

3. RECENT REGULATORY ACTIVITIES FOR COUNTERMEASURES OF AGING ISSUES

3.1. BASIC ROLES OF ORGANIZATIONS AND REGULATORY INSPECTIONS AND REVIEWS

Basic roles of responsible organizations are illustrated in Figure 2 from the planning stage to the operation of NPPs. In the operating stage, a main regulatory role is played by NISA in METI for inspections and reviews of operator's activities. The Japan Nuclear Energy Safety Organization (JNES) established in October 2003 technically supports NISA. NISA reports their regulatory action results to NSC periodically or timely depending on the subjects. NSC reviews and confirms NISA's reports, and if necessary, gives comments and recommendations to them. Further, in order to monitor and audit the NISA's actions NSC conducts "The Subsequent Regulation Review" for the selected subjects that require detailed investigations including hearing from NISA and operators and on-site visits.

It is to note that the fundamental responsibility for maintaining safety of NPP shall be taken by the organization operating and managing the facility, in other words, the operator. On the other hand, from the viewpoint of ensuring the safety of the public, the regulatory organizations have the responsibility of establishing laws, rules and guidelines for safety assurance and continuously monitoring and confirming the operator's activities and measures for ensuring safety.

Figure 3 illustrates the current regulatory inspection and review process for the operating stage of NPPs. Operator's safety management activities are regulated with the Safety Preservation Rules (SPRs) that shall be approved by NISA. NISA conducts the Safety Preservation Inspection every quarter to check operator's compliance with SPRs.

The Periodic Inspection and the Periodic Safety Management Review shall be conducted by NISA and JNES at the time not over 13 months after passing the previous Periodic Inspection, which are focused on operator's maintenance activities of structures, systems and components (SSCs) of the facility. The Periodic Inspection is imposed on the safety-significant SSCs, for example, belonging to the reactor shutdown system, the reactor coolant pressure boundary, the residual heat removal system and the containment system.

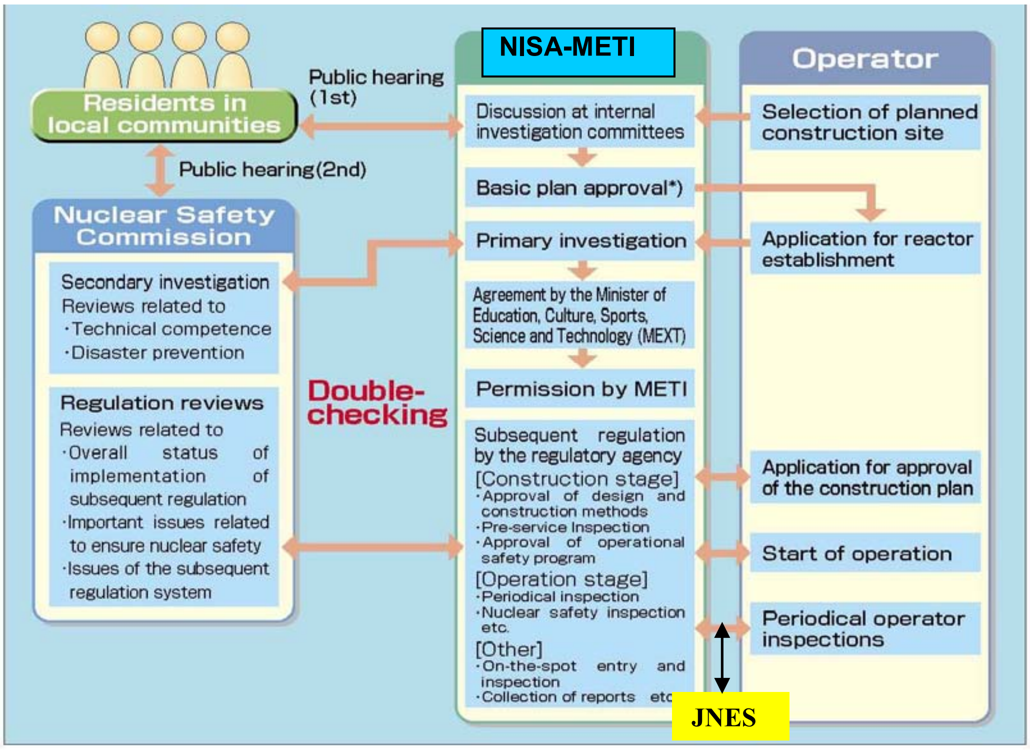

Fig. 2. Basic roles of Responsible Organizations from Planning Stage to Operation Stage of NPPs.

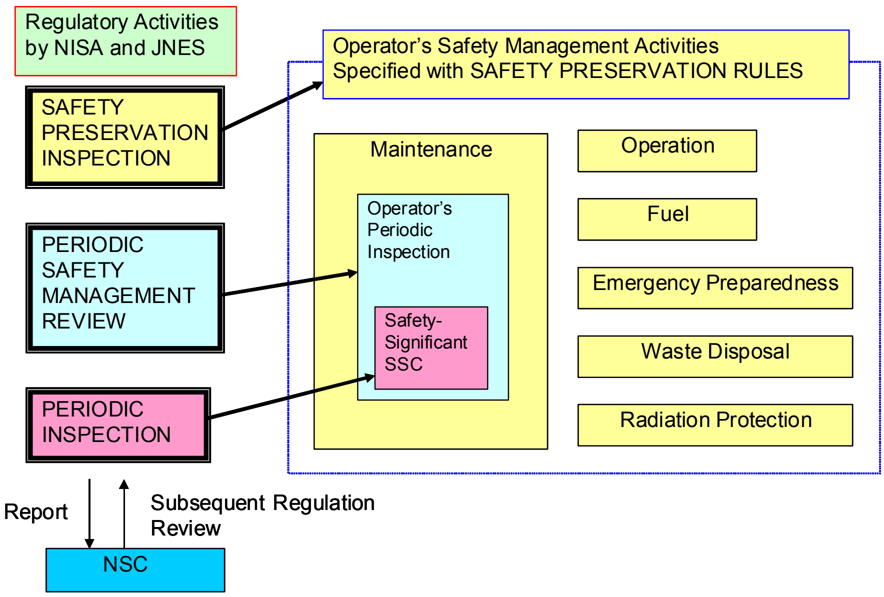

Fig. 3. Regulatory Inspections and Reviews for Operation Stage.

3.2. REVISION OF LAWS AND REGULATIONS FOR AGING COUNTERMEASURES

To intensify regulation for aging countermeasures, relevant laws were revised in 2002 and 2003 to oblige the following operator's activities that had been carried out voluntarily before:

- Operator's Periodic Inspection (OPI):

The operators shall carry out the OPI at the time not over 13 months after the end of the previous OPI. Implementation of the OPI is examined by JNES in the Periodic Safety Management Review in the oversight type approach. NISA evaluates the review result.

- Periodic Safety Review (PSR):

The operators shall carry out the PSR every ten years after commissioning. The PSR consists of three subjects:

  - Comprehensive evaluation of operating experience
  - Incorporation of latest technical information and knowledge
  - Aging management review

  In addition the probabilistic safety assessment is requested on an optional basis as before. NISA checks the operator's implementation status of the PSR in the Safety Preservation Inspection. To date the first PSR has been carried out for 34 units, and the second PSR for 2 units.

- Technical Evaluation on Plant Aging and 10-Year Program for Maintenance:

The operators shall carry out the Technical Evaluation on Plant Aging within 30 years after commissioning and every ten years after that. This evaluation covers the functional soundness of selected components and the maintenance management status of checkout/inspection, functional surveillance test, monitoring and repair/replacement of the portions to be evaluated. Based on the evaluation results, they shall develop a 10-year program for maintenance management. NISA and JNES check the operator's implementation status of this long-term maintenance program in the Periodic Inspection and the Safety Management Review.

To date the Technical Evaluation on Plant Aging has been carried out for 9 units. Figure 4 illustrates the time of the inspections and reviews for the aging countermeasures stated above.

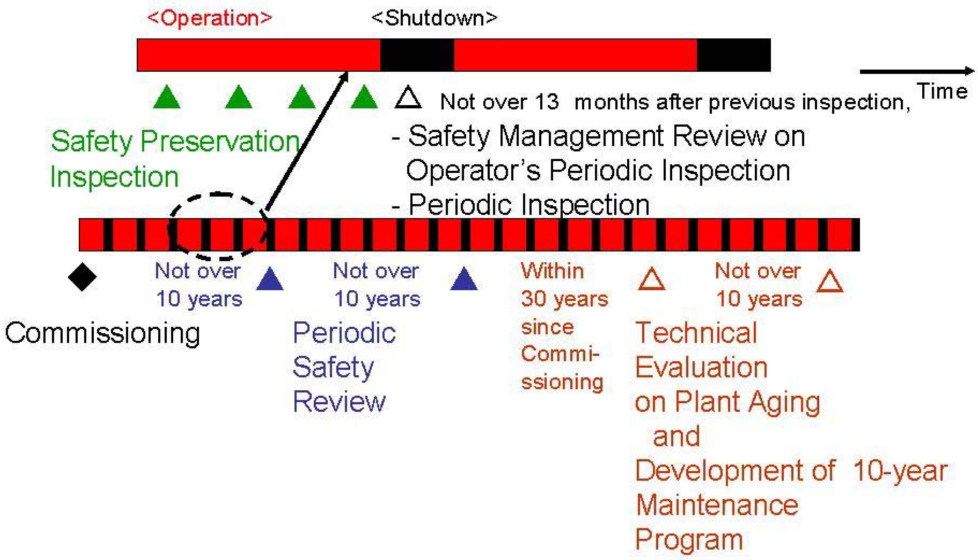

Fig. 4. Time of Inspections and Reviews for Aging Countermeasures.

3.3. PARTICIPATION WITH ACADEMIC SOCIETIES IN DEVELOPMENT OF RELEVANT TECHNICAL CODES AND STANDARDS

The following four technical codes and standards was developed or are under development for aging countermeasures by academic societies as a third party with technical experts: (The rules on Fitness-for-Service for NPPs, developed by the Japan Society of Mechanical Engineers (JSME) in 2002 and endorsed by NISA). (These rules are applied to the evaluation on component integrity to be carried out in the Operator's Periodic Inspection.)

- The standard of the Periodic Safety Review, being in progress by the Atomic Energy Society of Japan (AESJ)
- The standard of plant life management, being in progress by AESJ. (This standard will be applied to the Technical Evaluation on Plant Aging. The documents issued by IAEA regarding aging issues [7-8] etc. will be referred to.
- The pipe wall thinning management codes, being in progress by JSME (This standard will be applied to the pipe thinning management in the Operator's Periodic Inspection.)

NISA participates with AESJ and JSME in these developments, and will endorse them for regulatory application after detailed examination.

3.4 ADVISORY COMMITTEE ON AGING COUNTERMEASURES

A study committee on aging countermeasures started in METI last December. Its secretariat is placed in NISA. This committee was arranged to discuss and upgrade the aging countermeasures taking into consideration the lessons from the pipe rupture accident at Mihama-3. The committee members consist of technical experts and intellectuals from academia and societies, industry, local government, mass media and others. Major topics for study are:

- Adoption of latest domestic and overseas information and knowledge
- Clarification of criteria and guidelines for aging countermeasures
- Rational inspection approach taken by regulatory organizations.

The committee report will be issued by August 2005 and thereafter reflected to regulatory activities.

3.5. INCORPORATION IN NUCLEAR SAFETY RESEARCH PROGRAMME

NSC proposed a research program "Prioritized Plan for Nuclear Safety Research" in July 2004 for development and upgrade of safety concepts, guidelines, standards and criteria, and responses in an event of accidents and malfunctions. The following subjects of research and development (R&D) regarding aging issues are incorporated in the program:

- Understanding of various aging phenomena and those prediction methods (For example, the Irradiation Assisted Stress Corrosion Cracking (IASCC) is an important subject to be researched further.)

- Evaluation method of crack propagation and crack sizing technology

- Detection and measurement of crack and degradation of material

- Evaluation method of structural reliability

Those R&D subjects will be implemented by R&D organizations and universities in and after the 2005 fiscal year with developing detailed research plans.

4. CONCLUDING REMARKS

- Aging issues are of importance for safety assurance of NPPs in Japan on the background of increasing aged plants and aging effects observed in the recent events and cases.

- Upgrading of regulatory activities is in progress in establishing legislation and conducting reviews and inspections to ensure operator's appropriate aging countermeasures.

- One of the most and effective measures responding to aging issues is to transfer and share experiences and knowledge on aging problems among generations, organizations and countries. Hence, international engagement on the issues should be most encouraged.

ABBREVIATIONS

AESJ: The Atomic Energy Society of Japan

BWR: Boiling water reactor

INES: International nuclear event scale

JNES: The Japan Nuclear Energy Safety Organization

JSME: The Japan Society of Mechanical Engineers

KEPCO: Kansai Electric Power Company

METI: The Ministry of Economy, Trade and Industry

NISA: The Nuclear and Industrial Safety Agency

NPP: Nuclear power plant

NSC: The Nuclear Safety Commission

OPI: Operator's Periodic Inspection

PSR: Periodic Safety Review

PWR: Pressurized water reactor

R&D: Research and development

TEPCO: Tokyo Electric Power Company

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CANADIAN REGULATORY APPROACH TOWARDS AGEING MANAGEMENT PROGRAMS AND CRITICAL COMPONENT CONDITION MONITORING AND EVALUATION

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Abstract Effective ageing management programs and condition monitoring of key safety-related structures, systems, and components are an essential aspect for ensuring the long-term safety and reliability of nuclear power plants (NPP). This paper presents the Canadian Nuclear Safety Commission's (CNSC) approach towards ensuring that licensees operate and maintain their NPPs in a safe condition. It describes the processes and requirements in place that ensure prompt notification is given to the regulator following the discovery of previously unconsidered ageing phenomena through in-service and periodic inspections or through in-service failures. The paper goes on to describe the regulatory response towards these events, which involves requiring the licensee to investigate the cause of the failure, re-assess the safety of the facility, adjust the controls on plant operation and surveillance, and implement measures to monitor the condition of the structure, system, or component (SSC). The paper also briefly discusses the known degradation mechanisms of key CANDU SSCs and describes the requirements in place to ensure licensees sufficiently monitor the condition of these SSCs and appropriately disposition the results of these inspections. Finally, the paper describes the current and planned initiatives to improve the Canadian regulatory requirements for ageing management programs, as well as the oversight for the surveillance of critical NPP SSCs, including the use of probabilistic tools for condition evaluations and in-service inspections, and discusses the need to increase these efforts in order to account for the increasing effects of degradation mechanisms as Canada's power reactors approach their end of design life.

1. INTRODUCTION

CANDU NPPs have been supplying electricity to the Ontario power grid since 1962 and to New Brunswick and Quebec power grids since 1983. At present, there are 15 operational CANDU units in Ontario, representing 40% of the installed generating capacity, and one unit each in New Brunswick and Quebec. Canadian CANDU NPPs had an excellent safety and reliability track record in their early (typically 12) years of operation but performance at some plants declined due to cumulative effects of ageing degradation.

The early high level performance of CANDU NPPs resulted from solid design and construction, efficient operation by well trained and experienced staff, expert technical support as well as effective regulation by the AECB (now CNSC). Ageing of SSCs was being addressed proactively through a number of programs, such as a 'durability program' for major SSCs initiated at NPD nuclear generating station in 1970's and the Nuclear Plant Life Assurance program of the Nuclear Generation Division of Ontario Hydro initiated in 1980's.

This paper deals with ageing management of Canadian CANDU NPPs from the regulatory perspective. The paper also reviews main safety related ageing concerns and mitigation strategies for key SSCs. It then describes Canadian regulatory approach to ageing management and licensees' ageing management programs. Finally, the paper indicates a path forward involving a proactive ageing management approach.

2. CANADIAN REGULATORY APPROACH TO AGEING MANAGEMENT

In Canada, the CNSC has been the driver for dealing with many of the ageing concerns discussed below. In response to early signs of NPP ageing, CNSC staff implemented a "regulation-by-feedback" process (Fig. 1). This process ensured that when component degradation was discovered, either through inspection results or component in-service failures, licensees investigated the degradation, assessed its safety impact, and adjusted controls to mitigate further degradation. Subsequent inspections verified the adequacy of the mitigating measures. This process was applied on a case-by-case basis, as new degradation mechanisms were identified.

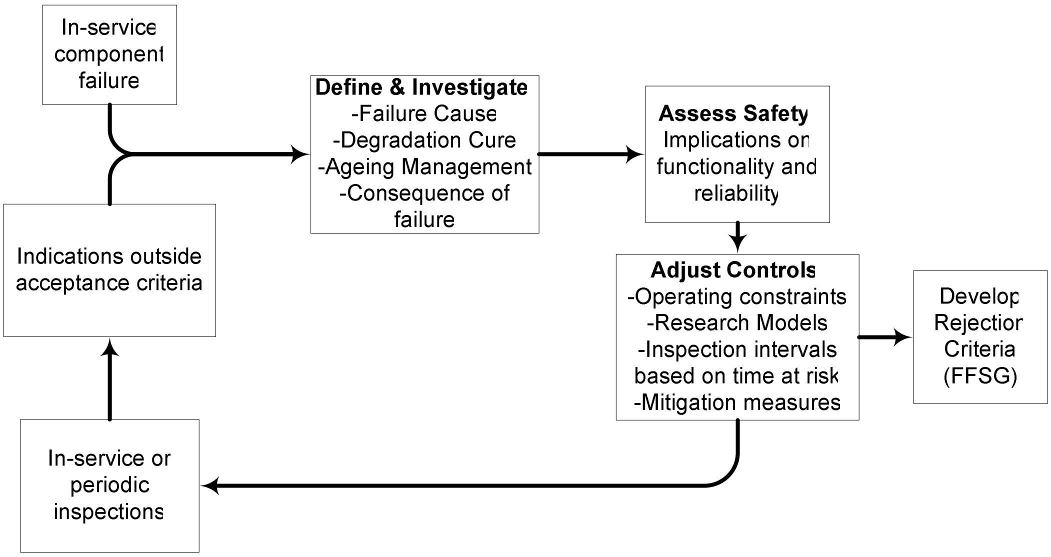

Fig. 1. Regulation by feedback process for managing component degradation.

In general, new forms of material degradation have been discovered through in-service inspections, and occasionally through in-service failures, such as pressure tube delayed-hydride cracking (DHC) or feeder stress-corrosion cracking (SCC). Through operating licences, the CNSC requires licensees to comply with in-service inspection standards, which provide extensive inspection requirements for nuclear safety related systems. In-service inspections have served to identify, at an early stage, degradation of safety-critical SSCs. Cracking on the extrados of feeder piping, discussed in section 4, is one such degradation mechanism that was identified through in-service inspections.

Recently, staff identified the need to further augment inspection requirements for high-energy non-nuclear safety important systems. Failure of these systems would not have significant radiological consequences, and therefore had not been included in nuclear in-service inspection standard; however these systems have the potential to affect conventional worker health & safety. CNSC staff are now evaluating the available means to incorporate additional inspection requirements in order to ensure that licensees are effectively monitoring the condition of high-energy conventional SSCs.

Through NPP operating licences, licensees are required to comply to Regulatory Standard S-991, which describes extensive reporting requirements for events at NPPs. In addition, through requirements for in-service inspections, licensees must report all in-service inspection indications that do not meet defined acceptance criteria. These reports allow the CNSC to remain abreast of the overall plant condition at our licensees' sites.

Having identified a previously unknown material degradation, staff require licensees to investigate extensively the cause of the failure, and to assess the implications of this failure on overall plant safety and on the existing safety case. Similar systems, which may be subject to this form of degradation, must also be inspected. Both staff and licensees use the information from these investigations to make a risk-informed decision whether to continue operation of the plant. This information is also shared with other operators to ensure that they also remain abreast of recent developments in reactor ageing.

Licensees are required to examine and implement measures to reduce the likelihood of further failures. In general, these measures may include increased surveillance, operating constraints such as reduced channel power limits, research projects and mitigating measures such as chemistry control. The approval to restart is granted only when the CNSC staff are satisfied that the licensee has a clear understanding of the causes of the degradation and that appropriate measures to mitigate further failures have been implemented.

Taking into account the information gained from the studies described above, rejection criteria for future inspection indications are also developed. These criteria are specified in Fitness-for-Service-Guidelines (FFSG). FFSGs include the maximum indication size for flaws based on the predicted inspection interval. This maximum size is based on the predicted growth rate of the flaw and ensures that the flaw will not propagate to failure prior to its next inspection.

The knowledge gained through these studies is used in the development of ageing models and modelling methodologies to predict component lifetimes. The operating histories of failed components also aided in determining the projected service life of similar components. As new failures occurred, it was recognized that a more preventive approach towards component ageing was also needed.

At the end of the 1980’s, CNSC recognized that although NPP licensees had programs in place related to ageing, they had not yet adequately integrated them into a comprehensive and systematic ageing management strategy. As a result, in 1990, CNSC Staff raised a Generic Action Item “Assurance of Continuing Nuclear Station Safety”, which required licensees to demonstrate that:

- potentially detrimental changes in the plant condition are being identified and dealt with before challenging the defense-in-depth philosophy;
- ageing related programs are being effectively integrated to result in a disciplined overall review of safety;
- steady state and dynamic analyses are, and will remain, valid;
- a review of component degradation mechanisms is being conducted;
- reliability assessments remain valid in light of operating experience; and
- planned maintenance programs are adequate to ensure the safe operation of the plant.

The scope of work and of the associated ageing management programs covered SSCs important to safety, including such systems as: special safety and safety-related systems; SSCs whose failure could prevent a safety-related or a special safety system from fulfilling its function or cause a safety system actuation; SSCs used in emergency operating procedures; and SSCs relied upon for protection from fire and seismic events.

In response to the above Generic Action Item, licensees' submissions listed and described a number of ageing related surveillance and maintenance activities that they were carrying out to ensure continuing nuclear safety. However, licensees did not demonstrate having a systematic and integrated approach to ageing management. CNSC staff, therefore, recommended that the licensees use the International Atomic Energy Agency (IAEA) guideline "Implementation and Review of a Nuclear Power Plant Ageing Management Programme" 2 as an appropriate framework for such a program.

Licensees have since developed or modified existing programs based on these guidelines that cover ageing management of the above systems. In 2003, letters were sent to each power reactor licensee informing them of CNSC staff's decision to close the GAI on the basis of the submissions provided and to monitor licensee program performance through the CNSC's ongoing compliance program.

The CNSC has not issued explicit regulatory requirements on ageing management. However, a number of age-related regulatory requirements are included in the following regulatory documents: Class I Nuclear Facilities Regulations3 (requiring licensees to describe "the proposed measures, policies, methods and procedures for operating and maintaining the nuclear facility"); R-7, Requirements for Containment Systems for CANDU Nuclear Power Plants4, R-8, Requirements for Shutdown Systems for CANDU Nuclear Power Plants5, and R-9, Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants6 (requiring that safety systems are available to operate when called upon); regulatory standard S-98, Reliability Programs for Nuclear Power Plants7 (requiring development of system availability limits and minimum functional requirements, and description of the inspection, monitoring, and testing activities designed to ensure system availability); specific conditions of an NPP operating license.

In order to address ageing, the licensees are required to inspect and perform material surveillance according to the technical requirements of CSA standards N285.48 (Periodic inspection of CANDU nuclear power plant components), N285.59 (Periodic inspection of CANDU nuclear power plant containment components), and N287.710 (In-service examination and testing requirements for concrete containment structures for CANDU nuclear power plants). These requirements include inspection techniques, procedures, frequency of inspection, evaluation of inspection results, disposition, and repair.

Maintenance programs are required for the purpose of limiting the risks related to the failure or unavailability of any significant SSC. For so-called "destiny components" (pressure tubes, feeder piping, and steam generator tubes), in addition to the standards' minimum requirements, the CNSC requires NPP licensees to develop fitness-for-service guidelines and life cycle management plans/programs.

3. LICENSEES' AGEING MANAGEMENT PROGRAMS

Canadian licensees have come a long way towards implementing systematic ageing management programs. They have always had in place many programs and activities that contribute to ageing management of NPPs, such as maintenance programs, in-service inspection, surveillance, testing and monitoring programs, equipment qualification, chemistry programs, feedback of operating experience and R&D results, and procurement programs.

As a result of the Generic Action Item on Assurance of Continuing Nuclear Station Safety, the Canadian licensees established systematic ageing management programs through the coordination and integration of the above programs and activities. The specific processes and procedures developed varied from licensee to licensee, although in general, they undertook efforts to identify gaps in their operating policies and procedures with regards to the ageing management of critical SSCs in accordance with IAEA guidance.11

Programs were developed that considered the known degradation mechanisms of the selected components. Licensees also considered operating experience to ensure that all mechanisms that had previously caused failures were addressed. The programs already in place to deal with known degradation mechanisms were evaluated to determine their effectiveness.

Coincident with the above activities, licensees developed, on their own or in conjunction with the plant designer, generic procedures for evaluating component and system ageing. Along with these, condition assessments of the major plant components were and are being performed. These assessments evaluated the feasibility, from a safety standpoint, of continued use of the components.

CNSC staff recognize that the current level of licensees' ageing management effort may need to be further augmented in order to ensure plant safety as Canadian NPPs continue to age. This will require strengthening the role of proactive ageing management utilizing a systematic ageing management process. Section 5 describes some initiatives that the CNSC is undertaking to address this concern.

4. AGEING CONCERNS IN CANDU NUCLEAR POWER PLANTS

This section presents in Table 1 an SSC-oriented summary of main ageing concerns that challenge CANDU plant safety as it ages; some of these concerns are unique to CANDU and some are applicable to nuclear plants in general.

Table 1. Summary of Ageing Concerns in CANDU Power Plants

SSC	Degradation Mechanisms & Effects	Safety Concern	Regulatory Requirements	Mitigation Strategies
Pressure tube (PT)	Irradiation-enhanced deformation of PT (sag, axial creep, diametral creep & wall thinning), DHC, material property changes	Failure of PT, (small LOCA), inadequate fuel cooling	N285.4-95 FFSGs	Design/material/manufacturing improvements (replacement PTs), chemistry control, improved leak detection, trip set-point reductions, inspection
Calandria tube (CT)	Irradiation-enhanced deformation of CT: sag	Impairment of SDS 2 (LISS nozzles)	PROL License Condition 3.5 CSA N285.4	Monitor CT-nozzle interference, reposition nozzle, replace FC
Feeder pipe	Wall thinning due to Flow Accelerated Corrosion, Stress Corrosion Cracking, Low-T Creep Cracking	Failure of feeder pipes (small LOCA), primary coolant leakage	CSA N285.4 FFSGs, Life cycle mgmt plan	Chemistry control, addition of chemical inhibitors, repair/replace, inspection
Steam generator tube	Corrosion (SCC, IGA, pitting, wastage), fretting, denting, erosion	Tube leaking or rupture, possible releases	CSA N285.4 OP&P Limits	Inspection and tube plugging. Chemistry control, water-lancing and secondary side chemical cleaning, installing additional bar supports to reduce vibration
PVC cable	Radiation and temperature-induced embrittlement	Insulation failure leading to current leaks and short circuits	R-7. R-8, R-9, L.C. 7.1	Develop effective EQ programs, procedural controls, test plans, visual inspection
Containment structure	Thermal cycling, periodic pressurizing, fabricating defects, stress relaxation, corrosion, embrittlement	Loss of leak tightness, structural integrity leading to possible releases	CSA N287 series, CSA N285.5, R-7	Pressure testing, visual inspections, concrete coating
Reactor assembly	Corrosion (SCC), erosion, fatigue, creep, embrittlement	Loss of moderator containment, shielding	CSA N285.4	Visual inspection, leak monitoring, lifetime predictions
Battery	Oxidation of grids and top conductors	Loss of power to essential systems	L.C. 4, CSA N286 series	Maintenance, operating experience trending, new battery designs
Orifice	Flow erosion, material deposition	Loss of monitoring capabilities, consequent loss of control	R-8	Condition monitoring, alternative flow measurements

A distinct feature of the CANDU reactor is that heat generation occurs in 380–480 horizontal Zr-Nb pressure tubes (PTs), rather than a large pressure vessel (Fig. 2). There are three significant pressure tube degradation mechanisms: irradiation-enhanced deformation (axial creep, sag, radial creep and wall thinning), delayed hydride cracking and irradiation-induced changes in PT material properties. Operating experience and ongoing R&D have enhanced substantially their understanding and predictability.12–16 Related safety concerns include the following potential problems: CT contact with reactivity mechanisms and associated impairment of their function; PT rupture and associated impairment of fuel cooling; reduced margins to demonstrate LBB and increased risk of PT rupture.

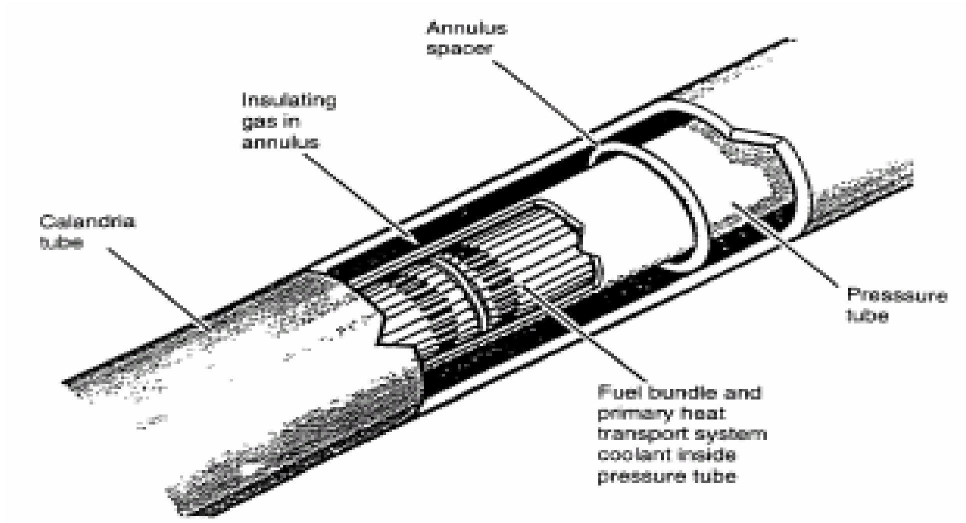

Fig. 2. Schematic of a CANDU fuel channel.

The cornerstone of the strategy for mitigating degradation of PTs is monitoring and characterization which is mandatory under Power Reactor Operating Licenses. CSA Standard N285.4-948 establishes minimum requirements for PT inspections, including the scope and schedule for various inspections (inaugural, periodic and material surveillance), as well as acceptance criteria for inspection findings (or "indications"); it defines specific inspections to detect and characterize indications resulting from each of the above PT degradation mechanisms. Should there be an indication that does not meet the acceptance criteria, the licensee is obliged to follow CNSC-approved "fitness-for-service guidelines" (FFSG) to demonstrate that the PT remains fit for continued service; there are three possible courses of action: returning the tube to service (usually with certain restrictions); repairing the indication; or replacing the PT.

Current pressure tube FFSGs allow for the use of probabilistic methods for PT condition monitoring, operational assessments, and inspections. In order to make use of these methods, licensees must demonstrate that all existing regulatory requirements are met, that the principles of defense-in-depth are maintained, that sufficient safety margins are ensured, and that the proposed increase in risk and cumulative effects are small and do not exceed CNSC safety goals. The current uses of these tools include: risk-informed inspection scoping (inspection size and frequency determination), statistically based estimation of parameters for fitness assessments (i.e. a percentile of a cumulated distribution of measured properties), and as an alternative for DHC and hydride blister predictions (although this option has not yet been explored).

In addition, licensees have recently proposed an update to the existing PT FFSGs. A part of these FFSGs include a requirement for a core assessment, for which the CNSC has been a driver. Under this requirement, licensees are required to assess the cumulative effects of PT degradation mechanisms on the integrity of pressure tubes throughout the entire core. This involves:

- evaluating the adequacy of material fracture toughness;
- evaluating the identified degradation mechanisms for the balance of pressure tubes that were not inspected;
- assessing leak-before-cracking in cases where the hydrogen equivalent concentration is greater than the terminal solid solubility for hydrogen dissolution (TSSD) at sustained hot condition; and,
- assessing the change of PT properties from surveillance information, including hydrogen equivalent concentration, fracture toughness, DHC growth rate, and threshold intensity factor for onset of DHC from a crack.

Probabilistic methods for these evaluations, i.e. analysis methods that determine the distributed output of engineering analyses based on probabilistic representations of distributed input parameters, are an important and powerful tool. Prior to making full use of these tools, however, CNSC staff foresee the need for further refining these methods, including the completion and/or further verification of probabilistic representations of distributed input parameters, and further justification of the proposed probability acceptance levels for the results of a probabilistic analysis.

Feeder piping, made from seamless cold drawn carbon steel, is used to supply fuel coolant to individual pressure tubes (Fig.3). There are two significant ageing mechanisms of feeder pipes: excessive wall thinning and cracking. Excessive wall thinning is due to Flow Accelerated Corrsoion (FAC) and cracking is speculated to be caused by Stress Corrosion Cracking (SCC) and creep cracking. These mechanisms have been prominent in outlet feeders, which are subject to harsher operating conditions than inlet feeders. The mechanistic understanding of FAC wall thinning is commonly taken to mean corrosion caused by the flow accelerated dissolution of magnetite (Fe_3O_4) on the inside surface of the outlet feeder pipe 17. The rate of feeder pipe wall thinning depends on water chemistry, particularly the coolant pH and on flow characteristics such as velocity and turbulence. Axial cracking has been observed in outlet feeder pipes at both the inside surface and the outside surface of the feeder pipe bend downstream of the PT connection. One through-wall circumferential crack was also discovered at the inside surface at a repaired field weld on the feeder. All of the cracks in the feeder pipe show intergranular paths.

The understanding about the cracking mechanism is still in the speculation stage. The cracking initiated at the inside surface is believed to be a Intergranular Stress Corrosion Cracking (IGSCC) due to an oxidizing water environment. The most likely mechanism of the outside surface cracking is argued to be a hydrogen assisted creep cracking. High residual stresses in feeder bends (which vary from reactor to reactor due to different manufacturing techniques) accompanied by the chemical environment produces favourable conditions for this cracking. The related safety concern is a potential pressure boundary failure and leakage of reactor primary coolant, if FAC and cracking are allowed to progress unchecked.

The strategy for mitigating wall thinning of feeder pipes due to FAC consists of chemistry control (reduced pH) and corrosion inhibitors to reduce the rate of degradation, inspection and

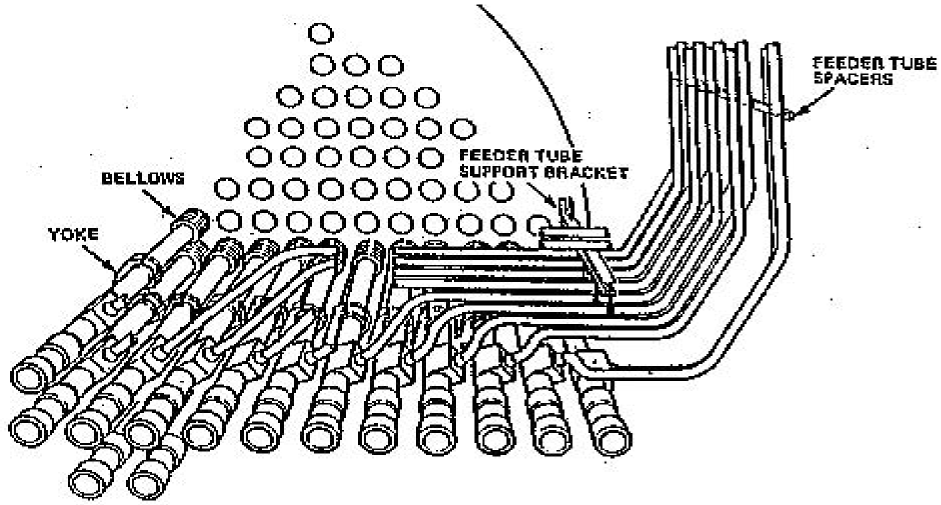

Fig. 3. Schematic showing typical feeder

monitoring to detect cracking and wall thinning of feeders, and repair/replacement of feeder pipes when the wall thickness is bellow an acceptable limit. Following the discovery of FAC thinning, CNSC staff required licensees to develop improved techniques, methodology, and accuracy for feeder thickness measurements and thinning rate assessments; the CNSC has approved licensee developed Feeder Piping Life Cycle Management Plan and Fitness-for-Service Guidelines. CNSC has also approved a methodology for determining the minimum required feeder thickness based on ASME Sections III and XI. CNSC reviews the inspection results and wall thinning assessments at each planned Outage and grants approval of reactor operation based on the predicted thickness of the feeder piping at the next planned inspection and the minimum required thickness. Pipe sections with confirmed cracks must be removed or replaced until an effective methodology to determine the acceptability of cracks in feeders is developed. For the stations where cracking has been discovered, CNSC has required the licensee to expand their inspection scope to cover all the high-risk sites such as tight radius bends and repaired field welds. CNSC has also requested the licensee to replace feeders which had been dispositioned due to wall thinning, to ensure sufficient safety margin in the event of a crack initiating and propagating at the thinned location of the feeder. CNSC has also asked licensees to improve non-destructive examination (NDE) for detecting feeder cracks which can be difficult to detect due to their characteristics such as a scalloped surface, secondary cracking, multiple surface cracks and discontinuities of cracks causing ultrasound reflections.

Licensees have made efforts to improve leak-detection systems, which provide early detection of leaking cracks. In addition, R&D effort administered by CANDU Owner's Group (COG) aims to develop more effective ageing management for feeder pipes. Recently COG performed a feasibility study of the application of probabilistic methods to feeder cracking. The feasibility study examined whether a probabilistic model for feeder cracking and feeder rupture can be developed, while ensuring that the frequency of feeder failure remains sufficiently low so as not to affect the reactor safety case. The study concluded that it is possible to develop such a model for estimating the probability of feeder cracking and to support life-cycle management decisions. CNSC staff feel that research to increase the level of mechanistic understanding is needed to reduce model uncertainty due to the limited understanding about feeder degradation mechanisms. COG has also begun research efforts to demonstrate that the feeder cracking failure mechanism would be leak-before-break (LBB). CNSC staff considers this effort to ensure the defence-in-depth safety concept rather than to disposition detected crack.

CANDU steam generators (SGs) are similar in construction to PWR steam generators and suffer from similar ageing degradation mechanisms and effects, such as corrosion (SCC, IGA, pitting, wastage), fretting, denting, and erosion of SG tubes.18-19 Comparable regulatory controls and ageing management actions are being used in Canada, however to date, no SGs have been replaced. The SG FFSG, accepted in 1999, make use of probabilistic tools for performing LBB simulations for the determination of flaw stability within the entire tube population and for determining flaw size distributions for condition monitoring and operational assessments. The SG FFSG considers the probabilistic approach as an alternative to deterministic methods. Shortcomings of current probabilistic methods and SG FFSG include:

- the use of “pattern-based” variables, which do not make use of mechanistic or physics of degradation tools;

- the lack of a clear basis for probabilities and distribution parameters;

- the tools are primarily reactive, rather than proactive; and,

- FFSG does not address inspection scoping.

Most of the other ageing concerns of Table I are generic nuclear plant problems. For example, the use of PVC-insulated cables inside reactor buildings has been a major ageing concern for all NPPs, including CANDU reactors. Similar regulatory controls and ageing management actions are being used in Canada, including cable replacement in connection with NPP upgrading/refurbishment 20.

5. PATH FORWARD

CNSC staff recognizes that the current level of ageing management effort may need to be increased to ensure plant safety as Canada’s NPPs continue to age. Due to the fact that the Canadian regulatory process to address ageing evolved on a case-by-case basis, the current regulatory approach is reactive rather than proactive, and lacks consistency by focussing on individual cases. CNSC staff are implementing measures to strengthen the role of proactive ageing management by focusing on important SSCs susceptible to ageing degradation and greater application of the systematic ageing management process utilizing Deming’s Plan-Do-Check-Act cycle2.

Proactive ageing management means being in control of SSC ageing while, in contrast, reactive ageing management means using a run-to-failure strategy. Proactive ageing management also involves providing for adequate understanding and predictability of SSC ageing, minimizing premature ageing (that is caused by errors in design, installation, operation, maintenance, inadequate communication between design, technical support, operations and maintenance functions, and unforeseen ageing phenomena), adjusting the use of proactive and reactive ageing management strategies based on existing understanding and predictability of SSC ageing, and continuous improvement of SSC specific ageing management programs.

To strengthen the role of proactive ageing management at Canadian NPPs, CNSC will continue maintaining and improving regulatory documents, standards and compliance program activities, and encourage further research on ageing degradation of SSCs important to safety, as needed.

Currently, effective regulatory oversight of licensees’ ageing management programs is hampered by the lack of explicit regulatory requirements on ageing management. Without common benchmarks it is difficult to ensure consistency and uniformity of compliance assessments of ageing management programs at different licensee sites. As a result, CNSC staff have undertaken the production of a regulatory standard outlining the regulatory requirements for NPP licensees' ageing management programs.

The objectives of this regulatory standard are to:

- describe the organizational characteristics of an effective ageing management program;
- describe the general attributes of an effective ageing management program for managing specific ageing mechanisms and their effects on particular SSCs or types of SSCs;
- inform NPP licensees of CNSC expectations and recommendations relating to ageing management of SSCs important to safety; and to,
- facilitate CNSC evaluations of the effectiveness of NPP ageing management programs within the framework of CNSC's compliance program.

CNSC staff are planning for the regulatory standard to include, as part of licensees' overall ageing management programs, such requirements as:

- Plant Reviews: involving a systematic review of the plant to identify all the SSCs which must be addressed by the program and the potentially detrimental effects of ageing on the ability of each of the SSCs to meet their design requirements;
- Gap Analyses: involving an assessment of the adequacy and effectiveness of existing activities already in place to manage each SSC's ageing, and to identify enhancements or additions to these activities; and,
- Documentation: including the governing ageing management programs procedures and the requirements for continuous improvement of the program, as well as the procedures for all supporting programs and activities.

The regulatory standard is intended for both CNSC staff and NPP licensees: NPP licensees will use the document as a benchmark for self-assessments of their ageing management programs, and CNSC staff will use the document as a regulatory basis for ongoing compliance inspection of ageing management programs and for comprehensive licensing assessments of licensee long-term operation applications. CNSC staff expect that this standard will result in an increased effectiveness of licensee ageing management programs and an increased reliability of SSCs important to safety, thus improving NPP safety.

The CNSC also foresees the need to further develop and improve probabilistic tools for condition assessments and condition monitoring of critical SSCs. Some specific uses of these tools are described in section 4, and will result in a more risk-informed approach towards managing the ageing of Canadian NPPs. In addition, the CNSC is moving towards the use of process-based approvals (PBA) for dispositions of certain well-understood ageing phenomena. PBAs will allow licensees to self-disposition low-risk inspection indications provided the disposition is performed in accordance with accepted FFSGs and with an approved and regularly audited procedure. CNSC staff foresee that an increased use of PBAs will result in improved regulatory effectiveness and efficiency, while reinforcing the CNSC's policy that licensees bear primary responsibility for ensuring the safe operation of NPPs.

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IMPACT OF REACTOR PRESSURE VESSEL DEGRADATION

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Abstract. This paper discusses the impact of the reactor pressure vessel (RPV) degradation that was discovered at the Davis-Besse Nuclear Power Station in March 2002. On February 16, 2002, the Davis-Besse Nuclear Power Station near Oak Harbor, Ohio, began a refueling outage that included a special inspection of the Alloy 600 nozzles that enter the closure head of the low alloy carbon steel reactor pressure vessel (RPV). The inspection focused on the nozzles associated with the mechanism that drives the control rods, known as the control rod drive mechanism (CRDM). This inspection was consistent with the plant's commitments made in response to United States Nuclear Regulatory Commission Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," which was issued on August 3, 2001. In conducting this inspection, it was discovered that three CRDM nozzles had indications of axial cracking (one with circumferential cracking also), which had resulted in leakage of the RPV's pressure boundary. Specifically, these indications were in CRDM nozzles 1, 2, and 3, which are located near the center of the RPV head. Significant degradation of the RPV's head was found adjacent to CRDM nozzle 3 and the plant entered an extended repair shutdown.

1. INTRODUCTION

Davis-Besse is a raised loop pressurized water reactor (PWR), manufactured by Babcock and Wilcox. The reactor licensed thermal power output is 2772 megawatts. The architectural engineer and constructor was Bechtel Power Corporation. The owner operator of the facility is the FirstEnergy Nuclear Operating Company (FENOC). The facility is located near Oak Harbor, Ohio on the western shores of one of the Great Lakes- Lake Erie.

The plant began commercial operation in August, 1978 and is currently licensed to operate until April 2017. The reactor pressure vessel has an operating pressure of 2155 psig (151.50 kg/square cm) and a design pressure of 2500 psig (175.75 kg/square cm). Davis-Besse had accumulated 15.8 effective full power years (EFPY) of operation when the plant shut down for its thirteenth refueling outage on February 16, 2002. During that refueling outage, while performing reactor pressure (RPV) vessel closure head inspections required by the United States Nuclear Regulatory Commission, workers discovered a large cavity in the 6 inch (15.24 cm) thick low-alloy carbon steel RPV head material. The cavity was about 6.6 inches (16.67 cm) long and 4 to 5inches (10.16 to 12.70 cm) at the widest point extending down to the 0.25 inch (0.635 cm) thick Type 308 stainless steel cladding.

FENOC promptly commissioned a root cause team to evaluate what had caused the corrosion. After initially considering a repair of the RPV head, FENOC purchased an unused head of the same design. The facility was out of service for over 2 years while the head was replaced and other wide-scale evaluations and improvements were made to the physical plant, programs, and staffing organization.

2. ROOT CAUSE FINDINGS

The technical root cause determined that the corrosion was the result of boric acid interaction with the RPV head's low alloy carbon steel (SA-533 Grade B). The source of the boric acid was via a through wall crack in a control rod drive mechanism Alloy 600 nozzle (Figure 1) and Alloy 82/182 J-groove welds. This crack was initiated as a result of Primary Water Stress

Corrosion Cracking. The Davis-Besse organization had believed that the boron accumulated on the RPV head was due to leaking control rod drive mechanism flanges above the RPV head and that such accumulation would not cause corrosion due to the elevated temperatures at that location.

This boron accumulation on the top of the RPV head was not fully removed during refueling outages and it masked the typical "popcorn" boron indication from a nozzle containing a crack (see Figure 2). Accordingly, the boric acid emitted through the nozzle crack was allowed to corrode the carbon steel head creating a cavity. Following the discovery of the cavity and the removal of the head, a disk containing the cavity was cut out and sent to a laboratory for examination of the degradation in detail. A complete copy of the technical root cause report and the degradation examination report are available on the Nuclear Regulatory Commission's web-based Agency-wide Documents Access and Management System (ADAMS) at http://www.nrc.gov.

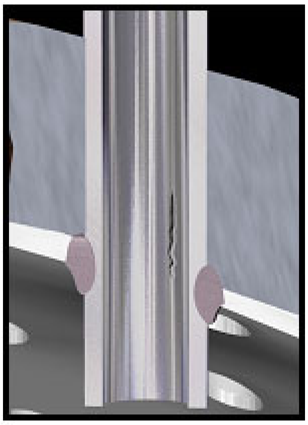

Fig. 1. Depiction of Through Wall.

Fig. 2. Typical Appearance of Leakage Due to Nozzle Crack through Wall Cracking.

The management organizational root cause identified four primary causes: a less than adequate nuclear safety focus (a production focus), Corrective Action Program implementation shortcomings (addressing symptoms rather than causes), ineffective use of industry information in addressing plant conditions (including RPV aging issues), and Boric Acid Program implementation weaknesses.

3. RESTART OVERSIGHT PROCESS

As a result of the degradation of the RPV head, Davis-Besse was placed under the Nuclear Regulatory Commission's Inspection Manual Chapter (IMC) 0350 process for restart. The purpose of the IMC 0350 process is:

- To establish a criteria for the oversight of licensees performance for licensees that are in a shutdown conditions as a result of significant performance problems or operational event (s).
- To ensure that when a plant is in a shutdown condition as a result of performance problems, the NRC communicates a unified and consistent position in a clear and predictable manner to the licensee, public, and other stakeholders.
- To establish a record of the major regulatory and licensee actions taken and technical issues resolved leading to approval for restart and to the eventual return of the plant to the routine Reactor Oversight Process (ROP)

- To verify that licensee corrective actions are sufficient prior to restart.
- To provide assurance that following restart the plant will be operated in a manner that provides adequate protection of the public health and safety.
- The Nuclear Regulatory Commission staff established a special IMC 0350 oversight panel to coordinate the Commission's activities in assessing the performance problems associated with the corrosion damage to the RPV head at Davis-Besse, monitoring corrective actions, and evaluating the readiness of the plant to resume operations.

In addition, the Nuclear Regulatory Commission's actions included the following:

- Issuance of a Confirmatory Action Letter and subsequent revisions
- Addressing the Advisory Committee on Reactor Safeguards interest
- Establishing a Lessons-Learned Task Force
- Developing and issuing generic correspondence to the industry
- Issuance of NRC Confirmatory Order for the restart of Davis-Besse
- Performing follow-up plant inspections and assessments

On March 18, 2002, the Nuclear Regulatory Commission issued Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity". On August 9, 2002, the Nuclear Regulatory Commission issued Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs". Orders were issued on February 11, 2003, and February 20, 2004, to pressurized water reactor-type nuclear power plants establishing interim inspection requirement for RPV heads.

4. EFFECTS

As a result of the degradation of the RPV head and the extended outage of two years, there were numerous effects on FENOC. These included:

- Loss of plant generation/replacement cost
- Extensive inspection of plant and materiel
- Plant materiel improvements (Reactor head replacement, Containment Sump, High
- Pressure Injection Pumps)
- Improvements in management, programs, and processes
- Impact on balance of FENOC fleet through the resources and
- Demands on employees (extended hours)
- Reduction of community support
- Loss of respect within the nuclear industry
- Loss of the regulator's confidence
- Impact on Corporate finances
- Congressional hearings, Department of Justice investigations, and negative publicity

Effects of the degradation of the RPV head on the nuclear industry included:

- Added outage costs due to increased inspections
- Revisiting Containment Sump/Debris issue
- Accelerated RPV head replacements
- Increased Nuclear Regulatory Commission review of licensing action requests
- Increased public interest
- Diversion of Nuclear Regulatory Commission resources
- Addressing Lessons-Learned from Davis-Besse

In addition, the Nuclear Regulatory Commission was affected by the degradation of the RPV head. These effects included:

- Issuance of a Confirmatory Action Letter and subsequent revisions
- Implementation of the IMC 0350 Process
- Establishment of an IMC 0350 Oversight Panel
- Increased inspections at Davis-Besse
- Public Meetings
- Advisory Committee on Reactor Safeguards Interest
- Congressional requests and hearings
- Office of Inspector General investigations
- Office of Investigation/Department of Justice investigations
- Establishment of a Lessons-Learned Task Force
- Generic correspondence to Industry (NRC Bulletin 2002-01, 2002-02, 2003-2, and Orders)
- Responding to petition Documents from public
- Issuance of NRC Confirmatory Order for Restart
- Follow-up plant inspections and assessments

5. CONCLUSION

The extended repair outage at Davis-Besse was costly and lasted over 2 years. Today, the plant's materiel condition has been refurbished and is operated with a focus on SAFETY, including the materiel condition of the plant. Davis-Besse has operated at approximately 100% power from April 2004–January 2005 and successfully completed a Mid-Cycle Steam Generator Inspection Outage in February 2005. The Nuclear Regulatory Commission IMC 0350 process continues in effect. As a result of the Commission's Confirmatory Order for Restart, Davis-Besse is required to perform independent assessments for the next five years (2004–2008) in the areas of Operations Performance, Corrective Action Program Implementation, Engineering Program Effectiveness, and Organizational Safety Culture, including Safety Conscious Work Environment. In addition, Davis-Besse has procured a new RPV head with Alloy 690 nozzles for installation in future.

REGULATORY ASPECTS OF AGING MANAGEMENT IN ARGENTINA

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Abstract. The nuclear power plant Atucha I is in operation from 1974. It is a power station of the type PHWR with a power of 357 MW and with a temperature of operation of 262°C. The nucleus this formed at the present time by 252 fuels channels. The fuels are of 37 bars, one structural and 36 fuels bars of natural uranium with 0,85% enrichment and of a longitude of 6,18 meters. This nucleus this inside a vessel of a steel 22NiMoCr37 and built with badges of different cast. For the characteristics of their design the top of the reactor cannot be retired without a high radiological and economic cost. This takes to that took the decision of placing the samples of the surveillance program in the bottom part of the fuels channels. Initially they were placed two set of test of samples that they should be extracted to 50% and the end of the life of the power plant. This situation takes to design a program of irradiations in experimental reactors, we irradiates material in SCK-CEN in Belgium, with an acceleration of 700 times, showing the material a similar behaviour to the VAK. We irradiates material in Lovissa, Finland, with a factor of advance of 24 times if being necessary it will be irradiated in the reactor RA1, in Argentina. Everything with the purpose of knows the current state of the recipient. In this work this irradiations are analysed, being compared the results and the importance of this knowledge was commented for the design of the surveillance programme of Atucha II.

1. MATERIALS AND EXPERIMENTAL PROGRAMME

In the panorama of the operative nuclear power stations in the world, the CNA-1 are only in their type, a PHWR with a RPV of 12 m of height, 5 m of diameter and 22 cm of thickness, made of a steel 22 NiMoCr 3 7 (similar to the ASTM A508-64 class 2) the composition of the steels is observed in the Table I. The temperature of operation of the wall of the RPV is of 265°C and the pressure of 121 atm. From the 2001, it operates with a fuel cycle of slightly enriched uranium (0.85% 235U).

Table 1. Composition of steel (wt %)

	C	Mn	**P**	**S**	Si	**Ni**	Cr	Mo	Cu
MB41.1	0.22	0.75	**0.016**	**0.010**	0.26	**0.82**	0.40	0.65	**0.14**
MB31.3	0.19	0.69	**0.009**	**0.012**	0.21	**0.81**	0.41	0.63	**0.12**
JF	0.18	1.35	**0.007**	**0.005**	0.27	**0.76**	0.51	0.12	**0.04**
HSST03	0.20	1.26	**0.011**	**0.018**	0.25	**0.56**	0.45	0.10	**0.12**

Over decades the CNA-1 RPV Surveillance Programme has monitored the irradiation embrittlement of the CNA-1 RPV material. The programme started in April 1974, during commissioning of the NPP. It was designed as the most complete research program based on the scientific knowledge existing at that time. Two sets of 15 capsules with surveillance material were installed in the reactor: SET-1 and SET-2 contained ~256 steel specimens, taken from archive material corresponding to the base metal of the upper and lower rings for the RPV wall and to the weld and heat affected zones.

Two materials from the RPV steel were selected to build the “Welding Coupon”: base material BM 41.1 and BM 31.3. From the point of view of radiation damage, base material BM 41.1 is the crucial CNA-1 RPV material, with a higher content of impurities and located at the lower circumferential ring of the CNA-1 RPV, exposed to a higher fast flux.

The design peculiar of the power station doesn't allow irradiations in near quick positions to the beltline of the RPV. For that reason, the irradiations of the program of surveillance were carried out in the only accessible position that is at the bottom of the coolant channels, under the fuels elements (see Figure 1).

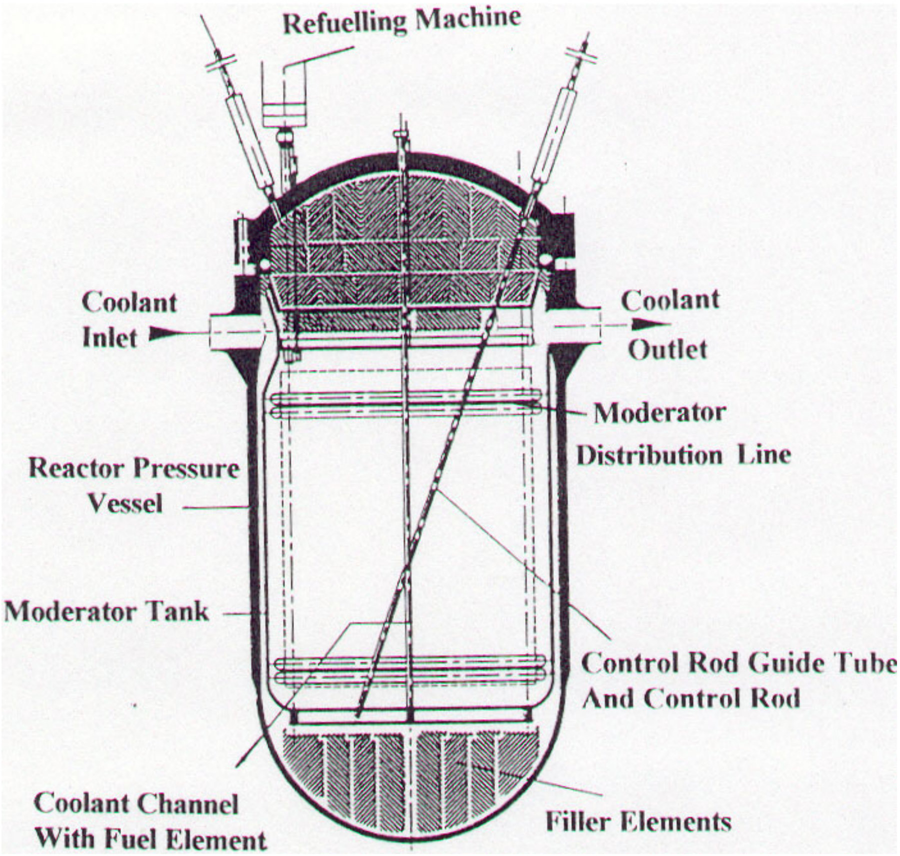

Fig. 1. Atucha I, RPV.

In this position the temperature is similar to that of the internal wall of the vessel, but the neutron spectrum is extremely different to that of the wall of the vessel. The capsules of SET-1 accumulated 6 years of irradiation and those of the SET-2 13 years of irradiation.

Tested the SET-1 values of the transition temperature shift were obtained were non-compatible with the operation of the power station [1, 2]. If the German norm KTA 3203 is applied [3, 4], for a similar steel to that of RPV of Atucha is found that the temperature transition shift at the end of the life is of 78°C and an accumulated fluencia of 1.28×10^{19} n cm^{-2} (see Figure 2).

Usually, the NPP RPV embrittlement is based on the prediction of international trend curves and if possible on the analysis of experimental values obtained within the surveillance program. As we will see in what follows, trend lines cannot be directly applied to the CNA-1 RPV case. The trend lines were developed for NPPs with RPVs that operate at higher temperatures, in the range of 272–299°C. The operating temperature of the CNA-1 RPV is 265°C, lower than that of typical PWRs.

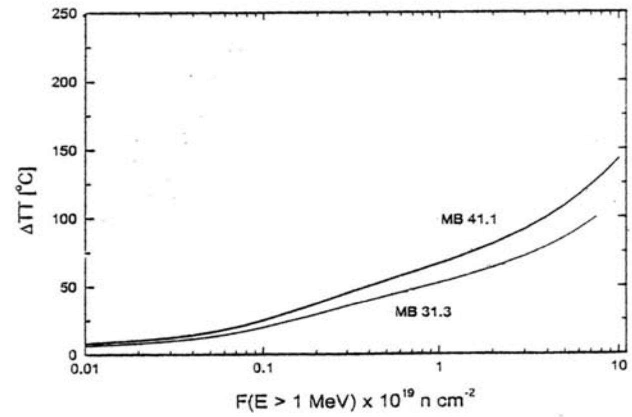

Fig. 2. German norm KTA 3203 applied to the RPV steel.

If these data are compared with those obtained in the SET-1 it is observed that the ΔTT (°C) is much bigger to the one predicted, Figure 4. This takes to the realization of new experiment in the VAK experimental reactor, the VAK is a German reactor, BWR of 15 MW. In this reactor two capsules were irradiated with materials from RPV of Atucha and of the reference material JF. These irradiations were carried out respectively during 50 and 140 days. This corresponds to a quick irradiation with a fast fluxes of 2.09×10^{12} n cm^{-2} s^{-1} and lead factor LF ~ 180 times [5]. The Charpy result are observed in the Figure 4 compared with the tendencies, showing a very good agreement and considering the composition of the MB 31.3 their behavior is also better than the MB 41.1.

At the same time was instaled in Atucha I the SET 3 with JF reference materials. These data obtained by the VAK for the MB of the RPV agree well with those obtained in the program of IAEA CRP-II for the steel HSST03 which adjust well with a law of powers ΔTT (°C) = 34 $F^{0.55}$, where F is the fast fluence expressed in units of 10^{19}n cm^{-2} [7]. On the other hand the points obtained for the SET-1 respond to an own law different to that of the VAK. In the year 1993 was tested the SET-2 [8], confirming the results obtained in the SET-1, if this group of results is placed in oneself graph, like is observed in the Figure 3 and Figure 4.

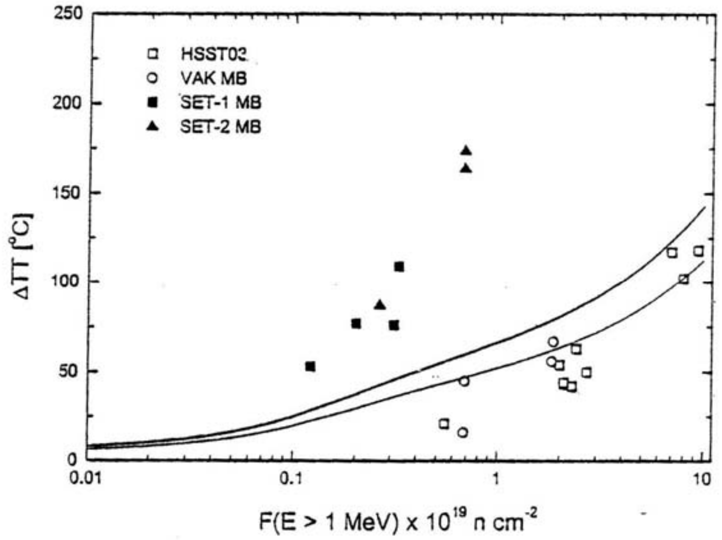

Fig. 3. German norm KTA 3203 applied to the RPV steel. SET1, SET2 and VAK for Atucha materials.

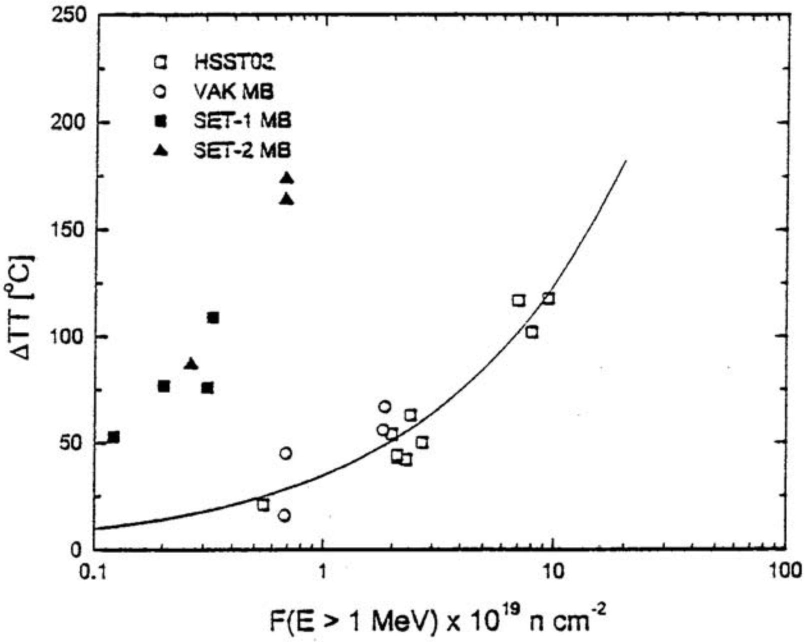

Fig. 4. ΔTT (°C) = 34.7 $F^{0.55}$.SET1, SET2 and VAK for Atucha materials.

This same it happens for the material JF irradiated in the VAK with a law of the type ΔTT (°C) = 23.6 $F^{0.47}$.(see Figure 5). To confirm these differences in 1998 the samples of the reference material JF of SET3 were tested [8].

They had a completely different behaviour (see Figure 6), what confirms that the irradiation conditions were completely different and they didn't allow exptrapolar to the wall of the recipient. The cause of this difference is attributable to the thermal component of the spectrum. The relationship between thermal and fast neutrons [$\Phi_{thermal} / \Phi_{fast}$] for the different irradiations performed are:

For the beltline position of Atucha-I RPV = 11

For the specimen position of Atucha-I surveillance program = 949

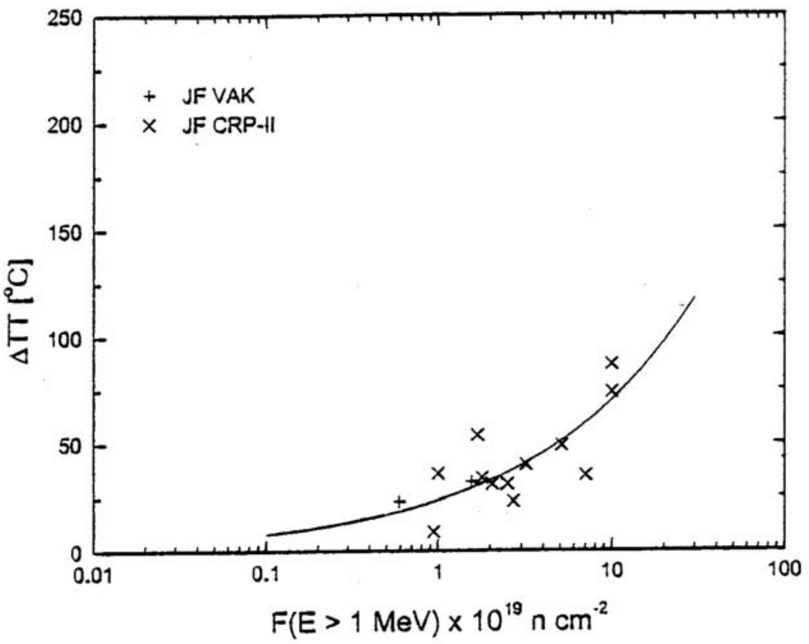

Fig. 5. JF materials. ΔTT (°C) = 23.6 $F^{0.47}$. CRP-II and VAK.

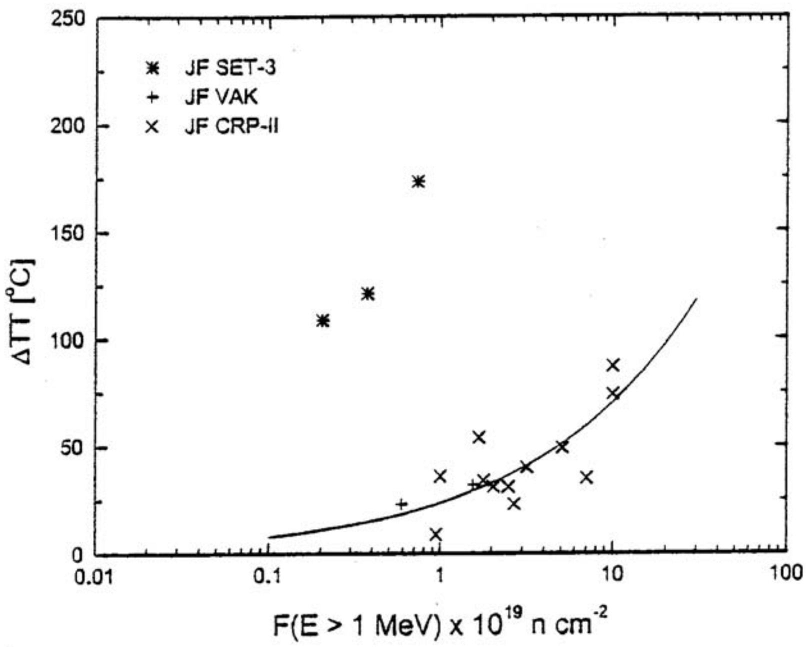

Fig. 6. JF materials. ΔTT (°C) = 23.6 $F^{0.47}$. CRP-II ,VAK and SET3.

For the irradiation position at VAK reactor = 4

Where $\Phi_{thermal}$ corresponds to the flux of neutrons with energies E < 0.4 eV and Φ_{fast} to the flux of neutrons with energies E > 1 MeV. The VAK irradiation results are questioned due to the presence of a possible dose-rate effect, which in turns depends on the end fluence. The VAK fast neutron flux was about two orders of magnitude higher than that of the CNA-1 RPV, the embrittlement obtained in the experiment could be too low, i.e. not representative of the real embrittlement of the CNA-1 RPV wall.

Regulations usually recommend the use of a lead factor LF ~ 3-5 for surveillance capsules, where LF is defined as the ratio between fast fluxes at the surveillance position and at the RPV wall. The results of VAK are questioned because of the associated LF ~ 180. After all of the uncertainties which have been presented, there was consensus about the necessity to make a new irradiation using both base materials (31.3 and 41.1) as well as the weld material, trying to reproduce the conditions of neutron spectrum at the beltline of Atucha-I RPV. This irradiation could not be made at Atucha-I NPP, because even if we could simulate the neutron spectrum of the beltline by shielding the capsules, the small lead factor that would be obtained at the bottom of the fuel channels would require many years to get useful values. To know the state of degradation of the recipient indeed they were designed a group of irradiations for the reference material and for the Atucha I materials, in chronological order the carried out irradiations and programmed, those carried out are those of CNAI, VAK, BR2, that of Loviisa still was not carried out the results of the test samples and that of RA1 is conditioned by these last results. In the Figure 7 it is observed in comparative form the characteristics of the spectrum of reactors [6].

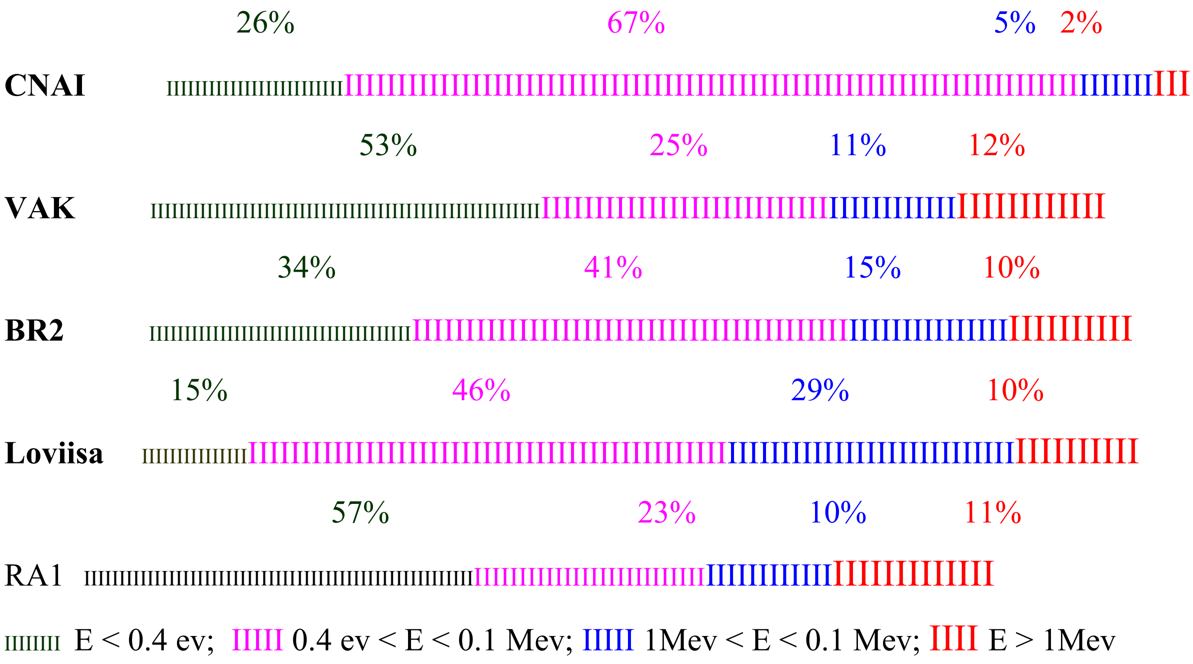

Fig. 7. Percent contribution of 4 main energy groups to the total flux.

2. COMPLEMENTARY IRRADIATION IN HOST REACTOR BR2

Within the framework of the Belgo-Argentinian Bilateral Cooperation, the Belgian Nuclear Research Centre (SCK•CEN) and the National Atomic Energy Commission (CNEA), a collaboration agreement was focuses on a new irradiation program, aimed at confirming past results, extending our surveillance database. The irradiations, carried out at SCK•CEN in the BR2 reactor, included a dosimetry program and accurate irradiation temperature control.

The BR2 irradiation experiment will be conducted under a higher lead factor LF ~ 800. Neutron damage at both facilities, VAK and BR2, do not significantly differ from that found in a typical PWR. VAK and BR2 embrittlement results should be more representative of the CNA-1 RPV wall. Neutrons with E > 0.1 MeV produce 95% of the damage.

The task is to perform the irradiation and evaluation of Atucha-I base material. In addition the JF material was selected as supplementary reference material. The JF material is a modern steel that is almost unaffected by neutron irradiation [9]. The data obtained from the irradiated JF material are very useful. They can be used to confirm the results obtained in the past and to assess the effect of flux differences between the different reactors. The JF material has been irradiated in Atucha-I at surveillance positions in SET-3 and in a complementary irradiation that KWU carried out at the VAK reactor in Germany in 1983. It should be noted that the specimens used in the BR2 experiment were extracted from the same coupon that was used for SET-3 and at VAK.

DPA was used to correlate the damage produced among specimens irradiated under different neutron environments. DPA is an integral parameter that takes in account the neutron spectrum and it is this capability that allows us to use it as exposure parameter for the comparison of different types of irradiation facilities. In the case of the surveillance positions, 83% of the total damage is produced by thermal neutrons. The dpa spectrum at the beltline position, is completely different, 82% of the total dpa comes from neutrons with E > 0.1 MeV. At VAK, most of the damage (~97%) originates at neutron energies above 0.1 MeV. The same happens at BR2. Neutron damage at both facilities, VAK and BR2, do not significantly differ from that found in a typical PWR. VAK and BR2 embrittlement results should be more representative of the CNA-1 RPV wall [6] (see Figure 8).

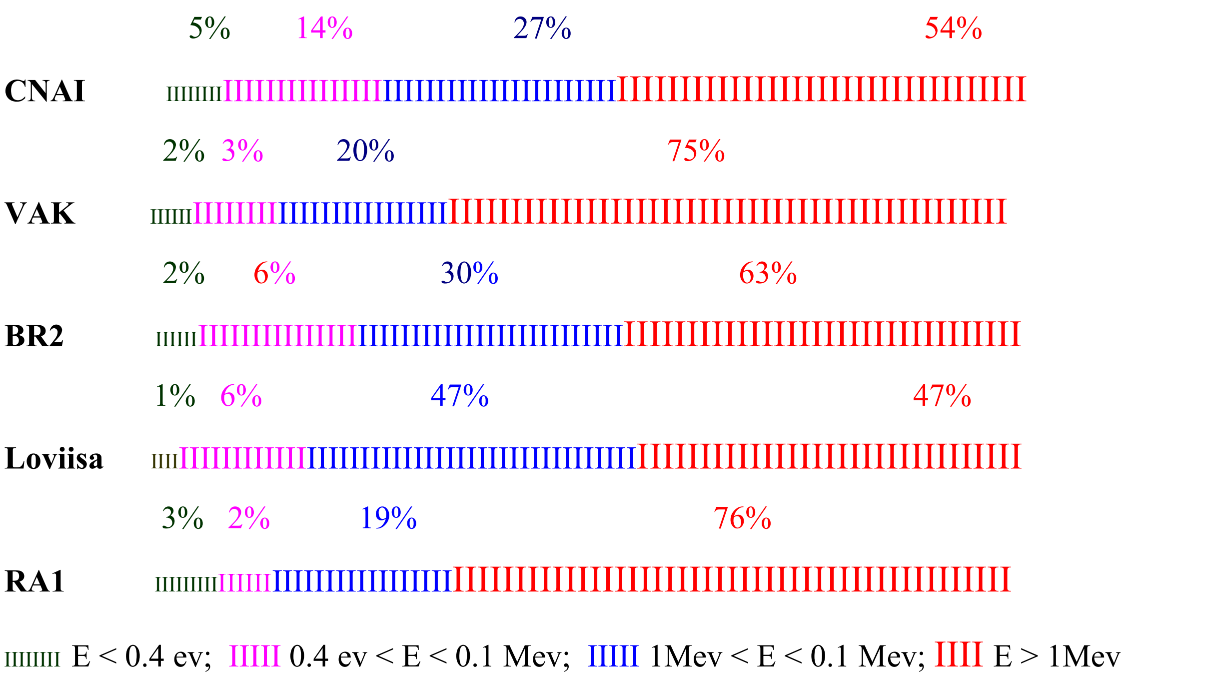

Fig. 8. DPA spectrum at the inner surface of the CNA-1 RPV wall, as compared to that found at VAK, BR2, Loviisa and RA1 irradiation position.

The irradiation had to be carried out at 265°C, a temperature similar to the CNA-1 RPV operating temperature. Both BR2 results obtained at a fluence of 1.97×10^{19}n cm^{-2} and VAK results obtained at a fluence value of 1.8×10^{19}n cm^{-2} overlap for the Atucha materials and for the JF materials [10].

3. CONCLUSIONS

The integrity of the CNA-1 RPV beyond the designed lifetime has to be demonstrated. Usually surveillance data provides all the elements needed to assess the RPV response to neutron irradiation. The atypical irradiation environment prevailing at the CNA-1 surveillance positions introduced severe difficulties in the assessment of the CNA-1 RPV steel embrittlement. Standard regulations fail to predict the radiation damage accumulated by the specimens irradiated at CNA-1 surveillance position. A high thermal component, which is not present at the RPV wall, has to be taken into account when evaluating the damage of the samples, when spectral differences are small, DPA can be used as exposure parameter to correlate embrittlement results. This is the case of the change in spectrum inside the CNA-1 RPV wall and is also that of transferring the results of complementary irradiations carried out at VAK and BR2 to the CNA-1 RPV beltline. It is not the case for the results obtained at the surveillance position.

To be able to make a correct prediction of the RPV embrittlement, irradiations have to be representative of the actual RPV beltline conditions. In general, irradiations performed at research facilities are accelerated irradiations, and the question is raised whether dose-rate effects affect or not these results. The BR2 experiment provides results at fast neutron flux $\varphi_f \sim 9.71 \times 10^{19}$n cm^{-2} s^{-1} and high acceleration factor (LF ~ 800). As expected, BR2 results match those obtained at VAK with an acceleration factor (LF ~180) and fast neutron flux $\varphi_f \sim 1.54 \times 10^{12}$ n cm^{-2} s^{-1}. BR2 and VAK experimental results give the same steel irradiation response independent of the neutron flux. The results of new mechanical tests that will be obtained at Loviisa with lower acceleration factor (LF~25) and fast neutron flux ($\varphi_f \sim 3.27$ 10^{11} n cm^{-2} s^{-1}) will provide additional insight into this matter.

Mitigation of irradiation damage to the CNA-1 vessel should be viewed as one of the major issues within CNA-1 RPV Life Management. One of the measures already in use worldwide, consist of operations reducing the flux at the RPV, so that the material properties remain under the regulatory limits for the extended operating time. The first actions in this direction were already taken at CNA-1 in 2003, when some peripheral fuel channels were removed leading to a flux reduction.

The existing CNA-1 Ex-Vessel Dosimetry Program was able to monitor the effect. Other very important issues, like in-service inspections, non-destructive testing, pressurized thermal shock conditions PTS, etc., come into consideration when qualifying the RPV for future operation. All this accumulated experience is of supreme importance since the completion of construction and start-up of the paralyzed 745 MWe Atucha-II project, also a PHWR type with a RPV made of JF steel, is now scheduled for 2009.

For the elaboration of the surveillance programme of the RPV, given the characteristics of the Atucha II, similar to Atucha I, we could simulate de neutron spectrum of the beltline by shielding the capsules and place that at the bottom of the fuel channels.

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SPECIAL SESSION

(Session 5)

INTEGRATED SURVEILLANCE SPECIMEN PROGRAM FOR WWER-1000/V-320 REACTOR PRESSURE VESSELS

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Abstract. Surveillance specimen programs play an important role in reactor pressure vessel lifetimeassessment as they should monitor changes in pressure vessel materials mainly their irradiation embrittlement. Standard surveillance programs in WWER-1000/V-320 reactor pressure vessels have some deficiencies resulting from their design – nonuniformity of neutron field and even within individual specimen sets, large gradient in neutron flux between specimens and containers, lack of neutron monitors in most of containers and no suitable temperature monitors. Moreover, location of surveillance specimens does not assure similar conditions as the beltline region of reactor pressure vessels. Thus, Modified surveillance program for WWER-1000/V-320C type reactors was designed and realized in two units of NPP Temelin, Czech Republic. In this program, large flat type containers are located on inner wall of reactor pressure vessel in the beltline region that assures their practically identical irradiation conditions with critical vessel materials. These containers with inner dimensions of 210×300 mm have two layers of specimens; using inserts (10×10×14 mm) instead of fully Charpy size specimens allows irradiation of materials from several pressure vessels at once in one container. Irradiation of these archive materials together with the IAEA reference steel JRQ (of ASTM A 533-B type) and reference steel VVER-1000 will allow to compare irradiation embrittlement of these materials and to obtain more reliable and objective results as no reliable predictive formulae exist up to no due to a higher content of nickel in welds. Irradiation of specimens from cladding region will help in the evaluation of resistance of pressure vessels against PTS regimes.

1. INTRODUCTION

Reactor pressure vessels (RPV) are components with the highest importance for the reactor safety and operation as they contain practically whole inventory of fission material but they are damaged/aged during their operation by an intensive reactor radiation. Surveillance specimen programs are the best method for monitoring changes in mechanical properties of reactor pressure vessel materials if they are designed and operated in such a way that they are located in conditions close to those of the vessels. Reactor Codes and standards usually included requirements and conditions for such programs to assure proper vessel monitoring [2, 3, 4].

WWER reactor pressure vessels are designed according to former Russian Codes and rules with somewhat different requirements using different materials comparing e.g. with ASME Code. Standard surveillance programs in WWER-1000/V-320 reactor pressure vessels have some deficiencies resulting from their design – nonuniformity of neutron field and even within individual specimen sets, large gradient in neutron flux between specimens and containers, lack of neutron monitors in most of containers and no suitable temperature monitors. Moreover, location of surveillance specimens does not assure similar conditions as the beltline region of reactor pressure vessels.

Prediction of radiation damage/embrittlement in weld metals of these type of vessels has been put into great interest when first results from Standard surveillance programs (SSP) were obtained – it looks that some of these weld metals showed higher irradiation embrittlement than was predicted with the use of the standard [1]. One of the reasons could be a fact that weld metals in most of these vessels contain higher content of nickel as it was tested within the Qualification tests of this vessel material – 15Kh2NMFA(A). In these tests nickel content was lower than 1.5 mass % but later Technical specification for the weld metal was changed and some of weld have as much as 1.9 mass % of nickel while no representative irradiation tests were performed. This situation can be seen in Figure 1 where results from some first tests of SSP specimen are summarized.

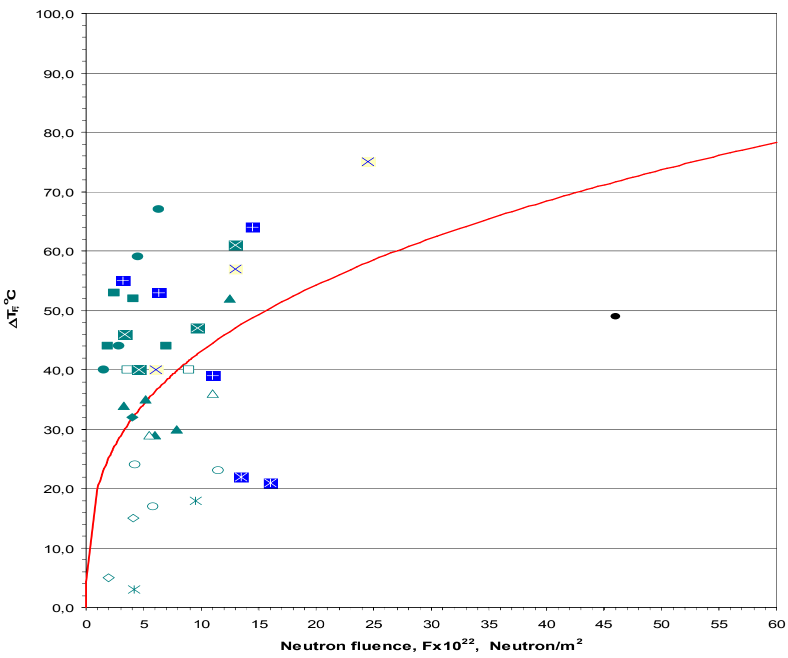


Fig. 1. Shift of ductile-to-brittle transition temperature of WWER-1000 RPV weld material due to irradiation. Results of surveillance specimen investigation.

1.1. STANDARD SURVEILLANCE PROGRAM

Standard surveillance program design was based on the experience with WWER-440 RPVs (design of cylindrical containers) but tried to decrease their high lead factor. Thus, new location of containers was put into design – over the reactor active core.

Containers

Specimens of one type are put in stainless steel containers identical to the ones in the SSP of WWER-440 type, i.e. either two Charpy type (impact of pre-cracked), or six tensile, resp. six fatigue type specimens. Six, resp. twelve (in two floors) these containers are accumulated into assemblies with one or two floors. Containers are pressed together by a special spring but they can practically free rotate within an assembly.

Location of containers

Five assemblies create one neutron embrittlement set. One set of assemblies was planned to be withdrawn at the same time. These sets are located in the upper part above the active core shroud near its outer diameter, i.e. above reactor active core – see Fig. 2.

The neutron field in the location of neutron embrittlement assemblies in the RPV as well as containers within assemblies is very complicated. Due to their location above the reactor core, neutron flux gradient is substantial not only between upper and lower floor in assemblies but also between individual assemblies within one set. Moreover, half of sets contains assemblies only with upper floor of containers where neutron flux is even lower than in RPV beltline, i.e. lead factor is lower than one.

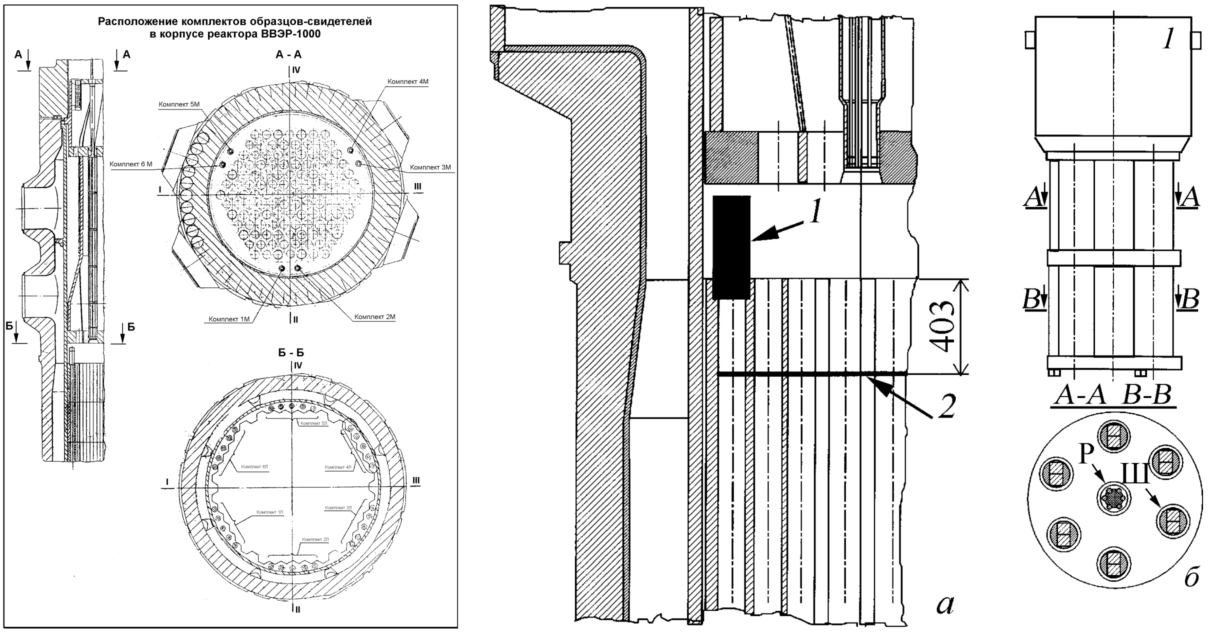


Fig. 2. Scheme of location of containers and containers assemblies of the SSP in WWER-1000/320 RPVs.

2. TECHNICAL ISSUES OF A STANDARD SURVEILLANCE PROGRAM

The review of the existing surveillance program of the WWER-1000/320 units confirmed the following facts:

General design

Design of assemblies and their positioning above the core result in nonuniform irradiation conditions and the number of specimens irradiated to similar neutron fluence is not sufficient for a reliable determination of the transition temperature shift.

Irradiation temperature

Irradiation temperature of the surveillance is higher (by about 10 to 15°C) than the RPV wall temperature.

Temperature monitoring

Temperature monitoring by diamond powder is not adequate for determination of the irradiation temperature since the results show far too large scatter and mostly even unrealistic results (lower temperature than inlet water temperature).

Neutron dosimetry

The quantity of neutron fluence monitors (3 sets) and variety in individual assemblies is insufficient to characterize fully the distribution of the neutron flux within the assembly and in individual surveillance specimens.

- The choice of neutron activation monitors does not enable to monitor fluences on surveillance specimens properly throughout the entire reactor lifetime.

- The lead factor in surveillance specimens is in upper floors lower than one and therefore the results cannot be used for prediction of irradiation embrittlement of RPV.

- The design of surveillance assemblies and containers inside of the assemblies does not allow clear determination of their orientation (moreover, they can rotate during reactor operation) with respect to reactor core center, which, together with small number of neutron monitors, does not ensure a proper determination of neutron fluence in individual surveillance specimens without direct autodosimetry (gamma-scanning) on each specimen.

3. MODIFICATION OF THE STANDARD SURVEILLANCE PROGRAM

Main disadvantage of the original SSP is that it is not capable to provide the monitoring of RPV material properties in a reliable way. Therefore, a modification of the program was elaborated in SKODA Nuclear Machinery, Plzen, Czech Republic for NPP with WWER-1000/V-320C type reactors for Belene (Bulgaria) and Temelin (Czech republic).

Main principles of the design was chosen in such a way to solve problems of the Standard Surveillance Program, mainly:

- location of containers should well monitor the conditions of reactor pressure vessel wall in beltline region, i.e. specimens temperature should be as close as possible (containers must be washed by a cold inlet water) and lead factor should be less than 5,
- whole set of specimens for one testing curve should be located in identical neutron fluence position
- as much as possible sets of specimens should be located in similar/close neutron fluence to be able to compare behavior of different materials
- withdrawal scheme of containers should assure monitoring pressure vessel material as well as neutron fluence during the whole RPV lifetime
- neutron monitoring should assure determination of neutron fluence to each of test specimens for every container
- temperature monitoring should be performed using melting temperature monitors with a appropriate range of melting temperatures
- cladding materials should be also included in the containers
- reference material should be added for an objective comparison of results
- spare containers should be added to monitor vessel annealing as well as further re-embrittlement if necessary

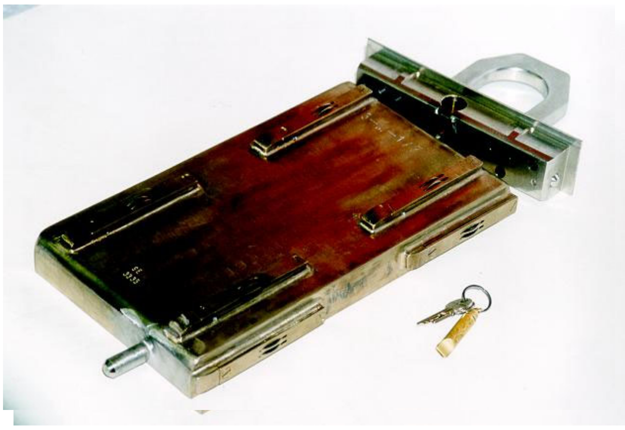
Design of such a program was performed and supported by a set of calculations (neutron physics, thermal-hydraulics) as well as experiments in a scale 1:1 (thermal-hydraulic characteristics measured in a hydraulic channel of a pressure loop in SKODA, thermal fatigue tests of container holders on pressure vessel wall).

Main characteristics of this Modified Surveillance Program are as follows:

Containers

Containers are of flat type with inner dimensions approx. 200 × 300 × 25 mm, are made from austenitic stainless steels plates welded on a frame. They contain special holders for location on pressure vessel wall – see Fig. 3.

All specimens for one withdrawal time are located in one irradiation container – specimens are in two layers, specimens of type and one set are touched each other in layer, only.



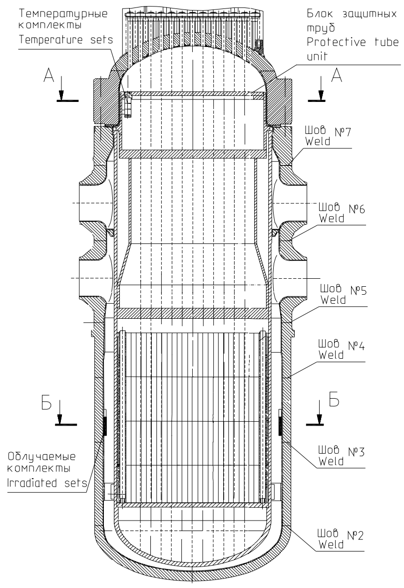


Fig. 3. Container of the Modified/Integrated Surveillance Program in NPP Temelin.

Fig. 4. Location of containers of the Modified Integrated Surveillance Program.

Location of containers

Containers are located in special holders that are welded on inner surface of reactor pressure vessel wall approx. 400 mm below the centerline of beltline region – see Fig. 4.

Containers are located symmetrically in maximum neutron fluence on vessel wall, i.e. in hexagonal corner positions.

Two additional identical containers are located between upper nozzles for monitoring possible thermal ageing effects.

Neutron monitors

Two types of monitors are used:

- 5 spectrometric sets of monitors in each container located close to both surfaces of the container and in different ends for absolute dosimetry of the container

    activation monitors - Co, Nb, Ni, Fe, Ti, Cu and Mn as foils and

    fission monitors - 237Np a 238U with and without Gd shielding

- two sets of wires – Cu and Fe – located on both surfaces in diagonal directions for relative dosimetry of each specimen
- scanning of specimens is prepared to check the neutron dosimetry
- continuous measurement of neutron fluences on outer pressure vessel wall (in the cavity) is a mandatory part of the program
- detailed calculation of neutron fields within assemblies and the reactor,

Temperature monitors

Several sets of melting temperature monitors are located either in specimens or in container filling:

Pb - 10% In	melting temperature	291°C
Pb - 8% In		300°C
Pb - 2,5% Ag		304.5°C
Pb - 1,75% Ag - 0,75% Sn		309°C
Pb		327°C

Withdrawal schedule

The following scheme is proposed:

2, 6, 10, 18, 26 + x years for radiation damage containers

14, 34 years for thermal ageing containers

one container for thermal annealing effect

one container for re-embrittlement rate effect.

This program has to be loaded into both pressure vessels on NPP Temelin, and was also prepared for the pressure vessel of unit 1 in NPP Belene, Bulgaria.

4. INTEGRATED SURVEILLANCE PROGRAM FOR WWER-1000/320 TYPE RPVs

In principle, it exists a possibility to use this reactor of WWER-1000/V-320C as a “host” reactor for those V-1000 units that are supplied by the Standard Surveillance Program and thus reliability of obtained results is not very high. Possibility of incorporation materials also from other reactors is given by the fact that containers of flat type are sufficiently large as they were designed for full size Charpy type specimens but now, application of reconstitution technique allows to include practically four times more specimens if inserts of dimensions 10×10×14 mm are used- see Fig. 5.

Fig. 5. Container of the Modified/Integrated Surveillance Program with inserts.

Integrated surveillance program for several similar reactors can be realized in accordance with the [2] if the following main requirements are fulfilled:

- reactors are similar in design and operation,
- neutron fluence determination on all RPV wall is assured for the whole reactor lifetime,
- operation of the "host reactor" is assured for the whole operation of reactors within the family.

A proper and reliable monitoring radiation damage in materials for WWER-1000/320 units is now under high study and interest as it was determined that in some welds with high nickel content (in some cases up to 1.88 mass %) radiation embrittlement can be much larger than that obtained from predicted formula given in [1]. Qualification tests for materials of WWER-1000 RPVs were performed on welds with nickel content below 1.5 mass %, but later the nickel content was increased (in most of V-1000 units) to get better fracture toughness properties but no further study of radiation embrittlement was performed.

Thus, using the opportunity that NPP was delayed in its start-up due to changes in I&C system, it was possible to modified content of some containers (for Unit 2) in such a way that specimens from archive materials of the following units were incorporated into the program: Khmelnitsky Unit No. 2, Rovno Units No. 3 and No. 4, Zaporozhye Unit No.6 (Ukraine) and Kalinin Unit 3 (Russia), as nickel content in all these weldments is well over 1.5 mass %. In this first part of the program only weld metals from this RPVs were included. From all materials, 12 specimens for impact notch toughness and 12 specimens for static fracture toughness tests are included. It is necessary to mention that all these RPVs contain their original Standard surveillance program.

In this time, second part of this Integrated surveillance program is under final realization. New six containers are manufactured that will replace containers from the first part in both units in NPP Temelin (design of container holders and containers itself allows inserting of new containers during reactors shut down where reactor internals are removed). Base metals from all abovementioned RPVs will be included in these containers together with base and weld metals from the NPP Belene. Moreover, standard IAEA reference material JRQ as well as IAEA reference V-1000 materials are also included for mutual comparison with results of the first part as well as for better and more objective evaluation of results (there exist a large database of the behaviour of JRQ steel, e.g. within the IAEA Co-ordinated programs and its database).

Realization of such Integrated Surveillance Program will substantially improve knowledge about behavior of WWER-1000 RPV materials during their operation, i.e. about radiation damage – embrittlement. Comparison of results from different RPVs also allows to assess the behavior of materials from other RPVs with only Standard surveillance program – based on comparison of chemical composition and operational conditions. It also allows comparison and analysis of results from testing their SSP and propose a correction coefficients (taking into account different irradiation conditions) if necessary. Results from this Integrated Surveillance Program also will enlarge existing database of radiation embrittlement data of this type of materials in a more objective manner.

5. CONCLUSION

Modified Surveillance Program for reactor pressure vessels of NPP Temelin with WWER-1000/V-320C type reactors is used for the Integrated Surveillance Program for several RPVs of NPPs in Ukraine, Russia, Bulgaria and Czech Republic as the Standard Surveillance Programs in WWER-1000/V-320 type reactors do not fulfill requirements given by codes and standards. Such Integrated Surveillance Program allows to obtain reliable information about radiation embrittlement of materials in tested reactors pressure vessels that will be also correlated with the IAEA reference steel JRQ to get more objective results.

Realization of this Integrated Surveillance Program increases information about the behavior of RPV materials of this type of reactors that have only Standard Surveillance Program. Moreover, it allows correlation of results from these Standard Surveillance Programs with those from other vessels not included in this Program that also increase reliability of such results. Generally, this Integrated Surveillance Program will increase safety of operating WWER-1000/V-320 type reactors operated in these countries.

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INVESTIGATIONS ON THE CORROSION BEHAVIOUR OF STRUCTURAL MATERIALS OF PWR PRIMARY CIRCUITS

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Abstract. The today main phenomena relating to the corrosion behaviour of structural materials in PWR primary circuits are mainly connected with stress corrosion cracking. Recently, cracks observed on some vessel head penetrations in US plants increase the interest of studies on Alloy 600 and Alloy 182. The extension of the live time of PWR plants in many countries including France induces to investigate the stress corrosion cracking behaviour of stainless steels materials. We describe the experimental procedures and tests performed for investigating the stress corrosion cracking behaviour of nickel base alloys and stainless steels materials, from constant deformation tests with C-rings or Reverse U-bends, to the crack growth rate monitoring by electrical or acoustic emission techniques. Following the objectives of the investigations, specific tests and results exploitation procedures have to be chosen: constant deformation or constant load tests are used for the evaluation of the sensitivity of materials to SCC, while SCC initiation or propagation may be studied with constant elongation tests. Examples illustrate these today investigations on stress corrosion cracking of Alloys 600, 182 and 690 in PWR primary conditions. Up to now, data acquisition has been the main objective of SCC experiments and semi-empirical correlations have been obtained and are used to predict material behaviour in PWR primary systems. New theoretical developments are needed to increase the confidence in the life time predictions.

1. INTRODUCTION

The investigations related to structural materials of Nuclear Power Plants (NPP) include corrosion behaviour which is very often a key issue related to life time prediction, particularly when NPP life duration is extended to fifty or even sixty years. However, we observe also occasionally problems associated to quite well known corrosion phenomena like Flow Assisted Corrosion (FAC) or Stress Corrosion Cracking (SCC). These events associated with corrosion phenomena had often similar root causes which continue to recur.

This paper focuses on the today main phenomena which are connected with the corrosion behaviour of structural materials in pressurised water reactors (PWR) primary circuits, and more particularly on stress corrosion cracking of Alloy 600 and its welding material, Alloy 182. Recently, cracks observed on some vessel head penetrations in US plants increase the interest of the studies on these Alloys 600 and 182. Illustrations are given to explain how SCC sensibility of materials is investigated. Experimental procedures and exploitation of results performed for investigating the SCC initiation or propagation parameters are also described for nickel base alloys. The determination of crack growth rates by electrical techniques or acoustic emission techniques is specified and examples are given related to the influence of the cyclic stresses.

It seems important to precise that the SCC of Alloy 600 is not a new phenomena: first report has been made by Coriou. At the end of the fifties, Alloy 600, a nickel base alloy, is found prone to stress corrosion cracking in pure water at 350°C. Many laboratory experiments have been made during the following 20 years which lead to the confirmation of Alloy 600 SCC in pure & in nominal primary water (hence the acronym PWSCC for Pure or Primary Water Stress Corrosion Cracking, called also “Coriou effect”). At the beginning of the eighties, SCC of Alloy 600 becomes a generic problem on highly stress alloy 600 components in PWRs, particularly SG tubes (rolling zones). The replacement of nearly all the SGs made with alloy 600MA tubes by SGs with Alloy 690 tubing has been generally the applied solution. Further, at the beginning of the nineties in France (1991) and ten years latter in US and in Japan, cracks are found due to PWSCC on vessel head penetrations (Alloy 600 tubing and Alloy 182 welds), which are the reasons of the today replacement of many PWR reactor vessel heads.

2. SENSITIVITY to SCC

To investigate the SCC sensitivity of a material, constant deformation specimens are generally used. It could be C-rings when plates are available, U-bends or more frequently reverse U-bends (RUBs) when materials are available as tubes. Figure 1 gives schematic drawing of these specific specimens and figure 2 illustrates a reverse U-bend before test and after cracking in PWR primary conditions. The objective of the important deformation and of the associated stress level is to increase the rate of occurrence of the phenomena. Another way to increase the phenomena is to conduct experiments at higher temperature (generally 350°C or 360°C), as SCC is generally a thermally activated phenomena.

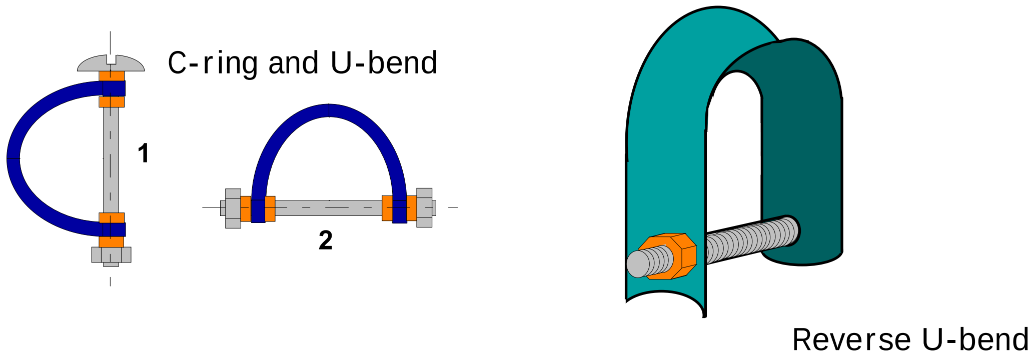

Fig. 1. Specimens used to investigate SCC susceptibility.

Fig. 2. Reverse U-bends made of Alloy 600 before (right) and after test (left) in PWR primary conditions.

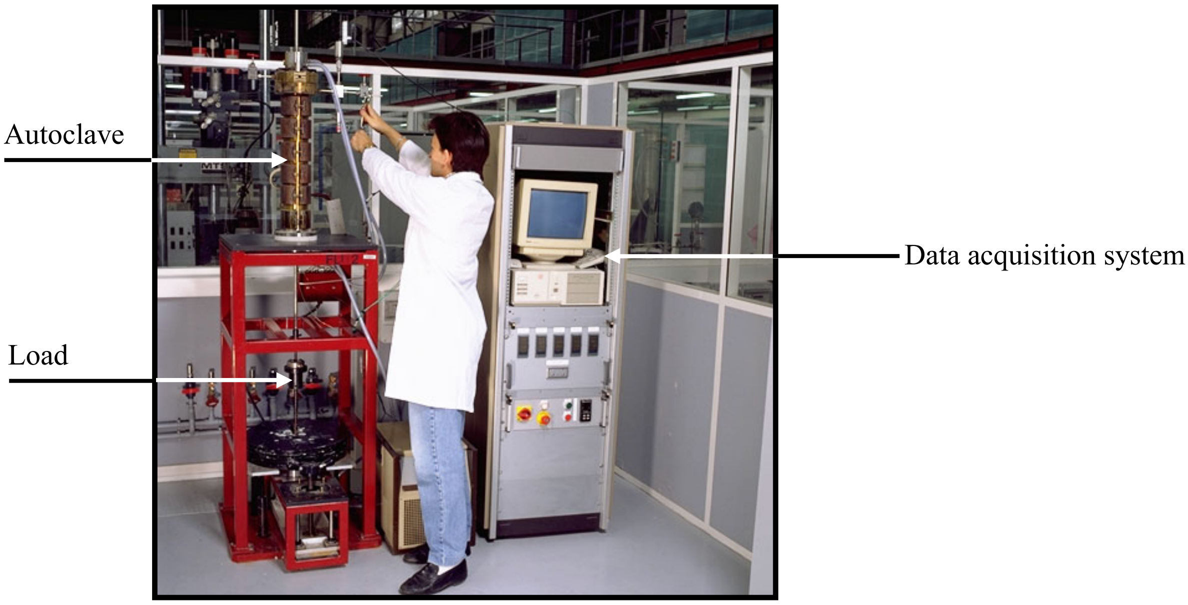

Fig. 3. Constant load device.

Constant load tests are also performed to investigate SCC sensitivity. These tests could be more representative of in-service conditions and have the advantages of the knowledge and the control of applied stresses. They are used to estimate the time to cracking as function of the applied stress for instance. One of the limits of constant load tests is that their duration could be very long and that only few tensile specimens could be used during the same test and required devoted devices as the one shown in Figure 3.

Of course, all these specimens have to be exposed to PWR primary conditions, and particularly, the chemical conditions have to be reproduced carefully, including the nominal hydrogen concentrations which directly influence the electrochemical potential of the specimens. The pollutants like chloride or sulphate have to be within the PWR specifications.

Figure 4 gives an idea of the typical results obtained with such experiments: the number of cracked reverse U-bends made of Alloy 600 is given as a function of the exposure time in PWR conditions: primary water with 1000 mg.L-1 of boron introduced as boric acid, with 2 mg.L-1 of lithium introduced as lithia (LiOH) with 35 cm^3 of hydrogen per L of water at 360°C. It shows that after 10 000 hours, about half of the reverse U-bends made of Alloy 600 are cracked. Similar tests performed with Alloy 690 RUBs did not evidence any cracks after 90 000 hours of exposure. It shows that Alloy 690 is immune related to SCC under test conditions, at least during 90 000 hours at 360°C, while Alloy 600 MA exhibits a high susceptibility to SCC under the same conditions.

The same kind of tests has been conducted with alloy 182 weld specimens (C-rings specimens) in primary conditions at lower temperature (330°C) to investigate the effect of thermal treatment: it has been put in evidence that 90% of the alloy 182 specimens are cracked when they are without thermal treatment (Mill Annealed conditions), after 5000 hours of exposure, while only 50% of thermally treated (TT) specimens are cracked after 30000 hours of exposure under the same conditions.

These quite simple tests are good ways to investigate susceptibility of materials to SCCC under primary conditions, if some precautions are taken, like to use quite a large number of specimens and/or heats in order to obtained statistical results. If they give ideas about the susceptibility, the deformation (and the associated stress) is so high that they have been often considered as non representative during the seventies. Even if a stress threshold could be determine (with constant load tests), it is now considered that high deformations or stress levels are ways to accelerated SCC, as temperature.

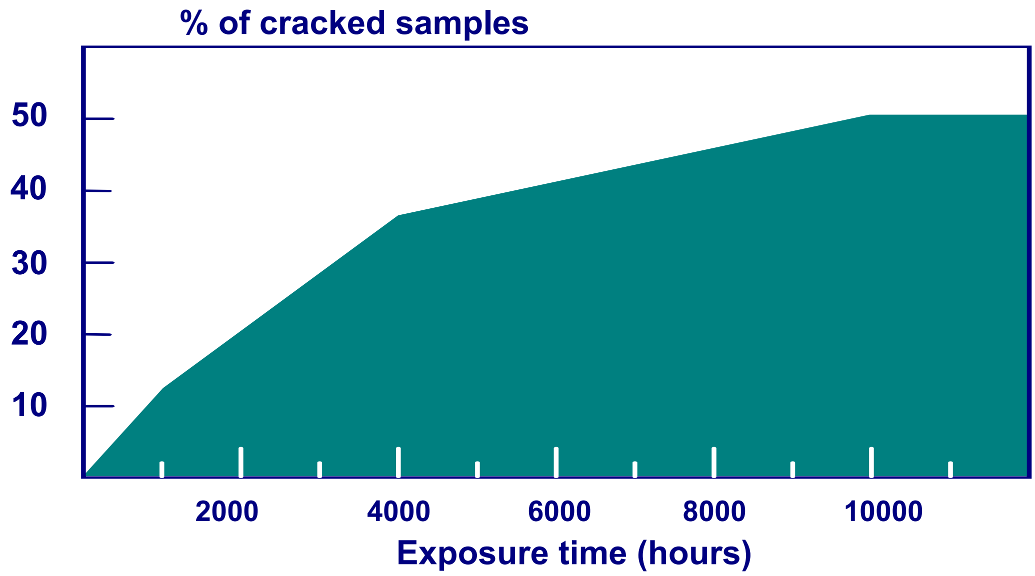

Fig. 4. Percentage of cracked Alloy 600 reverse U-bend as function of exposure time (PWR primary chemistry at 360°C).

3. INITIATION and PROPAGATION of SCC

Constant Elongation Rate Tests (CERTs) are generally performed to investigate the initiation and the propagation mechanisms of SCC. These tests need to have a tensile testing machine coupled with an autoclave system, and more often with a refreshed autoclave facility to control the water chemistry and particularly the pollutant concentrations, as they may play a predominant role. Only one specimen could be tested during each experiment. We would like to focus on the statistical exploitation of the number and the length of cracks which is performed at the end of the experiments at CEA corrosion laboratory: number and length of observed cracks are counted at different cut levels of the specimen. From the crack trace distribution Zt(lt,t) obtained on the section plane and with a function characterizing the average shape of a crack of depth l at time t, the function Z(l,t), which is the density Z of cracks deeper than l at time t, is obtained. Quantitative information concerning initiation and propagation of the cracks during the CERT is deduced from this function Z.

Illustration of this method could be made with the study of the influence of the cold worked layer of Alloy 600 on SCC: the same specimens of wrought Alloy 600 have been tested under the same experimental conditions (primary chemistry with lithine, boric acid and hydrogen additions, at 360°C) but with three surface conditions: without cold worked layer (electroplolished specimen), and with two depths of surface worked layer (130 µm and 280 µm) but with roughly the same surface Vickers Microhardness (450–460 VM). For each specimen, the density Z of cracks deeper than l at rupture time for the 3 worked layer conditions is reported figure 5. Such data lead to two main observations. The first one is that the number of cracks deeper than 25 µm is nearly the same whatever are the specimens, meaning that cold worked layer has no influence on crack initiation. But crack depth is function of the depth of the cold worked layers: deeper is the worked layer, longer are the cracks. So, the influence of the cold worked layer is on the crack propagation rate, as it could be illustrated with the same data reported in figure 6 where the average propagation rate (crack depth divided by the time to rupture) is reported as function of the crack density.

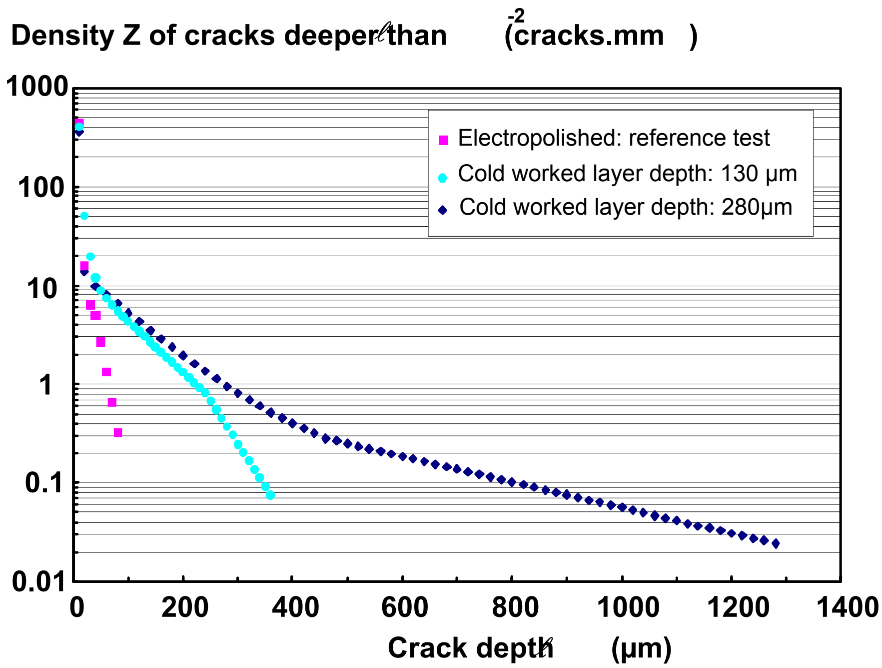

Fig. 5. Density of cracks as function of crack depth (Constant Elongation Tests, Alloy 600, 10-7 s-1, Primary chemistry, 360°C, tests performed to rupture).

In figure 6, two transitions could be observed related to the SCC propagation rate. The first one, observed on the 3 specimens, is located at crack density of 15 cracks.mm-2 and at a crack depth of 25 µm which is the location of the first triple line (FTL) between the grain boundaries on this material, showing that microstructural parameters are of key importance at the initial stage of cracks propagation. The second transition is observed only with the deepest cold worked layer at the same depth than the cold worked layer depth (280 µm) which put in evidence its role related to the kinetic transition towards rapid propagation rates. So a plastically strained cold worked layer of large depth strongly promotes IGSCC. It was found that when the transition in rate occurs in the cold worked layer, the critical value of the stress intensity factor obtained is about 9 MPa.m1/2, similar to the one obtained with single crack specimens like CT.

The main limit of CERTs experiments is due to the dynamic stress evolution during each test: large discussions are today on the application of the observed results to constant stress conditions.

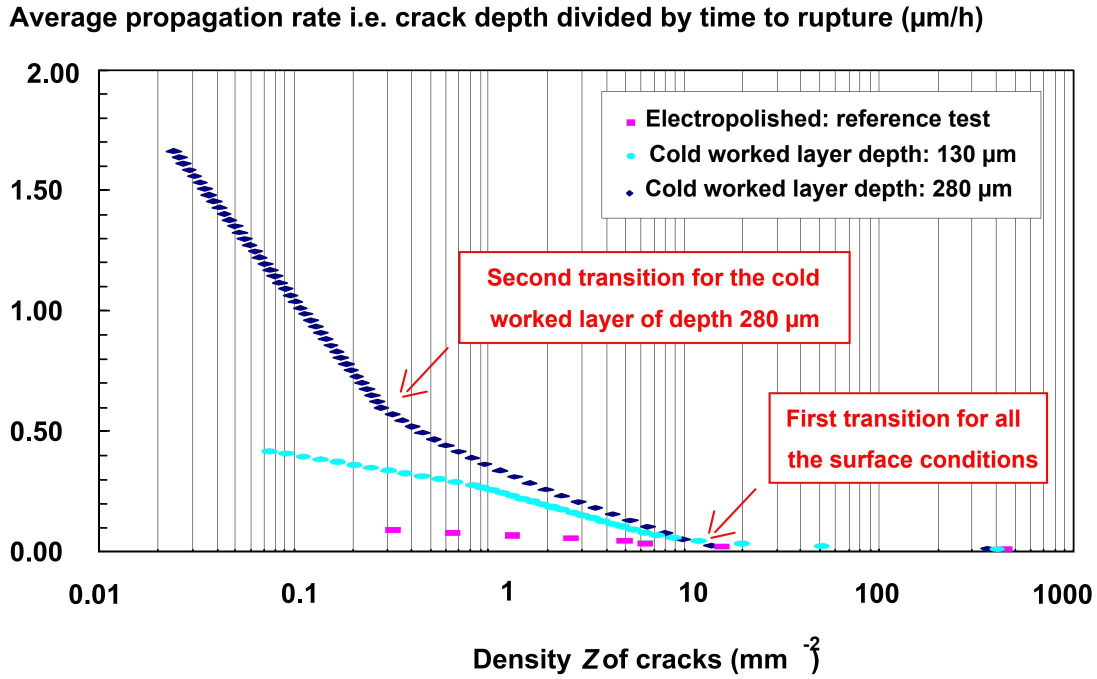

Fig. 6. Average propagation rate as function of the crack density without cold worked layer (electroplished conditions), and with two depths of cold working layer (Constant Elongation Tests, Alloy 600, 10-7 s-1, Primary chemistry, 360°C, tests performed to rupture).

4. CRACK GROWTH (PROPAGATION)

Measurements of crack growth rates under representative conditions is of key importance for operating PWRs. International efforts are made to obtained these data and the parameters with may influence these rates of crack propagation. Specific equipments are needed to make these measurements. The crack propagation is sometimes followed with the Acoustic Emission (AE) technique, but more generally followed with the RDCPD method (Reverse Direct Current Potential Drop system). Fatigue air pre-cracked Compact Tensile (CT) specimens in Alloy 600 are tested in PWR primary conditions, which is the real difficulty of methods which are used in air without troubles. For instance, in PWR primary water conditions, electrical insulation are more difficult to assume and the specimen could be polarised via the RDCPD method to corrosion potentials which are not representative of PWR conditions. When crack growth rates are monitored in-situ by AE or RDCPD methods, these methods are validated by post-mortem observations. Fracture surfaces are characterized by macroscopic and microscopic observations. Comparison of the crack growth rate and of the fracture features demonstrated that these post mortem observations are necessary to validate the crack growth rates and acts like a post standardisation method.

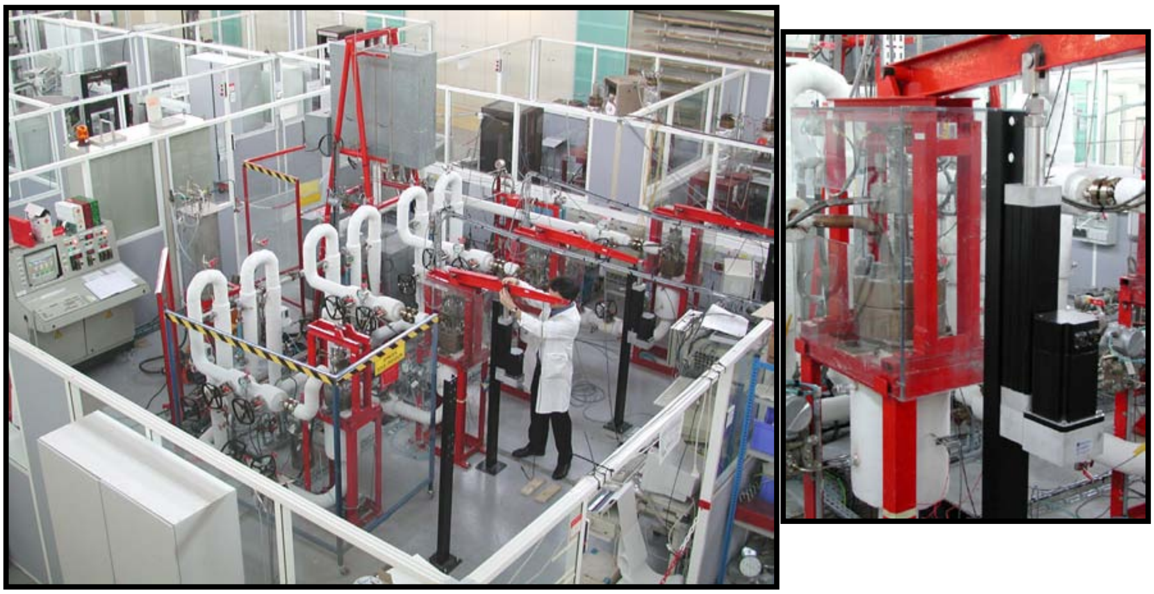

Fig. 7. View of the VENUS facility and of one of the four autoclaves equipped with a cyclic loading machine.

At CEA corrosion laboratories these crack growth rates are measured on the Venus facility (figure 7) which includes 4 refreshed autoclaves, each one equipped with one to 4 CT specimens, following the size of the specimens and the used monitoring crack method: with acoustic emission, only one CT-specimen could be use, while with the RDCPD method, up to 4 CT specimens could be put together in the same autoclave, even if more generally only 3 specimens are used. The water chemistry is controlled by ion exchange resins and via samples analysed by ionic chromatography and by plasma impedance spectrometry. The chlorides, fluorides and sulphates concentrations could be adapted to the expected conditions via exchange resins and the hydrogen partial pressure is measured in each autoclave with a hydrogen probe, while the oxygen could be monitored if necessary on the main circuit of the facility.

Since several years, there is a growing interest to assess the role and the effects of cyclic loadings or stress transients on the SCC behaviour of Alloy 600. Nuclear Power Plants have thermal cycles that may generate cyclic loadings (low frequencies of course) on the components. The determination of the effects of these cyclic loadings on crack propagation rates could be of great importance for the live time of components on which initiation of SCC has been detected. Illustration of obtained results is given by the observations performed on two heats of Alloy 600 which were tested under four different mechanical loading conditions. The specimens of one heat (3110439) exhibit an irregular crack front for the SCC crack but also for the air fatigue precrack (figure 8). This irregular crack front is explained by the heterogeneous microstructure of the material and, more particularly, by the distribution of the intragranular precipitates (high densities of intragranular precipitates near the surface of the material). On the contrary, the other Alloy 600 heat (WL344) specimens exhibit a regular crack front for the air fatigue precrack as well as for the SCC test (Figure 8). The distribution of carbides is homogenous for this heat. For the evaluation of the crack growth rate, it is of paramount importance to know if a mean value of the crack length has been taken (and how it has been done) or if the maximum length is used. These differences in the procedure of determining the crack length and so the crack growth rates are one of the explanations of the discrepancies of literature data.

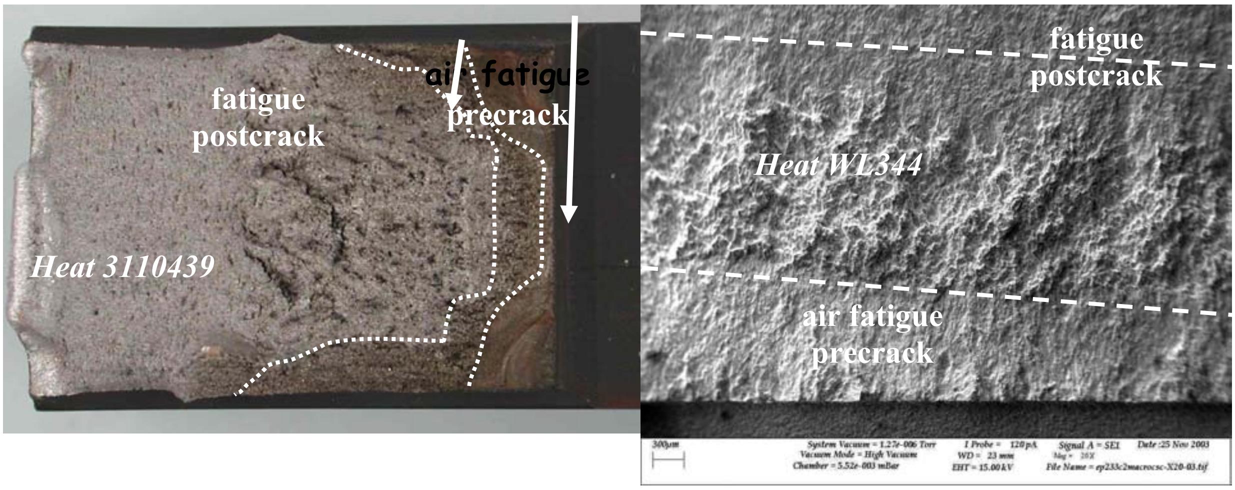

Fig. 8. Fracture surface observed on Alloy 600 for two different heats (PWR primary chemistry, 325°C, cyclic loading).

According to the crack growth measured on the samples, the two alloys tested appear to be similarly sensitive to SCC at this temperature. Concerning the influence of cycling mechanical loading, for tested conditions, the crack growth is higher for mechanical cycling for the 3110439 heat (about three times). But opposite observations are made for the WL344 heat. The crack growth rates obtained for these two heats are compared to the literature data on Alloy 600 (Figure 9). Our data are in the upper part and put in evidence that the waveform of the cyclic stress may accelerate or decrease the crack growth rates following the microstructure characteristics of the alloy 600 heats. On the heterogeneous heat (regarding the carbide precipitation), the cyclic stress (triangular or saw teeth forms) had an accelerator effect while the opposite is observed with the homogenous heat.

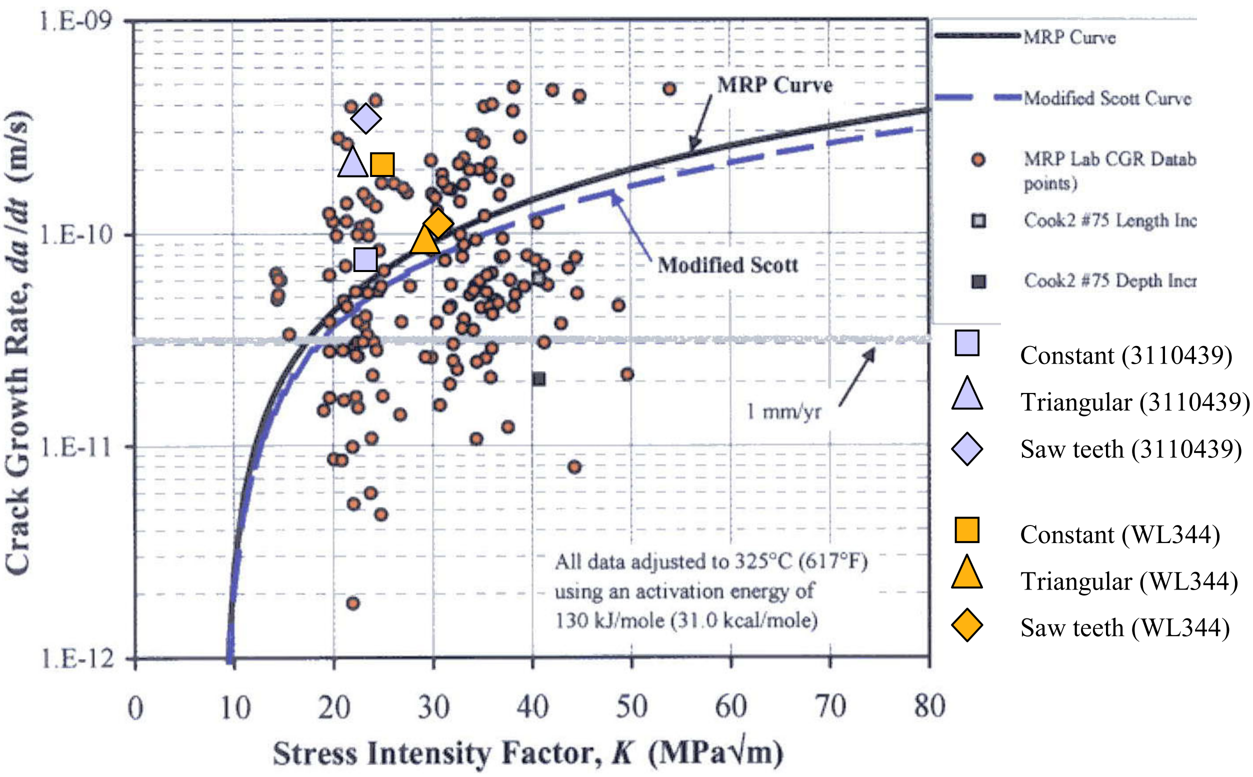

Fig. 9. Comparison of the crack growth rates on Alloy 600 as function of the stress intensity factor: heat and cyclic stress effects (PWR primary chemistry, 325°C, cyclic and uniform loading).

5. CONCLUSIVE COMMENTS

Investigations on the corrosion behaviour of structural materials of PWR primary circuits are performed since quite a long time, as primary stress corrosion cracking of Alloy 600 has been reported since more than fifty years. Nevertheless, PWSCC phenomena still occur on primary components like vessel head penetration of PWRs. International needs include a correct evaluation of the crack growth rates under PWR conditions in order to obtained agreement on a data base for reliable prediction of crack evolutions. If illustrations have been given on Alloy 600 where the effects of material parameters, temperature, cyclic stress and chemistry (hydrogen concentrations) are still under investigations, other materials are also studied in the framework of the extension of life time of PWRs. Up to now, Alloy 690 has exhibited a very good behaviour related to PWSCC: more investigations are performed to know if the Alloy 690 is completely immune to PWSC, even with very long exposure time. For stainless steels, some plant data show that they could be sensitive to PWSCC under specific exposure conditions (like high hardness). Investigations are mainly performed to find limits of the phenomena (stress level, temperature, cold worked layer, hardness, …), and to evaluate the influence of cyclic stress and of pollutants with eventual synergetic effects.

For nickel base alloys and stainless steels, the main efforts are related to the knowledge of physico-chemical mechanisms of PWSCC. Today, SCC of stainless steels in primary water conditions is studied. The influence of cold work hardening and pollutants on SCC susceptibility is characterized. A model will be developed. Welded materials, as Alloy 182 and 82, are also studied. Studies are concerning SCC initiation and propagation but also hydrogen embrittlement.

These studies lead to improve the PWCSS models and to obtain a reliable method for a good prediction of the behaviour of these passive materials in pressurised water reactors.

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FEEDBACK FOR GOOD PRACTICES AND RECOMMENDATIONS.
(NOT IN ORDER OF IMPORTANCE)

	Good Practices	Recommendations
1.	Address causes of problems instead of treating symptoms.	Develop guidelines on how to separate technical from managerial issues. Recommendations on self-assessment.
2.	Address all indicators that the NPP is giving. (Small indicators that are ignored may give rise to large events).	Address the issue of thermal ageing in heat affected zones (HAZ) with coarse gained material structure (stainless steels and low alloy carbon steel RPV.
3.	Implement strong corrective actions and check effectiveness by an independent review.	Hold dedicated meeting or workshops to address progress in flow assisted corrosion (FAC) or erosion assisted corrosion (EC) in piping every 2 years.
4.	Maintain confidence in NPP workforce. Show appreciation for a good job done.	Develop a guideline on how to select the best inspection and monitoring method for a given degradation mechanism.
5.	Avoid covering systems, structures and components (SSCs) with insulation material or ensure checks are made under it. (See 7. below).	Buried pipes (essential service water systems): Degradation, inspection and mitigation methods guideline needed.
6.	Avoid chlorine-containing insulation, especially when stainless steel piping is involved.	
7.	Ease of inspection is needed at all times.	(Note: NPP design may hinder this. There is a need to design for inspection/PLiM).
8.	Use a global safety approach.	(Note: Take in all aspects for safe operation, including secondary SSCs).
9.	Tailor in-service inspections for each specific task (e.g. Baffle bolts in PWR).	(Note: Dedicated methods for special problems need developing/improvement).
10.	Where possible, use NPP actual materials in tests and research programmes.	(Note: Real materials reflect the best state of the NPP).
11.	Always put safety targets before economic ones.	(Note: High safety levels are associated with reliable operation/profitability).
12.	Avoid pressure transients especially when thinned piping exists in the NPP. Operate “softly”.	
13.	Identify all areas in the NPP where orifices/bends in piping change flow direction due to FAC-EC enhancement.	(Note: Design aspects should avoid such features. Improved materials should be chosen).
14.	Do not rely on critical thickness criteria in piping. Measure and re-assess.	(Note: Verification and benchmarking allow ease of following thinning rates. This will aid timely replacement decisions).

15.	Replace thinned carbon steel piping with better-alloyed piping (>0.5% chromium).	
16.	Flanges and hangers on piping can influence the FAC/EC probability due to local cooling or flow-2 phase flow conditions ("flashing" to steam).	
17.	Inspect daily if a problem is suspected in a given area.	
18.	Create a systematic list of SSCs for the ageing management programme (list all known and possible mechanisms).	(Note: General practice as recommended in IAEA TECDOCs)
19.	Recognize geographical features (nearness to sea) and allow for increased corrosion attack. (Chloride/chlorine).	
20.	Develop "living knowledge" database concerning ageing in SSCs.	(Note: Databases also serve knowledge management-saving of expertise for others).
21.	Create multidisciplinary approaches (engineers, metallurgists, chemist, and managers).	(Note: Global approach gives a better chance to solve problems).
22.	Focus inspections on piping (carbon steel) operating between 130-1650C (critical/maximum for FAC/EC).	
23.	Create a strong utility ownership attitude with less reliance on outside contractors.	(Note: With deregulated markets and less own personnel, this could be a problem area for NPPs).
24.	Avoid complacency due to high availability factors in previous cycles.	(Note: Vigilance at all times is needed).
25.	Define persons or groups responsible for verification of test results.	
26.	For pipe thinning do not rely on computer predictions but also confirm all by verifying results on the piping.	
27.	For replacement piping: plan well ahead for procurement due to delay times. For RPV heads also.	Develop methods to improve estimation of residual life in SSCs (benchmarks, uncertainties and small specimen effects).
28.	Adjust inspection of frequency according to previous indications and findings.	High cycle fatigue and thermal ageing needs further study/CRP.
29.	Manufacturing methods create surface tensile stresses. Avoid if possible. QA documentation from manufacturer must be high. Be aware of mechanically formed bends.	Make sure all documentation in available and maintain the record of the NPP configuration.
30.	Always plan work to ensure lowest possible radiation doses ALARA.	Practice on Mock-up.

31.	Avoid subjective reporting-always use verifiable facts and data.	
32.	A bottom –up Approach in NPP management is preferred. Personnel should be able to transmit their findings easily to management.	
33.	Avoid complicated work processes. Make ease and transparency of work process a goal.	(Note: The more simple the process, the less likelihood of a mistake).
34.	Use ageing management assessment team (AMAT) through the IAEA.	(Note: Consult existing data and apply lessons learned).
35.	Avoid prolonged shutdowns of NPP. (Corrosion protection).	
36.	Management and inspection techniques must evolve as the NPP ages. Improve inspection reliability of data.	Recommendations on how to create "regulations by feedback" in NPPs (Workshop).
37.	NPPs should aim to always exceed the minimum regulator safety and technical requirements on SSCs.	
38.	Create a "regulation by feedback" system as this responds to in-service inspection results and failures.	
39.	Maintain risk-informed research activities.	Examine the impact of open markets (reducing operating costs) on the levels of safety. Identify methods to counteract this. (Specialist meeting)
40.	Recognize the importance of both mechanical/chemical state and chemical environment in which SSCs operate.	Try to create the optimum environment (chemical, neutron and flow, …)
41.	Use monitoring systems based on the principle of surveillance (real material) specimens.	
42.	Avoid dissimilar metal welds in main coolant times if at all possible.	
43.	Include enough specimens for RPV annealing/re-irradiation studies. Anticipate revision/reassessment of mechanical properties from surveillance.	Increase analyses of root causes in NPP accidents/events/component degradation and also management issues (safety culture) and make discussions on international levels.
44.	Recognize the stainless steel cladding as being an important issue in PTS.	Develop guides of major damage/degradations and improve or prepare adapted codes and standards.
45.	The importance of having comprehensive manufacturing information (material, heat treatment, mechanical shaping, chemical composition etc.) is recognized. Keep updated documentation. Assure knowledge transfer to younger generation NPP operators.	Develop a guide on "good practices in the management of ageing of SSCs in NPPs".

46.	Maintain NPP personnel training at the highest possible levels.	The IAEA should draw attention to the fact that even high-level NPP management can be unaware of or underestimate the risk solely by their confidence in the large number of safety systems and margins of operation. Technicians may not be able to communicate the issues to a market-oriented shareholder management.
47.	Promote questioning attitudes in NPP personnel at all levels of jobs.	
48.	Pro-active replacement of SSCS "Do not postpone". It is better to replace before the SSC fails. Plan replacements to coincide with other outages (e.g. refuelling).	A system to review a NPP's response to significant events should be instituted.
49.	Use information sources on SSC degradations/ (materials), even those occurring outside nuclear applications.	
50.	Use the Plan-Do-Check-Act (Demel) approach. IAEA report "Implementation and review of AMP" (1999).	A CRP could be developed to identify future research needs, including advanced reactor types.
51.	Openness in discussing and showing of NPP problems is important. Make personnel aware of consequences of major events.	Identify all safety aspects concerned with long-term-operation ("SALTO").
52.	Eliminate known "problem materials" (e.g. Alloy 600/182, low carbon steel piping) wherever possible.	
53.	Start ageing management programmes as early as possible, together with PSR. NPPs should implement plant-life management with a view to long-term safe operation (LTO) already at start-up. LTO is a utility decision. LTO will depend on the safety assessments and economics. Keep documentation to show continual improvements to SSCs to aid regulatory decision in safety assessments.	IAEA should develop TECDOC of international experiences and knowledge regarding accidents, malfunctions, ageing effects, etc., root causes, which is useful as a textbook for educating and training personal such as designers, operation staff and workers beyond generations and countries. A database like the INIS type is needed but for education and training of personnel, database is not directly applicable due to a huge amount of information.
54.	Monitor at appropriate conditions (erosion-corrosion and neutron flux conditions) for surveillance programmes. Make sure neutron spectrums are represented with due regard for the thermal neutron contributions.	Neutron flux-rate effects are still only partially studied. More planned experiments are needed.
55.	NPP plant managers should always be included in IAEA workshops concerned with SSC ageing/degradation. Often managers have different appreciation (economics, production goals) of problems than technicians.	
56.	Ageing plant issues will become events if management issues exist. Therefore address management issues.	

57.	Create a NPP personnel culture to encourage discussion of failures and mistakes. Reward those who show honesty in admitting errors, but sanction those who hide errors or falsify data, documents and control sheets.	
58.	Hard copy documents must be short (Maximum 10 pages). Make information access “user friendly”.	The role of PSA/risk informed regulation/inspection needs investigation and focus in a guideline to take into account aged SSCs and aged management.
59.	Identify design-dependent problems and find best solutions regarding safety and then economics. Maintain information flow in operator’s groups and check generic issues.	Issues concerned with “Aged NPPs and young NPP personnel who will operate the NPPs in future” may have to be investigated more. Training aspects are essential.
60.	Continue risk-informed safety classification methods.	Cross-linking databases OECD Pipe failure/AIRS and others is desirable.
61.	Anticipate synergistic effects of 2 or more minor mechanisms to be causing a large problem.	Recommendations of a systematic development of human resources needed.
62.	Fully appreciate all consequences of power up rates on SSCs. (Increased pressure, temperature, flow-through rates etc.)	Good practices to follow for long-term shut down of NPPs due to major refurbishment activities etc.

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