Limitations of the inspection and testing concepts for pressurised components from the viewpoint of operating experience

H. Schulz

(GRS)

Abstract: The role of in-service inspection and testing is to contribute to a safe and reliable operation of systems, structures and components. It is therefore the objective of inspections and tests to identify malfunctions and degradations at a stage early enough to avoid detrimental impacts on safety as well as on the reliability of the plant. Taking mostly the pressure boundary of German light-water reactors as an example, it is the intention of this paper to analyse how successful present inspection and testing requirements are and to discuss limitations.

Based on a review of the world-wide operating experience the following questions of a more generic nature are addressed:

- Are the relevant damage mechanisms being addressed in our codes and standards?
- What are the criteria to develop a representative scope of inspection?
- How to maintain a sufficient level of information for a decreasing number of nuclear power plants in operation?

It can be concluded that the revision of codes and standards according to lessons learned from operating experience remains as an ongoing process. Furthermore, the criteria applied to derive a representative scope of inspection need to be addressed in more detail, specifically with respect to corrosion. The continuous evaluation of operating experience of a large numer of plants is the most valuable source to identify beginning degradations.

1. INTRODUCTION

The role of in-service inspection and testing is to contribute to a safe and reliable operation of systems, structures and components. It is therefore the objective of inspections and tests to identify malfunctions and degradations at a stage early enough to avoid detrimental impacts on safety as well as on the reliability of the plant. Taking the pressure boundary of German light-water reactors as an example, it is the intention of this paper to analyse how successful present inspection and testing requirements are and to discuss limitations.

Looking to the primary pressure boundary of light-water-cooled reactors, the reactor pressure vessel is the only component which supports all three safety functions directly. Therefore, the integrity of the reactor pressure vessel was and is the component receiving utmost attention. But, as safety assessments demonstrate, the integrity of other large primary circuit components like steam generators and pressurisers are of equal importance if their failure can compromise the barrier function of the containment. As a consequence, the safety requirements are similar for all large components of the primary pressure boundary. The fundamental safety functions are also the basis for a classification or grouping of systems and components. Accordingly, the components of the primary pressure boundary are all classified as the highest class of components.

Over the last decades we have accumulated more than 7 000 calendar years of operation with about 350 light-water –reactors world wide. Besides a few of the very early plants which used natural circulation, all systems use forced circulation to remove the heat from the core. The system pressure differs with respect to the general principle applied in the design (PWR or BWR). The operating temperatures are less than 350 °C. With respect to damage mechanism relevant to the integrity of the components of different designs and the potential failure modes there are considerable similarities. For this reason, a broad information exchange is beneficial to all.

Together with the utilities, the vendors of the nuclear steam supply systems (NSSS) are organising the information exchange in so-called "Owners Groups". Furthermore, information is exchanged at an international level via the World Association of Nuclear Operators (WANO). All information relevant to safety is exchanged via the Incident Reporting System (IRS) which is jointly operated by the IAEA and the NEA of the OECD.

2. DESIGN APPROACH

Pressurised components of nuclear power plants are designed according to the respective system and safety requirements. For pressurised components, there is a long tradition in all industrialised countries to specify detailed requirements for the design, manufacturing and operation of pressurised equipment. These requirements shall ensure a high level of reliability and protection of the plant personnel and the public. Besides the use of high-quality components, safety measures like safety valves, burst discs etc. are applied. For components in nuclear power plants, the respective expertise has been used in the development of technical requirements and codes & standards. There are considerable differences regarding the level of prescriptive details in codes & standards developed in the different countries, but the overall safety requirements are following the same principles as outlined in [1, 2].

The general philosophy in the design of pressurised components, structures and internals is to demonstrate that the function and the integrity of the component or structure is within the limits of the technical specifications or codes and standards applied.

As far as the pressure boundary is concerned, adequate precaution against the following failure mechanisms is deemed to have been taken, for example when the criteria specified in the corresponding safety standards (ASME, RCC-M, KTA) have been complied with:

(a) failure due to plastic collapse,
(b) failure due to excessive deformation,
(c) failure due to incremental deformation,
(d) failure due to instability,
(d) failure due to fatigue,
(e) failure due to brittle fracture,
(f) failure due to the effects of corrosion.

In the case of mechanisms (a), (b) and (c), the prevention of rupture and/or functional failure is of particular importance. In the case of precautions against fatigue (e), an attempt is made to prevent the initiation of a crack which, in most cases, with undetected further crack propagation, would lead to leakages.
Issues of instability (d) are generally of less importance in the case of thick-walled components and will not be considered any further here.
As a rule, precautions against the brittle fracture failure mechanism (e) are ensured by means of toughness requirements for materials and parts, processing requirements and the limitation of irradiation influences during operation. In individual cases, additional proof of brittle fracture safety is demanded, such as in the case of the reactor pressure vessel.

Protection against the effects of corrosion (f) is ensured by the choice of material, design requirements, individual specifications in the material and processing requirements as well as appropriate specifications for water chemistry and plant operation.

It is common practice that bounding loading conditions and their frequencies are specified and that the engineers apply the most practical methods in performing the necessary analyses and preparing the stress analyses report. With regard to safety relevant structures, codes and standards require an in-depth investigation covering all kinds of loading conditions specified in the design, defined e. g. as conditions of normal operation, abnormal operation, testing, and accidents which includes rare and very rare events. Major attention is normally paid to the operational and accidental loads, whereas loads arising from testing conditions sometimes receive less attention.

In performing the stress analyses, it is usual practice to determine a maximum stress level and range at a certain location or cross section of the component or structure and to neglect the time history of the stress. This practice was due to the limitations in computing time in the past which restricted the extensive calculation of the geometry-dependent distribution of loads (forces, moments etc.) and resulting stresses and strains. The same situation was more or less given for the fatigue analysis. Within the design stage, priority is given to the demonstration that the usage factor is below the criteria. In order to do this in the most efficient way, stress cycles have been grouped together in an intentional conservative way by taking the maximum stress range also for cycles with gradually or even considerably lower stress ranges. This practice is sufficient and effective to demonstrate conformance with the design criteria. However, it is not the best approach to characterise the real system and component with respect to the fraction of life time used.

3. APPROACH OF IN-SERVICE INSPECTION (ISI), TESTING, MONITORING AND SURVEILLANCE

The objective of in-service inspection and testing is to check the performance and to identify malfunction and degradation of systems, structures and components.

The common safety codes and standards have adopted similar approaches to fulfil the objectives by a mixture of tests and inspections. There are tests addressing the status of the pressure boundary in a global sense, proving adequate leak tightness and integrity. Typical tests of that type are the leak tightness test and the hydro test which is usually performed at a higher pressure than the operating pressure. Furthermore, interaction of the pressure boundary with the supports and the outside environment are checked visually by walk-downs during the outage for refuelling and partly during service. Complementary to the general tests, detailed non-destructive inspections are required at selected locations to detect indications to local degradation at an early stage. Most of the resources are allocated to the non-destructive inspections. To limit the corresponding efforts, the locations to be inspected are selected according to different criteria to achieve a so-called representative scope of inspection.

The basic idea is that the locations selected for inspections represent the typical manufacturing quality and that they are exposed to the typical environmental conditions and loads. Regarding the loads and environmental conditions, the codes and standards normally guides the user to select locations which are exposed to conditions which are more challenging compared to mean values. Regarding the manufacturing quality, weld regions are selected as locations of prior interest. Locations with limited degradations but accepted manufacturing quality are addressed specifically. In the ideal world, all important parameters are covered by the representative scope of inspection with equal statistical significance.
In the last ten years, risk-based methods have been increasingly applied in the selection of locations to be inspected by non-destructive techniques.

Complementary to this concept of ISI, which is limited in the final result to a yes or no situation (signals above or below registration level), monitoring and surveillance methods have been applied to follow the load characteristic of the component and the system and the material condition. With this additional information, available margins can be assessed before local damage occurs and measures can be taken or developed to avoid or reduce detrimental conditions.

In reality, it is quite difficult to define a representative scope of ISI. Constraints are given by the incompleteness of information regarding the local condition of the component, by the environment and loading, by inherent limitations of the inspection methods to differentiate between signals originated from the metallurgical structure or real defects, by limitations in the accessibility for inspection, and by radiation dose consideration besides the question related to the economic effort.

In the scope of inspection, the requirements for ISI, as outlined in the KTA Safety Standards [2, 3], distinguish between the safety classes, the materials and the components in a similar way as it is done in other safety codes and standards.

4. REVIEW OF OPERATING EXPERIENCE

4.1. General aspects

The evaluation of operating experience with light-water reactors demonstrates that the provisions taken to ensure the function and the integrity of the components have been successful in general, but the design provisions themselves are not sufficient to avoid degradations. Major factors contributing to the success are the proven technology available for pressurised components, the ability to inspect the components with non-destructive testing methods, to repair or exchange components where the safety margin is reduced to an unacceptable value and, last but not least, the awareness to address evolving problems in time wherever relevant information is distributed within the technical community.

The general approach followed in many countries to address relevant technical issues for long-term operation contains as major elements:

- continuous evaluation of operating experience to identify changes in the reliability of systems, structures and components,
- extended plant monitoring to enhance the understanding of system behaviour and load conditions of the components,
- evaluation of safety margins for lower bound conditions by experimental and / or analytical R&D programmes,
- generic studies to identify areas where knowledge gaps exist, which show a potential for future problems,
- replacement of components susceptible to degradation to enlarge safety margins,
- enforcement of technical requirements in codes and standards to avoid recurrence of failures due to non-optimised technical solutions.

In addition, it is important to mention the role of “good operating practice“ which is characterised by a spirit of caution and the awareness that any deviation from the proven path may be the origin of future problems and therefore needs to be carefully planned and studied before it is implemented. This spirit relies to a considerable extent on the open mindedness of all parties involved and a free exchange of technical information.

4.2. System-/ Component-oriented review

The review of the world wide operating experience regarding important observations on system and component behaviour is more or less restricted to the primary pressure boundary and parts of the steam and feedwater system inside the containment and safety-relevant systems connected to the primary pressure boundary of PWRs and BWRs. The major sources used for this review are [5, 6, 7].

Summary PWRs

- no failure of large components,
- limited number of pipe breaks and leaks at small-bore piping, generally with a diameter less than 50 mm (< 100 events),
- limited number of leakage events at residual heat removal and safety injection systems, leak rate generally less than 20 l/min,
- leakage events at reactor pressure vessel head control rod penetrations (Inconel 600 material), replacement of about 20 vessel heads,
- leakage events at seals and valve packings with boron acid corrosion attack at the outside of ferritic parts of the pressure boundary and bolt shafts at flanges,
- less than 20 steam generator tube ruptures,
- one leakage event at a steam generator secondary shell,
- many leakage events at steam generator tubes of Inconel 600 material due to stress corrosion cracking, a large percentage of steam generators (~ 40 %) using this material for the steam generator tubes have already been replaced,
- some reactor pressure vessels exhibit a considerable embrittlement in the beltline area (mainly restricted to weld material having higher content of Cu, P, Ni and fluences > 2 x 10^{19}),
- primary piping and housings using duplex-type or stand-cast unstabilised austenitic material exhibit a considerable reduction in fracture resistance properties due to precipitations.

Regarding cracks in the pressure retaining walls there is no reliable data base available covering the world wide experience. Some relevant events with respect to the integrity analyses are extended cracks in some steam generator secondary shells at the water / steam level, extended cracks at some steam generator primary side cladding, underclad cracks of limited size in the belt-line of some reactor pressel vessels, cracks in austenitic piping in location of thermal stratification and fluctuations due to fatigue.

Summary BWRs

- no failure of large components,
- limited number of pipe breaks and leaks at small-bore piping, generally with a diameter less than 50 mm,
- a considerable number of through-wall cracks at medium-size piping, leak rate generally very low,
- replacement of piping
  - primary piping being affected by intergranular stress corrosion cracking, extent of replacement depends on the strategy of mitigation measures and inspection efforts,
  - primary piping with low fracture resistance properties in welds of ferritic steels with higher strength (specific case for steam and feedwater piping of German BWRs),

- cracks in the feedwater reactor pressure vessel nozzle area of some designs which required repair action,
- cracks in the inner cladding of a few reactor pressure vessel heads due to stress corrosion (mostly solved by grinding out the affected area),
- for BWRs, the intergranular stress corrosion of austenitic material of the reactor pressure vessel internals is an area where further repair and replacement actions may be expected.

In general, it can be stated that the extent of repair and replacement is higher than expected during the design stage for some of the components of PWRs and BWRs. But the developments in the techniques applied for repair and replacement have compensated these deficiencies to a state that the overall plant reliability and safety is not compromised.

4.3. Damage-mode oriented review

All indications being detected in the course of in-service inspections, surveillance or maintenance work have to be assessed with respect to the type of defect, e. g. crack, wall thinning, surface attack. In most cases, a root cause analysis is necessary to get a clear picture of the damage mechanism causing the defect and to develop mitigation measures.

The major damage mechanisms acting on the systems and components are
- corrosion-assisted cracking and wall thinning involving a number of different mechanisms,
- embrittlement,
- fatigue, high- and low-cycle,

as well as interactions or synergism.

A summary covering the major international events and events for the German NPPs for the dominant damage mechanisms is given in Figures 1 to 4.

Reviewing the damage modes being active at reactor coolant pressure boundary components, it can be stated that the provisions taken in the selection of materials, the design as well as the manufacturing of the component did not address the corrosion aspects to a sufficient degree of detail to avoid degradations at various locations. Figure 5 illustrates this as an example for the piping of the reactor coolant pressure boundary of PWRs and BWRs being operated in Germany. The knowledge about corrosion mechanisms is mainly of an empirical nature and the capability of models - to predict the initiation and evolution of corrosion related defects - is still limited. This area will require considerable attention also in the future. The limitations in predictive capabilities and differences in expert opinions also contribute to the reluctance to feed the experience back to the design safety codes and standards.

5. REFLECTION OF THE PRESENT CONCEPTS OF ISI, MONITORING AND SURVEILLANCE

In view of the many technical issues being involved, only three questions of a more generic nature will be addressed. These are:
- Are the relevant damage mechanisms being addressed in our codes and standards?
- What are the criteria to develop a representative scope of inspection?
- How to maintain a sufficient level of information for a decreasing number of nuclear power plants in operation?

The discussion of these questions is based on the safety codes and standards, current practice and operational experience with the nuclear power plants in Germany. Most of the 19 nuclear power plants being presently operated are in the second half of their projected life time, this means that a broad base of operating experience is available.

Other interesting questions like capability of non-destructive testing techniques are left aside.

5.1. Are the relevant damage mechanisms addressed in our codes and standards?

The present potential and/or active damage mechanisms being relevant to the reactor coolant pressure boundary components are listed in table 1. This list is more or less being commonly used in the technical community for PWR and BWR nuclear steam supply systems. The second column in the table identifies to what extent these damage mechanisms are explicitly or implicitly addressed in the German design codes and standards. There is an obvious distinction showing clearly that the fracture modes and fatigue are explicitly addressed in the codes, whereas the corrosion-related mechanisms are more implicitly addressed. It needs to be recalled that certainly all stress- and strain-related corrosion mechanisms need a certain stress or strain value, and therefore the stress and strain limits applied to protect the component from different fracture modes are also relevant in the analysis of the potential corrosion mechanisms. In our more and more specialised world, mechanical engineers, material scientists, physicists and chemists are working on their individual tasks. The chain which should connect all the disciplines in the right way is sometimes not strong enough. The KTA safety standards as well as some of the other internationally used safety standards and codes have their strength in the chapters dealing with the specific requirements related to materials, design, manufacturing and ISI and give little guidance regarding the overall interdisciplinary integrity aspects. Expert systems which should have their strength to guide the user though all the links between the different disciplines are not yet developed to a sufficient degree. Therefore, the strength of the chain to connect all the disciplines in the right way with regard to integrity depends to a large extent on the availability of experienced professionals.

The third column of the table identifies to what extent the damage mechanisms are being addressed specifically in the codes and standards for in-service inspection, monitoring and surveillance. In a similar way, as we have seen it for the design codes and standards the level of stress and fatigue is more explicitly mentioned than the corrosion-related aspects. In this respect, again one has to remember that the level of stress or strain acting on the component is also important for the corrosion aspects. The surveillance of material properties for component sections above a certain neutron fluence is a standard requirement more or less from the beginning. The extension of the regular plant monitoring to the monitoring of local loads acting on the components was a developing process over the last 20 years. The change of the KTA safety standard to include requirements on local monitoring explicitly was part of the more recent changes of the safety standard. With this change, the incentive is given to take the option to stretch the inspection interval (from a stricter requirement to an interval which is closer to the international standard) in favour of a more extended local load monitoring.

Table 1Treatment of degradation mechanisms in the KTA safety standards for the reactor coolant pressure boundary (PWR + BWR)

degradation mechanism		**addressed by design standards**	**addressed by standards for ISI, monitoring, surveillance**
embrittlement	neutron activated	Explicitly	explicitly
	thermal activated	Explicitly	-
corrosion	IGSCC	implicitly partly explicitly	implicitly infrequently explicitly
	pitting +TGSCC		
	SICC		
	FAC		
	Boric acid corrosion		
fatigue	mechanical	Explicitly	explicitly
	thermal		
synergism	corrosion fatigue	-	-
	IASCC	-	-

GSCC: Intergranular Stress Corrosion Cracking
TGSCC:Transgranular Stress Corrosion Cracking
IASCC: Irradiation-Assisted Stress Corrosion Cracking
SICC: Strain-Induced Corrosion Cracking
FAC: Flow-Accelerated Corrosion

In summary, we can answer this question in the sense that more work has to be done to incorporate present knowledge on damage mechanisms in the codes and standards for design and in-service inspection and to strengthen the links between the different disciplines to formulate ISI and monitoring requirements for components more explicitly in view of the environmental conditions.

5.2. What are the criteria to develop a representative scope of inspection?

There are different governing criteria like safety, reliability or susceptibility of a component section to damage which could be used to develop a representative scope of inspection, monitoring and surveillance. In practice, all three criteria are used, but even at a very superficial level there is no consensus in the technical community how to weigh the different criteria in their quantity. Priority is given to safety, but also for this criteria there are differences in the practical application using either a fully integrated risk approach or looking mainly at the consequences of a failure. From the viewpoint of a practitioner dealing with component integrity issues, the main interest is to detect any damage at such an early stage that there are multiple choices of technical measures to solve the problem. It is quite easier to grind out a crack of a limited size before any weld repair action becomes necessary and to study in detail the technical and operational measures to avoid recurrence of the damage.

In the German KTA safety standard, the definition of the representative scope of inspection is governed by the safety importance of the component and the susceptibility of a certain component section to damage due to the operational loads. The importance to safety leads to different inspection scopes for the reactor coolant pressure boundary and the connected systems being isolated or isolable. The susceptibility of a component section to damage resulting from operation is addressed in a more general way. The tables in the safety standard which give guidance on the scope of inspection contain no specific explanations with regard to the criteria applied in the selection of the requirement. Looking at the tables as a whole, the governing criteria can be derived in the development of the inspection scope.

The governing criteria are the susceptibility of component sections and materials to local damage due to the operational conditions. The level of load is stated qualitatively. Individual damage mechanisms are addressed implicitly by requiring certain inspection, e.g. wall thickness measurements on elbows (erosion, corrosion) and enlarged percentage of welds in austenitic piping systems being operated at temperatures above 200 °C for boiling water reactors (intergranular stress corrosion). The fact that all necessary inspections have to be performed with non-destructive testing techniques at a very sensitive level, ensures that any damage can be detected at a very early stage. A further aspect which is implicitly contained in the code requirements is the distribution of information resulting from inspections over the time. The principle is that all components which are single components, like the reactor pressure vessel and the pressuriser, have to be inspected at a rather short interval of five years. For all components where a number of identical components exist at a plant, such as steam generator, piping and housings, half of the components have to be inspected at a time interval of five years. In general, components have to be inspected within two inspection intervals, that means every ten years, which is the more common inspection interval in other countries. These requirements assure that there will be no extended duration without any information coming from inspections.

For the manufactured components it is assumed that the welds and base metal in the neighbourhood of welds is more susceptible to initiation of damage as compared to the other base metal sections. For this reason, these areas are selected as representative sensitive locations for inspection. The safety standard does not contain further guidance which would allow distinctions in inspection requirements in relation to the manufacturing related aspects. This is done at a plant-specific level. By requiring inspections of all terminal ends (connection between the piping and the main components), the complexity of the load and the manufacturing conditions is implicitly addressed. For certain non-destructive inspections, the orientation of the defect, which is the most critical one in relation to the load condition, is also given and by prioritising this defect orientation for the inspection, the load condition is taken as a selective criteria.

In some countries, there is a strong development to re-arrange the in-service inspections requirements by taking the risk as a governing criteria in determining a representative scope of inspection. A quantitative risk estimate can be performed with more precision today than 20 years ago, taking the benefit of all the operating experience and information resulting from inspection, monitoring and surveillance. It is important to note that even today, a significant portion of damages at the pressure boundary components is not being discovered by the regular ISI but in the course of maintenance activities or non-regular inspection being initiated by information connected to events at other systems of plants. Figure 6 illustrates as an example how damages in piping have been detected by different means, ISI, walk downs, maintenance work, etc. Unexpected damages occuring still at components which are operated for a long time. In the last years we have seen a number of events where damage at pressure boundary components was not discovered by regular in-service inspections for various reasons. Either the damage location was not foreseen for inspection, the signals in the previous inspection were not interpreted correctly or the inspection technique was not qualified for the specific crack orientation or the damage mechanism involved.

The latest ASME code case for risk-informed inspection gives much better guidance in the selection of a risk-based scope of inspection by reflecting more intensively the operating experience with respect to the different damage mechanisms in relation to system and material parameters. But we are still on the learning curve and the input data required for a more quantitative assessment of the probability of a damage occurring at a certain location is difficult to establish at least for a number of damage mechanisms. The application of models which are able to quantitatively distinct between sections of low and high failure probability requires a more sophisticated technical discussion of the governing criteria in the development of a representative scope of inspection and is therefore strongly supported. But the limitation of the present models have to be seen with the necessary spirit of caution and awareness and the element of random sampling to supplement the scope of inspection derived by criteria should be kept at a minimum level.

In summary, we can answer the question regarding the criteria to develop a representative scope of inspection not as sufficiently as we would like to do it. From the viewpoint of a practitioner dealing with integrity issues, the criteria for the selection of representative locations should mainly be based on the lessons learned from the operating experience. This means that each system has to be analysed to identify all relevant damage mechanisms as discussed in relation to the previous question. In the selection of component sections, parameters influencing the manufacturing quality like residual stress pattern, hardness, sensitisation etc. should be equally rated as operation conditions, which are more challenging within the regular operating pattern like elevated temperatures, unfavourable water chemistry conditions, local load variations etc. Materials and material combinations should be selected on the basis that variations in chemical composition, mechanical and fracture properties and thermal treatment conditions are screened to identify conditions where the likelihood of local damage may be increased. These aspects are normally treated in the plant-specific inspection requirements and are specifically important if an active damage mechanism is present in the plant operation. A further important aspect is the information density available from the operating experience of light-water reactors. As we have seen in the past, a large percentage of relevant damages which occurred in light-water reactors are connected to a certain material, specific water chemistry, specific design conditions within one generation or to a certain manufacturing practice. It is therefore important to identify the level of information available on a plant-specific and a generic data base. The question of safety as a governing criteria is adequately covered by the classification system. Risk-based methods will play a more important role in the future. They have their specific strengths in areas where the models are verified by means of a large database. By definition, the “unknown” effect cannot be calculated. Therefore, the selection of a small number of locations to be inspected by random sampling is of value, but only if applied for a larger family of components.

5.3. How to maintain a sufficient level of information for a decreasing number of nuclear power plant components?

This question is of special importance to the nuclear power plants in Germany where, due to the governmental agreement with the utilities, almost half of the units will be taken out of service in the next ten years. In general, this means that in ten years from now out of two generations of boiling water reactors only one unit from the construction line 69 will be in service and two units of the construction line 72. For PWRs, only units of the third and fourth generation remain in service.

Assuming that the rates of abnormal events in the next ten years will be similar to those in the last ten years, they will decrease considerably faster than the number of plants in operation. Figure 7 illustrates this for passive mechanical components. Having very low event rates is positive in principal, but it also poses a considerable challenge to maintain an adequate level of trained professionals to assess each case adequately. Looking at the density of information which results from all of the inspections performed, we see a considerable change because the number of locations being inspected will be greatly reduced. This is illustrated in Figure 8 as a rough estimate for the reactor coolant pressure boundary piping. There are two main reasons for the accelerated reduction of information. In the optimised design of the PWRs of the third and fourth generation, the number of welds in the components is greatly reduced compared to similar components in the first and second generation of plants and the inspection requirements of the safety standard prescribe to inspect a certain percentage of all the welds . If one looks at the situation component by component, the picture is quite different. For the reactor pressure vessel, the decrease of information almost goes along with the decrease of the number of plants. The largest difference concerns the primary piping where the highest reduction in number of welds was achieved.

As discussed with the previous question, the available knowledge about damage mechanisms acting on pressure boundary components is mainly based on the world-wide operating experience and related research. The distribution of information is illustrated qualitatively in Figure 9. It is important to judge the relevance of the world-wide operating experience in relation to the specific plants being operated in one country. In the past, we have seen many times that due to the longer operation time and partly more unfavourable conditions existing at plants operated in the U.S.A. we could use the lessons learned for our plants. In order to complement the reduced national database by additional information, it is therefore important to identify these for all materials, specific design solutions and operational conditions of a system with the highest similarity being operated somewhere else. In cases where a single technical solution or specific operating condition has no similarity to a larger family of components, the scope of inspection, as presently required by the safety standards, may not be sufficient anymore. This is not a new issue at all and has been dealt with for the plant-specific scope of inspection in the past, but it is important to deal with this on a generic level in time. Another technical approach to limit the efforts regarding in-service inspections is the intensification of the plant monitoring. Drawing special attention to plant monitoring can provide information on plant system behaviour more precisely than non-destructive testing methods. Therefore, this option has already been incorporated in the latest version of the safety standard and could even be further developed.

In summary, there are different approaches available to maintain a sufficient level of information in a situation where the number of nuclear power plants is decreasing. The first choice is to intensify plant monitoring to balance the reduced level of generic information coming from similar components. In addition, the effort in the evaluation of the international operating experience should be strengthened to study the relevance of information available from events, root cause analyses and inspections. Furthermore, it is necessary to strengthen technical co-operations to ensure that all issues can be treated according to the state of the art.

6. CONCLUDING REMARKS

The objective of in-service inspection, testing, monitoring and surveillance is to check the performance and to identify malfunction and degradation of systems, structures and components. Reviewing the operating experience we can conclude that the design provisions taken are adequate to avoid gross structural failures but they are not sufficient to ensure a lifetime of all of the components coherent with the design goals. We see strong indications that some of the materials and manufacturing procedures used are not robust enough to avoid corrosion related degradations.

In general, it can be stated that the extent of repair and replacement is higher than expected during the design stage for some of the components of PWRs and BWRs. But the developments in the techniques applied for repair and replacement have compensated these deficiencies to a state that the overall plant reliability and safety is not compromised.

Regarding the codes and standards applied for in-service inspection, testing, monitoring and surveillance more work has to be done to incorporate present knowledge on damage mechanisms and to strengthen the links between the different disciplines to formulate ISI and monitoring requirements for components more explicitly in view of the environmental conditions.

From the viewpoint of a practitioner dealing with integrity issues, the criteria for the selection of representative locations should mainly be based on the lessons learned from the operating experience. In the selection of component sections, parameters influencing the manufacturing quality like residual stress pattern, hardness, sensitisation etc. should be equally rated as operation conditions, which are more challenging within the regular operating pattern like elevated temperatures, unfavourable water chemistry conditions, local load variations etc. Risk-based methods will play a more important role in the future. As experience show us the selection of a small number of locations to be inspected by random sampling is still of value.

To maintain a sufficient level of information in a situation where the number of nuclear power plants is decreasing it may be necessary to intensify plant monitoring to balance the reduced level of generic information coming from similar components. In addition, the effort in the evaluation of the international operating experience should be strengthened to study the relevance of information available from events, root cause analyses and inspections. Furthermore, it is necessary to strengthen technical co-operations to ensure that all issues can be treated according to the state of the art.

REFERENCES

1. INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Defence in Depth in Nuclear Safety, INSAG-10, IAEA, Vienna (1996)
2. INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safety Standards Series, Safety of Nuclear Power Plants: Operation, Requirements No. NS-R-2, Vienna (2000)
3. KTA Safety Standards 3201.1-.4
4. KTA Safety Standards 3211.1-.4
5. U.S. Nuclear Regulatory Commission, Office for Analysis and Evaluation of Operational Data, Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 - 1995, Washington, DC 20555-0001
6. Frank Michel, Hans Reck, Helmut Schulz, Experience with piping in German NPPs with respect to ageing-related aspects, Nuclear Engineering and Design 207 (2001) 307-316
7. INTERNATIONAL ATOMIC ENERGY AGENCY / NUCELAR ENERGY AGENCY, Nuclear Power Plant Operating Experiences, Incident Reporting System, 1996 - 1999

Titel:
9810804x.p65
Erstellt von:
PageMaker 6.5
Vorschau:
Diese EPS-Grafik wurde nicht gespeichert
mit einer enthaltenen Vorschau.
Kommentar:
Diese EPS-Grafik wird an einen
PostScript-Drucker gedruckt, aber nicht
an andere Druckertypen.

Fig. 1 Intergranular stress corrosion cracking of BWR austenitic piping

Titel:
9905505x.p65
Erstellt von:
PageMaker 6.5
Vorschau:
Diese EPS-Grafik wurde nicht gespeichert
mit einer enthaltenen Vorschau.
Kommentar:
Diese EPS-Grafik wird an einen
PostScript-Drucker gedruckt, aber nicht
an andere Druckertypen.

Fig. 2 Strain-induced corrosion cracking in ferritic steel piping of German NPPs with BWRs

Fig. 3 Thinning in PWR carbon steel piping due to flow-accelerated corrosion

Fig. 4 Thermal fatigue of piping in PWR plants

Fig. 5 Shares of the different damage mechanisms concerning safety-relevant piping in German PWR and BWR plants (1974-1998)

Fig. 6 BWR: Piping Events in Nuclear Heat Generation Systems and Reactor Auxiliary Systems (1974 - 1998)

Titel:
0006801x.p65
Erstellt von:
PageMaker 6.5
Vorschau:
Diese EPS-Grafik wurde nicht gespeichert
mit einer enthaltenen Vorschau.
Kommentar:
Diese EPS-Grafik wird an einen
PostScript-Drucker gedruckt, aber nicht
an andere Druckertypen.

Fig. 7 Reportable Events Affecting Passive Mechanical Components in German NPPs with PWR and BWR (as at December 1998)

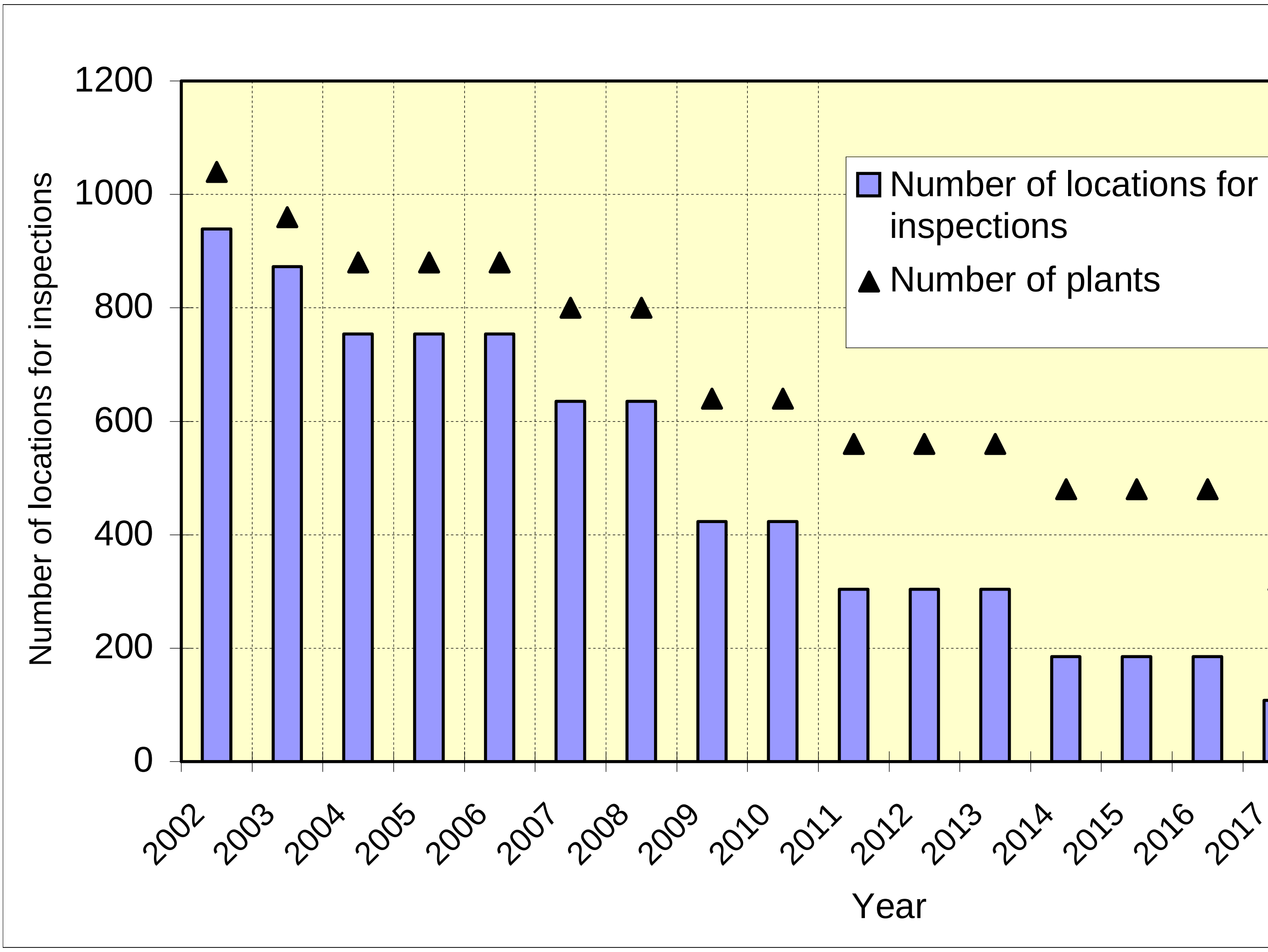

Fig. 8 Locations for inspections allocated to PWR plants in operation

Titel:
0113514x.p65
Erstellt von:
PageMaker 6.5
Vorschau:
Diese EPS-Grafik wurde nicht gespeichert
mit einer enthaltenen Vorschau.
Kommentar:
Diese EPS-Grafik wird an einen
PostScript-Drucker gedruckt, aber nicht
an andere Druckertypen.

Fig. 9 Sources of Information Regarding Operating Experience
Qualitative Distribution Today and Ten Years from Now