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Organisation de Coopération et de Développement Economiques
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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

NEA/CSNI/R(2006)8
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SURVEY ON PRIMARY WATER STRESS CORROSION CRACKING (PWSCC) AND NICKEL-BASED ALLOY

Final version - September 2006

English text only

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COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of senior scientists and engineers, with broad responsibilities for safety technology and research programmes, and representatives from regulatory authorities. It was set up in 1973 to develop and co-ordinate the activities of the NEA concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations.

The committee's purpose is to foster international co-operation in nuclear safety amongst the OECD member countries. The CSNI's main tasks are to exchange technical information and to promote collaboration between research, development, engineering and regulatory organisations; to review operating experience and the state of knowledge on selected topics of nuclear safety technology and safety assessment; to initiate and conduct programmes to overcome discrepancies, develop improvements and research consensus on technical issues; to promote the coordination of work that serve maintaining competence in the nuclear safety matters, including the establishment of joint undertakings.

The committee shall focus primarily on existing power reactors and other nuclear installations; it shall also consider the safety implications of scientific and technical developments of new reactor designs.

In implementing its programme, the CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA) responsible for the program of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health (CRPPH), NEA's Radioactive Waste Management Committee (RWMC) and NEA's Nuclear Science Committee (NSC) on matters of common interest.

FOREWORD

Nickel-based alloys are used in several primary pressure boundary components. Degradation of these components could lead to significant loss of safety margins as well as potential loss of coolant accidents (LOCA). For example, circumferential cracks discovered in the CRDM nozzles could result in a small to medium LOCA. The leakage of primary water could lead to significant degradation of the pressure boundary as seen in the wastage of low alloy steel at the Davis-Besse plant. Both the regulators and industry need data and information to develop effective inspection, repair, and mitigation strategies to avoid significant degradation and loss of safety margins. This is needed for alloys used in the existing components as well as alloys used for the replacements.

The Committee on the Safety of Nuclear Installations (CSNI), at its meeting in June 2003, agreed to conduct a survey to collect information that help to identify common needs and area of potential co-operative activities based on experiences, status of existing data and research programs, and regulatory requirements and practices in NEA member countries.

This report provides a summary of the results of the questionnaire on PWSCC and Ni-Based alloys. The main aim of the questionnaire was to obtain a comparison between the operating experience, inspection practices and acceptance criteria applied in the different countries. It was also considered of relevance to compare experience and practices with Alloy 690, which is being used to a large extent in replacement components, as well as research activities in this area. Participants in the questionnaire were encouraged to include any other information that they considered being useful for a better understanding of the PWSCC phenomenon.

The survey was focused on thick section components but any relevant information concerning for example steam generators or research activities which were not explicitly addressed in the questionnaire were included. The questionnaire is shown in full in annex 1.

The complete list of CSNI reports and the text of reports from 1993 onwards, is available on http://www.nea.fr/html/nsd/docs/

ACKNOWLEDGEMENTS

Gratitude is expressed to the delegates of the CSNI Working Group on the Integrity of Components and Structures for providing answer to the survey on PWSCC and Ni-Based Alloys. Special thanks to Dr. Kjell Pettersson (SKI, SWD) for synthesizing the responses and writing the report.

EXECUTIVE SUMMARY

The Integrity and Ageing of Components and Structures Working Group of the CSNI is responsible for work related to the development and use of methods, data and information to assess the behaviour of materials and structures. It has three sub-groups, dealing with the integrity of metal components and structures, ageing of concrete structures, and the seismic behaviour of structures

The CSNI has an active program addressing technical aspects of integrity and ageing of metal components and structures. Plant ageing management aspects gained increasing attention over the last years. The CSNI has developed numerous technical studies to assess the impact of ageing mechanism on the safe and reliable operation of nuclear power plants. Considerable information has been generated on Stress Corrosion Cracking related events, including piping and component failure- OPDE Project, and methodologies exist for SCC inspections and some mitigating measures. Currently, the OECD-NEA SCC and Cable Ageing Project (SCAP) will consolidate the acquired knowledge and experience into commendable practices.

The majority of the nuclear power plants currently in operation are Light Water Reactors. There have been several instances of primary water stress corrosion cracking in the nickel-based alloys and weldments. Recent examples include cracking at safe ends of primary loop piping and at reactor vessel head penetrations. While considerable research work has been ongoing for the steam generator tubing, there is an incomplete understanding of susceptibility of the thick sections. In addition to the understanding of degradation mechanisms, there is a need for data on crack initiation, crack growth rates, stress analysis of welded assemblies of nickel-based components, and efficacy of NDE techniques. This is vital for defining appropriate inspection techniques and frequency to avoid potential breach of the primary pressure boundary. This information is also essential for consideration of probability of crack detection, leak-before-break concepts, leakage-detection requirements, and risk assessments.

Nickel-based alloys are used in several primary pressure boundary components. Degradation of these components could lead to significant loss of safety margins as well as potential loss of coolant accidents (LOCA). For example, circumferential cracks discovered in the CRDM nozzles could result in a small to medium LOCA. The leakage of primary water could lead to significant degradation of the pressure boundary as seen in the wastage of low alloy steel at the Davis-Besse plant. Both the regulators and industry need data and information to develop effective inspection, repair, and mitigation strategies to avoid significant degradation and loss of safety margins. This is needed for alloys used in the existing components as well as alloys used for the replacements.

At the meeting of the Metal Subgroup of the Integrity and Ageing of Components and Structures (IAGE) Working Group, held in Paris on 15th – 16th of April 2003, presentations made by the US NRC and CSN representatives highlighted the large concerns about PWSCC in nickel based alloys used in the reactor vessel head penetrations and other components, in particular Alloy 600 and its associated welds. Consequently, the CSNI, at its meeting in June 2003, agreed to conduct a survey to collect information that help to identify common needs and area of potential co-operative activities based on experiences, status of existing data and research programs, and regulatory requirements and practices in NEA member countries. It was also considered of relevance to compare experience and practices with Alloy 690, which is being used to a large extent in replacement components, as well as research activities in this area. Participants in the questionnaire were encouraged to include any other information that they considered being useful for a better understanding of the PWSCC phenomenon.

The survey was focused on thick section components but any relevant information concerning for example steam generators or research activities which were not explicitly addressed in the questionnaire were included. The questionnaire is shown in full in annex 1.

Broadly speaking there are two types of pressurized water reactors in the world, the western PWRs and the VVER reactors developed in the Soviet Union. As indicated in answers to the questionnaire from Finland, the Czech Republic, and the Slovak Republic where VVERs are used, there are no nickel based alloys in these reactors, at least not in the locations covered by the questionnaire.

The following general conclusions can be drawn from answers to the questionnaire:

It is clear from the questionnaire that many of the older pressure vessel heads with Alloy 600/182 head penetrations have been replaced and that more replacements are anticipated in the future. The majority of replacement heads have or will have Alloy 690/152 head penetrations. The reason for the choice is given as either recommendation from others, literature, or the good experience with Alloy 690 as a steam generator tube material. In addition laboratory testing has shown that Alloy 690 is much more resistant to PWSCC than Alloy 600. In most of the few cases when Alloy 600/182 have been chosen for the head penetrations no reason for the choice has been given, probably because that particular question was missing in the questionnaire. In one case, Davis Besse, the choice was opportunistic, there was a head available on short notice.

Inspection practices vary between countries. As a general rule most countries apply ASME XI rules for the components with Alloy 600/182 material combinations. However since experience of cracking accumulated most countries have reduced the inspection intervals for the most critical components. In many cases replacement heads will be inspected according to the same rules as the older heads.

ASME XI is also the basis for acceptance criteria in most countries although they may have been slightly modified when they have been included in national regulation. Especially for head penetrations there has been a need to apply more stringent criteria or standards for acceptance of cracks. This need is expressed in a rather detailed US instruction on how to deal with cracks in vessel head penetrations.

Several questions dealt with *crack initiation*. It is clear from the answers that nobody attempts to predict crack initiation as such, except perhaps in a small number of cases. However the American approach of calculating EDY, effective degradation years is an indirect way of expressing the probability that cracks might have initiated and it is based on an Arrhenius approach. It does not contain any stress dependence. Therefore as a general rule residual stresses do not play a role in estimates of *crack initiation.* However residual stresses are often calculated but then perhaps for the purpose of flaw tolerance analysis or risk informed inspection.

For flaw tolerance analysis a crack growth law is needed. Most countries use the so called Scott equation both for Alloy 600 and Alloy 182. However for the latter a multiplication factor of five is used in comparison with values for Alloy 600. It seems to be a general feeling that more data is needed in order to get more confidence in the crack growth laws used. A couple of countries have indicated that they use another crack growth law than the Scott equation.

A main objective of the questionnaire was to identify common needs and potential co-operative activities. For the utility side such activities are already going on through the various owners groups and EPRI. The question is whether or not similar activities are needed for the national regulators. One could have in mind both a coordination of regulation and a collaboration of research. With regard to regulation the questionnaire indicates that the regulation already is fairly well coordinated. The small differences that exist may well be present in order to serve special national needs. With regard to research those who have answered the questionnaire have not indicated much need for more research. This may however be a consequence of the fact that most questions on research specifically addressed *crack initiation* while the equally important question on crack growth data for Alloys 690/152 was never asked.

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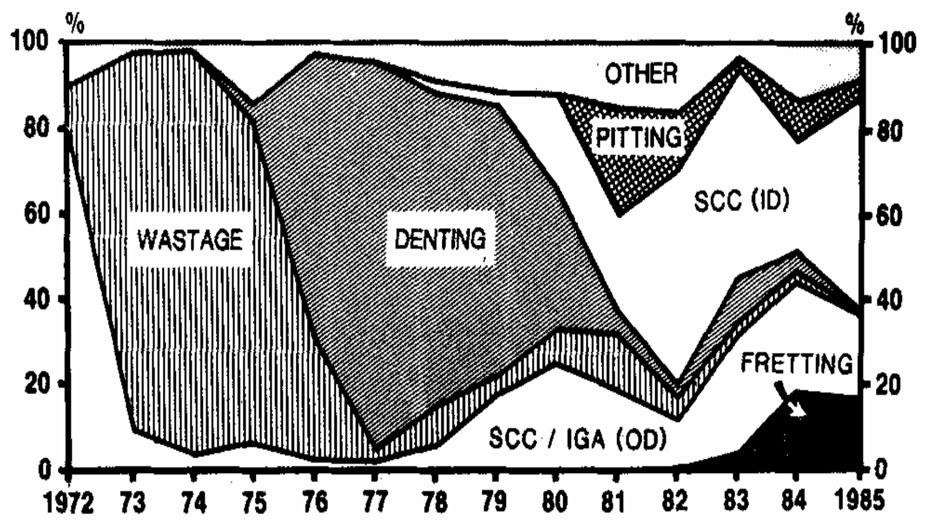
1. BACKGROUND

Despite early reports on SCC susceptibility of Alloy 600 to stress corrosion cracking (SCC) failures in water environments similar to the environment of the primary system of pressurized water reactors (PWR) [2], the alloy was used almost universally as steam generator tube material in early PWRs. As is clear from a review by Berge in 1987 [3] it also took some time before primary water stress corrosion cracking became a major cause of steam generator tube failures in PWRs, Figure 1.

Figure 1. **Causes of steam generator tube plugging in Western countries up to 1985** [3].



But alloy 600 and its corresponding weld metal alloys, Alloy 82 and Alloy 182, were also used in several other locations in the PWR. In 1991 two reports on cracking in Alloy 600 components were published at the Fifth Symposium on Environmental Degradation in Nuclear Power Systems – Water Reactors. In France cracking had been observed in instrumentation nozzles made of Alloy 600 in the pressurizer [4]. The report from the United States dealt with cracks in heater sleeves of Alloy 600, also in the pressurizer [5]. The American report also mentioned cracks in instrumentation nozzles.

At about the same time as these reports were presented, September 1991, a leak was detected in the vessel head of the French reactor Bugey 3 in connection with a hydro test [6]. The location of the leak was one of the penetrations for the control rod drive mechanism, Figure 2. Subsequent examination of the inside of the penetration showed numerous cracks in the Alloy 600 tube which were the likely cause of the leak. The likely cause of the cracks was determined to be primary water stress corrosion cracking (PWSCC) of Alloy 600 based on the appearance of the fracture surface and the intergranular crack path observed in metallographic cross section. The cracks were axial and it was subsequently determined that the residual stresses caused by welding the penetration to the vessel head were larger in the circumferential direction than in the axial direction. Cracks were also observed in the Alloy 182 weld, the J-groove weld, and this cracking was identified as an alternative leak path.

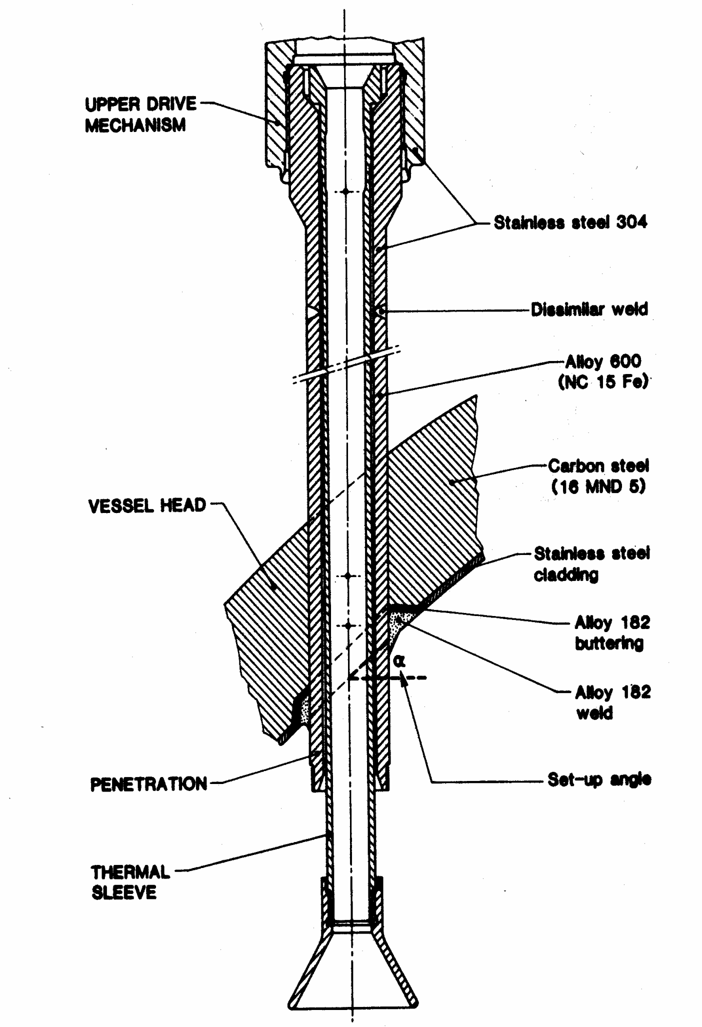


Figure 2. **Vessel head penetration**

The observed leak in Bugey 2 led to inspection of penetrations in other vessel heads both in France and abroad. It soon became clear that the vessel head penetration cracking was a generic problem for PWRs with head penetrations of Alloy 600 welded with Alloy 182. A large number of vessel heads have subsequently been replaced by new heads, often because they were found to have cracks, but also in many cases as a preventive measure. A few years after the first leak was detected in Bugey 3 circumferential cracks in the nozzles were detected in a couple of American PWRs. These cracks are more of a safety concern than the axial cracks because they can more easily lead to unstable failure of the nozzle [7].

Another important application of Alloy 82 and Alloy 182 is in the welding of so called safe ends of piping to the nozzles of the pressurized components. The weld typically consists of a buttering of Alloy 182 to the low alloy ferritic steel of the nozzle which is welded with Alloy 82 or Alloy 182 as filler material to the safe end made of austenitic stainless steel. There are a number of recent cases of PWSCC in this type of weld where the crack has followed an interdendritic path in the weld material [8, 9]. One of the problems with the assessment of the cause of failure in welds of Alloys 82 and 182 is that they are prone to hot cracking during welding and it can sometimes be difficult to distinguish between hot cracking and the interdendritic path of PWSCC in these materials.

An important point in the assessment of the safety of components made of Alloys 82, 182 and 600 is the temperature of operation. It has been demonstrated conclusively that crack growth rates during PWSCC of nickel based alloys is temperature dependent as shown in Figure 3 and that the temperature dependence can be characterized by an Arrhenius expression with a constant activation energy [1].

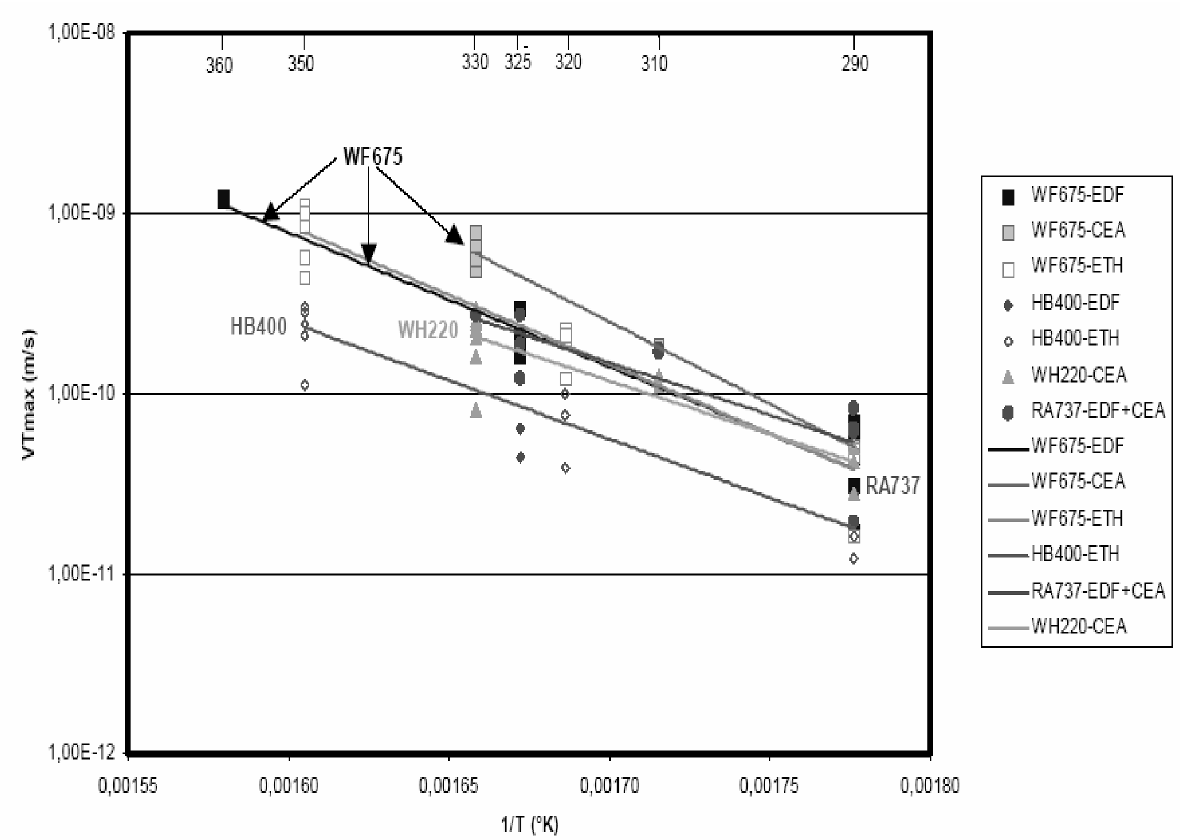


Figure 3. **Results of crack growth tests on different variants of Alloy 600** [1]

Figure 3 gives an activation energy of 130±20 kJ/mol in agreement with other determinations. A similar activation energy has been determined for crack growth in Alloy 182.

Figure 4 from a recent review [10] of primary water cracking of Alloys 82/182 in PWRs summarizes the locations in the system where PWSCC of nickel based alloys is a concern. A more extensive review on the problem of PWSCC from a USA perspective has recently been issued by the NRC as a NUREG report [11].

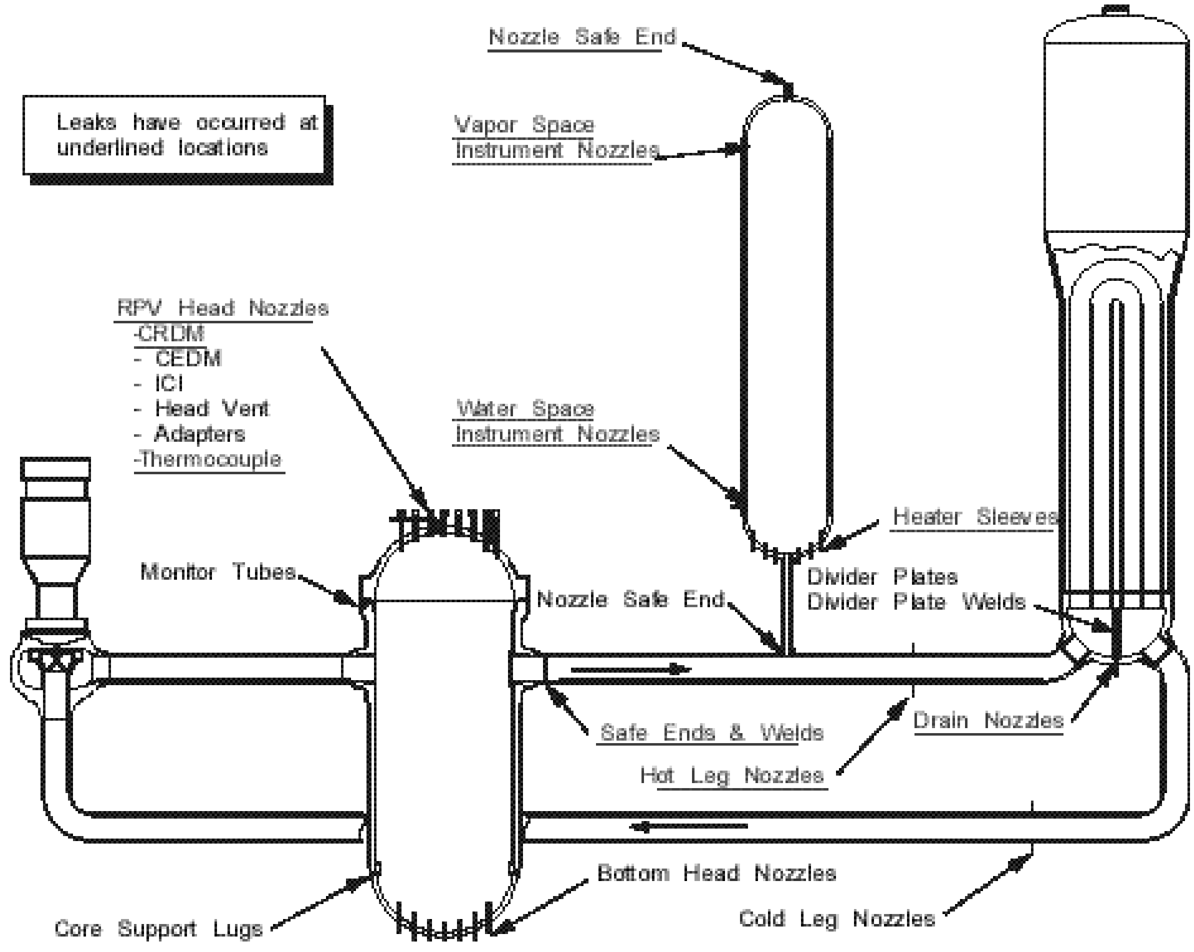


Figure 4. **Locations in the PWR where PWSCC may occur**

2. THE QUESTIONNAIRE

At the meeting of the Metal Subgroup of the IAGE Main Group, held in Paris on 15th – 16th of April 2003, presentations made by the US NRC and CSN representatives highlighted the large concerns about PWSCC in nickel based alloys used in the reactor vessel head penetrations and other components, in particular Alloy 600 and its associated welds. It was agreed that a questionnaire be prepared to collect information to help and identify the common needs and area of potential co-operative activities based on the experience, status of existing data and research, and regulatory practices in the various member countries. While considerable research work has been ongoing for the steam generator tubing, there is still incomplete understanding of susceptibility of the thick sections. In addition to the understanding of degradation mechanisms, data is needed on crack initiation, crack growth rates, stress analysis of welded assemblies of nickel-based components, and efficacy of NDE techniques. This is vital for defining appropriate inspection techniques and frequency to avoid potential breach of the primary pressure boundary. This information is also essential for consideration of probability of crack detection, leak-before-break concepts, leakage-detection requirements, and risk assessments.

A main aim of the questionnaire was to obtain a comparison between the operating experience, inspection practices and acceptance criteria applied in the different countries. It was also considered of relevance to compare experience and practices with Alloy 690, which is being used to a large extent in replacement components, as well as research activities in this area. Participants in the questionnaire were encouraged to include any other information that they considered to be useful for a better understanding of the PWSCC phenomenon.

The survey was focused on thick section components but any relevant information concerning for example steam generators or research activities which were not explicitly addressed in the questionnaire could be included. The questionnaire is shown in full in Appendix A. Alternatively the participants could use an EXCEL spreadsheet which was provided with the questionnaire.

3. A FEW GENERAL COMMENTS ON THE RESULTS OF THE QUESTIONNAIRE

Broadly speaking there are two types of pressurized water reactors in the world, the western PWRs and the VVER reactors developed in the Soviet Union. As indicated in answers to the questionnaire from Finland, the Czech Republic, and the Slovak Republic where VVERs are used there are no nickel based alloys in these reactors, at least not in the locations covered by the questionnaire. From the Power Reactor Information System (PRIS) of the International Atomic Energy Agency (IAEA) Table 1 of operating PWRs of the western type in the world has been compiled.

Country	Number of operating PWRs	Ni based alloys	NEA member
Belgium	7	Yes	Yes
Brazil	2	Not available	No
P. R. of China	7	Not available	No
France	58	Yes	Yes
Germany	13	No	Yes
Japan	23	Yes	Yes
R. of Korea	16	Not available	Yes
The Netherlands	1	No	Yes
Slovenia	1	Not available	No
South Africa	2	Not available	No
Spain	7	Yes	Yes
Sweden	3	Yes	Yes
Switzerland	3	No	Yes
Taiwan	3	Not available	No
United Kingdom	1	Yes	Yes
USA	69	Yes	Yes

The questionnaire was sent out to NEA member countries. Replies with information on Ni based alloys were received from Belgium, Japan, the Netherlands, Spain, Sweden and USA. However since the answer from the Netherlands only concerned one reactor which is soon to be shut down it will not be dealt with further in the present report.

4. SUMMARY OF THE RESULTS OF THE QUESTIONNAIRE

4.1 Indicate the pressure boundary components (not including steam generators tubes) that are made of Alloy 600 or Alloy 182 in the PWRs in your country. Also indicate the operating temperatures, and whether these are estimated or measured:

As with most of the questions in the questionnaire separate answers should be given for the following components:

Reactor Vessel Head Penetrations (base metal)

J-groove welds RVH Penetrations

Reactor Bottom Vessel Head Penetrations (base metal)

J-groove welds RBVH Penetrations

Vessel to Reactor Coolant System Nozzles (Hot leg)

Vessel to Reactor Coolant System Nozzles (Cold leg)

Pressurizer Nozzles

If it was considered relevant other components could be included in the replies.

In Belgium two 2 loop units have Alloy 600 in the penetrations welded with Alloy 182. The operating temperature of the head is 307 ˚C while the bottom head operates at 284 ˚C. The RCS nozzles do not have any Inconel. However the safety injection nozzles contain both Alloy 600 and Alloy 182 and operate at 284 ˚C.

Only one 3 loop unit has had the top head replaced and now have penetrations of Alloy 690 welded with Alloy 152. The operating temperature is 317 ˚C. The bottom head has the original Alloy 600 welded with 182 and operates at 287 ˚C. The nozzles do not contain any Ni based alloys.

Two 3 loop units have the original heads with Alloy 600 penetrations welded with Alloy 182. The operating temperature is 318 ˚C. The bottom heads are the original 600/182 combination operating at 299 ˚C. The RCS nozzles are welded with 182 and the hot leg operates at 330 ˚C while the cold leg operates at 294 ˚C. The pressurizer nozzles are welded with 182 with an operating temperature of 345 ˚C.

The two last 3 loop units have again had the original heads with Alloys 600/182. However these heads operate at 287 ˚C. The RCS nozzles are welded with 182 with a hot leg temperature of 324 ˚C and cold leg temperature of 282.5 ˚C. The pressurizer nozzles welded with 182 operate at 345 ˚C.

The 23 Japanese PWRs are all of MHI design with MHI as vendor. 11 vessel heads have penetrations made of Alloy 600 and operate at an estimated temperature of 289 - 294 ˚C. 11 replacement heads with Alloy 690 penetrations operate at an estimated temperature of 308-321 ˚C. There was one head with alloy 690 penetrations used since operation.

The J-groove welds are welded with Alloy 132 in the older heads and with Alloy 152 in the replacement heads. Alloy 132 has also been used for the other welds of the questionnaire, sometimes in combination with Alloy 82. The bottom head penetrations are all made of Alloy 600. Estimated operating temperatures are 284-289 ˚C for the bottom head and cold leg nozzle, 317-325 ˚C for the hot leg nozzle and 345 ˚C for the pressurizer.

Answers were given concerning 5 Spanish PWR, all of Westinghouse 3 loop design. 3 plants with Alloy 600 heads and Alloy 182 J-groove welds operate at an estimated temperature of 290 ˚C for the head while two plants with replacement heads with Alloy 690 penetrations and Alloy 52/152 J-groove welds operate at an estimated temperature of 318 ˚C. In the bottom head all plants have Alloy 600 penetrations with Alloy 182 J-groove welds. The bottom heads operate at about 290 ˚C. The RCS nozzles are all welded with Alloy 182 at the safe ends with the cold leg nozzle operating at 290-292 ˚C and the hot leg nozzle at 327 ˚C. The pressurizer nozzles are all welded with Alloy 82/182 and operate at about 345 ˚C.

The three Swedish PWRs are all Westinghouse 3 loop designs. Two of the units have had the top head replaced by a new head with 690/152 combination instead of the original 600/182. All three operate with 320 ˚C in the head. The bottom head has Alloy 600 penetrations welded with Alloy 182 and operates at a temperature of 286 ˚C. No material is indicated for the RCS nozzles but as mentioned previously in this report welds of Alloy 182 are involved. The hot leg temperature is 320 ˚C and cold leg temperature is 286 ˚C. It is indicated that alloy 600 is used in the pressurizer with an operating temperature of 343 ˚C.

In view of the large number of PWRs in the USA the answer to the questionnaire is quite extensive. Here it is summarized component for component since that is the format used in the answer to the questionnaire.

The PWR designs termed B&W Raised and B&W Lowered numbers six and one respectively. Both have Alloy 600 penetrations and operate at 316.5 and 318 ˚C respectively.

There are 29 Westinghouse 4 loop reactors which all have Alloy 600 head penetrations. Operating temperatures vary between 286 ˚C to 316 ˚C. 9 of the 13 Westinghouse 3 loop reactors also have Alloy 600 penetrations with operating temperatures between 292 and 314 ˚C. The other 4 have replacement heads with 690/152 penetrations. The five 2 loop reactors again have Alloy 600 penetrations and the range of operating temperatures is 305-311 ˚C.

14 reactors are of Combustion Engineering design, 11 of which are termed just Combustion Engineering and the other 3 Combustion Engineering Standard design. They all have Alloy 600 penetrations which operate at 308 – 313 ˚C.

In all the US PWRs the top head penetrations are welded with Alloy 182 and the operating temperatures are as given above.

All the US PWRs have the 600/182 combination in the penetrations of the bottom head. None of the answers included the operating temperature of the bottom head but it is presumably close to the cold leg inlet temperature of about 285 ˚C.

There are no US answers concerning the reactor coolant system nozzles.

All the reactors have Alloy 600 in parts of the pressurizer. The operating temperature is not given.

In the category “Others” examples are listed where failure has occurred. These cases are repeated under question 16.

4.2 Indicate if these components are inspected on a regular basis (frequency), or randomly and the percentage inspected each time. Indicate the inspection method used:

In Belgium the head penetrations are inspected every 2 or 3 years with UT+VT+ET for the plants operating with a hot head. The plants with a cold head are inspected every 5 years. For the plant with a replaced head no inspection program is in place yet. For all plants the J-groove weld is inspected once with mainly VT, in the form of high resolution video.

The bottom head penetrations are typically inspected visually by Bare Metal Visual inspections, they have been inspected once and the frequency of the following inspection is to be decided, except in 2 units where they have been inspected once with UT methods including weld/tube interface and for one of the units partially a second time, while the J-groove weld is not inspected at all

In the cases where the RCS nozzles have been welded with 182 the hot leg nozzle is inspected to 100 % every 5 years by UT from the inside. In the cold leg the same inspection is done every 10 years.

Pressurizer nozzles of Alloy 600 are inspected every 3 years by UT from the outside, except for small nozzles which are inspected every 5 years.

Inspection frequency for the head penetrations and J-groove weld vary between utilities in Japan. For 17 plants inspections are performed every fifth year on 100 % of the objects, while the other 6 are inspected to 100 % at every refueling outage. The method is in all cases bare metal visual inspection (BMV). In the bottom head the inspection frequency is generally 1 in 5 years, again with BMV. The RCS nozzles are inspected every tenth year with UT, PT and VT. The pressurizer nozzles are inspected visually every fifth year and with UT and PT every tenth year.

In Spain the reactor head penetrations are inspected according to NRC Order EA-03-009 which is described later. The J-groove welds are not inspected. The bottom head penetrations are inspected according to NRC Bulletin 2003-02. For the RCS nozzles and pressurizer ASME XI is used which requires inspection every tenth year. The methods used are VT and UT.

In Sweden the head penetrations and the J-groove weld in the top head are inspected every year by VT. The J-groove weld in the bottom head is not yet inspected but an inspection programme is in the planning stage. For the RCS nozzles the cold leg nozzle is inspected every 5 years by UT and ET. The pressurizer nozzles are inspected by UT every 3 years while the heater sleeves and instrument penetrations are inspected every year by VT.

In the USA all plants are treated in the same way. The inspection program varies depending upon a plant's PWSCC susceptibility category. Interim inspection requirements for reactor pressure vessel heads at pressurized water reactors were established in the NRC issued Order EA-03-009. The order indicated what type of calculation had to be performed to determine if a plant was in either a high, moderate, or low susceptibility category. The specific inspection plan that a plant must follow depends upon their susceptibility category as presented below. For those plants in the High category, RPV head and head penetration nozzle inspections shall be performed using the following techniques every refueling outage.

(a) Bare metal visual examination of 100% of the RPV head surface (including 360° around each RPV head penetration nozzle).

(b) Either:

(i) Ultrasonic testing of each RPV head penetration nozzle (i.e., nozzle base material) from two (2) inches above the J-groove weld to the bottom of the nozzle and an assessment to determine if leakage has occurred into the interference fit zone,

or

(ii) Eddy current testing or dye penetrant testing of the wetted surface of each J-Groove weld and RPV head penetration nozzle base material to at least two inches above the J-groove weld.

For those plants in the Moderate category, RPV head and head penetration inspections shall be performed such that at least the requirements of (c) or (d) shown below are performed each refuelling outage. In addition the requirements of (c) and (d) shown below shall be performed at least once over the course of every two refuelling outages.

(c) Bare metal visual examination of 100% of the RPV head surface (including 360˚ around each RPV head penetration nozzle).

(d) Either:

(i) Ultrasonic testing of each RPV head penetration nozzle (i.e., nozzle base material) from two (2) inches above the J-groove weld to the bottom of the nozzle and an assessment to determine if leakage has occurred into the interference fit zone,

or

(ii) Eddy current testing or dye penetrant testing of the wetted surface of each J-Groove weld and RPV head penetration nozzle base material to at least two inches above the J-groove weld.

For those plants in the Low category, RPV head and head penetration inspections shall be performed such that at least the requirements of (e) must be completed at least every third refueling outage or every five years, whichever occurs first. The requirements of (f) shown below must be completed at least once over the course of five years after the issuance of this order and thereafter at least every four refueling outages or every seven years, whichever occurs first.

(e) Bare metal visual examination of 100% of the RPV head surface (including 360˚ around each RPV head penetration nozzle)

(f) Either:

(i) Ultrasonic testing of each RPV head penetration nozzle (i.e., nozzle base material) from two (2) inches above the J-groove weld to the bottom of the nozzle and an assessment to determine if leakage has occurred into the interference fit zone,

or

(ii) Eddy current testing or dye penetrant testing of the wetted surface of each J-Groove weld and RPV head penetration nozzle base material to at least two inches above the J-groove weld.

For all plants in every category, visual inspection shall be performed during each refueling outage to identify potential boric acid leaks from pressure retaining components above the RPV head.

4.3 Indicate the Code, Regulation, Specific Regulatory Body Requirement, domestic or foreign operating experience, etc., governing inspection of these components:

In Belgium the inspection requirements on the components with Ni-based alloys, with the exception of the J-groove welds, are based on proposal from utility approved by safety authorities, based on previous inspection results and specific evaluations. It is also noted that US rules are generally applicable in Belgium (ASME XI ed. 1992 for ISI) but deviations are possible in agreement with Belgian safety authorities (sometimes less severe, sometimes more).

In Japan all answers refer to Specific Regulatory Body Requirement (NISA-163-a-03-1, 2003)[1)], (JSME S NA1-2002)[2)], etc. As of June 2005, the revised Regulatory Body Requirement of NISA-163-a-03-1 was issued as the NISA-163-a-05-2[3)].

1) NISA-163-a-03-1, "Inspections of components fabricated with Ni based alloy in reactor coolant pressure boundary in PWR", Nuclear and Industrial Safety Agency (NISA), December 2003

2) JSME S NA1-2002, "Code for Nuclear Power Generation Facilities, Rule on Fitness-for-Service for Nuclear Power Plants", The Japan Society of Mechanical of Mechanical Engineers, October 2002

3) NISA-163-a-05-2, "Inspection of components fabricated with Ni based alloy in reactor coolant pressure boundary in PWR", Nuclear and Industrial Safety Agency (NISA), June 2005

For Spain the answer was given in section 4.2.

In Sweden the applicable regulation is found in a document issued by the Swedish Nuclear Inspectorate called SKIFS 2000:2. Like in the Belgian system it does not regulate in detail. It is rather up to the utility to suggest appropriate procedures for subsequent approval by the authority.

For pressure vessel heads in the USA the applicable regulation is Order EA-03-009 establishing interim inspection requirements for reactor pressure vessel heads at pressurized water reactors issued on February 11, 2003 by NRC.

For the bottom head the inspections governing bottom mounted instrument nozzles are from Section XI of the ASME Code and also NRC Bulletin 2003-02.

Inspection of the pressurizer bottom penetrations is governed by Article IWB-2000 of Section XI of ASME Pressure and Boiler Vessel Code. The same code is applicable to other components.

4.4 What acceptance criteria are used for cracks in these components (codes, etc)?

In Belgium for the head penetrations, base metal, acceptance criteria have previously been based on own fracture mechanics analyses based on ASME XI App.A principles, in future rules proposed by NRC for RPH head penetrations and introduced as ASME App.X will be used. For the J-groove weld no analyses are expected to be needed or applicable.

For the other components is used ASME XI App. C for acceptable size, App.A for fracture mechanics analysis, using crack growth law based on info from literature.

In Japan the acceptance criteria are specified in JSME S NA1-2002,etc.

No cracks have yet been detected in Spain but it is anticipated that any cracks will be treated on a case-by-case basis. It is noted that the opinion of the Regulator is that criteria should be established for this degradation mechanism.

The Swedish questionnaire has the answer “flaw tolerance analysis” on all points. This is of course identical to the Belgian approach although details may differ with regard to data and calculation methods.

In USA the vessel head penetrations with J-grooves are treated as follows. Acceptance criteria of cracks are evaluated accordingly. Guidance was given in a letter from Richard Barrett to Alex Marion dated April 11, 2003. This letter was directed towards industry to aid in flaw evaluation. This guidance is shown below. While these guidelines are a tool that industry can use, most times if a crack is detected, no matter if the crack meets or does not meet the guidelines, it is usually repaired. There are some situations where cracks are not repaired. These situations are when the crack is subsurface or very shallow. Other than the guidance presented in the letter below there are acceptance criteria presented in Article IWB-3000 of Section XI of the ASME Boiler and Pressure Vessel Code.

Guideline:

A flaw growth analysis shall be performed on each detected flaw to determine its maximum growth due to fatigue, corrosion cracking, or both mechanisms, when applicable during a specified evaluation period. The minimum time interval for the flaw growth evaluation shall be until the next inspection.

All applicable loading shall be considered, including welding residual stress, in calculating crack growth.

The Flaw shall be characterized in accordance with the requirements of IWA-3400 of section XI including the proximity rules of Fig. IWA-3400-1 for surface flaws

A flaw shall be projected into both axial and circumferential orientations, and each orientation shall be evaluated. The axial orientation is the same for each nozzle, but the circumferential orientation will vary depending on the angle of intersection of the penetration nozzle with the head. As illustrated in Fig I. any flaws with ±10º of the plane formed by the J-groove weld root shall be considered pure circumferential flaws.

The location of the flaw, relative to both the top and bottom of the J-groove attachment weld, shall be determined.

The flaw shall be evaluated using analytical procedures to calculate the following critical flaw parameters:

a_f = the maximum depth to which the detected flaw is calculated to grow at the end of the evaluation period

a_i = the maximum length to which the detected flaw is calculated to grow at the end of the evaluation period

Acceptance Criteria

The calculated maximum flaw dimension at the end of the evaluation period shall be compared with the maximum allowable flaw dimensions in the table below.

Reactor vessel upper head penetration nozzle acceptance criteria (1) (3)				
Location	*Axial*		*Circumferential*	
	a_f	I_f	a_f	I_f
Below weld (ID) (2)	t	No limit	t	0.75 Circ. (4)
At and above weld (ID)	0.75t	No limit	repair	repair
Below weld (OD) (2)	t	No limit	t	0.75 Circ. (4)
At and above weld (OD)	repair	repair	repair	repair

Notes:

(1) Surface flaws of any size in the attachment weld are not acceptable

(2) Intersecting axial and circumferential flaws in the nozzle are not acceptable.

(3) t = wall thickness of head penetration nozzle

(4) 75 percent of the circumference

For all other components the Acceptance Standard is indicated in the Section XI ASME Boiler and Pressure Code in Article IWB-3000 Acceptance Standards.

4.5 For a unit that has been operating for some years at $T > T_{ref}$ (hot head) and then changes to cold head ($T < T_{ref}$), how do you consider this affects its PWSCC susceptibility?

To this question Belgium has answered “Not applicable”

Incidents of leakage from reactor pressure head penetration at Ohi unit 3 occurred May 2004 implied that insufficient buff polish on the surface of weld metal have induced the occurrence of stress corrosion cracking at relatively lower temperature. When there were no clear evidences that the buff polish has been done, reactor pressure vessel replacement were decided. They do not think that T cold conversion at the earlier stage is the only effective measure.

The Spanish answer is that in Spain the units that have changed to cold head operation have also replaced their reactor vessel heads so they do not have to cope with this scenario.

Sweden has answered “With a cold head the stratification risk increases with thermal variations as consequence.”

In USA the susceptibility is categorized as either high, moderate, or low as described in NRC's EA-03-009 order. An equation is utilized in order to determine a plant category. This equation is shown below:

$$EDY = \sum_{j=1}^{n} \left\{ \Delta EFPY_j \exp\left[-\frac{Q_i}{R}\left(\frac{1}{T_{head,j}} - \frac{1}{T_{ref}} \right) \right] \right\}$$

Where:

EDY = total effective degradation years, normalized to a reference temperature of 600 ˚F

$\Delta EFPT_j$ = operating temperature in years at $T_{head,j}$

Q_i = activation energy for crack initiation (50 kcal/mole)

R = universal gas constant (1.987cal/mole,K)

$T_{head,j}$ = 100% power head temperature during time period j (K = ˚C+ 273.15)

T_{ref} = reference temperature (600 ˚F = 588.8 K)

N = number of different head temperatures during plant history

All licensees use the following criteria to assign the RPV head at their facility to the appropriate PWSCC susceptibility category:

High (1) Plants with a calculated value of EDY greater that 12; OR

(2) Plants with an RPV head that has experienced cracking in a penetration nozzle or J-groove weld due to PWSCC

Moderate (1) Plants with a calculated value of EDY less than or equal to 12 and greater than or equal to 8 AND no previous inspection findings requiring classification as High

Low (1) Plants with a calculated value of EDY less than 8 AND no previous inspection findings requiring classification as High

The same classification is used for both Alloy 600 and Alloy 182.

4.6 Is the Arrhenius formula applied for the *crack initiation* determination of Alloy 600 and Alloy 182? Do you use another formula, or is this aspect not considered? Indicate the values of the different parameters in the formula(e).

In Belgium both the Arrehenius formula and other formulae are use in evaluations by the utility. The calculations are not used for any regulatory justification.

In both Spain and Japan the Arrhenius formula is used. In Spain the use is indirect in the sense that they rely on US regulation which uses the Arrhenius formula.

The formulae are not used in Sweden because no credit is currently taken for the crack initiation phase when considering the ISI programme.

In USA the Arrhenius formula is used for the vessel head penetrations as described under question 5. It is not considered for any other component.

4.7 On which of the following are these values based? (Several answers could apply):

(See Appendix A for the alternatives)

Belgium refers to the alternative “Bibliography” with the comment: Own evaluation of what is openly available (Publications in open literature, conferences, NRC guides or documents, non proprietary version of MRP documents or presentations)

No answer from Sweden on this point.

In Japan values are determined by joint research between utilities and the reactor vendor. Also bibliographic data is used.

In Spain the source for the values are codes and standards used in the country.

USA refers to research activities in the country with the comment: 50 kcal/mole is an accepted industry best estimate activation energy for SCC initiation in primary water environments based on research in this area.

4.8 For *crack initiation*, are the residual stresses measured, estimated, known, or not considered?

Belgium answered that the stresses are estimated based on fabrication procedure (stress relieved or not) and typical distributions from literature + limited own finite element analyses

In Japan the residual stresses are measured on mockups or calculated by elastic-plastic finite element analysis.

Residual stresses are not considered for crack initiation in Spain

No answer from Sweden.

For the vessel top head in USA residual stresses are estimated using finite element calculations. These results have been used to determine theoretically the likely location for crack initiation and growth. The results have also been used to determine if a repair technique is a viable solution. However, residual stresses are not taken into consideration when determining the susceptibility of a plant for PWSCC.

A slightly different formulation is used for the J-groove welds in the top head: Residual stresses have been used to theoretically determine the likely location for crack formation and growth. They have also been used to determine if a repair technique is a viable solution. However, residual stresses are not taken into consideration when determining the susceptibility of a plant for PWSCC.

For other components residual stresses are not considered.

4.9 Are you confident in the formula and values used for the *crack initiation* of Alloy 600 and Alloy 182, or do you think that further research activities should be carried out?

The feeling in Belgium is that further research is needed.

The Japanese are confident in their evaluation method but recognize the need for further research. It is said that there is a trend of increasing initiation frequency for PWSCC with operating year, operating temperature and applied stress and that the trend for Alloy 600 base metal seem to be relatively significant rather than for weld metals. There are a few equations for SCC initiation prediction. However, some of them are established by using data obtained from both base materials and weld metals and the equations are applied to both materials. Therefore, it is necessary to collect the knowledge and field data on the initiation of the SCC in the world and to promote the research activities on initiation time evaluation methods for both base metals and weld metals, as one of the countermeasures for ageing nuclear power plants [1)].

1) NISA Report, "Improvement of Ageing Management for Nuclear Power Plants (in Japanese)"issued by Nuclear and Industrial Safety Agency (NISA), August 2005

The Spanish regulator has the opinion that further research is needed for Alloys 82 and 182.

No comment from Sweden.

In the USA questionnaire the question is answered with regard to the vessel top head and it is noted that the crack initiation activation energy was provided in question 5. Its value has been evaluated in multiple experiments. Therefore, it seems to be a highly reliable value.

4.10 Are there any research activities being carried out in your country to refine on these values? Please, describe

In Belgium there is a joint program with the Belgian research center SCK.CEN on I 600 and I 182 (metallurgical investigation, tensile, SSRT, in future crack growth) + participation to FROG SCC program.

There is no research on this point in Sweden.

In Japan research activities with constant load SCC tests are carried out in collaboration between utilities and reactor vendor [1)].

1) "Technical Study Report on Countermeasures for Ageing Nuclear Power Plant (in Japanese)" issued by JAPEIC, March 2003

The Spanish have no specific research. But, they are performing a review of the state of the art, which includes the specific project "Resistance to PWSCC of Alloys 690, 52, 152 in PWRs (MRP-111)"of the EPRI Materials Reliability Program.

Those who answered the questionnaire in USA are not aware of any current activities.

4.11 If this aspect is not considered in your country, so far, and, on the view of the incidents that have happened with these materials in several plants in different countries, do you think that some research activities should be initiated for a better knowledge of their behaviour?

No answer from Belgium

In Japan it is considered that new finding may justify some new research. Please refer to the answer to question item 4.9.

Spain refers to their answer to question 9.

Sweden: We are working with initiation as a phenomenon to get a better understanding of PWSCC, however there are no plans to include initiation in the ISI-program.

For the reactor top head and pressurizer USA states that the question is not applicable.

For the J-groove welds and bottom head penetration the answer given is: Crack initiation has been studied for Alloys 600/182 in this country. The accepted activation energy associated with crack initiation is roughly 50/45 kcal/Mole. Even though crack initiation has been examined, it is not considered when determining the inspection schedule and criteria.

4.12 What formula(e) is/are applied for *crack growth* for Alloy 600 and Alloy 182? Indicate the values of the different parameters of the formula(e).

This question was given with the option of giving separate answers for different components. However Belgium applies the same laws on all components. For Alloy 600 previous evaluations were based on different laws available at the time. Now use law proposed by NRC flaw evaluations guidelines:

$$da/dt(m/s) = 2.67.10^{-12}(K-9)^{1.16}\exp\left[\frac{-130000}{8.314}\left(\frac{1}{T}-\frac{1}{T_{ref}}\right)\right]$$

(This equation is sometimes called the Scott equation or Scott model.) For Alloy 182 the question is under discussion (several available laws used: EDF, Sweden, NRC…). The law imposed by USNRC for VC Summer found too conservative (exponent 1.16 not realistic for high K_I). Upper bound law from EDF showing saturation at high K_I more applicable.

Also in Japan the Scott equation, in what is said to be a modified version, is used for all components.

In Spain there is no specific formula. It is the intention to adopt one after the review of the state of the art, see question 10, has been finished.

The Swedish answer applies to the vessel head:

Alloy 600: 320°C; K_I<25.8 MPa(m)$^{1/2}$ da/dt=3.45E-21$K_I^{9.5}$ mm/s,
K_I>25.8 MPa(m)$^{1/2}$ da/dt=8.9E-8 mm/s

Alloy 182: 320°C; K_I<26.7 MPa(m)$^{1/2}$ da/dt=3.61E-15$K_I^{5.76}$ mm/s,
K_I>26.7 MPa(m)$^{1/2}$ da/dt=6E-7 mm/s

In USA for alloy 600 in the vessel heads and pressurizer the following formula is used:

$$\dot{a} = \exp\left[-\frac{Q_g}{R}\left(\frac{1}{T} - \frac{1}{T_{ref}}\right)\right]\alpha\left(K - K_{ref}\right)^{\beta}$$

This is of course the same formula as for Belgium above. For values of the constants the USA answer refers to the EPRI MRP-55 paper.

For Alloy 182 it is noted that crack growth data has been reported in literature at rates roughly five times as fast as those found for Alloy 600 reported in the Scott Model. Some type of Scott model adjustment may therefore be used as a crack growth formula.

4.13 On which of the following are these values based? (Several answers could apply):

(See Appendix A for the alternatives)

Belgium refers to recently published NRC guidelines. For Alloy 182 they also refer to own evaluation of what is openly available (Publications in open literature, conferences, NRC guides or documents, non proprietary version of MRP documents or presentations)

Japan refers to an ongoing JNES research project and to the Scott equation and NRC orders. In this case, JNES research project is the "Evaluation Technology for Stress Corrosion Cracking Growth of Ni-Based Alloys" (NiSCC project) conducted by JNES. Recently, the results of NiSCC project shows that the crack growth rate of base metal for the cold worked nozzle materials of Alloy 600 are lower than that evaluated by either the modified Scott curve or the ASME Sec XI Appendix O curve [1)].

1) Y. Yamamot et al., "Evaluation of Crack Growth rate for Alloy 600 Vessel penetrations in Primary Water Environment", 12th International Conference on Environmental Degradation of Materials in Nuclear Systems – Water Reactors, 2005

No answer from Spain.

Sweden refers to several sources see for ex. "P Efsing and C Jansson in Eleventh Int. Conf. Env. Deg. 2003" [12].

The USA answer primarily applies to Alloy 600: The crack growth equation presented in question 12 was fitted with a crack growth rate (CGR) data set. The final database of CGRs included 158 data points from tests performed by Westinghouse, Studsvik, EDF, CEA, and CIEMAT covering the full temperature range tested (190 ˚C to 363 ˚C). The CGR data points were adjusted to the most typical test temperature of 325 ˚C using an activation energy of 130 kJ/mole. The 130kJ/mole was the recommended value for activation energy for crack growth of Alloy 600 materials in primary water environments. The stress intensity factor dependence of the Scott equation, developed from steam generator tubing data, was adopted. This results in a power-law exponent, b, of 1.16 and an apparent crack tip stress intensity factor threshold, K_{th}, of (MPa*sqrt(m)). The crack growth amplitude is derived by fitting of all the data points in the set.

4.14 Are you confident in the formula and values used for the *crack growth* of Alloy 600 and Alloy 182, or do you think that further research activities should be carried out?

The Belgian view on this question is that further research is needed.

Currently the Japanese refers to the Scott equation. However they think that PWSCC crack growth rate data should be obtained. Therefore, JNES is performing the project for PWSCC crack growth. A comment from JNES: Further research needs to be carried out. A lot of research items are remained: especially correlation between laboratory data and field phenomena.

In Spain it is felt that for Alloy 600 there are different approaches which seem to be accepted by the nuclear community. For Alloys 82/182 some further research activities are needed from the point of view of the regulator.

The Swedish answer states that they are in the middle of an extensive research program that aims at describing the crack advance in different types of materials in PWR-environment.

The USA answer with regard to Alloy 600 is that the crack growth equation, described in question 12, is based on controlled testing of fracture mechanics specimens fabricated using 22 heats of CRDM nozzle, thick -wall tube, rolled bar, forged bar material, and four heats of plate materials. A lot of details were taken into consideration during the analysis of this data set. The present crack growth equation does not separate different heats of materials, and is a best estimate for the combination of different materials heats.

For Alloy 182 the growth rate data is preliminary and more data points need to be experimentally acquired.

4.15 Are there any research activities being carried out in your country to refine on these values? Please, describe:

In Belgium there is a joint program with the Belgian research center SCK-CEN on I 600 and I 182 (metallurgical investigation, tensile, SSRT, in future crack growth) + participation in FROG SCC program.

Currently in Japan the following two research projects, relating to crack growth in various Ni-Based alloys and their weld metals, are conducted by JNES, being supported by METI in Japan.

- Project of Evaluation Technology for Stress Corrosion Crack Growth of Ni- Base Alloy; [including constant load SCC tests], (NiSCC)
- Project of Evaluation Technology for Stress Corrosion Crack Growth of Ni Nased Alloy; [including constant displacement SCC tests], (NiSCC)

These data will be applied in the assessment evaluation of pressure boundaries in PWR and BWR reactors.

There is no work performed in Spain.

Sweden refers to answer to question 14.

For Alloy 600 the NRC has sponsored research at Argonne National Laboratories to examine crack growth rates. There is also work underway at the University of Michigan headed by Gary Was to understand the mechanisms contributing the measured crack growth rates of Alloy 600.

The MRP industry task group is currently examining the issue of a crack growth rate in 182 welds.

4.16 Have any "events" occurred in any of the units in your country?

In Belgium one vessel head has been replaced due to cracking. One more head has slowly growing cracks which are being followed.

In the bottom head penetrations indications assumed to be fabrication defects have been found. There is one small indication in an RCS nozzle not confirmed as a crack. In the pressurizer several indications have been found in one unit, not confirmed as cracks. In two units the safety injection nozzles have indications at weld/base metal interface not exposed to coolant.

In Japan one leak in the vessel head penetration J-groove has been observed in Ohi-3. In Takahama-1 an indication in the bottom head penetration has been observed. A leakage from a crack in a pressurizer nozzle has occurred in Tsuruga-2.

There are no events reported from Spain.

In Sweden two heads have been replaced since several minor indications were detected in the base metal of the penetrations in two units and in the J-groove weld in three units. In two units defects have been discovered in the hot leg nozzle to safe-end weld. In one pressurizer there has been leakage in an instrument penetration. There has also been a leak in a steam generator drainage pipe on the primary side in one unit.

The answer from USA is very extensive and contains a lot of interesting detail which can not be repeated here. Most of the cases are however described in NUREG-1823 [11]. For B&W reactors 7 units have been affected by vessel head penetration cracking, some of them several times. To the extent that J-groove cracking can be distinguished from penetration cracking this phenomenon has affected 2 reactors. In view of this extensive cracking several vessel heads have been replaced. There are no reports on bottom head cracks for the B&W reactors, but there are three cases of cracking in pressurizer nozzles.

For the Westinghouse reactors there are five plants with penetration cracks, two of which without leakage. There are also 4 cases of J-groove cracks. There is one case each of a leak in the bottom head penetration and the bottom head J-groove weld. One case of a crack in a reactor coolant system nozzle is reported under the category "Others". (As mentioned above no separate report has been prepared on RCS nozzles by NRC.)

For the Combustion Engineering reactors four cases of penetration cracks or indications without leakage are reported. For the J-groove weld there are two cases with one leak. No indications are reported for the bottom head. The pressurizer nozzles with heater sleeves and heater sheaths have apparently been quite a big problem for CE reactors. 13 plants have had leaks, some of them several times. Reactor coolant system nozzles have also been a problem, but only for instrumentation and temperature measurement nozzles. Problems with leaks have been reported from 8 plants.

4.17. Have any of the large components been ***replaced*** or ***repaired*** as a result of events or defects?

Belgium reports the replacement of one vessel head.

In Japan one vessel head and one bottom head in a unit have been repaired as a result of defects. There is also one case of repair of two pressurizer nozzles.

There are no cases in Spain.

Two vessel heads have been replaced in Sweden and two hot leg nozzles have been repaired. There is also one case of repair of an instrumentation penetration in the pressurizer.

6 vessel heads in B&W reactors and 3 vessel heads in Westinghouse reactors have been replaced as result of defects in USA. 6 vessel heads in Combustion Engineering plants are planned to be replaced in 2005-2007. There is only one case of repair of bottom head penetrations where two penetrations in one unit have been repaired. One RCS hot leg nozzle has been repaied after detection of a crack in the weld. The USA answer also contains detailed information on repair on all small nozzles and instrumentation penetrations which have been repaired.

4.18. Have any of the large components been *replaced preventively* (without indications or defects been detected on them)?

Japan reports reactor vessel head replacement in eleven units and the replacement of steam generators in eleven units as of end of March 2005..

In Spain four vessel heads have been replaced.

In USA one B&W reactor and two Westinghouse reactors have had the head replaced as a preventive measure.

4.19. If the answer to either of the two previous questions is yes, then: What materials have been chosen for the new components? What is the operating temperature? Has the operating temperature been changed since replacement?

The answers to the questionnaire show that with a few exceptions all head replacements and repairs are carried out with Alloy 690 as base material and Alloys 52/152 as weld material. With this material combination it is generally assumed that no reduction in operating temperature is needed.

One of the Spanish replacement head uses Alloy 600/182 and operates at a reduced temperature of 289 ˚C. The other reactor at the same site, Ascó, with an Alloy 690/152 head also operates with the same reduced temperature.

One B&W reactor, Davis Besse, also has an Alloy 600/182 replacement head. That particular choice was justified as due to a convenience and timeliness of restart of the plant. No information on temperature is available for that case. It is listed as Not Applicable.

4.20. What are the reasons for having chosen these materials (several answers could apply):

(see Appendix A for the alternatives)

Belgium refers to NSSS recommendation and foreign experience.

The Japanese answers differ between utilities. While four cite NSSS recommendation, one cites others also including Japanese research and foreign experience.

In Spain the basis for the choice has been recommendation from the material supplier and foreign experience. One utility also refers to the good experience with Alloy 690 in steam generator tubes.

In Sweden reference is made to Swedish research, experience with the material in other industries and foreign experience.

Regarding the vessel heads the USA answer states that Alloy 690 has been chosen as the material for use in replacement heads due to its superior SCC properties over alloy 600. This is based on research conducted and also plant experience seen overseas. Alloy 52/152 was developed to be used with alloy 690. The composition of the weld material is similar to that of alloy 690 once it is welded. The same comments apply to other replaced or repaired components.

4.21. Are the new materials considered to be susceptible to PWSCC, immune to this degradation mechanism or has this still to be determined? Please comment on the reason for this decision.

Belgium notes that there have been no failures of Alloy 690 in plant so far, but warns that it is necessary to be careful about materials supposed to be immune. They also note that 690/152 certainly is much better with respect to PWSCC than 600/182.

The Japanese attitude is different. They answer an affirmative yes to the question of Alloy 690 immunity to PWSCC.

In Spain and Sweden the question of immunity is considered to be unresolved.

In USA it is noted that Alloy 690 susceptibility to SCC is very small compared to alloy 600, however, it is not immune to SCC. Alloy 690 has been shown to SCC under the right circumstances. Regarding Alloys 52/152 the question of immunity still has to be determined.

4.22. Is the Arrhenius formula applied for the *crack initiation* determination of the new alloys, do you apply another formula, or this aspect is not considered? Indicate the values of the different parameters in the formula(e).

This question is not considered for the vessel heads in Belgium and it is not applicable to other components.

In Japan two utilities use Alloy 600 activation energy while others do not consider crack initiation.

In Spain no crack initiation scenario has been established yet for Alloy 690 and a definitive approach awaits ongoing research.

In Sweden this question is not considered.

In USA for the reactors with head penetrations to which this question applies the same formula is applied as in question 5 and 6. However, the EDY is reset to zero. This question is not considered for the J-groove welds.

4.23. On which of the following are these values based? (several answers could apply):

(see Appendix A for the alternatives)

In view of the answers to the previous question Belgium, Spain and Sweden abstained from answering.

One Japanese utility refers to information from the NSSS supplier while another refers to Japanese research.

USA refers to the answer to question 7.

4.24. For *crack initiation*, are the residual stresses in the new components measured, estimated, known, or not considered?

No answer from Belgium, Spain and Sweden.

One Japanese utility measures stresses while another makes estimates. The others did not answer.

In USA the new heads are considered to be similar to the old heads in this respect and the stresses are estimated in the same way.

4.25. Are you confident in the formula and values used for *crack initiation* of the new materials, or do you think that further research activities should be carried out?

No answer from Belgium and Sweden.

The Spanish do not use any formula for Alloy 690 crack initiation. Therefore it is the opinion of the regulator that further research is needed to establish a formula or to determine Alloy 690 immunity to PWSCC.

The Japanese are confident in their evaluation method, but consider that it is better to obtain more PWSCC data.

In USA, currently, the same value of crack initiation activation energy used for the old Alloy 600 heads is being used to determine the susceptibility of the new Alloy 690 heads. In the future it may be appropriate to adjust the number to consider changes to alloy 690 in the reactor vessel heads. This comment applies to other components as well.

4.26. Are there any research activities being carried out in your country to refine these values? Please, describe:

No research is carried out in Belgium or Sweden.

Spain repeats their answer to question 10.

The Japanese PWR utilities and NSSS supplier are performing PWSCC constant load tests for not only alloy 600 materials but also Alloy 690 materials.

Those who have answered the question for USA are not aware of any ongoing research.

4.27. Indicate, for each unit, the number of years that the new components have been in operation? If there are several units with the same number of operating years, indicate the number of units.

For the reactor heads the situation is summarized in the following table:

Country	No. years	No. of units
Belgium	6	1
Japan	4-8	7
	4	3
	3	1
Spain	9	1
	8	1
	2	1
	1	1
Sweden	10	1
	1	1
USA	1	1
	≤2	9
	2	1

Details of smaller replacement/repairs are available on the original spreadsheets from USA.

The Japanese have listed steam generator nozzles with 11 years since replacement for 1 unit and 4 years for one other unit.

Sweden has listed two RCS nozzle repairs with 1 and 2 years of operation after replacement.

4.28. Indicate if the *new components* are inspected on a regular basis (frequency), randomly, and the percentage inspected each time, or if there are no current plans for inspection. Indicate the inspection method used:

In Belgium the new head will be inspected to 100 % by UT every tenth year.

In Japan the new heads will be tested at each outage. The methods vary. While one utility will do leakage tests others will use visual inspection to 100 %. For the replaced SG nozzles the frequency will be 1/5 for VT and 1/10 for UT and PT.

In Spain one utilty will inspect the new heads with a frequency of 1/4 with VT and EC/UT. NRC Order EA-03-009 requirements will be followed.

Sweden has answered that inspection of the new components will be done with a frequency according to flaw tolerance assessment with UT and VT. This means following SKIFS 2000:2 but adjusting the frequency in case a sizeable defect is detected.

In USA the new components are inspected according to the same regulation which applied to the old components, see question 2.

4.29. Are you using the same guidelines and acceptance criteria for new components as for those that were replaced? If not, please describe the differences.

The way this question is formulated and the absence of answers from Belgium and Spain can be interpreted to mean that they as well as the other answering countries use the same guidelines and acceptance criteria for the new components as for those that were replaced.

4.30. Have any indications or defects been detected in these inspections of the *new components*?

The answer is a general No with one exception. In USA mechanical nozzle seal assembly (MNSA) clamps installed on hot leg nozzles as temporary repairs till a permanent repair could be made have leaked. This however has little to do with PWSCC so one can safely say that so far there are no known cases of PWSCC of the Alloy 690/152/52 combinations.

5. CONCLUSIONS FROM THE QUESTIONNAIRE

It is clear from the questionnaire that many of the older pressure vessel heads with Alloy 600/182 head penetrations have been replaced and that more replacements are anticipated in the future. The majority of replacement heads have or will have Alloy 690/152 head penetrations. The reason for the choice is given as either recommendation from others, literature, or the good experience with Alloy 690 as a steam generator tube material. In addition laboratory testing has shown that Alloy 690 is much more resistant to PWSCC than Alloy 600. In most of the few cases when Alloy 600/182 have been chosen for the head penetrations no reason for the choice has been given, probably because that particular question was missing in the questionnaire. In one case, Davis Besse, the choice was opportunistic, there was a head available on short notice.

Inspection practices vary between countries. As a general rule most countries apply ASME XI rules for the components with Alloy 600/182 material combinations. However since experience of cracking accumulated most countries have reduced the inspection intervals for the most critical components. In many cases replacement heads will be inspected according to the same rules as the older heads.

ASME XI is also the basis for acceptance criteria in most countries although they may have been slightly modified when they have been included in national regulation. Especially for head penetrations there has been a need to apply more stringent criteria or standards for acceptance of cracks. This need is expressed in a rather detailed US instruction on how to deal with cracks in vessel head penetrations.

Several questions dealt with *crack initiation*. It is clear from the answers that nobody attempts to predict crack initiation as such, except perhaps in a small number of cases. However the American approach of calculating EDY, effective degradation years, is an indirect way of expressing the probability that cracks might have initiated and it is based on an Arrhenius approach. It does not contain any stress dependence. Therefore as a general rule residual stresses do not play a rule in estimates of *crack initiation.* However residual stresses are often calculated but then perhaps for the purpose of flaw tolerance analysis or risk informed inspection.

For flaw tolerance analysis a crack growth law is needed. Most countries use the so called Scott equation both for Alloy 600 and Alloy 182. However for the latter a multiplication factor of five is used in comparison with values for Alloy 600. It seems to be a general feeling that more data is needed in order to get more confidence in the crack growth laws used. A couple of countries have indicated that they use another crack growth law than the Scott equation

A main objective of the questionnaire was to identify common needs and potential co-operative activities. For the utility side such activities are already going on through the various owners groups and EPRI. The question is whether or not similar activities are needed for the national regulators. One could have in mind both a coordination of regulation or a collaboration of research. With regard to regulation the questionnaire indicates that the regulation already is fairly well coordinated. The small differences that exist may well be present in order to serve special national needs. With regard to research those who have answered the questionnaire have not indicated much need for more research. This may however be a consequence of the fact that most questions on research specifically addressed *crack initiation* while the equally important question on crack growth data for Alloys 690/152 was never asked. Therefore the questionnaire may give a somewhat misleading indication that no research cooperation is needed.

6. REFERENCES

1. Vaillant, P., et al. *Crack growth rates in thick materials of Alloy 600 and weld metals of Alloy 182 in laboratory primary water comparison with field experience*. in *Fontevraud 5 International Symosium*. 2002. Fontevraud, France.

2. Coriou, H., et al. in *3rd Saclay Metallurgy Colloquium*. 1959. Saclay, France.

3. Berge, P. *Can PWR U-tube steam generators work safely?* in *Third International Symposium on Environmental Degradation in Nuclear Power Systems - Water Reactors*. 1987. Traverse City: The Metallurgical Society.

4. Alter, D., et al. *Stress corrosion cracking of pressurizer instrumentation nozzles in the French 1300 MWe units*. in *Fifth International Symposium on Environmental Degradation in Nuclear Power Systems - Water Reactors*. 1991. Monterey, USA: American Nuclear Society.

5. Hall., J.F., et al. *Cracking of Alloy 600 heater sleeves and nozzles in PWR pressurizers*. in *Fifth International Symposium on Environmental Degradation in Nuclear Power Systems - Water Reactors*. 1991. Monterey, USA: American Nuclear Society.

6. Buisine, D. and et. al. *Stress corrosion cracking in the vessel closure head penetrations of French PWR's*. in *Sixth International Symposium on Environmental Degradation in Nuclear Power Systems - Water Reactors*. 1993. San Diego, USA: TMS.

7. *NRC Bulletin 2001-1. Circumferential cracking of reactor pressure vessel head penetration nozzles.* 2001.

8. Jenssen, A., et al. *Structural assessment of defected nozzle to safe-end welds in Ringhals 3 and 4*. in *Fontevraud 5 International Symosium*. 2002. Fontevraud, France.

9. Rao, G., G. Moffat, and A.R. McIlree. *Metallurgical investigation of cracking in the reactor vessel Alpha loop hot leg nozzle to pipe weld at the V. C. Summer station*. in *Fontevraud 5 International Symposium*. 2002. Fontevraud, France.

10. Pathania, R.S., A.R. McIlree, and J. Hickling. *Overview of primary water cracking of Alloys 182/82 in PWRs*. in *Fontevraud 5 International Symosium*. 2002. Fontevraud, France.

11. Grimmel, B., *U.S. Plant Experience With Alloy 600 Cracking and Boric Acid Corrosion of Light-Water Reactor Pressure Vessel Materials. NUREG-1823, April 2005*

12. Efsing, P. and C. Jansson. *Screening of Crack Growth Data and the Relevance from an End-Users Perspective*. in *Eleventh International Symposium on Environmental Degradation in Nuclear Power Systems - Water Reactors*. 2003. Stevenson, USA: ANS.

APPENDIX 1

THE QUESTIONNAIRE IN MS WORD FORMAT

(Note that in the version sent out to the participants questions 20-30 were mistakenly numbered 21-31. This error did not occur in the Excel version)

Questionnaire on PWSCC and Nickel-Based Alloys

At the last meeting of the Metal Subgroup of the IAGE Main Group, held in Paris on 15th – 16th of April 2003, presentations made by the US NRC and CSN representatives highlighted the large concerns about PWSCC in nickel based alloys used in the reactor vessel head penetrations and other components, in particular Alloy 600 and its associated welds. It was agreed that a questionnaire be prepared to collect information to help and identify the common needs and area of potential co-operative activities based on the experience, status of existing data and research, and regulatory practices in the various member countries. While considerable research work has been ongoing for the steam generator tubing, we still have incomplete understanding of susceptibility of the thick sections. In addition to the understanding of degradation mechanisms, we need data on crack initiation, crack growth rates, stress analysis of welded assemblies of nickel-based components, and efficacy of NDE techniques. This is vital for defining appropriate inspection techniques and frequency to avoid potential breach of the primary pressure boundary. This information is also essential for consideration of probability of crack detection, leak-before-break concepts, leakage-detection requirements, and risk assessments.

In order to compare the operating experience, inspection practices and acceptance criteria applied in the different countries you are asked to complete this questionnaire. It is also of relevance to compare experience and practices with Alloy 690, which is being used to a large extent in replacement components, as well as research activities in this area. Should you have any other information that you consider could be useful for a better understanding of the PWSCC phenomenon, please do not hesitate to include it.

This survey is focused on thick section components but any relevant information concerning for example steam generators or research activities which are not explicitly addressed in this questionnaire would be appreciated.

An EXCEL spreadsheet is also provided for your convenience, and one sheet per design and vendor should be used.

Design: ______________________ **Vendor:** ______________________

A. CURRENT PLANT STATUS

1.- Indicate the pressure boundary components (not including steam generators tubes) that are made of Alloy 600 or Alloy 182 in the PWRs in your country. Also indicate the operating temperatures, and whether these are estimated or measured:

	Alloy	**Number of units**	**Temperature**	Estimated/ Measured
Reactor Vessel Head Penetrations (base metal)	______	______	______	______
J-groove welds RVH Penetrations	______	______	______	______
Reactor Bottom Vessel Head Penetrations (base metal)	______	______	______	______
J-groove welds RBVH Penetrations	______	______	______	______
Vessel to Reactor Coolant System Nozzles (Hot leg)	______	______	______	______
Vessel to Reactor Coolant System Nozzles (Cold leg)	______	______	______	______
Pressurizer Nozzles	______	______	______	______

Others (describe) ______________________

Comments

2.- Indicate if these components are inspected on a regular basis (frequency), or randomly and the percentage inspected each time. Indicate the inspection method used:

	Yes/No	**Frequency**	**Percentage**	**Inspection Method (VT, UT, etc.)**
RVH Penetrations (base metal)	______	______	______	______
J-groove welds RVH Penetrations	______	______	______	______
Reactor Bottom VH Penetrations (base metal)	______	______	______	______
J-groove welds RBVH Penetrations	______	______	______	______
Vessel to Reactor Coolant System Nozzles	______	______	______	______
Pressurizer Nozzles	______	______	______	______

Others (describe) ______________________

Comments

3.- Indicate the Code, Regulation, Specific Regulatory Body Requirement, domestic or foreign operating experience, etc., governing inspection of these components:

Reactor Vessel Head Penetrations (base metal) ____________________
J-groove welds RVH Penetrations ____________________
Reactor Bottom VH Penetrations (base metal) ____________________
J-groove welds RBVH Penetrations ____________________
Vessel to Reactor Coolant System Nozzles ____________________
Pressurizer Nozzles ____________________
Others (describe) ____________________
Comments

4.- What acceptance criteria are used for cracks in these components (codes, etc)?

Reactor Vessel Head Penetrations (base metal) ____________________
J-groove welds RVH Penetrations ____________________
Reactor Bottom VH Penetrations (base metal) ____________________
J-groove welds RBVH Penetrations ____________________
Vessel to Reactor Coolant System Nozzles ____________________
Pressurizer Nozzles ____________________
Others (describe) ____________________
Comments

5.- For a unit that has been operating for some years at $T > T_{ref}$ (hot head) and then changes to cold head ($T < T_{ref}$), how do you consider this affects its PWSCC susceptibility?

6.- Is the Arrhenius formula applied for the *crack initiation* determination of Alloy 600 and Alloy 182? Do you use another formula, or is this aspect not considered? Indicate the values of the different parameters in the formula(e).

	Arrhenius	Another(*)	Not considered
RVH Penetrations (base metal)			
J-groove welds RVH Penetrations			
Reactor Bottom VH Penetrations (base metal)			
J-groove welds RBVH Penetrations			
Vessel to Reactor Coolant System Nozzles			
Pressurizer Nozzles			

Others (describe) ____________________

Comments

(*) What parameters are included? ______________________________

Comments

7.- On which of the following are these values based? (several answers could apply):

a.-	Research activities carried on in the country	
b.-	Values given for the codes and standards in force in the country	
c.-	Values given by the NSSS supplier	
d.-	Values given by consulting engineering company	
e.-	Values given by the materials supplier	
f.-	Bibliography (please, specify)	
g.-	Other	

Comments

8.- For *crack initiation*, are the residual stresses measured, estimated, known, or not considered?

Measured	
Estimated	
Known from construction documents	
Not considered	

Other ______________________________

9.- Are you confident in the formula and values used for the *crack initiation* of Alloy 600 and Alloy 182, or do you think that further research activities should be carried out?

10.- Are there any research activities being carried out in your country to refine on these values? Please, describe:

11.- If this aspect is not considered in your country, so far, and, on the view of the incidents that have happened with these materials in several plants in different countries, do you think that some research activities should be initiated for a better knowledge of their behaviour?

12.- What formula(e) is/are applied for *crack growth* for Alloy 600 and Alloy 182? Indicate the values of the different parameters of the formula(e).

	Alloy 600	Alloy 182
Reactor Vessel Head Penetrations (base metal)		
J-groove welds RVH Penetrations		
Reactor Bottom Vessel Head Penetrations (base material)		
J-groove welds RBVH Penetrations		
Vessel to Reactor Coolant System Nozzles		
Pressurizer Nozzles		
Others (describe)		

Comments

13.- On which of the following are these values based? (several answers could apply):

a.- Research activities carried on in the country

b.- Values given for the codes and standards in force in the country

c.- Values given by the NSSS supplier

d.- Values given by consulting engineering company

e.- Values given by the materials supplier

f.- Bibliography (please, specify)

g.- Other

14.- Are you confident in the formula and values used for the *crack growth* of Alloy 600 and Alloy 182, or do you think that further research activities should be carried out?

15.- Are there any research activities being carried out in your country to refine on these values? Please, describe:

16.- Have any "events" occurred in any of the units in your country?

	Yes/No	Specify (leakage, crack,…)	No. of units
RVH Penetrations (base metal)			
J-groove welds RVH Penetrations			
Reactor Bottom VH Penetrations (base metal)			
J-groove welds RBVH Penetrations			
Vessel to Reactor Coolant System Nozzles			
Pressurizer Nozzles			

Others (describe) ______

Comments

B. REPLACEMENT COMPONENTS

17.- Have any of the large components been *replaced* or *repaired* as a result of events or defects?

	Replaced/Repaired	Number of units
Reactor Vessel Head Penetrations (whole head)		
Reactor Bottom Vessel Head Penetrations		
Vessel to Reactor Coolant System Nozzles		
Pressurizer Nozzles		
Others (describe)		

Comments

18.- Have any of the large components been *replaced preventively* (without indications or defects been detected on them)?

	Yes/No	Number of units
Reactor Vessel Head Penetrations (whole head)		
Reactor Bottom Vessel Head Penetrations		
Vessel to Reactor Coolant System Nozzles		

Pressurizer Nozzles	______	______
Others (describe)	______	______

Comments (reasons for the replacement)

19.- If the answer to either of the two previous questions is yes, then: What materials have been chosen for the new components? What is the operating temperature? Has the operating temperature been changed since replacement?

	Material	**Operating T: Measured/Estimate**	**Reduced?**
Reactor Vessel Head Penetrations (base metal)	______	______	______
J-groove welds RVH Penetrations	______	______	______
Reactor Bottom Vessel Head Penetrations	______	______	______
J-groove welds RBVH Penetrations	______	______	______
Vessel to RCS Nozzles	______	______	______
Pressurizer Nozzles	______	______	______
Others (describe)	______	______	______

Comments

20.- What are the reasons for having chosen these materials (several answers could apply):

Research activities carried on in the country	
Experience with these materials in other industries of the country	
NSSS recommendation	
Materials supplier recommendation	
Foreign experience	
Other	

Please, describe your answer

21.- Are the new materials considered to be susceptible to PWSCC, immune to this degradation mechanism or has this still to be determined? Please comment on the reason for this decision.

a.-	Susceptible to PWSCC	☐
b.-	Immune to PWSCC	☐
c.-	To be determined	☐
d.-	Other	☐

Comments

22.- Is the Arrhenius formula applied for the *crack initiation* determination of the new alloys, do you apply another formula, or this aspect is not considered? Indicate the values of the different parameters in the formula(e).

	Arrhenius	**Another(*)**	**Not considered**
Reactor Vessel Head Penetrations (base metal)	☐	☐	☐
J-groove welds RVH Penetrations	☐	☐	☐
Reactor Bottom Vessel Head Penetrations	☐	☐	☐
J-groove welds RBVH Penetrations	☐	☐	☐
Vessel to Reactor Coolant System Nozzles	☐	☐	☐
Pressurizer Nozzles	☐	☐	☐

Others (describe) ______________________

Comments

(*) What parameters intervene in this formula? ______________________

Comments

23.- On which of the following are these values based? (several answers could apply):

a.-	Research activities carried on in the country	☐
b.-	Values given for the codes and standards in force in the country	☐
c.-	Values given by the NSSS supplier	☐
d.-	Values given by consulting engineering company	☐

e.- Values given by the materials supplier ☐

f.- Bibliography (please, specify) ☐

g.- Other ☐

Comments

☐

24.- For *crack initiation*, are the residual stresses in the new components measured, estimated, known, or not considered?

Measured ☐

Estimated ☐

Known from construction documents ☐

Not considered ☐

Other ____________________

25.- Are you confident in the formula and values used for *crack initiation* of the new materials, or do you think that further research activities should be carried out?

26.- Are there any research activities being carried out in your country to refine these values? Please, describe:

27.- Indicate, for each unit, the number of years that the new components have been in operation? If there are several units with the same number of operating years, indicate the number of units.

	No. of years	**No. of units**	**No. of years**	**No. of units**
Reactor Vessel Head Penetrations (whole head)	________	________	________	________
Reactor Bottom Vessel Head Penetrations	________	________	________	________
Vessel to Reactor Coolant System Nozzles	________	________	________	________
Pressurizer Nozzles	________	________	________	________
Others (describe)	________	________	________	________

Comments

☐

28.- Indicate if the *new components* are inspected on a regular basis (frequency), randomly, and the percentage inspected each time, or if there are no current plans for inspection. Indicate the inspection method used:

	Yes/No	Frequency	Percentage	Insp. Method (VT, UT, etc.)
Reactor Vessel Head Penetrations (base metal)				
J-groove welds RVH Penetrations				
Reactor Bottom Vessel Head Penetrations				
J-groove welds RBVH Penetrations				
Vessel to Reactor Coolant System Nozzles				
Pressurizer Nozzles				
Others (describe)				

Comments

29. Are you using the same guidelines and acceptance criteria for new components as for those that were replaced? If not, please describe the differences.

30.- Have any indications or defects been detected in these inspections of the *new components*?

	Yes/No	Specify (leakage, crack)	Repair (Yes/No)
Reactor Vessel Head Penetrations (base metal)			
J-groove welds RVH Penetrations			
Reactor Bottom Vessel Head Penetrations			
J-groove welds RBVH Penetrations			
Vessel to Reactor Coolant System Nozzles			
Pressurizer Nozzles			
Others (describe)			

Comments

C. ADDITIONAL INFORMATION