IAEA Extrabudgetary Programme on Mitigation of Intergranular Stress Corrosion Cracking in RBMK Reactors

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Abstract: Intergranular stress corrosion cracking of austenitic stainless steel piping in BWRs has been a major safety concern since the early seventies. Similar degradation was found in RBMK reactors' piping in 1997. Early in 1998 the IAEA responded to requests for assistance from RBMK operating countries through its ongoing activities. In 2000 the IAEA initiated a dedicated Extrabudgetary Programme on Mitigation of Intergranular Stress Corrosion Cracking in RBMK Reactors to assist in addressing the issue. The scope of the Programme included in-service inspection, assessment, repair and mitigation, and water chemistry and decontamination. The Programme was pursued by means of exchange of experience, formulation of guidance, transfer of technology, and training, which will assist the RBMK operators to address related safety concerns. The Programme implementation relied on voluntary extrabudgetary contributions from a number of countries and was implemented in close co-ordination with ongoing national and bilateral activities. The Programme was successfully completed in 2002. The Programme had a positive effect on the safety of RBMK plants, but open issues still remain to be resolved. Status of technology transfer to RBMK operators has been summarized in detail, and a number of recommendations to address unresolved safety issues was provided. This paper summarizes main information provided in the Final Steering Committee Report, which was prepared by the whole Steering Committee of the Programme.

1. INTRODUCTION

Since 1997 RBMK reactors have experienced cracking in portions of their stabilized stainless steel piping systems, which has the characteristics of intergranular stress corrosion cracking (IGSCC), similar to what had been experienced by western boiling water reactors (BWRs). While no through wall cracks have been detected to date, such cracking incidents pose a threat to the integrity of the primary pressure boundary that if not managed and mitigated, can increase the likelihood of a loss of coolant accident. Although rupture of these 300 mm diameter pipes is within the parameters of a design basis accident, there is still the potential for damage to the reactor core and the release of radioactivity to the atmosphere.

In 1998, Member States operating RBMK reactors requested assistance in managing and mitigating IGSCC. The IAEA initially responded to these requests by organizing a workshop on the issue in Slavutych, Ukraine, and two experts' meetings to discuss the issue and exchange related experience. A dedicated Extrabudgetary Programme on Mitigation of Intergranular Stress Corrosion Cracking in RBMK Reactors was launched in 2000. The Programme was undertaken as a complement to existing national, bilateral and international activities, taking full advantage of their results and ongoing efforts, and paying specific attention to co-ordination in order to avoid duplication and obtain achievable leverage effects. The Programme concluded in mid-2002.

The objective of the Programme was to assist RBMK operating countries in mitigating IGSCC in austenitic stainless steel piping, emphasizing four technical areas:

- improvements in in-service inspection performance and qualification ;
- comprehensive assessment techniques ;

- qualification of repair techniques ;
- water chemistry and decontamination methods.

Throughout the Programme, the activities were carried out with the main focus on ensuring reactor coolant system integrity and without consideration of generation specific design safety features.

Programme implementation was based on the IAEA practices and its developed infrastructure for providing nuclear safety assistance. The activities were carried out by four Working Groups that addressed the above technical areas, under guidance and co-ordination of a Programme Steering Committee. The main part of the work was performed by the individuals and organizations involved in the Working Groups. The Programme meetings served mainly for co-ordination. Strong elements of the Programme were training and technology transfer, which were supported by the ongoing bilateral and international activities of the involved Member States, in particular the Swedish International Project and the US DOE International Nuclear Safety Program.

Detailed information on the Programme history, implementation and its results is provided in the Final Report of the Programme's Steering Committee [1], which integrates and summarizes information prepared by each of the Working Groups. The report [1] presents a description of the problem, an overview of the history of the Programme, findings on the root cause for RBMK pipe cracking, overall conclusions and recommendations to mitigate the problem, and status of their implementation. Detailed list of programme participants, activities and reports prepared is also provided in [1]. Further information on the Programme could be obtained at http://www.iaea.org/ns/nusafe/ebpigscc.htm, where also all the reports prepared within the Programme could be downloaded.

The Final Programme's Steering Committee Report [1] was prepared by all its members : Brickstad, B., Det Norske Veritas, Sweden, Bryce, B., Mitsui Babcock, United Kingdom, Brylev, E., , Rosenergoatom, Russian Federation, Gryshchenko, V., Nuclear Regulatory Authority, Ukraine, Kaliberda, I., Gosatomnadzor, Russian Federation, Lance, J., Electric Power Research Institute, USA, Letzter, A., Det Norske Veritas, Sweden, Liszka, E., Swedish International Project, Sweden, Maksimovas, G., Permanent Mission, Lithuania, Mayfield, M., United States Nuclear Regulatory Commission, USA, Negrivoda, G., Ignalina NPP, Lithuania, Neretin, Yu., Chernobyl NPP, Ukraine, Petrov, A., Research and Development Institute of Power Engineering (NIKIET), Russian Federation, Poulter, L., Serco Assurance, United Kingdom, Sanchez, J., Tecnatom, Spain, Roberts, A., Battelle, USA, Shibuya, S., Fugen NPP, Japan, Staudt, U., VGB PowerTech e.V., Germany, Taylor, T., Pacific Northwest National Laboratory, USA, Tchernikov, O., Leningrad NPP, Russian Federation. Their contribution to the whole Programme as well as to summarizing the outcomes and outputs of the 2 years effort is acknowledged.

This paper is based on the Final Programme's Steering Committee Report [1] and summarizes main information provided in it.

2. BACKGROUND

RBMK reactors are pressure tube reactors that have been built in Russian Federation, Ukraine and Lithuania. There are three generations of RBMK reactors, with the first reactor having commenced operation in 1973 (Leningrad 1) and the most advanced version having commenced operation in 1990 (Smolensk 3).

In January 1997, during routine in-service radiographic inspections, cracks were found in 35% of the 974 welds of 300 mm diameter austenitic piping in Leningrad 3. None of these cracks were through wall cracks. The areas were removed for investigation and intergranular cracks were found in the heat affected zone. These had the classical characteristics of IGSCC studied extensively in western BWRs.

Inspection of sample welds of similar piping carried out later at Kursk 1 revealed defects in 5 welds out of 80 inspected, after which 100 percent inspection was performed.

By early 1998, it was clear that IGSCC posed a generic problem that was affecting all 14 operating RBMKs to some degree. However there are several aspects of the cracking which are yet unexplained and leave the "root cause" as an open question. For example, it is the later RBMKs such as Leningrad 3 and Chernobyl 3 that appear most affected, with some of the first generation RBMKs being among the least affected. Obviously, the explanation for cracking appears to be more complex than one purely focused on the age of the reactor.

In a similar manner, it has not been possible to make an easy correlation between factors which often affect the propensity for defects, such as whether the welds were made in shop (factory welds) or on-site at the reactor (assembly welds). Thus at Chernobyl 3, a greater proportion of on-site welds exhibit IGSCC; whereas at Leningrad 3 and at the two Ignalina reactors, it is the factory welds that appear worse.

Defects of crack lengths over 250 mm have been reported (26% pipe circumference), with depths in some cases up to 12 mm (75% of pipe wall thickness). However, the great majority of defects are smaller.

One reason that the occurrence of IGSCC in RBMKs is so significant is that, being pressure tube reactors, there is necessarily a large amount of piping, much more than in a pressure vessel type reactor. Therefore any problem with the pipe inevitably generates a large maintenance task, with associated economic and radiation dose issues. There are over 1500 welds in austenitic pipe in the intermediate diameter range of 300 mm and wall thickness of 15–16 mm per RBMK reactor. Most attention has been focused on these welds, although there are other components made of the same austenitic steel for which IGSCC may be of significance. Accordingly, any inspection, repair or mitigation technique that has to be applied individually to such a large number of welds will be expensive and lead to significant radiation doses being received by the staff.

3. ROOT CAUSE

Knowledge of root cause is central because it can help to:

- design in-service inspection plans with respect to both inspection locations and frequencies ;
- perform analyses aimed at exclusion of double ended guillotine break potential ;
- analyze the issues of crack initiation and growth (including re-initiation after repair) ;
- develop effective pipe repair procedures ;
- develop methods to eliminate the cracking phenomenon.

Cracking of 300 mm diameter titanium stabilized stainless steel pipe (08Ch18N10T) in RBMK reactors is the result of IGSCC, just as has been observed in non-stabilized and stabilized stainless steel piping in western BWRs. Overall, the highest number of cracking indications has occurred in the downcomer sections and group distribution headers, Fig. 1.

IGSCC arises with the simultaneous occurrence of three critical parameters: material condition, stress condition and water chemistry. Of all the factors that are involved in the interaction of these three conditions, the following three appear to be the critical ones for RBMK piping:

- Some level of thermal sensitization due to welding (chromium depleted grain boundaries). There is no true threshold sensitization level, but cracking is easier the more the pipe is sensitized, and vice versa. The sensitization may be further enhanced by the phenomenon of low temperature sensitization that occurs during long term operation.
- High tensile residual stresses, plastic strain and deformation derived from the pipe weld preparation and welding process.
- An oxidizing reactor water environment, with total concentrations of chloride and sulfate above 10 μg/kg, which strongly affects water conductivity.

All other variables (such as operating stresses, other water impurities and metallurgy) are considered to be secondary in nature and may account for the observed variability in cracking locations from plant-to-plant and different crack sizes for similar operating times.

Stress corrosion cracking is a two step process: (1.) crack initiation and (2.) sub critical crack propagation. IGSCC modelling for the case of sensitized AISI 304 type stainless steel clearly shows that crack propagation takes place with a variable crack growth rate, which is strongly correlated to parameters such as water chemistry, weld residual stresses and material characteristics. The metallurgical investigation of cracks removed from the RBMK piping suggests that preexisting surface features have led to early crack initiation and therefore it is believed that the life limiting factor for affected welds is crack propagation. However, early crack initiation does not necessarily predict a fast crack growth rate.

There were no through wall cracks observed in 300 mm piping in RBMKs. For the RBMK case, Russian specialists explain this to be due to reduction of sensitization along the fusion line from inner to outer surface, reduction of plastic strains introduced by multi-pass welding, and change of residual stresses from tensile in the weld root to compressive stresses close to the outer surface of the pipe. However, Western experts point out that there is no sensitization threshold, and so, if there is tensile stress, a crack has a finite probability of growing, albeit slowly, even in non-sensitized material. Cracking will stop in either a compressive stress field (which results in a negative stress intensity factor) or if the crack tip is deformed and hence blunted. But it is not clear how such conditions would routinely be sustained in the pipe wall. Therefore, it cannot be concluded that cracks would not ever reinitiate and grow through the pipe wall.

4. RESULTS

Main results of the Programme are focused on the 4 technical areas addressed : in-service inspection, flaw assessment and inspection intervals, repair techniques and other mitigation strategies, in particular plant water chemistry. Since RBMK operators have been working in parallel to mitigate the problem, it is important to provide the Programme results along with information on status of implementation of mitigation measures at the plants and to identify and describe the gaps that remain to be filled.

4.1 Status

In-service inspection

- Manual ultrasonic inspection techniques for IGSCC detection were transferred effectively to Ukraine and Lithuania; Russian specialists feel their semi-automatic system is adequate.
- Reliability of detection performance of NDE specialists at RBMK power plants can be equivalent to detection performance achieved at Western power plants.
- All inspectors could use additional practice in crack depth sizing.
- Criteria in the draft qualification document specifies a target flaw size of 4 mm.
- Supporting data from the qualification pilot study indicate that cracks with a through wall depth of 25% (4 mm) have approximately an 80% probability of detection.

Flaw assessment and inspection requirements

- Most national flaw assessment methods will give similar results regarding acceptable crack sizes (if one disregards the crack growth rate).
- Insufficient data on crack growth rates in RBMK materials has led to different approaches to its description.
  - Stress intensity dependent; $da/dt = 4.5 \times 10^{-12} K_I^3$mm/s (western position)
  - Constant crack growth rate of 1 mm per year in depth, regardless of the stress condition (Russian position)
- Inspect repaired welds on annual cycle, at least until evidence is available that no new cracking is occurring.
- Inspect cracked welds left without repair on annual cycle. However, with appropriate sizing capability and crack growth data, fracture mechanics methods may be used to determine a suitable future inspection interval to provide sufficient plant safety.
- For IGSCC susceptible welds where no defects have yet been detected, inspect all welds. Determination of inspection intervals should be carried out using a procedure based on fracture mechanics that allows for defect growth.

Repair techniques

- U.S. technology for low stress welding techniques, heat sink welding and narrow groove GTAW (automated argon arc welding with non-fusible electrode) were transferred to RBMK operators. These methods will provide a reduction of both material sensitization and weld residual stresses.
- Use of weld overlays for repair of cracked welds in 300 mm diameter pipes is being implemented.
- Mechanical stress improvement process (MSIP) is being qualified as a mitigation technology for new and repaired welds

Water chemistry and decontamination

- “Mild” decontamination methods currently used have no effect on IGSCC propensity.
- In-line water chemistry monitoring systems, chromatography equipment and automatic data acquisition systems are now part of formal water chemistry programs at most plants. Reactor water conductivity is routinely <0.1mS/cm
- Condenser monitoring, leak detection and remedial actions are being instituted to reduce probability of tube leaks and/or reduce the recovery time following leak detection.
- Deaeration will soon be a standard reactor start-up procedure to reduce oxygen levels in the coolant during critical operational periods.

4.2 Open issues

The following open issues, which need to be addressed in a timely manner, were identified :

- Inspection/inspector qualification.
- Optimum inspection intervals based on realistic crack growth and realistic assessment of weld inspection capability.
- Weld overlay repair optimization and welder qualification
- Expanded experience with MSIP.
- Less corrosive water chemistry.

4.3 Recommendations

- Ensure that qualification document contains specific flaw acceptance criteria needed for successful implementation of ISI.
- Develop training materials and conduct regular training classes to qualify inspectors on the specific ultrasonic procedures used to detect and characterize IGSCC.
- Develop flaw evaluation criteria based on crack growth characteristics of actual plant materials in realistic stress and environmental conditions.
- Develop optimum inspection intervals based on realistic crack growth and realistic assessment of weld inspection capability.
- Develop a procedure to demonstrate a sufficiently small probability of double-ended guillotine break (DEGB).
- Consider development and implementation of welding procedures using filler wire in the root (i.e. avoid autogenous welding of the root).
- Conduct further work to establish design margins, application of coolant filled pipes and software to ease design of weld overlay and justification of long-term operation.

- Assess effects of sensitization to pipe caused by weld overlay.
- Select optimal solutions on welding equipment and processes and finalization of welding procedures.
- Apply MSIP routinely to all new pipe welds and repaired pipe welds, as long as crack depth is less than x%? of wall, -once installed in plant, reassess inspection intervals.
- Establish goals of further reducing water conductivity below 0.065 mS/cm and ECP below –0.230V(SHE) for all plants.
- Improve efficiency of condensate polishing beds for sulfate and chloride removal by adopting regular bed cleaning methods and choice of resins similar to BWRs. Also, evaluate conversion of cation bed to a mixed bed.
- Maintain chloride and sulfate in reactor water bellow or at 5 µg/kg for each ion and institute plant action levels if these levels are exceeded.
- Reevaluate continued use of copper alloy condensers given regular tube leakage and high copper content in the water.
- Consider alternative water chemistries - hydrogen water chemistry and noble metal chemical additions (NMCA). NMCA process appears to be more compatible with current RBMK conditions.

5. CONCLUSIONS

The Programme has made a postive impact on RBMK safety, but there are open issues, which remain to be resolved by the operating plants.

Western techniques and know-how have been successfully transferred to RBMK operators.

The success criteria identified during the Programme implementation by RBMK operators were met by Programme outputs and outcomes.

The co-ordination with other ongoing activities had rather positive results, such as pilot installation of eletrochemical potential monitoring system at Ignalina 2, delivery of automated welding equipment to Ignalina and Leningrad plants, both based on input provided through respective Working Groups and other bilateral projects.

References

1. *International Atomic Energy Agency*, Mitigation of intergranular stress corrosion cracking in RBMK reactors. Final report of the Programme's Steering Committee. IAEA-EBP-IGSCC, Vienna (2002).