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VVER Operational Safety Improvements: Lessons Learnt from European Co-operation and Future Research Needs

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1. INTRODUCTION

Nuclear Research Institute Řež (NRI) has over the years participated in many EU projects (5th Framework Programme, PHARE, TACIS) designed to improve operational safety of PWR and VVER reactors. Within 5th Framework Programme the Institute takes part in more than 30 projects covering the following research areas:

- reactor safety
- components integrity
- waste management

This paper summarises our involvement within the first two areas which are directly related to reactors operational safety, and gives an idea how results of the projects can be used to enhance safety of VVER reactors. We would like to emphasize that the NRI contribution is a small part of the global effort of the EU, and the presented results are meaningful only in this the context.
(More detailed information of the subject has been published in Nuclear Engineering and Design 221/2003).

Key words: VVER, operational safety, improvements, European co-operation, research

2. REACTOR SAFETY PROJECTS :

Nine projects can be included into this group:

2.1. EVITA - European Validation of the Integral Code ASTEC

ASTEC - the system code for nuclear severe accident and source term analyses is one of the few such codes now in existence It has been created by merging several detailed codes dealing with specific phenomena, the basic ones were ESCADRE and RALOC/FIPLOC.

A large development and validation group consisting of 18 organizations from 8 countries and the European Joint Research Centre is dealing with the ASTEC code validation and testing, while the main code development task is born by the IPSN and GRS. NRI Řež participates in several Work Packages and co-ordinates the code platform comparison tests.

Testing of the ASTEC V0.3 on different computer platforms has been already finished. A good portability was indicated. The differences in the results of several validation cases for individual code modules and one reactor case testing of the whole code were minor. ASTEC validation against experiments performed by a large number of the partners has been very thorough, NRI is contributing in the two areas:

- aerosol transport in the primary system where the module SOPHAEROS will be tested against the experiments performed in the 5th Framework SGTR project and against the MELCOR 1.8.5 code
- validation of the IODE module against the OECD ISP 41.

The results of ASTEC will be compared against those of MELCOR 1.8.5 or 1.8.4 system code.

Since ASTEC code provides better insight into some physical phenomena taking place during severe accidents in different types of PWR plants, including VVER, continuation of its development based on validation against experiments is the main future objective. The NRI Řež expertise should allow to confirm the satisfactory performance of ASTEC for the fission product transport.

2.2. ARVI - Assessment of Reactor Vessel Integrity

This project was designed with the objective to reduce remaining uncertainties related to the possibility of the in-vessel debris retention during a nuclear accident with degraded core. NRI participates in the studies of in-vessel debris retention resulting from external vessel cooling by water - for VVER-440 reactors (together with VEIKI Budapest).

We have already estimated the debris masses and decay heat participating in the vessel load, as well as the accident timing. Capability of KTH Sweden MVITA code has been extended, by adding decay heat source time dependence (originally - constant source) and calculation of the margins to critical heat flux on the external vessel surface which is a crucial factor in the vessel wall survival

Results of the project will be directly applicable for smaller reactors, e.g. VVER-440, however this effort should be supported by more extensive validation of MVITA type codes on existing or newly performed experiments within this project. The main uncertainty, which prevents application of the external vessel cooling for larger reactors with small thermal margins, is the possible phase separation in the debris pool caused by chemical phenomena. This cannot be resolved without further experiments with prototypic core material like those in the OECD projects RASPLAV and MASCA.

SGTR – Steam Generator Tube Rupture Scenarios

Knowledge of the aerosol transport and retention in a steam generator (SG) in case of a severe accident with SG tube or collector rupture is still very limited, and the SGTR project was prepared to extend existing experimental data (for horizontal SG carried out by VTT and Fortum Finland and for vertical SG - by PSI Switzerland and CIEMAT Spain). The supporting analytical effort is provided by our Institute and NRG Netherlands. To define necessary experiments for horizontal SG, we have performed a number of analyses of Loviisa and Dukovany VVER-440 plants severe accident sequences with tube rupture or collector break using MELCOR 1.8.3 code.

Probably the most important finding is that that in the cases of SG tube rupture the radioactivity releases strongly depend on whether the primary system remains under pressure or not. For instance – the computed average aerosol flow through tube break (cases LO1, LO2, LO3, LO5) was between 0.00006 and 0.00003 kg/s, while in the cases without depressurization (LO1A, DUC1, DUC2) the average values were two orders of magnitude higher.

Thus, these analyses indicated that primary system depressurization is a very efficient way how to reduce radioactivity release to the environment and that it may be a good accident management tool.

Further analyses shall concentrate on the effect of fission products retention in the broken tube or tubes not considered previously. The corresponding model which is under development at VTT based on the experimental results of this project, should be incorporated into the MELCOR code.

In the future, more attention should be paid to the SG collector break accidents, since the collector break contribution to core damage is greater and such sequences are more difficult to handle because of much faster timing than SGTR; primary system depressurization is not applicable, since collector rupture causes spontaneous depressurisation.

2.3. ECOSTAR - Ex-Vessel Core Melt Stabilisation

Complex of phenomena occurring in molten corium during last stage of a severe accident in an European reactor is studied within this project. Programme includes several large-scale experiments and a series of small-scale ones with real (prototype) molten corium, and foresees improvement of the relevant computer codes and their validation.

NRI contributs to the Working Programmes: Data and Properties, and Real Material Solidification. Two new phenomena (fundamental from the viewpoint of melts properties at high temperatures) were found – namely, oxygen air-lift and miscibility gap in ZrO_2+Fe_2O_3+SiO_2 system, important for understanding of behaviour and crystallisation of corium in the conditions of a large scale experiment (French VULCANO) and in the course of a real accident.

To get an accurate picture of the distribution of the decay heat generating isotopes the long-term experiments covering corium cooling in the spreading bath are needed, since two observed phenomena have to be taken into account.

2.5. LPP - Late Phase Source Term Phenomena

NRI participates in the examination of radiological consequences of a severe accident (the source term) in terms of the plant behaviour (coolability of molten pools) in the light of the new experimental data. The results should allow to develop severe accident management strategies.

NRI effort has concentrated on the VVER-1000 reactors two units of this type are in trial operation at the Temelin site. Our main objectives are:

- improving understanding of how radioisotopes are released from degrading core (melt pools),
- optimising mitigation measures,
- better prediction of the source term (reducing uncertainties of severe accident consequences) which will allow realistic accident management strategies and off-site emergency planning.

The plant assessment calculations were performed for three VVER-1000 severe accident scenarios. All sequences were initiated by a large break LOCA, failure of all active Emergency Core Cooling systems resulted in development of the sequences leading towards the severe accident region. In the **first** scenario, core debris after Reactor Pressure Vessel (RPV) failure are captured in the reactor cavity. In both the **second** and **third** sequences corium is allowed to spread out of the cavity into neighbouring rooms. In addition, in the third scenario - operation of one leg of the LP ECC system is recovered after RPV failure (water was supplied to the primary system, and through the hole in the bottom of the reactor pressure vessel it entered the reactor cavity and flooded the melt pool). The 2nd and 3rd scenarios reflected the severe accident management (SAM) aspect: extending of the corium flooded area and corium water cooling should result in decrease of corium penetration rate in concrete and reduce the threat of containment failure. The relevant sensitivity study elucidated influence of ECC recovery timing on corium pool behaviour and the source term. It should be noted that the in-vessel phase was the same for all scenarios, the significant differences were observed in ex-vessel phase.

The fractions of released radionuclides belonging to the medium and low volatile classes are presented in Table 1. The modelled in-vessel phase lasted till the RPV failure and included formation of two melt pools in the reactor pressure vessel.

Table 1 . Fractions of radionuclides released during in-vessel phase

Radionuclide	Calculated Fraction
Ba, Sr	0. 2
Te	1. 0
Ru	0.6
Mo, Nb	0.1
Ce	0.02
La ,Y	0.05
U	0.04

Ex- vessel release of fission products

The fractions of radionuclides released from the corium pool in the reactor cavity during ex-vessel phase are presented and compared for all three calculated scenarios in the following Table 2 .

Table 2. Computed fractions of radionuclides released from corium pool for all studied scenarios

Radionuclide	Scenario		
	1st	2nd	3rd
Ba, Sr	0.2	0.2	0.2
Te	0.01	0.01	0.01
Ru	1.7×10^{-6}	2.1×10^{-6}	2.1×10^{-6}
Mo, Nb	9.3×10^{-7}	1.0×10^{-6}	1.0×10^{-6}
Ce	0.4	0.4	0.4
La, Y	0.04	0.04	0.04
U	0.001	0.001	0.001

Two important findings can be derived from these calculations: first - radionuclide releases are only slightly dependent on whether the corium pool is flooded and cooled with water or not, and second - they are not strongly dependent on the corium surface area.

Source term

In the course of all of the three analysed scenarios (24 hours), the containment did not fail, so the source terms were very low. The lowest source term was observed for the sequence with the recovery of the ECC system and melt pool cooling. The fractions of selected fission products released to the environment for all of the three studied scenarios are presented in the following Table 3.

Table 3 . Fractions of radionuclides released for all studied severe accident scenarios

Radionuclide	Scenario		
	1st	2nd	3rd
Ba, Sr	1.9×10^{-6}	1.8×10^{-6}	1.2×10^{-6}
Te	7.5×10^{-6}	7.5×10^{-6}	6.3×10^{-6}
Ru	1.5×10^{-6}	1.5×10^{-6}	1.5×10^{-6}
Mo, Nb	5.9×10^{-8}	5.9×10^{-8}	5.9×10^{-8}
Ce	3.1×10^{-7}	3.4×10^{-7}	3.4×10^{-7}
La	5.0×10^{-8}	5.1×10^{-8}	5.1×10^{-8}
U	9.5×10^{-9}	9.7×10^{-9}	9.6×10^{-9}

Thus, the MELCOR calculations of the VVER-1000 unit within the LPP Project made possible to:

(i) define the boundary and initial conditions for the release of relevant radionuclides from in-vessel and ex-vessel melt pools
(ii) assess the influence of SAM strategies (corium spreading out of the cavity and cooling) on melt pool behaviour and on source term,
(iii) compare MELCOR results (FP releases, source term) with those of advanced codes or correlations

Regarding the influence of SAM procedure on corium ex-vessel behaviour and on the source term, the following preliminary findings can be formulated:

(i) widening of the melt pool area (melted material was allowed to spread out of the cavity) and corium cooling with water supplied by the recovered ECC system resulted in reduction of the melt pool mass and volume by 19% and 27% respectively,
(ii) the maximum temperature of corium was reduced by 150°C,
(iii) the most important aspect of application of the SAM procedure was a significant reduction (by 74%) in the maximum depth of corium penetration through containment basemat concrete.

On the other hand, application of the SAM procedure did not significantly influence the radionuclide release and behaviour during the ex-vessel phase. The following findings can be formulated:

(i) release of all fission products from the corium pool in the cavity was observed only during the initial, very short period of corium pool history, when the melt temperature was high,
(ii) there are no significant differences between ex-vessel releases for scenarios with and without application of the SAM procedure,
(iii) water cooling of the corium pool resulted in a reduction of the source term. As the pool scrubbing effect was not significant, the underpressure in the containment (in the second half of the scenario) was the major cause of the reduction in the fission products release from the NPP.

2.6. OPTSAM - Optimisation of Severe Accident Management Strategies for the Control of Radiological Releases

Within our participation in OPTSAM project and with relation to VVER-1000 (Temelín NPP) three strategies were selected for detailed studies, based on:

1. By-pass release control,
2. Reactor cooling system depressurisation and
3. Hydrogen management system.

The base case sequence of the bypass strategy (1) was initiated by SG Head Cover rupture (D_{eq} = 40 mm) and loss of all feed water systems (without any personnel intervention). Two sensitivity calculations were defined with two different operator actions: case 1A including emergency feedwater system (EFW) recovery at the time when core degradation begins, defined by the core outlet temperature criterion (650°C); and case 1B including the primary circuit depressurization via Atmosphere Dump System at the time of core degradation, beginning with EFW recovery (5 minutes delayed). The results are as follows:

- Recovery of EFW results in a significant delay of the fission products (FP) release, but the total amount of FP released remains practically unchanged ,
- Depressurization of the primary circuit, which terminates the containment bypass, has a significant impact on the mass of released FP (reduction for minimum of two orders of magnitude.

The base case sequence of reactor cooling system depressurization strategy (2) was initiated by loss of all feedwater systems, including the emergency ones. The sensitivity cases used two different paths of atmosphere dump system with different maximum flow rates. The first path (2A) leads from pressurizer to bubble tank, and the second (2B) - from all SG inlet collectors also to bubble tank. Depressurization starts at the time when the core degradation begins (defined by the core outlet temperature criterion (650° C). Results related to the impact on the sequence progression showed that:

- depressurization via first path resulted in the initiation of high-pressure injection system (HPI) - one or all three trains, depending on the version of calculation, but
- using of the second path resulted in too slow depressurization with the lower head failure before the initiation of HPI system.

Calculations performed within the scope of the RCS depressurization strategy showed that the idea of applying the atmosphere dump system to prevent RPV failure and significantly reduce the source term was too optimistic. The release could be prevented only by earlier and faster primary circuit depressurization than that in 2A sequence.

The base case of the hydrogen management system strategy was initiated by medium LOCA (D_{eq} = 100 mm) on cold leg, loss of all ECCs and recovery of one train of the spray system 30 minutes after the RPV lower head failure. A sensitivity calculation performed took into account passive autocatalytic recombiners (PARs) system. Under severe accident conditions the hydrogen production rate is significantly higher than under DBA. The calculated sequences showed that recovery of one train of spray system, leads to a significant decrease of steam concentration in the containment atmosphere and to hydrogen deflagrations. Application of PARs in the Temelin NPP has a very positive effect, even if their performance under the severe accident conditions is not sufficient to prevent the hydrogen deflagration, the system can significantly reduce the amount of hydrogen which is burnt and to reduce the pressure and temperature peak values, thus reducing risk of the loss of containment integrity.

The results and conclusions of OPTSAM project provide the first example of the SAM preparation methodology, optimized by the radionuclide releases and risk reduction potential. Application of integral codes within the project scope showed both their capabilities and needs of their improvement.

2.7. ICHEMM - Iodine Chemistry and Mitigation Mechanisms.

The global effort involves experimental studies, development of kinetic models of iodine reactions based on the results of these studies and quantification of the studied mechanisms effects on the iodine source term. This quantification will be based on computational simulations with iodine containment codes such as IODE and IMPAIR for which our Institute is responsible. Obtained results could be used directly for the preparation of the iodine specific accident management measures for VVER/440/213, similar to those which had already been adopted in Britain or France.

2.8. EURO-FASTNET –Future Advances in Sciences and Technology for Nuclear Engineering Thermal-Hydraulics

The project should provide a review of the R&D needs in thermal hydraulics for all types of European reactors in operation, including VVERs as well as that for considered innovative reactors designs, addressing issues connected with reactor performance, availability, reactor life span, and also reactor safety, code validation and uncertainty evaluation. NRI which is responsible for VVER-related applications has contributed all necessary VVER specifics.

2.9 . DEEPSSI – Design and Development of a Steam Generator Emergency Feedwater Passive System for Existing and Future PWRs using Advanced Steam Injectors

The objective of the DEEPSSI project which has been recently launched is to design and to test an advanced Steam Injector (SI) apparatus and then to use this apparatus to develop a passive Steam Generator Emergency Feedwater System (SG-EFWS) which can replace the steam turbine driven pumps in western type PWR and provide an additional reliable water injecting tool in the existing VVER-440/213 steam generators (NRI responsibility). VVER-related applications are foreseen.

2.10 . Safety related conclusions and lessons learnt

Benefits and inputs for plant modifications

Information and knowledge acquired as a result of projects within 5th Framework Programme (and preceding ones) significantly complement the knowledge base applicable for VVER safety purposes. It was, for instance, shown that for VVER-440/213 the preferable severe accident management strategies are those that lead to significant reduction of the source term inside the containment and thus reduce radioactivity release to the environment. An important lesson learnt is related to the volatile iodine forms behaviour and possibility to influence their concentration in the containment (ICHEMM).

Obtained data on severe accident sequences with containment bypass (for instance – with steam generator tube break) in VVER-440/213 and experimental investigation of radionuclides retention in steam generator with vertical and horizontal tube arrangement will permit to assess both common features and differences of the classic and VVER arrangements (SGTR).

Applicability of some strategies preventing the reactor vessel failure and reducing melt-concrete interaction is for both VVER types different. In this respect, VVER-1000 belongs to the same category as a prevailing majority of other PWRs in operation, i.e. the selected strategy is based on timely flooding of the core debris within the vessel - before it is damaged, and the possibility to cool the melt outside the vessel (ECOSTAR). For VVER-440/213 reactors the situation is more favourable since there is a possibility to employ other, very promising strategy of melt retention within the vessel; results obtained within ARVI project supplement existing information on this strategy.

Long-term aspects of the source term are treated in LPP project to which we contribute the VVER-1000 related analyses. OPTSAM results are a significant step towards optimisation of the global SAMG strategy. Individual modules developed within EVITA project allow to clarify a number of source term aspects and the progressive code ASTEC and its modules are one of the basic tools for quantitative evaluation of the different SAM programmes efficiency, allowing the unifying and objective comparison.

It is obvious that results of the 5th FP projects reach the final user – NPP Operator and it is he (Operator) who takes decision on the implementation of rationally justified measures, or decides to request for additional analyses of cost-benefit type.

We have limited these conclusions to the results of ongoing 5th FP projects in which we are directly involved with the objective to enhance VVER safety, especially with respect to severe accidents. However, our international co-operation link-ups are much wider, a good example is recently finalised PHARE PR/TS/03 performed for the needs of Regulatory Bodies of Czech and Slovak Republics and Hungary in assessing mitigating measures during severe accidents.

An important step in increasing quality of the safety evaluation should be validation of thermal-hydraulic codes against results obtained on the experimental stand VVER-1000 at Elektrogorsk in Russia. The relevant project is being prepared in co-operation with EC-PHARE and OECD/NEA.

Results obtained within 5th FP projects can be complemented with results of other OECD projects oriented at both PWR and VVER reactors, as for instance – MASCA, SETH, MCCI as well as such PHARE projects as for instance – Bubble Condenser Qualification, Development of Crisis Tools and Application of Risk Based methods. These projects and projects under preparation bring together a number of organisations that can, within 6th FP, develop effective networks capable to solve VVER related pressing issues.

3. COMPONENT INTEGRITY PROJECTS

3.1. FRAME – Fracture Mechanics Based Embrittlement Modelling

This project is designed to provide more quantified data on the relation between irradiation induced transition temperature shifts determined by Charpy notch impact tests and by static fracture toughness tests. Several sets of characteristic materials have been already chosen for the irradiation programme – PWR base metals and welds with low and high content of impurities typical for European- and US- made reactor pressure vessels (RPV), the IAEA reference JRQ material as well as VVER-440 and VVER-1000 base metals and welds. Both types of tests are performed in different laboratories and their data will be compared and analysed. Results will then serve as a base for

potential changes in reactor codes prediction formulae and for the decision about using the static fracture toughness data of irradiated RPV materials. Round robin testing of unirradiated specimens is prepared.

3.2. PISA – Phosphorus Influence on Steel Ageing

The project is carried out with the objective to reduce uncertainties associated with brittle intergranular failure of stressed material (RPV steel) due to phosphorus segregation on grain boundaries. Impact of this failure mechanism on RPV steel properties during service and at the end-of-life will be established.

The multi-partner project includes the existing data survey, experimental and analytical parts and is focused on reactor type specific steels, including 15Kh2NMFA steel used for VVER-1000 RPVs.

Results from unirradiated test conditions have been summarized in a report, a literature survey and critical analysis of degradation mechanism due to phosphorus were also finished and published within the project.

The project findings will allow to improve management of the plant life extension by accurate prediction of the end-of-life properties of RPVs which will make a significant contribution to reactor safety throughout Europe. Special importance of the project for VVER reactors follows from the fact that phosphorus induced radiation embrittlement in their RPV materials can be not only a leading mechanism but also the governing one for lifetime assessment. Thus, the results of this project should improve nuclear safety in VVER plants.

3.3. GRETE – Evaluation of Non-Destructive Testing Techniques for Monitoring of Material Degradation

Within this project the capability and reliability of innovative inspection techniques applied to the critical components for the purposes connected with planning failures prevention actions and possibility of lifetime extension (for existing power plants) are assessed. Experimental part consists in a round robin exercise, when aged (irradiated and fatigue damaged) samples will be tested by various techniques (ultrasonic, magnetic, thermoelectricity and hardness indentation).

Detection of radiation damage in reactor pressure vessel is studied on the PWR RPV as well as VVER-1000 RPVs materials, two different austenitic steels have been selected as typical for the piping materials. In both cases, several damage stages (i.e. different neutron fluence, and different fatigue damage values) have been studied which should allow to evaluate not only sensitivity of the methods but also their reliability. Evaluation of possible technology transfer to the industry for practical use of these techniques is foreseen.

Application of reliable and technological methods for VVER plants will allow to assess the instantaneous state of tested material in critical components and thus prevent any large damage or even any failure of the component.

3.4. SPIQNAR - Signal Processing and Improved Qualification for Non-Destructive Testing of Ageing Reactors

The project is of a high importance for in-service inspection (ISI) of VVER-440 and VVER-1000 critical components of the primary circuit taking into account NDT qualification, risk informed ISI and signal processing aspects.

Realisation of qualified ISI in compliance with ENIQ and IAEA methodologies is required by the Czech Nuclear Authority starting 2002. Another significant aspect of the project is related to the applied materials since a substantial part of VVER-440 primary circuit welds are austenitic stainless or dissimilar welds, and VVER-1000 primary circuit components (like RPV or primary piping) contain austenitic cladding, dissimilar welds are also present in the piping of emergency cooling systems.

A part of the SPIQNAR effort has been concentrated on the development of a prototype for Inhomogeneous Anisotropic SAFT (InASAFT). For this purpose the design and implementation of an InASAFT-specific mesh generator has been finished. Furthermore, a basic InASAFT algorithm was designed and implemented.

Preliminary lessons learnt within the early phases of this and preceding project confirm the necessity to apply verified reliable manufacturing technologies of synthetic but realistic defects, the importance of balance between technical justification and practical trials. These conclusions are comparable to the lessons learnt within the First Pilot Study of ENIQ.

3.5. CASTOC – Crack Growth Behaviour of Low Alloy Steel for Pressure Boundary Components under Transient LWR Operating Conditions

The specific topic of this project is investigation of environmentally assisted cracking (EAC) as the major ageing mechanism in conjunction with transient conditions of water chemistry and loading. The main objective is to reach more detailed understanding of the acting mechanism and to generate data on crack growth rates under complex loading and to simulate transient operating conditions, in particular - transient loads and transients in water chemistry. NRI performs autoclave tests with western type and Russian type RPV steels under BWR and PWR (VVER) water conditions in order to compare the susceptibility to SCC crack growth under transient loading for different RPV materials.

Another NRI contribution to the CASTOC project are tests for the crack growth under static load and cyclic load under PWR water conditions with different oxygen concentration under defined transient loading conditions with two VVER type steels (15Kh2MFA and 15Kh2NMFA).

The foreseen round robin exercise will increase reliability of the VVER data and it will significantly contribute to more reliable comparison of PWR and VVER material behaviour.

3.6. LIRES - Light Water Reactor Reference Electrodes

The main objective of the LIRES project is to develop reference electrodes that are robust enough for the use inside LWR reactors and will be applicable for monitoring the corrosion performance of stainless steel core components, which over the time accumulate extensive irradiation damage and are therefore susceptible to irradiation-assisted stress corrosion cracking (IASCC). The corrosion potential measured against a reference electrode allows to distinguish between conditions where IASCC is likely to occur or not.

Eight European research laboratories, NRI including, take part in LIRES project, developing high temperature reference electrodes and carrying out round robin test.

High temperature reference electrode that will be developed and recommended for future experimental corrosion programmes, performed in reactors, should be available at all laboratories and ultimately - widely used for corrosion tests. This will require some additional measurements, checking of such electrodes lifetime and maximum operational parameters as well as direct application in several different programmes.

3.7. Lessons learnt from integrity-related projects

Co-operation in projects that unify problems of both types of reactors, i.e. PWR and VVER, allows a cross application of experience. The PISA project can serve as an example where experience from the embrittlement studies of VVER materials with high phosphorus content is applied to western type materials. At the same time, the researched phosphorus ageing mechanism will help the VVER utilities in their Ageing Management Programmes.

Comparison of the different approaches helps to harmonise the PWR and VVER assessment codes. Search for a better intercomparison between different parameters of RPV radiation embrittlement in the FRAME project will allow to include fully the fracture mechanics lifetime evaluation into appropriate VVER codes, the same as it is used in PWR.

Non-destructive determination of material damage during in-service inspections can be a very useful tool for precluding failure of tested components. Results obtained within the GRETE round robin exercise will help in the validation of used methods and their verification. These methods after their successful validation within the programme could be applied as a part of in-service inspections for VVER RPVs, internals and pipings.

Qualification of non-destructive procedures used in in-service inspections are mandatory for VVER plants starting 2002. Thus, experience from such qualifications performed in compliance with

ENIQ requirements within the SPIQNAR project and applied to VVER materials, especially to different dissimilar and austenitic welds in reactor primary circuit, represent a very useful tool for the test results assessment.

Corrosion damage can play a significant role in any reactor component. At the same time, corrosion testing can depend very much on testing conditions that should be relevant to the operational ones. Intercomparison of crack growth studies in similar materials within the CASTOC project is useful for validation of test results obtained in individual laboratories and also for comparison of the PWR and VVER type materials behaviour, which will be used in the preparation of appropriate VVER assessment codes.

Corrosion studies necessitate a precise definition of testing/operating conditions. One of the most important parameters is the corrosion potential but its measurement in primary water conditions is complicated due to non-existence of a suitable corrosion electrode. The electrode developed within the LIRES project will be used to study VVER materials not only in autoclaves but also directly in reactor loops.

Results from all these projects will be in a final stage used by the utilities for the lifetime assessment and safe operation of their plants. Some results will be also included into procedures for integrity assessment and/or into requirements of the Nuclear Regulatory Body.

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