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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(2007)18
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CSNI INTEGRITY AND AGEING WORKING GROUP

PROCEEDINGS OF THE CSNI WORKSHOP ON STRUCTURAL RELIABILITY EVALUATION AND MECHANICAL PROBABILISTIC APPROACHES OF NPP COMPONENTS

Hosted by Electricité de France - With support from USNRC(USA) - SKI(SWD) - GRS(GER)

Held on September 25-27, 2006 in Lyon, France

The enclosed CD-ROM contains full papers and presentations

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NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full member. NEA membership today consists of 28 OECD member countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, the Slovak Republic, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

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

The CSNI Workshop on Structural Reliability Evaluation and Mechanical Probabilistic Approaches of NPP Components was held from September 25 to 27, 2006 in Lyon, France hosted by Electricité de France, with the support from USNRC (USA), SKI (Sweden) and GRS (Germany).

Pressurised thermal shock continues to be an important issue for long-term operation. In the past, FALSIRE [1], [2] and ICAS [3] benchmarks (OCDE-NEA projects) assessed deterministic approaches and first highlighted probabilistic issues. Probabilistic approaches are more and more widely used.

The CSNI and its Integrity of Components and Structures Working Group (IAGE WG) considered there was a need to clearly understand how calculations were performed in the different NEA member countries. Furthermore, recommendations on improvements or on common basis that could be adopted were found to be very valuable. Consequently, in June 2001 the CSNI initiated a benchmark on the probabilistic approach to examine the reactor pressure vessel – PROSIR Benchmark.

The main objectives of the PROSIR benchmark on "Probabilistic Approaches of RPV" were to:

- confirm performance of probabilistic approaches for RPV structural integrity
- compare and improve probabilistic fracture mechanic tools
- identify the major parameters and uncertainties that play a role in these approaches (e.g. flaw-type and distribution, toughness models and uncertainties, crack arrest, warm pre-stressing)
- issue some recommendations on best practices

The main objective of the Workshop was originally to discuss and disseminate the results of the OECD/NEA Probabilistic Structural Integrity of a PWR Reactor Pressure Vessel Benchmark (PROSIR). However, the Workshop objective has been broadened to cover probabilistic approaches of mechanical components in nuclear power plants.

The Workshop will assess the current practices and the state of the art with respect to these approaches. Learning from others is a key to success and progress. Information obtained as a result of this assessment will be utilised to develop a consensus on these approaches and identify issues or "gaps" in the present knowledge for the purpose of formulating and prioritising needs in this area.

The Workshop gathered 51 participants from 12 countries including the EC. Key players from the utilities, researchers and academy contribute to the discussion and share their experience and techniques on probabilistic approaches of mechanical components in nuclear power plants.

The CD-Rom attached to this copy contains full papers and presentations made at the Workshop.

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1 NEA/CSNI/R(1994)12 FALSIRE: phase 1: CSNI Project for Fracture Analyses of Large-Scale International Reference Experiments (phase 1): comparison report, 1994.

2 NEA/CSNI/R(1996)1 FALSIRE: phase II: CSNI project for Fracture Analyses of Large- Scale International Reference Experiments, 1996.

3 NEA/CSNI/R(1999)3 Comparison Report of RPV Pressurised Thermal Shock International Comparative Assessment Study (PTS ICAS).

ACKNOWLEDGEMENTS

Gratitude is expressed to Electricité de France for hosting the Workshop and to Mr. Francoise Hedin, Deputy Director of EDF – SEPTEN, for introducing the Workshop and welcoming the participants. Special thanks to Mr. Claude Faidy, EDF – SEPTEN, Chairman of the Sub-Group on the Integrity of Metal Components of the CSNI Working Group on the Integrity of Components and Structures for his help, support and excellent organization.

Thanks are also expressed to the Workshop Scientific Committee and to the Chairpersons for their effort and co-operation.

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Session 1.2	M. Kozluk (AECL, CAN) E. Ardillon (EDF, FR)
Session 2.1	B. Brickstad (SKI, SWD) E. Roos (MPA, GER)
Session 2.2	G. Roussel (AVN, BLG) K. Onizawa (JAEA, JPN)
Session 3.1	M. Kirk (NRC, USA) S. Chapulliot (CEA, FR)
Session 3.2	C. Faidy (EDF, FR) A. Huerta (OECD-NEA)

TABLE OF CONTENT

GENERAL CONCLUSIONS AND RECOMMENDATIONS

1. Deterministic versus probabilistic for safety analysis

- Probabilistic tools are essential to highlight effects of uncertainties around the deterministic criteria of safety analyses, in particular for structural integrity.
- The level of integration in the decision process of probabilistic considerations on structural integrity of passive components:
  - is different by country (from decision based on risk level in some countries to no consideration in a few other countries)
  - need a large investment from all partners (Utilities, Safety Authorities and Technical Support Organisations) to obtain conclusion acceptance
  - can be completed by "economic" considerations
- Moving to probabilistic approaches need:
  - validation and verification of methods, tools, models, data
  - minimum international harmonization

These important points have been discussed all along the Workshop on the basis of 2 typical structural integrity examples, the RPV integrity under pressurized thermal shocks and the piping system integrity (degradation acceptance to continue operation or leak before break).

The more active country in this domain is the USA with a large involvement of USNRC research and US nuclear industry. It's accepted in some conditions to decide on risk level, after an extensive work program on many years with all the parties associated.

But, many other countries are active in this direction - Sweden, Canada, Japan, UK, Korea, Czech Republic. Spain is supporting the ideas with a limited investment. In France, different actions are on-going at the utility level but not discussed yet with the Safety Authority. Germany has a limited investment and no acceptance from the Safety Authority.

2. Conclusion and recommendation

2.1 PROSIR RR

2.1.1 Major conclusions

- With same data and same models and criteria the results are in a good agreement for crack initiation of a single crack, but not perfect for the moment for flaw distribution.
- The brittle rupture model and uncertainty effects cannot be analysed globally, but it will be better to look at them separately.
- Some "user" error or un-precise "specification" can lead to major final result errors.
- The link between deterministic sensitivity studies and different probabilistic analysis is essential.
- After some consideration of thermo-mechanical stresses, the K estimation scheme and the corresponding criteria has been discussed (plasticity effects, crack arrest, WPS).

- Intermediate validation of the PFM analysis is important for verification of the result validity.
- Crack arrest has not been covered in this part of the work, but will be in a complementary program.
- More parametric and sensitivity studies are necessary (in a complementary program).

2.1.2 Recommendations

- Finish the existing phase through a final report with all the results (after checking that all the data are in their last version for each partners) and major updated conclusions.
- Make a specific effort to understand the reasons of differences for crack initiation through a flaw distribution (RR4).
- Integrate NURBIM EC project and PROSIR recommendations.

2.1.3 Future activity

- Continuation of PROSIR in 2007
  - consider crack arrest and warm pre-stressing in few cases
  - perform complementary sensitivity studies
  - finalise the methodology in a best practice document for RPV crack initiation and crack arrest of one crack in a flaw distribution under Pressurized Thermal Shock
  - move to a similar work on other type of problems like piping degradation and LBB, or SG tubes (to be discussed at the next IAGE meeting in April 2007)

2.2 PTS analysis: practical application and consequences

2.2.1 Summary and conclusion

- The more complete analysis has been done in the USA, many other countries are in pilot studies phase and starting discussion with their own regulators.
- Different level of criteria:
  - from crack initiation to crack arrest, with or without warm pre-stressing or constraint correction
  - up to risk consideration (CDF and LERF) to define the acceptable risk contribution of the RPV failure
- Final results are very high in terms of $RT_{NDT,PTS}$ (more than 150°C for plate type of RPV with longitudinal welds, largely more than that for forge rings type of RPV).
- Now, USNRC is looking with ORNL on consequences in relaxing this PTS criteria:
  - on the initial design criteria (ASME III - appendix G) - design rules are not re-evaluated with deterministic rules during operation for level C and D transients (these transients are only considered in the probabilistic PTS screening criteria, with some beyond design transients)
  - on the P-T limit curve (*on going*)
    - some results show that crack smaller than 1/4t and the removal of Safety Factor 2 on pressure, did not increase risk level (RPV failure), if WPS is considered
    - the key parameter during the cool-down is the duration at maximum pressure
  - on the acceptance standard of flaws in operation (Section XI- IWB) - *to be done*
  - on flaw evaluation in operation (Section XI- Appendix A) - *to be done*

- In general, it's not a regulatory request to justify the ISI performance by a detailed deterministic approach; the USA probabilistic PTS screening criteria does not take into account any periodic ISI; in general, the PTS screening criteria covers all the possible transients with their own frequency (including re-pressurisation transients or cold overpressure).
- All these probabilistic approaches can affect the existing operation rules in many different directions.
- Concerning the other countries the discussion with the regulators on RPV is just starting, but they are on the way to develop their own PTS screening criteria for their own plants with similar criteria. Some, like France, are more concerned by forged ring and crack initiation, in order to develop new safety factors, including effects of uncertainties on the major influencing parameters.

2.2.2 Recommendations

- Finally, a global consistency between the different approaches are needed (design, P-T, PTS, flaw evaluation) to assure a safe behaviour of RPV.
- Transferability of results, like PTS screening criteria, has to be reviewed in detail.
- RPV PTS probabilistic approaches are efficient if a large investment is done by the different parties (Utility, Regulator and Technical Support Organisations).

2.3 LBB and piping application

2.3.1 Summary and conclusion

- Two complementary objectives on piping integrity probabilistic approaches - risk informed ISI and LBB (including LB-LOCA re-definition).
- All of them have a key request - good understanding of the different degradation mechanisms.
- Again, probabilistic tools are essential to highlight effects of uncertainties around the deterministic criteria of piping integrity analyses.
- All the decisions are done for the moment on the basis of deterministic approaches, but in some countries the risk evaluation can be done in a probabilistic way to justify limited time of operation.
- For piping system the analysis results are more expressed in relative risk than in absolute risk level.
- A complete review of probability of leak versus failure has been done in USA between USNRC and industry, for main coolant loop and connected lines. The complete detail probabilistic work is too complex and time consuming for the moment (due to the number of degradation mechanisms to analyze) and an “expert elicitation”, with specific and detailed procedures, has been used to quantify the risk level and define a more realistic size for the LB-LOCA on USA plants (between 8" and 12"); without consideration of some external events like seismic loads.
- Different other examples are presented from Sweden, Canada, Germany and Czech Republic.

2.3.2 Recommendations

- Probabilistic highlights of margins for structural integrity of piping systems (including LBB) is an essential tool for decision makers (Utility and Regulators).
- The development of methods, tools, models and data for different degradation mechanism and leak flow rate / leak detection capability are key issues to obtain results with a reasonable confidence level.
- For piping system it's generally easier to work on relative risk improvement than in absolute risk level.
- Precise procedure to perform "expert elicitation" has to be developed.

2.4 Other applications

2.4.1 Partial safety factors

- Example for flaw evaluation in ductile piping system has been presented by EDF.
- A procedure to optimise the safety factors on crack size, load and material properties (J_{IC} criteria) in piping systems has been described.
- The methodology has been applied and integrated in the flaw evaluation procedure of the French RSE M Code and accepted by the French Safety Authority.

2.4.2 Canister

- Example of a different application using probabilistic at the design level and to define an operation surveillance strategy.
- This particular point is largely under discussion at the ASME Code level - how probabilistic highlights can be introduced at the design level?

3. General conclusion

- Probabilistic approaches give largely more information than a pure deterministic evaluation with the same level of knowledge (methods, tools, models, data).
- Now, it is an essential tool to highlight effects of uncertainties around the deterministic criteria for structural integrity of safety class components.
- The connection between the probabilistic highlights and pure deterministic approaches seems to be clear in the USA and is under discussion between users and regulators in many other countries (for example in the USA, without any crack over the ISI performance criteria, the level C and D transients are only considered for the probabilistic PTS screening criteria).
- These connections can be derived in different manners:
  - just highlights when some regulatory margins are close to the criteria
  - risk informed analysis
  - fitting of safety factors with reliability targets
- Applications are different for a unique high safety concerned component like RPV or numerous components affected by different degradation mechanisms (piping systems or steam generator tubes), but, in all cases a reasonable knowledge of the degradation mechanism (empirical or theoretical) is essential to be confident in the results
- A large investment of all the parties is necessary to use the results - Utility, Regulator, Technical Support Organisations

WORKSHOP PROGRAMME

Session 1.1: Opening Session

Chairpersons: Mr. A. Huerta (OECD-NEA) – Mr. C Faidy (EDF-FR)

- Welcome by EDF-SEPTEN.
  Francoise Hedin, Deputy Director of EDF – SEPTEN

- Welcome from OECD and Workshop objective.
  A Huerta (OECD-NEA)

- PROSIR General Presentation.
  C Faidy (EDF-FR)

- PROSIR Deterministic Results and Recommendations.
  C Faidy (EDF-FR)

- PROSIR Comparison of K Estimation Schemes.
  S Chapuliot (CEA-FR)

Session 1.2:

Chairpersons: Mr. M Kosluk (AECL-CAN) – Mr. E. Ardillon (EDF-FR)

- PROSIR Detailed Presentation of Probabilistic Results.
  C Faidy (EDF-FR)

- PROSIR Discussions of Differences in PFM Solutions.
  T Dickson (ORNL-USA)

- PROSIR Parametric Studies.
  S Chapuliot (CEA-FR)

- Identification of the Key Parameters in the PROSIR Round Robin Exercise.
  P Dillström, (Inspecta Technology AB, SWD)

- PROSIR Recommendations – Conclusions and General Discussion.

Session 2.1:

Chairpersons: Mr. B Brickstad (SKI-SWD) – Mr. E Roos (MPA-GER)

- An Overview of the USA Pressurized Thermal Shock Evaluation.
  M Kirk (NRC-USA) and T Dickson (ORNL-USA)

- Structural Integrity Analysis of Reactor Pressure Vessel Using Probabilistic Fracture Mechanics Analysis Code: PASCAL.
  K Onizawa (JAEA-JPN)

- French Probabilistic Pilot Study on Reactor Pressure Vessel Integrity.
  E Meister (EDF-FR)

- Korean Probabilistic Evaluation of RV Integrity under Pressurized Thermal Shock.
  G Sohn (KOPEC-KOR)

Session 2.2:

Chairpersons: Mr. G Roussel (AVN-BLG) – Mr. K Onizawa (JAEA-JPN)

- Probabilistic Approach to PTS Evaluation for VVER RPVs.
  V Pistora (NRI-CR)

- Risk-based Fracture Evaluation of Reactor Vessels Subjected to Cool-down Transients Associated with Normal Shutdown.
  T Dickson (ORNL-USA)

- Fitting of 3-Parameters Weibull and Log-Normal Distributions on Fracture Toughness Data.
  M Marquès (CEA-FR)

- LBLOCA Redefinition to Support 50.46 Rulemaking.
  M Kirk (NRC-USA)

- A Probabilistic Approach to Leak before Break Demonstration.
  P Dillström, (Inspecta Technology AB, SWD)

- Application of the Simulation Based Reliability Assessment to LBB Concept.
  L Pecinka (NRI-CR)

Session 3.1:

Chairpersons: Mr. M Kirk (NRC-USA) – Mr. S Chapuliot (CEA-FR)

- The Need for a Probabilistic Approach to Manage Cracking of Carbon Steel Reactor Coolant Piping at the Point Lepreau Generating Station.
  T Gendron (AECL-CAN)

- Development and Applications of Analysis Methods for Evaluating Structure Reliability of Piping Components.
  H Grebner (GRS-GER)

- Probabilistic Approaches for Safety Margin Assessment on the Mechanical Components of the French Operating NPPs.
  E Ardillon (EDF-FR)

- Probabilistic Analysis of Canister Inserts for Spent Nuclear Fuel.
  P Dillström, (Inspecta Technology AB, SWD)

- Current and Potential Applications of Probabilistic Methods to Ageing Management in CANDU NPP Structural Components.
  K Dinnie (NSSL-CAN)

Session 3.2:

Chairpersons: Mr. C Faidy (EDF-FR) – Mr. A Huerta (OECD-NEA)

- Round Table on Probabilistic Approach and Safety Analysis.
- Workshop Recommendations and Conclusions.

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