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Organisation de Coopération et de Développement Economiques
Organisation for Economic Co-operation and Development

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English text only

NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

NEA/CSNI/R(2003)17/VOL2
Unclassified

PROCEEDINGS OF THE JOINT CSNI/CNRA WORKSHOP ON "REDEFINING THE LARGE BREAK LOCA: TECHNICAL ISSUES AND ITS IMPLICATIONS"

Zurich, Switzerland
June 23-24

English text only

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996), Korea (12th December 1996) and the Slovak Republic (14 December 2000). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

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, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, 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 NUCLEAR REGULATORY ACTIVITIES

The Committee on Nuclear Regulatory Activities (CNRA) of the OECD Nuclear Energy Agency (NEA) is an international committee made up primarily of senior nuclear regulators. It was set up in 1989 as a forum for the exchange of information and experience among regulatory organisations and for the review of developments that could affect regulatory requirements.

The Committee is responsible for the NEA programme, concerning the regulation, licensing and inspection of nuclear installations. The Committee reviews developments that could affect regulatory requirements with the objective of providing members with an understanding of the motivation for new regulatory requirements under consideration and an opportunity to offer suggestions that might improve them or avoid disparities among member countries. In particular, the Committee reviews current practices and operating experience.

The Committee focuses primarily on power reactors and other nuclear installations currently being built and operated. It also may consider the regulatory implications of new designs of power reactors and other types of nuclear installations.

In implementing its programme, the CNRA establishes co-operative mechanisms with the NEA Committee on the Safety of Nuclear Installations (CSNI), responsible for co-ordinating the activities of the Agency concerning the technical aspects of design, construction and operation of nuclear installations insofar as they affect the safety of such installations. It also co-operates with the NEA Committee on Radiation Protection and Public Health (CRPPH) and the NEA Radioactive Waste Management Committee (RWMC) on matters of common interest.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA) is an international committee made up of senior scientists and engineers. It was set up in 1973 to develop, and co-ordinate the activities of the Nuclear Energy Agency 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 among the OECD Member countries.

The CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations, which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of the programme of work. It also reviews the state of knowledge on selected topics on nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus on technical issues of common interest. It promotes the co-ordination of work in different Member countries including the establishment of co-operative research projects and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups, and organisation of conferences and specialist meetings.

The greater part of the CSNI's current programme is concerned with the technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment, and severe accidents. The Committee also studies the safety of the nuclear fuel cycle, conducts periodic surveys of the reactor safety research programmes and operates an international mechanism for exchanging reports on safety related nuclear power plant accidents.

In implementing its programme, the CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities 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 and NEA's Radioactive Waste Management Committee on matters of common interest.

A. Foreword

The Committee on the Safety of Nuclear Installations (CSNI) of the OECD-NEA co-ordinates the NEA activities concerning the technical aspects of design, construction and operation of nuclear installations insofar as they affect the safety of such installations.

The Committee on the Nuclear Regulatory Activities (CNRA) of the OECD-NEA co-ordinates the NEA activities concerning the regulation, licensing and inspection of nuclear installations with regard to safety.

In December 2002, the CNRA and the CSNI jointly requested the NEA to organize a workshop on “Redefining the Large Break LOCA: Technical basis and its implications".
The Workshop was held on June 23-24, 2003 in Zurich, Switzerland hosted by HSK (Swiss Federal Nuclear Safety Inspectorate), PSI (Paul Scherrer Institut) and the OECD/NEA.

The objective of the Workshop was to facilitate an exchange of information on a topic, which could potentially impact both the operation of current reactors and the design of future reactors. A number of OECD countries were actively working in this area at the moment.
Regulators, Researchers and Industry representatives needed to exchange information on the current regulation and technical issues associated with the Large Break LOCA (LB-LOCA), and to further discuss rationales and motives which could lead to a redefinition of the LB-LOCA. The focus was on design and safety implications. Policy issues were not discussed but the workshop provided technical inputs for policy makers. The workshop covered different reactor designs (CANDUs, VVERs, LWRs).

The workshop was articulated over three questions:

- What drives the need to redefine the LB-LOCA?
- Does an adequate technical basis exist to support a redefinition of the LB-LOCA?
- What are possible new definitions for the LB-LOCA? What are their implications on current and future reactors?

A survey, completed by member countries, gave the participants a clear view on the current regulatory status and issues. The survey was intended to complement the workshop's discussions and provide general background information. It is published in a separate volume under the reference NEA/CSNI/R(2003)16

“Responses to the survey on "Redefining the Large Break LOCA: Technical basis and its implications"”.

These proceedings are divided into 2 volumes referenced NEA/CSNI/R(2003)17/VOL 1 and VOL 2.

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

B. Acknowledgement

Gratitude is expressed to HSK (Swiss Federal Nuclear Safety Inspectorate) and PSI (Paul Scherrer Institut), Switzerland for hosting the Workshop. In particular, special thanks to Dr. Sabyasachi CHAKRABORTY (HSK) and Mr. Martin ZIMMERMANN (PSI) for their help and commitments.

Thanks are also expressed to chairmen and moderators of the sessions and to the Organizing Committee for their effort and co-operation.

Organizing Committee:

Mr. C. Harwood	CNSC	CAN
Dr. J. Macek	NRI	CZ
Dr. J. Dusek	SUJB	CZ
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Mr. T. Kliment	VUJE	SK
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Mr. E. Mathet	OECD-NEA	International, Secretary

Chairmen and moderators of the sessions:

Dr. U. Schmocker	Director (HSK, Switzerland)
Dr. A. Thadani,	Research Director (US NRC, USA)
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	Dr. J. Sugimoto (JAERI, Japan)
Session 2:	Prof A. Alonso (University of Madrid, Spain)
	Dr. N.Chokshi (USNRC, USA)
Session 3:	Dr. J. Hyvarinen (STUK, Finland)
	Dr. V. Snell (AECL, Canada)

The Organizing Committee and the NEA would also like to express gratitude to Chairman Diaz of the US Nuclear Regulatory Commission for his participation in this workshop.

Joint CSNI/CNRA Workshop on
"Redefining the Large Break LOCA:
Technical basis and its implications",
Zurich, Switzerland, June 23-24, 2003

Volume 1

Volume 2

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Volume 1

Name of the speaker is in italic.

Volume 2

SESSION 2:

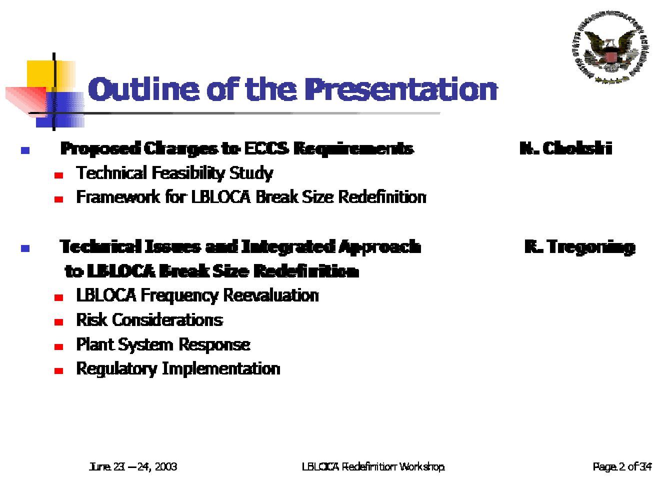
Does adequate technical basis exist to support a redefinition of the LB-LOCA?

Chairman: Dr. A. Thadani, (RES Director, US NRC, USA)
Moderators: Prof. A. Alonso (Univ. Madrid, Spain)
Dr. N. Chokshi (US NRC, USA)

Risk-Informing LBLOCA Requirements Critical Issues and Technical Approach

Dr. N. Chokshi – Dr. R. Tregoning, (USNRC, USA)

Joint CSNI/CNRA Workshop on
"Redefining the Large Break LOCA
Technical basis and its implications",
Zurich, Switzerland, June 23-24, 2003



Technical Feasibility Study

- **Feasibility assessment performed with respect to specific changes to 50.46**
- **Feasibility study included:**
  - Evaluation of current requirements, their basis and evolution
  - Review of related regulations and implementing documents
  - Review of risk information relevant to 50.46 and related accident
  - Development and comparison of options for risk-informing current requirements
  - Development of recommendations for changes
- **Feasibility study observed that:**
  - ECCS requirements are not commensurate with the risk significance of the various LOCA sizes and unnecessary conservatisms exist in the requirements

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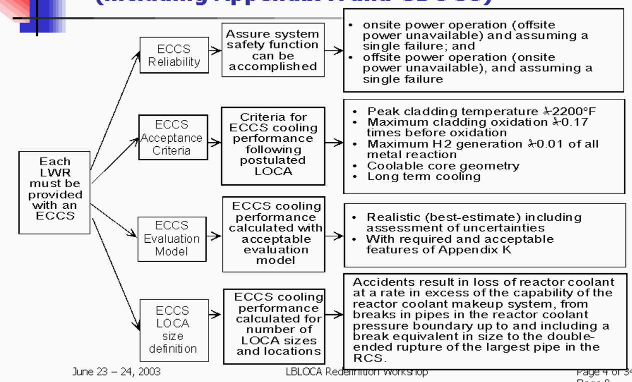
Overview of 50.46 (including Appendix K and GDC 35)



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Results and Directions

- ECCS reliability:
  - **As an option, replace LOCA/LOOP requirement with functional ECCS reliability requirement that is commensurate with the LOCA frequency**
- ECCS acceptance criteria:
  - **Provide a performance-based acceptance criteria for fuel rod integrity, maintenance of coolable core geometry, and long-term cooling capability.**
- ECCS evaluation model:
  - **All future applicants should use best-estimate codes for LOCA analyses**
  - **For current licensees, any changes that redefine the design basis LBLOCA should use best-estimate codes**
- ECCS spectrum of break sizes and locations:
  - **Provide a comprehensive "LOCA failure analysis and frequency estimation"**
  - **Prepare a proposed rule that allows for a risk-informed alternative to the present maximum LOCA break size**

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Framework for LBLOCA Break Size Redefinition

- Voluntary Risk-Informed Alternative
- Any proposed changes should be risk-informed and consistent with the principles of RG1.174
- ECCS functional reliability should be commensurate with the frequency of accident in which ECCS success would prevent core damage or a large early release.
- No changes to functional requirements unless fully risk-informed (For example, no change to ECCS coolant flow rates or containment capabilities to mitigate accidents).
- Only the non-significant contributions to risk are handled through severe risk accident management
- Realistically conservative estimates, with appropriate margin for uncertainty
- High quality PRA including low power and shutdown operations.
- Use a 10-year period for the estimation of LOCA frequency redistributions, with re-estimation every 10 years and review of new type of failures every 5 years.
- Operational changes should be reversible

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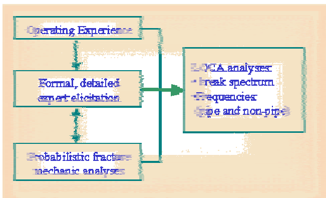
LB-LOCA Frequency Reevaluation

- LBLOCA Frequency Reevaluation
- Risk Considerations
- Plant System Response
- Regulatory Implementation



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LOCA Frequency Reevaluation: Historical Operating Experience Basis

- **Advantages**
  - Consistent with methodology used to develop failure rates of active components and easily implemented into a probabilistic risk assessment framework.
  - Identifies piping systems and degradation mechanisms which are most important contributors to current predicted failure rates.
  - Provides information on the effectiveness of mitigation techniques for known degradation mechanisms.
  - Provides an indication of the importance of precursor leaks prior to rupture for some failure mechanisms.
  - Identifies trends and possible importance of emerging degradation mechanisms.

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LOCA Frequency Reevaluation: Limitations of Operating Experience

- **Comprehensive database required to accurately assess importance of rare events (No LBLOCA in LWR operating history).**
  - Reporting requirements and accuracy are variable and depend on the degradation mechanism and piping system.
  - Passive system failures are reported using various mechanisms.
- **Not necessarily representative of future system performance.**
  - Material aging and environmental effects are not always accurately captured in the experience base.
  - No similar maintenance plan as with active components to ensure applicability historical failure rates.
- **Methodology based on existence of precursor event prior to failure.**
  - Not all degradation mechanisms exhibit a precursor leak prior to failure.
  - Development of the conditional failure probability given a precursor event is mechanism specific and has historically been poorly known.

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LOCA Frequency Reevaluation: Historical Role of PFM

- **Use**
  - Future risk associated with known degradation mechanisms.
  - Relative risk ranking of known mechanisms (Risk-Informed ISI).
- **Advantages**
  - Predicts future piping system performance for particular degradation mechanisms. Mitigation effectiveness can be considered and compared.
  - Provides qualitative severity ranking among known degradation mechanisms that are not as sensitive to model inaccuracies.
- **Limitations**
  - Requires physically-based models to simulate all active degradation mechanisms.
  - Results are extremely sensitive to model formulation and input variable assumptions. Variability of several orders of magnitude are typical.
  - Applications often are not benchmarked against operating experience.

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LOCA Frequency Reevaluation: Technical Issues

- **LOCA contribution variables span broad technical fields**
  - PFM, piping design, piping fabrication, operating experience, materials, degradation mechanisms, thermo hydraulics, operating mitigation practices, stress analysis, nondestructive evaluation, human factors, etc.
  - Integrated, multidisciplinary approach required.
- **Impact of material aging and environmental effects must be considered and benchmarked against operating experience.**
- **Contribution of both precursor leaks and nonprecursor LOCA contributors must be considered.**
  - Credit leak detection (leak-before-break) as appropriate.
  - Identify and quantify contributions of non-leaking cracks prior to failure.

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LOCA Frequency Reevaluation: Technical Issues, cont.

- **Large pipe breaks have justified operating margin for non-design basis passive failures.**
  - Ensure that mitigation flexibility is maintained for accidents exceeding any relaxed design basis.
  - Better understanding of other LOCA contributors is necessary.
- **LOCA contribution of non-piping passive failures must be accounted for to accurately gauge LOCA risk.**
  - SG manway failure.
  - Vessel penetration cracking.
- **Consider likelihood and severity of future, surprise degradation mechanisms.**
  - Difficult to assess the unknown or unexpected events.
  - Necessary to guard against appearance of new mechanisms by a LOCA.

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LOCA Frequency Reevaluation: Technical Issues, cont.

- **LOCA estimates will account for uncertainties.**
  - Quantify uncertainty resulting from analysis.
  - Identify major causes of uncertainty which can potentially be evaluated in future.
- **Need to assess new mechanisms and trends which erode or improve LOCA basis.**
  - Present knowledge of future LOCA frequencies will always have uncertainties.
  - Monitoring programs must be developed to allow continual assessment of LOCA precursor events.
  - Research must evaluate mechanisms and trends which may become important and continually estimate their severity.

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LOCA Frequency Reevaluation: General Approach

- **Operating Experience**
- **Expert Elicitation**
  - Re-evaluate historical LOCA frequencies.
  - Develop relationship between leak rate/break size and expected frequency for LBLOCA events.
  - Provide input to probabilistic LOCA computer code development.
- **Probabilistic LOCA Code Development**
  - More rigorously combine operating experience and PFM insights.
  - Explicitly consider contributions from piping and non-piping components, and the evolution of new degradation mechanisms.
- **Continuous LOCA Assessment**
  - Develop and maintain LOCA precursor database through expansion of existing pipe failure database.
  - Identify emerging degradation mechanisms and conduct anticipatory research to assess LOCA significance.

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LOCA Frequency Reevaluation: Expert Elicitation Process

- **Expert opinion (elicitation) is a formal process for providing quantitative estimates for the frequency of physical phenomena when the required data is sparse or when the subject is too complex to adequately model.**
  - The rarity of LOCA events (data sparseness) is evident.
  - Complexity is evident in the enormous pipe system variables which must be considered to model from first principles.
  - Complexity also exists due to the many potential non-pipe LOCA failure modes which contribute to the LOCA spectrum.
- **Elicitation has been used successfully to solve similar problems.**
  - Development of seismic hazard curves.
  - Performance assessments for high-level radioactive waste repository.
  - Determination of reactor pressure vessel flaw distributions.

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LOCA Frequency Reevaluation: Expert Elicitation Approach

- **Pilot Elicitation (Complete).**
  - Conducted last year using 11 internal (NRC) experts with broad knowledge-base.
  - Provided interim results for evaluation of ECCS reliability.
  - Developed possible framework for subsequent elicitation and identified strengths and weaknesses to address in formal elicitation.
  - Identified technical issues for consideration within formal elicitation.
- **Formal Elicitation (Ongoing).**
  - Individual elicitations conducted for each expert that is monitored by a facilitation team.
  - Twelve external experts assembled from nuclear industry, DOE laboratories, consultants, and international regulatory agencies with broad knowledge-base.
  - Facilitation team is comprised largely of NRC personnel.

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LOCA Frequency Reevaluation: Formal Elicitation Approach

- **Issue Development.**
  - Define scope and objectives of elicitation.
  - Construct approach for determining baseline LOCA frequencies.
  - Determine significant issues affecting baseline LOCA frequencies.
  - Develop framework for elicitation questions.
- **Base case quantitative estimation.**
  - Develop quantitative estimates for well-defined piping conditions.
  - Two estimates using PFM and two estimates from service history analysis.
- **Elicitation question formulation.**
- **Individual elicitations**
  - Provide answers to questions and justification for answers.
  - Discuss significant issues which impact LOCA frequency estimation.
- **Analyze quantitative results and qualitative rationale.**
- **Results summary**
  - Summarize quantitative and qualitative results for elicitation questions.
  - Summarize analysis methodology and LOCA results.
  - Obtain feedback from the expert panel.

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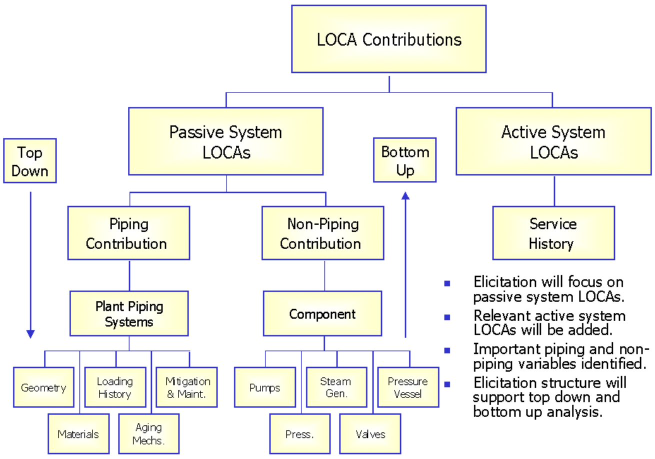

- Elicitation will focus on passive system LOCAs.
- Relevant active system LOCAs will be added.
- Important piping and non-piping variables identified.
- Elicitation structure will support top down and bottom up analysis.

LOCA Frequency Reevaluation: Passive LOCA Code Development

- **Objectives:**
  - Determine the relationship between break size and expected event frequency for large primary system pipes (>150 mm diameter).
  - Provide confirmatory analysis of elicitation results.
  - Develop tool which can be used for subsequent frequency re-evaluation.
- **Approach**
  - Construct separate modules to consider piping, non-piping contributions, and future surprise mechanisms.
  - Modules will couple state-of-the-art PFM modeling with understanding of operating experience historical, recent, and potential degradation mechanisms to determine frequency partitioning.
  - Scale modeling frequencies using expert judgment to determine the LBLOCA frequency.
  - Use insights from elicitation to initially focus on the most important systems and mechanisms.
  - Monitor and interact with the NURBIM program which has similar objectives.

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LOCA Frequency Reevaluation: Continuous Assessment of LOCA Challenges

- **Objectives:**
  - Develop framework for evaluating operating experience for LOCA precursor events in piping and non-piping components.
  - Evaluate mechanisms or trends which could be detrimental affect future LOCA frequencies.
- **Approach**
  - **Participate in the CSNI-sponsored OECD Piping Database Exchange (OPDE) project to expand international operating experience.**
    - Twelve participating countries: Belgium, Canada, Czech. Republic, France, Finland, Germany, Japan, Korea, Spain, Sweden, Switzerland, U.S.
    - The SKI-pipe SLAP database serves as the baseline.
    - Year 1 of 3 year effort is focusing on events between 1998 and 2001. Year 2 on events from 1995 – 1998 and 2002.
  - **Continue to evaluate international operating experience and research results to identify trends and mechanisms which warrant further research.**

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Risk Considerations: Technical Issues

- LBLOCA Frequency Reevaluation
- Risk Considerations
- Plant System Response
- Regulatory Implementation

Risk Considerations: PRA Scope and Quality

The plant-Specific PRA model needs to reflect

- All contributors
  - Internal events: transients, LOCAs, floods, fires
  - External events: seismic, high winds, floods
- All operating
  - Full-power
  - Low-power
  - Shutdown
- The as-built and as-operated plant
- The operating history of the plant
- The plant-specific response to an event
- The key sources and impacts of the uncertainties

Risk Considerations: PRA Scope and Quality

- Coordinate with nuclear industry to develop appropriate PRA standards.
- Two documents for assessing Level 1 and Level 2 Internal Events (Excluding fire) at full power operations.
  - ASME RA-S-2002, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications*.
  - Draft regulatory guide 1122, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*.
- American Nuclear Society (ANS) is currently developing additional external event and low power/shutdown standards.
  - Standard for *Probabilistic Risk Assessment for Nuclear Power Plant Applications: External Events*.
  - Standard for *Probabilistic Risk Assessment for and Nuclear Power Plant Applications: Low Power and Shutdown and internal fires*

Risk Considerations: Improved Modeling

- Study risk associated with various break sizes among LBLOCA events.
  - Response time may vary with break size.
    - Slower diesel start times for smaller LOCA sizes.
    - Increased time for operator actions.
  - Functional requirements may also be a function of size.
    - Lower peak pressure in containment.
    - Reduced maximum containment heat removal.
- Investigate impact of non-piping LBLOCAs.

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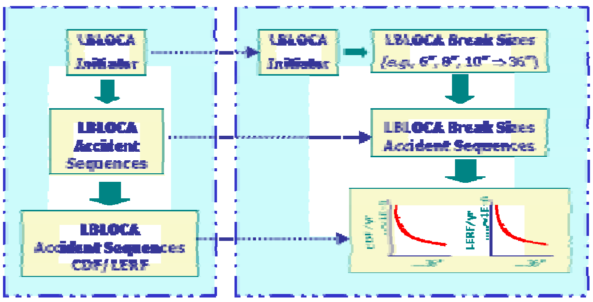
Risk Considerations: PRA Modeling



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Risk Considerations: Design Basis Rationale

- Systematically investigate the risk significance of LBLOCA sequences outside the proposed design basis.
- Systematically investigate the integrated risk impact of proposed design basis changes on non-LBLOCA sequences.
- Identify sequences outside the proposed design basis that require best estimate thermal-hydraulic analysis.
- Identify severe accident management strategies for sequences outside the proposed design basis.

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Risk Considerations: Long-Term Risk Monitoring

- Periodically reevaluate impact of new mechanisms and other factors that could change the LBLOCA frequency.
- Periodically reevaluate the risk impact of plant operational changes.
- Incorporate the observed equipment reliability to ensure that implemented plant changes retain acceptable risk significance.
- Investigate impact of revised accident sequence timing from best estimate thermal-hydraulic analysis.

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Plant System Response: General Technical Issues

- LBLOCA Frequency Reevaluation
- Risk Considerations
- → Plant System Response
- Regulatory Implementation

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Plant System Response: General Technical Issues

- **Must understand performance implications associated with specific operational configuration changes.**
- **Plant transients may change considerably depending on allowed operational changes.**
  - Single failure assumption.
  - Off-site power availability.
- **Adequacy of current evaluation codes and methods.**
  - Most best estimate methods developed for LBLOCA, but not SBLOCA.
  - Experimental data range of conditions may not be adequate for new scenarios.
- **Need to ensure that operational changes do not significantly alter the expected LOCA frequencies.**

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Plant System Response: Approach

- **Determine impact of design basis revision.**
  - Evaluate acceptance criteria margin for revised design basis with no plant changes (baseline).
  - Evaluate acceptance criteria margin for anticipated plant changes relative to baseline.
  - Examine consequences of LBLOCAs outside the revised design basis with anticipated plant changes (defense-in-depth).
- **Perform calculations for surrogate design basis accidents**
  - Steam generator manway failure.
  - Upper and lower head penetration failure.
- **Evaluate applicability of existing experimental data base for predicting plant response under anticipated operating changes.**
- **Revise and update regulatory guides for LOCA analysis.**

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Integrated Approach: Regulatory Implementation Plan

- LBLOCA Frequency Reevaluation
- Risk Considerations
- Plant System Response
- Regulatory Implementation

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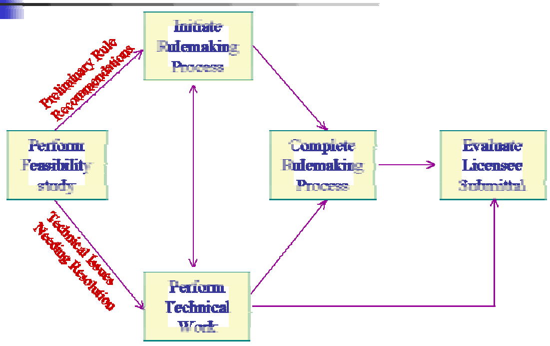
LBLOCA Break Size Redefinition: Rulemaking Process



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Integrated Approach: Regulatory Implementation Plan

- **Technical basis**
- **Establish decision criteria.**
  - Determine risk criteria and appropriate metrics.
  - Examine defense-in-depth sufficiency.
- **Implement revisions.**
  - Rulemaking process.
  - Revise existing license requirements.

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Integrated Approach: Regulatory Rulemaking Steps

- **Define rule requirements and formulate rule language.**
- **Solicit early stakeholder interaction.**
- **Develop supporting documentation.**
  - Technical basis.
  - Licensee guidance.
  - Regulatory analysis.
- **Seek Commission approval for proposed rule.**
- **Solicit and incorporate public comment.**
- **Establish final rule.**

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Preventing failures by in-service inspection

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1. Introduction

The concept of defence in depth is fundamental to the safety of nuclear installations. Defence in depth consists in a hierarchical deployment of different levels of equipment and procedures in order to maintain the effectiveness of physical barriers placed between radioactive material and workers, the public or environment in normal operation, anticipated operational occurrences and in accidents at the plant.

The primary way of preventing accidents is to achieve a high quality in design, construction and operation of the plant, and thereby to ensure that deviations from normal operation are infrequent. According to international guidance on defence in depth[1] design relies on deterministic assumptions and procedures without explicit consideration of probabilities. Conservatism, including safety margins, is a basic prerequisite, which apply to the first three level of the defence.

In-service inspection and testing are important means on the second level to prevent incidents and accidents by detecting any degradation of components and equipment before it can affect the safety of the plant. Moreover, in-service inspection and testing of structures, systems and components important to safety need to be of such a standard and frequency as to ensure that levels of reliability and effectiveness remain in accordance with the design assumptions and intent, and that the safety of the plant has not been compromised since beginning of operation.

2. The early in-service inspection strategies

The design arrangements of components in fossil-fuelled power plants has always taken into account the need for ready access to all components to facilitate inspection, maintenance, repairs, and replacement of components as required. With the development of nuclear power plants in the early 1960s, system designers initially believed that periodic in-service inspection of passive components in the reactor coolant pressure boundary would be impractical due to the radioactivity of the systems. The system designers also assumed that in-service inspection would be unnecessary provided the components were designed, constructed and manufactured to higher quality standards than those applied to fossil-fuelled power plants. Consequently, very limited attention was given to the need for in-service inspection in the early nuclear power plants and the systems were not provided with adequate access for the inspection of many important components. As the number of nuclear power plants in service increased concerns about the possibility of service induced defects were however raised. Therefore, the nuclear industry and the regulatory body in USA began to

[1] Defence In Depth in Nuclear Safety, INS 10. A report by the International Nuclear Safety Advisory Group. International Atomic Energy Agency, 1996.

develop criteria for in-service inspection, and in late 1960s the first draft code for in-service inspection of nuclear reactor coolant systems was published.

The code, ASME Section XI, were formally accepted by U.S. Atomic Energy Commission (predecessor to the Nuclear Regulatory Commission, NRC) in early the 1970s, and had considerable influence on in-service inspection requirements in many other countries. The code defined a classification system, similar to safety classification, for identification of those components whose malfunction or structural failure could potentially impair the safe continued operation and thereby would be merited for surveillance measures throughout the service lifetime of the plant. Different examination categories were also defined in the code. The selection of specific areas to be inspected in each component category was to be based on factors such as:

- environmental condition, such as irradiation that could cause embrittlement of the reactor vessel belt-line;
- operational transients, such as system start-ups and shutdowns, which could induce fatigue as a result of cyclic strains;
- component design configurations identified with higher stress field, such as vessel nozzles, weld joints, and structural discontinuities between piping, pumps and valves;
- material properties of dissimilar metal joints, such as weld joints between austenitic stainless steel and ferritic steels that may be subject to additional thermal strains in service.

Initially considerations, discussed by Chockie et al [2], led to the suggestion to perform 100 percent inspections of the most critical areas during each scheduled shut down period. Because the time required to complete such inspections proved to be excessive and costly a more practical approach was taken. This was based on the frequency of anticipated refuelling outages and inspection programme, which relay on representative sampling. The developed code therefore came to include sampling plans with fixed percentages of components for each of the code class and component examination categories, to be inspected during a given inspection interval. In the earlier editions of the Section XI code the inspection interval was 10 years. At least three inspection periods corresponding to one-third of the inspection interval were also required. Some probabilistic optimisation studies[3] led however to the introduction of an alternative approach in the 1976 edition of the code for determining inspection interval. With the assumptions and data used in the study, marked advantages of performing in-service inspections early during the operation lifetime rather than equal-spaced inspection interval were revealed.

3. Augmented inspections and NDT qualifications

In the late 1970s and early 1980s intergranular stress corrosion cracking was observed in many boiling water reactors (BWR) piping systems. The cracking problems were in many cases discovered by leakage, and not by in-service inspections. These observations led to concerns about the efficiency of the inspection programmes that were based on ASME Section XI or similar codes, and which prescribed the selection of specific areas to be inspected, more or less on the basis of design stresses. When the root causes of the cracking mechanism had been identified was it possible to develop relatively effective augmented in-service inspection programmes for early crack detection with non-destructive testing (NDT) systems.

[2] L.J. Chockie, S.H Bush, R.R. Maccary, *Extended rules of the 1974 ASME Section XI Code "In-service inspection of nuclear power plant components"*. Presented at IMechE Conference held in London 1976.

[3] L.M. Arnette, *Optimisation of in-service inspection of pressure vessels*. Presented at the ASM Conference on Non-destructive Testing in the Nuclear Industry held, in Denver USA 1975.

At the same time service degradation of pressurised water reactor (PWR) steam generator tubing began to occur. The degradation of this tubing, which still is ongoing in some PWR plants, is caused by many different mechanisms; e.g. wear, wastage, vibrational fatigue, intergranular corrosion attack, stress corrosion cracking, etc. Many nuclear power plants, both BWR and PWR, have also been faced with the problem of thermal fatigue cracking in different components, such as reactor pressure vessel nozzles and piping. Erosion-corrosion in PWR piping system has over the years caused several failures, of which some have lead to accidents in which people have been injured. Service degradation of reactor pressure vessel internals that are necessary for, e.g. maintaining core geometry and emergency core cooling, has also been observed.

All these degradation incidents have, in a similar way as the piping stress corrosion problem in the 1970s and 1980s, been followed by extensive work to develop augmented inspection programmes for periodic control of susceptible components, and diagnose affected parts.

However, several inspection failures have also occurred over the years, and international round-robin studies, such as the international PISC (Programme for Inspection of Steel Components) programme, and similar national exercises have therefore been performed to measure or estimate NDT performance. Many of these exercises have shown discouraging defect detection and defect sizing performance plus a large variability between participating inspection teams, even between teams using similar inspection procedures. Consequently, with the degradation and inspection failure history that the nuclear industry has been faced with so far together with the awareness that further degradation will occur over the years to come it was obviously necessary to develop and implement performances based qualification methodologies for assessment of the effectiveness of NDT systems. Such NDT qualification requirements have now been introduced or are on the way of being introduced in many countries. These requirements are either in the form of regulations or regulatory guides or in the form of individual plant licence conditions. Transition periods still apply in some countries, but the requirements will become fully effective in the near future.

4. Experience and further developments

Since the introduction of augmented inspection, and other similar programmes developed for specific degradation mechanisms, together with more stringent requirements for demonstration of NDT performance relatively few cases of serious degradation not detected by inspections have been reported. This is a general observation based on failure experience world wide. A detailed evaluation of SKI's database STRYK, which include information of all degradation occurred in the Swedish nuclear power plants between 1972 and 2002, also shows that more than 90% of the 900 incidents with cracks and other defects reported have been detected by in-service inspections.

However, many of the present inspection programmes used in different countries requires extensive inspections. In a situation when the nuclear power generation industry are facing an increasing economic pressure due to fierce competition in deregulated electricity markets this has led to discussions on the need to optimise the in-service inspections. Many nuclear utilities have therefore started to develop and implement risk-based or risk-informed inspection approaches for these optimisations. Also regulatory bodies have recognised that probabilistic risk assessments have evolved to the point that it can be used increasingly as a tool in regulatory decision making. As a result a number of risk-informed approaches are now evolving in the nuclear industry in different countries. Examples are as follows:

In USA the regulatory body, NRC, has developed regulatory guidance for the use of risk-informed in-service inspection and the nuclear industry have developed a both quantitative and a semi-quantitative approach for practical applications. These applications are now rapidly being introduced in U.S. nuclear power plants. More than half of the U.S. plants apply of risk-informed in-service inspection in partial or full scope and many others are expected to apply for application in the near future.

The application is restricted to in-service inspection of piping systems and the NRC reviews the licensee's bases for the assessment that the proposed change meets the intent of the ASME Code requirements. Additional augmented inspection programs to address generic piping degradation problems have been recommended by the NRC to preclude piping failure and implemented by the industry. Notable examples of augmented programmes for piping inspections are to address intergrannular stress corrosion cracking (IGSCC) of stainless steel piping in boiling water reactors (BWR) (NRC Generic Letter 88-01), thermal fatigue (NRC Bulletin 88-08, NRC Bulletins 88-11, NRC Information Notice 93-020), stress corrosion cracking in pressurised water reactors (PWR) (IE Bulletin 79-17), Service Water Integrity Program (NRC Generic Letter 89-13) and flow accelerated corrosion (FAC) in the balance of plant for both PWRs and BWRs (NRC Generic Letter 89-08). The manner in which the augmented inspection programs for piping are addressed is also reviewed.

In Spain the regulatory body, CSN, also has developed regulatory guidance for the use of risk-informed in-service inspection for piping systems. Many Spanish plants now apply of risk-informed in-service inspection in partial or full scope, and in which they have included previous augmented inspection programmes.

In Sweden a qualitative risk oriented approach was introduced in 1987 for piping and other components in the piping system. After a transition period of five years it become mandatory in 1992. It was also extended to all components and parts thereof, except reactor pressure vessels and some steam generator parts. For the reactor pressure vessel and for steam generator tubes special rules still apply. As in Spain these risk-oriented programmes include previous augmented inspection programmes.

5. Discussion

As stated in the introduction to this paper in-service inspection and testing are important means in the defence in depth strategy to prevent incidents and accidents by detecting any degradation of components and equipment before it can affect the safety of nuclear power plants. The efficiency of applied in-service inspection programmes, and thereby how well they can fulfil their role, depends on many factors such as strategies and procedures for

- identification and ranking of reactor system parts and components in which failure can have an impact on the safety level;

- identification of components which can be degraded by different kinds of operating conditions;

- determining inspection intervals and the extent and frequency of inspections;

- demonstration of the performance of non-destructive testing systems to be used during inspections.

In many in-service inspection strategies components are consequence ranked via safety or quality grade classification systems. This is a transparent and robust but relatively rough approach. The consequence index grading in the present Swedish risk oriented system represents an attempt to differentiate more closely between possible consequences, in terms of loss of reactor coolant. More accurate estimates of the possible consequences can be achieved by the use of PRA. The strength of PRA is recognised as lying in its ability to address the importance of various components, systems, safety functions and structures. PRA has turned out to be effective in ranking the importance of components independent of the complexity of the accident sequences they are included in. However, as recommended by a

European nuclear regulatory task force[4], a living plant specific PRA is an important prerequisite for any application of risk oriented inspection. Full-scope Level 1 PRA is certainly required, and Level 2 PRA provides insights in the risk to the environment. Highly detailed PRA model construction is also emphasised, in order to maintain the PRA in a living fashion. The PRA models and data have to be updated with a certain frequency and every time there are substantial changes to plant configuration or in-data. Other important aspects are e.g. the use of plant specific data in PRA, and a related continuous plant specific data collection and processing system needs to be set up and maintained. Sensitivity studies and uncertainty analyses are also necessary to identify the impact of the lack of knowledge about data, assumptions and phenomena on the quantitative analysis results.

In any in-service inspection strategy, based on either purely traditional deterministic approaches or on qualitative or quantitative risk oriented approaches, its efficiency will strongly depend on how well components possible susceptibility to different kinds of degradation mechanisms can be assessed or determined. In the early in-service inspection strategies influential factors for degradation considered, were more or less focussed on design stresses. Since then substantial knowledge has been accumulated about degradation influential environmental, loading and other operating condition in relation to components design and material composition. Many of these more realistic degradation influential factors were introduced for component selection via the different augmented inspection programmes. However, degradation history shows clearly that such factors must be continuously updated based on research results and detailed damage analyses, which often reveal other circumstances than those expected.

Many quantitative risk oriented inspection strategies also require defect distributions and probabilities for a certain type of defect occurring as input data to the analyses. Such estimates have to be based on comprehensive failure databases, with detailed information of not only the type of degradation, but also actual environmental and other relevant plant conditions. SKI's STRYK database is one example of a database with information that can provide input data to quantitative risk analysis. However, further work is needed to compile failure data for such quantitative risk oriented approaches. An international project, OPDE, has recently started to establish a data base with world wide failure data. In this work concerns have been raised about the difficulties to get enough detailed background information of individual cracking and degradation events. Such detailed information is necessary to judge whether a set of data is relevant to a specific plant or not.

It should also be noted that the accumulated failure and degradation history contained in this type of data bases to a large extent is a result of the extensive inspections that have been performed over the last two to three decades. Estimation of pipe rupture frequencies from such data can therefore be biased.

Furthermore, the quantitative risk oriented inspection strategies require fracture mechanics models that can be used for reasonably accurate estimation of leak and break probabilities. Several such models have been developed, but there are still questions about their accuracy and validation status.

Other aspects to be considered in the further development of risk oriented in-service inspection approaches relate to the extent of inspection and sample size determination. It is obvious that for components with a dominating influence on risk, the sample size should be greater than for less risk significant components. In the present qualitative system used in Sweden the sample size required for high-risk components are 100%. US NRC also

[4] *Report on risk-informed in-service inspection and in-service testing.* Prepared by the Nuclear Regulator´s Working Group (NRWG) Task Force on Risk Based In-service Inspection. June 1999. EUR 19153.

recommends in one of their guides[5] 100% inspections for high-risk regions. For medium and low risk components sampling is necessary. The fixed percentages prescribed in many inspections regulations and codes are based on expert judgement. However, statistical models exist for determining the sample sizes and performing sensitivity analysis. A general observation when applying such models is that the sample sizes will be relatively large depending on inspection effectiveness required, how the populations are defined, assumptions made on defect distributions, and detection reliability of the non-destructive testing systems to be used.

There are also some fundamental aspects to be discussed during the process of further development. All risk-oriented philosophies are built on the extensive experience that has been gained over the large number of operational years and use of this knowledge to predict future events. Consequently questions must be raised about how we can be sure, that current experience and knowledge of e.g. degradation mechanisms that occur in nuclear plant is complete. The answer is of course that our knowledge not is complete. A recent example of this is the Davis Besse reactor vessel head degradation. Therefore can not in-service inspection programmes or other means in the defence in depth strategy be purely risk based. Also future in-service inspection programmes have to include high consequence components or regions but with low risks due to estimated low failure probability based on current knowledge. Examples of such components are reactor pressure vessels. Only then can in-service inspection programmes fulfil their two similar but slightly different roles:

- to ensure that the probability of failure of safety significant components is low or negligible and

- to provide confidence that the probability of failure of such components is low or negligible.

6. Summary and conclusions

The basic purpose of an in-service inspection programme is to decrease the plant risk by decreasing the probability of failure, inspection having obviously no impact on the consequences of a failure.

Development of in-service inspection strategies for passive components in nuclear power plants has always involved implied, but unquantified, risk assessments. In the early in-service inspection strategies components were consequence ranked by a safety or quality grade classification system.

In these strategies influential factors for degradation considered, were more or less focussed on design stresses.

During the late 1970s and early 1980s component failures due to different types of cracking and other types of degradation were observed in many nuclear power plants in several countries. The cracking and degradation problems were often discovered by leakage, and not by in-service inspections. These observations led to concerns about the efficiency of the inspection programmes that were based on ASME Section XI or similar codes.

Since the introduction of augmented inspections, and other programmes developed for specific degradation mechanisms, together with more stringent requirements for demonstration of NDT performance relatively few cases of serious degradation not detected by inspection have been reported.

Substantial knowledge has been accumulated over the years about the degradation mechanisms that can cause component failures and how this type of degradation can occur.

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[5] *An approach for plant specific risk-informed decision making: In-service inspection of piping*. Draft Regulatory Guide DG-1063. U.S. Nuclear Regulatory Commission. January 1998.

Developments and sophistication levels of PRA have increased the risk insights. The experience and knowledge gained have successively been used to further improve the efficiency of the in-service inspection programs.

A number of risk-informed approaches are now evolving in the nuclear industry in different countries. Opportunities for further improvements and optimisations clearly exist. Such improvements, based either on qualitative or quantitative risk approaches, have however to consider the fact that in-service inspection and testing are important means in the plants defence in depth strategy. Any risk-informed approach should therefore, as far as possible, consider the fact that unexpected degradation also will occur in the future. Any risk-informed in-service inspection approach also has to be balanced with other optimisation efforts in the plants and that are based on risk considerations.

Leak Detection and Monitoring

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General

Leaks can lead to the release of radiation within the containment as well as loss of coolant accidents if such leaks increase in size. Large leaks are controlled by the reactor protection system which forms part of the safety system of every nuclear power plant. This system reacts to signals issued in response to pressure increases or activity increases in the containment and level drops in the pressurizer.

Public discussions in Germany on “pipe cracking” in nuclear power plants have brought the subject of leak detection to the fore. Demands from licensing authorities have led nuclear power plant operators to request the design and implementation of a concept for upgrading existing leak detection systems. Independent of this, international organizations as IEC, see a need for action in this area.

Work on U.S.A. standards started in the 1973’s finally leading to the international paper ISA-67.03-1982 with the title “Standard for Light Water Reactor Coolant Pressure Boundary Leak detection”. The purpose of this standard is to standardize criteria, methods, and procedures for assuring the design and operational adequacy of reactor coolant pressure boundary leak detection systems used in light water cooled nuclear power plants. A further objective of this paper finalized in 1982 is to encourage design improvements yielding increased utility and reliability of reactor coolant leakage detection systems.

This paper will summarize the basic aspects of leak detection as described in ISA-67.03-1982, gives an overview on the standard technique called LDS in Germany and then, concentrates on a rather new sensitive leak detection systems called FLÜS, based on local humidity monitoring inside the insulation of components to be monitored. The actuating situation of this system was the requirement of authorities for a sensitive leak monitoring system at the closure head of the reactor pressure vessel at a German nuclear power plant. Since that time several systems of this type were installed in the frame of updating leak detection systems according to the ISA standard or to apply special demands in WWER reactors as well as in other light water reactors in Europe and America.

International Standard

The significance of leakage from the Reactor Coolant Pressure Boundary (RCPB) will depend upon the leak location, the leakage rate, duration, and the nature of the flow path permitting leakage. Through-wall cracks or flaws are the most difficult to detect and monitor because they can occur at any RCPB location. This type of leak is also of most concern because the leak may develop from some unpredicted combination of internal defects and external stresses in a non-isolatable portion of the RCPB.

The standard distinguishes between those critical “unidentified leak“ and “identifiable leak” at known locations such as mechanical seals, reactor pressure vessel (RPV) seals, body overpressure protection fittings, pressure relief valves or leakage into collection systems, e.g. pump seal or valve packing leakages. The intent is to avoid identifiable leaks and provide measures for a most sensitive monitoring of unidentified leaks.

According to ISA-67.03-1982 at least ***three dissimilar, diverse, and independent principal methods*** of monitoring coolant leakage from the RCPB to the containment shall be provided. One of these methods shall be sump level and/or sump flow monitoring. A summary of all acceptable methods are mentioned in the following:

- *Sump level change*
  - *Required sensitivity of 1 gpm*
  - *Separate measurement of identified leakage (valve stem packing glands or other sources)*
- *Air radioparticulate and radiogas activity monitors*
  - *Specification of sampling system design*
  - *Assumptions for design computations*
- *Containment air cooler condensate flow collection*
  - *Sensitivity of 1 gpm*
- *Humidity monitoring of containment air*
- *Monitoring of the integral parameters as temperature, pressure*
- *Visual Inspection and tape moisture sensors*

But, the standard explicitly points to further leak detection systems in future, which might fulfill and/or surpass the standard capabilities. The main items of capabilities are:

The sensitivity and response characteristics for each of three principal leak detection monitoring systems shall be shown by design calculations or performance tests to be capable of indicating and alarming **1 gpm (228 liters/h) leakage increase within one hour**. The systems according to the standard fall into the safety category of non-nuclear safety systems or monitoring instrument systems.

All systems shall be designed to operate whenever the plant is not in cold shutdown condition and operate at the range of actual environmental conditions (with respect to temperature, humidity, radiation, pressure, humidity etc.) around the system components.

Seals, relief systems, and other probable sources of leakage shall be identified. Leakage to the primary reactor containment from identified sources shall be collected or otherwise isolated so that flow rates from identified leaks are monitored separately from unidentified leaks.

Provisions shall be made to monitor systems connected to RCPB through passive barriers for indication of intersystem leakage. Acceptable methods include radioactivity monitoring and water inventory monitoring.

The specific leakage detection methods (see above) were explained in more detail in ISA-67.03-1982. Especially the task, special system design criteria, procedures are mentioned such as

- sampling system design and assumptions for design computation for air radioparticulate and radiogas activity monitors,
- baseline of normal condensate flow for air cooler condensate flow collection or
- normal fluctuations of humidity monitoring of leakage detection.

All leak detection methods mentioned in the standard can be graded into further aspects:

- Is it an integral method to monitor many potential unidentified leakages with unknown locations? Does the arrangement of sensors cover all potential unidentified leakages?
- Is the method being able to quantify the leak rate?
- Can the method be used to get sufficient information with respect to location, thus, potential cracks are found?

Looking to the basics it becomes clear that the aspects are not fulfilled by all methods to the same extent. Radiation measurements might be a proper leak detection system, but is not directly designed to measure leak rate. Sump level is simple to use to measure the integral leak rate, but not the right method to localize. Thus, it is clear that the proposed combination of at least three methods is a supplementation of the necessary properties, too:

- To detect,
- To localize/ identify the leak and

- To quantify the leakage rate.

Standard LDS in Germany

The standard leak detection system LDS is used for detecting leaks in pressure-retaining components in support of the leak-before-break (LBB) concept for the containment, main steam and feedwater, auxiliary system building, annulus and valve compartment. The LBB concept means that design and materials selection ensure that a through–wall crack will not cause failure of the component without prior existence of a stable leak for a prolonged duration. A detectable leak size below the response threshold of the reactor protection system means that a sufficiently long period of time is available for a leak to be evaluated and repaired where necessary. The system falls into the category of operational information systems without initiating automatic intervention in the process.
The advantage of LDS lies in its global leak detection ability with a reasonable effort. LDS comprises

- An relative humidity measurement systems of the compartment air at selected positions including room air temperature monitoring,
- The condensate mass flow measurement system, forming part of the recirculation system,
- The level monitoring system in the building sumps,
- Evaluation and display system.

Humidity measurement reacts most sensitively to leaks. Absolute humidity has proven to be the best method, as the increase is proportional to leak rate. Absolute humidity is determined by dew point sensors or by its calculation of relative humidity and air temperature.

A mass flow measurement was developed to measure condensate buildup in the ventilation systems. The sensitivity can be adjusted to cover the range of interest. The water level in the building sumps are determined by capacitance level meters.

The design-specific sensitivity of the system is determined during planning with a computer model. A guideline for the sensitivity threshold which has proven itself in practice is 30 g/s (108 Kg/h). A properly designed LDS permits reliable detection of leaks of a mixture of water and steam on this order of magnitude within several minutes.

Acoustic Leak Monitoring Systems

As already mentioned in the ISA-67.03-1982 as a potential method *acoustic leak detection systems* have been developed and installed in nuclear power plants serving as one of the three diverse methods of the standard. This is for instance the ALÜS system of Framatome ANP installed in several WWER reactors in Eastern Europe.

ALÜS uses the fact that leakages in pressure retaining components cause high-frequency sound waves (100 to 500 kHz), which spread throughout the structure of the pressure boundary in the form of structure-borne noise. ALÜS detects and locates the leak by recording differences in the intensity measured at different places vs. time. Within a given range of variation, the leakage rate can be deduced from the leak noise for leaks involving cracks.
An ALÜS monitoring leak detection system comprises

- Ultrasonic pick-ups
- Preamplifier
- Signal station (filter, RMS formation, digitization, digital signal transfer)
- Ultrasonic transmitter for leak simulation
- Data concentrator and evaluation computer

The availability of the system is continuously checked by the evaluation of the background signal during operation and periodically by activating the transmitter leak simulation.

ALÜS is a *prompt* leak detection system reacting within around 1 minute. It is able to cover specific areas as the main components of the whole primary circuit and serves as a direct method to determine the *location* of an unidentified leakage by comparing the level of sound at adjacent *point sensors*. For this purpose, a computer model is used, experimental attenuation measures are done in the outage of the plant and their characteristic parameters are implemented to the model.

The sensitivity of ALÜS depends mainly on the level of background noise. It comprises the leakage range from 25 kg/h at the pressurizer and the adjacent safety valves up to 220 kg/h at the main coolant pumps. Its sensitivity and other properties fulfill the requirements of ISA-standard.

ALÜS can be used to detect and monitor intrinsic leaks at safety valves.

FLÜS – a sensitive *local* and integral humidity leak detection system

FLÜS is an innovative leakage monitoring system. By detecting leaks of only 1 kg/h (= 0,004 gpm) it surpasses the sensitivity of ISA-67.03-1982 by **two orders of magnitude**.
FLÜS has been designed for nuclear power plants to cover all questions mentioned above, e.g.

- To detect leakages at a very early stage,
- To give a sufficient location and/or identification of the leak position and
- To quantify the leak rate.

The system was firstly installed in a German PWR reactor in 1994[6] for monitoring the closure head of the RPV ensuring the integrity of the penetration nozzles. This area started to be under special discussions since cracks and a leak had been found in a French reactor in 1991 (Bugey 2) and later in many other reactors of the same type. Recently a rather "big hole" at one control rod drive mechanism (CRDM) nozzle at Davis-Besse, U.S.A. was under public pressure indicating the general crack problem of adjacent materials selected for the penetrations of this type of RPV's.

The leak found in Bugey 2 was only in the range of 2 kg/h measured during a pressure test7. Consequently, the German authorities required an adequate very sensitive leak detection system, in the range of 1 kg/h, although some differences of fabrication were pointed out with respect to the procedure of heat treatment of the closure head after welding.
FLÜS was firstly installed in 1993 and finally set in 1994 for continuous operation at the German power plant. Further projects included the extension of the method to cover larger areas including the method of global humidity monitoring of the compartment rooms.

Key notes of FLÜS

FLÜS is the only method being able to measure the absolute humidity inside the insulation of components at elevated temperatures up to 400 °C and more. This explains its very high **sensitivity**:

- As the vapour of a leak is firstly detained by the insulation in a comparable low volume, even very small leak rates lead to a clear increase of humidity in this area.
- Due to the high temperature inside insulation the absolute humidity level increases up to levels of 100 °C dew point. That is at least 1 to 2 orders of magnitude (in terms of vapour mass per volume) more than normally achieved, when humidity escapes to the environmental air in the containment and condensation takes place.

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6 At nuclear power plant KWO, Obrigheim

7 by the acoustic leak detection method. The method is not sensitive enough during operation of the plant (see above).

FLÜS does not use point sensors, but a distributed sensor tube along the component to be monitored, and measures the humidity along the component as a spatial profile. That makes it efficient to cover longer distances up to more than 1 km with one central system.
FLÜS does not use any electronic component at the components to be monitored. It uses only a metallic tube with some defined small porosity at selected intervals, where the vapour is penetrating into the tube by diffusion process. This leads to a very reliable method, whereas necessary electronic components (one central humidity sensor, pressure gauge, mass flow meter etc.) can be placed in accessible areas or even outside of the containment.

Basics of FLÜS

FLÜS measures this increase of local humidity by a sensor tube running inside the insulation, preferably mounted along the pipe's or pressure vessel outer surface. The metallic sensor tube is resistant to high temperature and high radiation level.

Initially dry instrument air is injected into the tube. Moisture emanating from a leak diffuses through porous (0.5µ) sintered metal elements placed at selected interval (standard length: 1 m) along the sensor tube. The air inside the tube is moved in fixed time intervals, the measuring cycle, through one central moisture detector which measures the absolute humidity level, the dew point τ, as a function of time. A record of the dew point τ vs. time, starting when air movement begins and showing a humidity peak, indicates the position of a leak along the sensor tube from the known air flow velocity inside the tube. With each measuring cycle new instrument air is fed into the tube, thus, the same preconditions for the next measuring cycle is fulfilled. The measuring cycle is typically in the range of 15 to 20 min and the air flow velocity in the range of approx. 1 m/sec.

For an automatic self check of the system, a fixed amount of vapor - as a test gas - is injected directly into the sensor tube for each measuring cycle. It moves through the tube's complete length L, resulting in an indication (in the following called as „test peak") measured by FLÜS and indicating the length by its time of arrival. The amplitude and the time of arrival of the test peak are checked to be within given ranges (typically 10 %): deviation causes an alarm indicating a system fault. In addition, the arrival time of the test peak is measured for each measuring cycle and correlated to the precisely measured length L. Thus, the location of the dew point signal along the tube is calibrated for small deviations in the flow meter (mass flow of air).

The patented method measures a defined fraction of the absolute humidity around the sensor tube determined by the surrounding temperature and diffusion time. Without any leak, background profiles are measured serving as reference curves.

Beside the method of local monitoring inside the insulation of a component, FLÜS can monitor globally the air in the compartment room either adjacent to the component to be monitored or systematically distributed in the Containment. In this case special room sensitive tubes (RST), small sections of a sensor tube (length of only 30 cm) are used.
The modular design of FLÜS enables flexible applications to all kinds of monitoring tasks. The main components comprise

- Sensor tubes and non-sensitive tubes for connecting purpose to FLÜS monitoring lines. Up to 8 lines of each 300 m sensor tubes are possible.
- Compressor unit for providing dry air, if not available in the plant.
- The analog station of the system including sensor module (humidity , air mass flow, pressure and temperature sensor), calibration module (device for short injection of a defined amount of vapor), valve modules (for distribution to several FLÜS lines , if needed), and data interface,
- The processing station for control, evaluate and store measurement data.

Field Experiences

The main parameter determining the sensitivity of FLÜS is the construction/geometry of the insulation. Leak simulation procedures during plant's operation are applied after commissioning of the system proving the capability to detect specified leakages. For this purpose, distilled water is injected to the position of the sensor tube being at elevated temperature > 100 °C. For low leak rates the water completely evaporates and serves as a defined source of vapor. By varying the leak rate a calibration curve can be provided for quantifying a potential leak.

Field tests show that leak rates of less than 1 kg/h can be detected and a sufficient assessment of leak rate can be given. No leak was undetected including leak simulation tests and one actual leak.

FLÜS location accuracy of a leak is better than 2 % of the total length of the concerned monitoring line whereas the temperature profile along the line have to be considered. This provides a good measure to identify a potential leak.

FLÜS has firstly been qualified in Germany, later in other countries during commissioning procedures. In one situation a leak was detected one month after installation of the system. It was a flange leak at the Upper Block of the RPV in a WWER reactor. The leak rate was estimated to be around 5 to 10 kg/ h. Shortly after detection the plant was shut down, the leak identified and repaired. After 3 days the plant was back to operation. FLÜS has proven to be a reliable and efficient leak monitoring system. In this case, the system monitors the complete primary circuit including local and integral humidity monitoring using 570 m sensor tubes and 12 RST.

The track record of operational experience can be summarized as follows:

- 100 % reliability of installed sensor tube
- Non- availability < 0,003 (according to Mil-Standard)
- Automatic continuous monitoring
- No false alarms
- Low maintenance scope

For a Canadian customer a design study was made for calculating the non-availability according to military standards.

Summary

ISA-67.03-1982 standardize criteria, methods, and procedures for assuring the design and operational adequacy of reactor coolant pressure boundary leak detection systems used in light water cooled nuclear power plants. It indicates that at least three diverse methods should be applied. The leak detection sensitivity recommended is rather moderate with 1 gpm. Experiences with cracks and consequential leaks show that more sensitive methods are necessary in special cases. A rather new method based on local humidity measurement is able to cover this problem and is able to provide an integral monitoring of the plant, too.

SESSION 2 (continued):

Does adequate technical basis exist to support a redefinition of the LB-LOCA?

Chairman: Dr. A. Thadani, (RES Director, US NRC)
Moderators: Prof. A. Alonso (Univ. Madrid, Spain)
Dr. N. Chokshi (USNRC, USA)

Does adequate technical basis exist to support a redefinition of the LB-LOCA?

Presentations on
Ageing management of PWR piping systems

Claude FAIDY (EDF-SEPTEN, France)

Joint CSNI/CNRA Workshop on
"Redefining the Large Break LOCA:
Technical basis and its implications",
Zurich, Switzerland, June 23-24, 2003

1. Introduction

The structural integrity of the reactor coolant system (RCS) of PWR's is a key safety issue. As high-energy piping and essential piping system for cooling the reactor core, the design basis considered different hypothetical double end guillotine break ruptures, generally in 11 locations [fig. 1]. The consequences of that importance are the design, fabrication rules and the surveillance programs of these lines in operation.
The field experience is in accordance with these precautions and limited degradations have been encountered up to now.
Different evolution of these initial design bases and different practices of surveillance program are used in a case by case application process in different countries:

- leak before break
- realistic break opening section and break opening time [table 1]
- risk informed in-service inspection

2. General description of MCL

In USA, different piping designs are encountered between Westinghouse [fig. 2], B&W and CE plants [fig. 3]. The Westinghouse plants have 2, 3 or 4 loops with one reactor pump (RCP) in each loop and the B&W and CE plants have 2 RCS loops with 2 RCP in each loop. The main connected lines are: a surge line, safety injection lines, charging line and residual heat removal lines.

The designs for French, Japanese and German PWR's are similar to that in Westinghouse-designed PWRs, except that the material of German PWRs are different (ferritic instead of stainless steel).

The designs of VVERs are slightly different (up to 6 loops and some have motor operating valves included in the hot and cold leg) with the 2 types of materials [fig. 3].
Inside diameter of these lines are between 711mm (28") and 1066mm (42") and corresponding thickness from 56mm (2.2") to 104mm (4.1").

The design pressure is in general less than 18 MPa and design temperature less than 350°C, for a lower temperature of 10°C during safety injection.

The major differences are:

- Straight pipes: centrifugally cast or forged stainless steel, cladded ferritic steel, wrought or rolled plate with longitudinal welds
- Elbows: 2 shells with longitudinal weld, bend without welds, cast

- Nozzles/tees: set-in or set-through, with or without dissimilar welds, with or without shop welded safe end, cast or forged [fig. 4]
- Dissimilar metal welds: 308-309SS or Alloy 82:182 NiCrFe alloy

The different materials encountered are: 316/304/08Cr18Ni10T, CF8A/CF8M/CF8, A106/SA516/SA508.

Specific requirements are imposed to the water chemistry:

- metal release rate has to be minimised
- corrosion has to be avoided
- deposition of heat transfer surfaces has to be minimised
- the dose rate build-up has to be reduced
- radiological oxygen formation has to be suppressed as far as possible.

The important parameters are the boric acid, lithium hydroxide and hydrogen concentrations, and the resulting pH level. For current PWR operation, typical range of $pH_{300°C}$ is 6.9 to 7.4. The boric acid concentration is decreased along a reactor cycle and start at a relatively high level (1000 to 2500 ppm). The lithium hydroxide is co-ordinated with the boric acid concentration to achieve the desired pH. The typical concentration at the beginning of the cycle is in the range of 1.8 to 4 ppm, and then is reduced with the boron concentration reduces. Hydrogen is added to the primary coolant with typical concentration of 25 to 50 cm^3/kg.

3. Design basis: regulations, codes and guides

The RCS piping are subjected to regulations, codes and standards of the countries where the different plants are operated.

In USA, the Code of Federal Regulations, Title 10, Part 50 (10CFR part 50) contains rules for the design, construction, operation and inspection of Nuclear Power Plants (NPPs). Only ASME Code Section III and XI [5,6] are endorsed by this regulation. Before 1963, pipings were designed to ANSI/ASME B31.1 "Power piping" or B31.7 "Nuclear piping".

French and German have developed their own codes (RCC-M/RSE-M [3,4] and KTA [7]) in order to fulfil specific regulatory requirements and to fit with their own industrial organisation. Similar development has been done in Russia for VVER plants.

The Code requirements concern:

- design
- fabrication/control
- over-pressure protection
- pressure test
- operation
- in-service inspection
- repair-replacement
- quality assurance

The different damages considered in these Codes are mainly:

- plastic deformation
- collapse load
- buckling
- fatigue on the basis of a design transient list
- partially, rupture and corrosion

For different damages the safety factors are clearly expressed in the Code, for some others they are not. Some other potential damages are not necessary strictly covered by the existing Codes.

Different levels of criteria are proposed by the Code in accordance with the frequency of the transient. For level D criteria a combination of RCS rupture with seismic event are generally considered.

The design lifetime is generally of 20-30 or 40 years of operation. Different requirements are under discussion in each country to define how this initial design life could be extended to 50 or 60 years. Periodic safety reviews are proposed in different country, except in USA for the moment where License Renewal process is proposed.

4. Ageing mechanism and operating experience

Six ageing mechanisms have to be analysed for RCS piping [from 1, 2]: thermal fatigue, vibration fatigue, thermal ageing, primary water stress corrosion (PWSCC) cracking, boric acid and atmospheric corrosion.
Some of these damages have been encountered:

- low cycle thermal fatigue, mainly in connected lines and nozzles
- vibration fatigue, mainly on small connected lines
- thermal ageing of cast duplex stainless steel elbows and nozzles [fig. 5]
- PWSCC of dissimilar metal welds
- Boric acid corrosion, mainly on outer surface and connected to a leak
- Atmospheric corrosion on some DMW

Some are not encountered, or not completely sure, just possible through laboratory tests:

- high cycle fatigue close to high ΔT mixing tees (as charging line nozzle)
- thermal ageing of welds and dissimilar metal welds

5. Assessment methods and ageing management

All these degradation mechanisms are not covered by design code rules, except low cycle fatigue. All the others are considered to be not active by material choices, fabrication qualification and installation and pre-test analyses.

Nevertheless, all the design code rules considered mainly initial material properties and not necessary end of "real" life values.

Concerning the capability of managing these degration mechanisms, general assessment methods are available or under development through R&D programs in order to define the threshold of activation of this mechanism, its kinetic to predict potential degradation rate and fitness for service criteria.

Operation specification and surveillance programs are used to assure high safety level of these corresponding lines: fatigue monitoring systems, leak detection systems, in-service inspection of more sensitive areas (with some limitation of performance for some materials)…

In some cases, repair and replacement techniques are available and could be implemented.

6. Recent evolution in RCS piping reliability

2 major concerns of RCS piping are:

- leak before break
- risk-informed in-service inspection

6.1 Leak before break

In order to replace double end guillotine break in 1 ms of RCS piping, US-NRC has proposed to replace that event by the larger connected line (no LBB) break, in order to simplify the RCS piping supports. And consequently, improve ISI quality and dose rate to perform any maintenance action by improving access to these piping systems.

The background of this approach is different probabilistic approaches done in the early 80's that have confirmed higher probability of "leak before break" than "break before leak".

As an example, in these studies for a Westinghouse plant [8], on the West cost, leak probability was 1.5 10^{-7} per reactor-year with 90% confidence and break probability was 1. 10^{-9} per reactor-year with 90% confidence. For East cost, the probabilities are reduced to 10^{-8} and 10^{-11} per reactor-year.

In all case the probability of rupture by indirect causes are greater.

A Piping Review Committee" [9] review in this period the RCS piping reliability for PWRs and BWRs and issued some requirement to mechanistically justify LBB application, including safety factors.
Different requirements are proposed:

- no water hammer, no vibration, no fatigue, no corrosion
- justification of material properties necessary for justification (mainly toughness)
- demonstration of large critical crack size in accidental situation (seismic loads) regarding the leak detection capability in normal operation with safety factors (10 on leak detection capabilities and 2 on critical crack sizes)

All the US Westinghouse PWR plants have justified LBB application and have simplified the corresponding supports in the early 90's.

Different countries, on a case by case process, have adopted this procedure, in connection with local regulation [10]. Specific procedures have been developed in different countries as in Germany the "Basic Safety Concept".
In all cases:

- the design and fabrication rules lead to "high quality" components with corresponding justification;
- the seismic loads have to be limited to design rule criteria;
- all degradation mechanisms have to be under control by reliable models, monitoring and ISI;
- leak detection system performances have to be in accordance with the mechanical justifications.

6.2 Risk-informed ISI

A lot of pilot studies done in different countries and mainly in USA in the past 10 years, have analysed the RCS reliability for different applications, and specifically to optimise the ISI program with risk considerations. The risk is defined as the product of degradation probability and consequences as Core Damage Frequency (CDF).

The results of these different studies confirm the very low probability of DEGB of PWR RCS systems, including major degradation mechanism (with very simple models), without ISI except for DMW connected to major components.

7. Conclusions

The RCS piping systems of existing PWRs have generally a very high quality standard for design, fabrication and operation rules. The corresponding field experience confirms very limited degradations encountered in these systems.

Nevertheless, some degradations appears recently (SCC of DMW in VC SUMMER and RINGHALS), some have no direct consequences encountered (as loss of toughness by thermal ageing), some are potential and not encountered for the moment (but no ISI can justify absence of early degradation due to the thickness of these piping systems); some other questions are not completely covered in this presentation, as risk of brittle fracture for some cladded ferritic pipings.

The RCS reliability remains very high, but an important effort has to be maintain to understand, case by case, encountered and potential degradation mechanisms is an essential contribution to assure long term high safety level of PWR plants.

8. References

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[10] LLBB 95 conference, Report OCDE/GD(96) 11, Lyon, France, June 1995

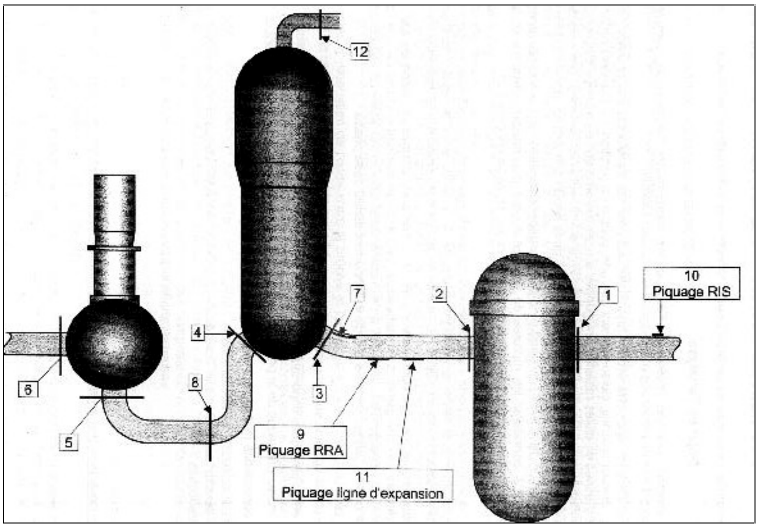


Figure 1 : DEGB locations on French PWRs

Ruptures	Aire maximale de brèche (cm²)			Temps d'ouverture brèche		
	Brèches convent.	Brèches réalistes	**Brèches FAR**	Brèches convent.	Brèches réalistes	**Brèches FAR**
R1	930	900				
R2	930	550				
R3	1350					
R4	2500		**sans objet**	1 ms	20 ms	**sans objet**
R5	2900					
R6	2 × 3910	700				
R8	790					
R7	4890	4890 (1)		1 ms	1 ms (1)	
R9	**670**			**1 ms**		
R10	**560**					
R11	**635**					
R12	**4400**					

Table 1 : Realistic break areas and breal opening time used for realistic applications

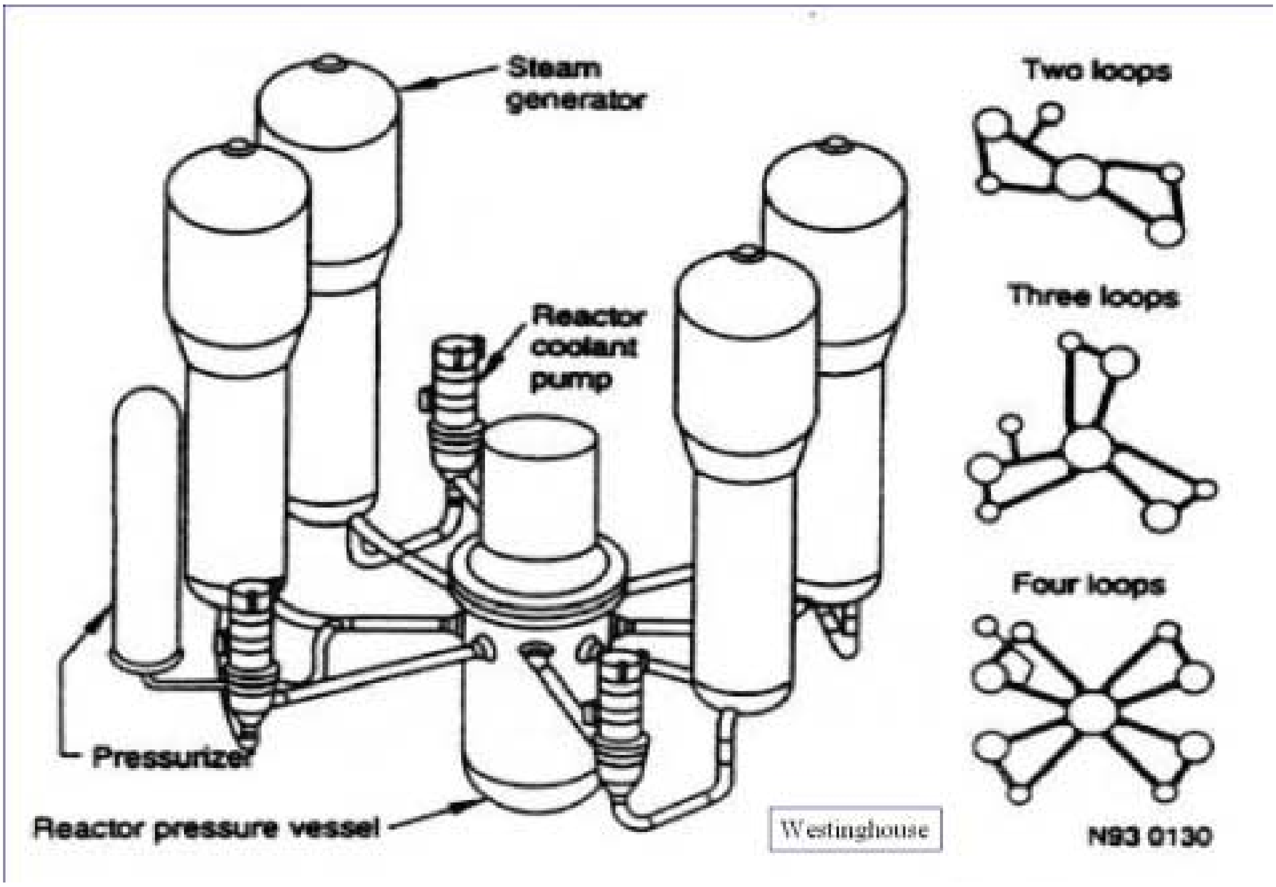
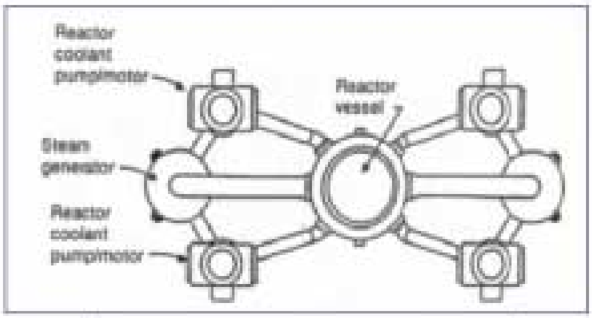
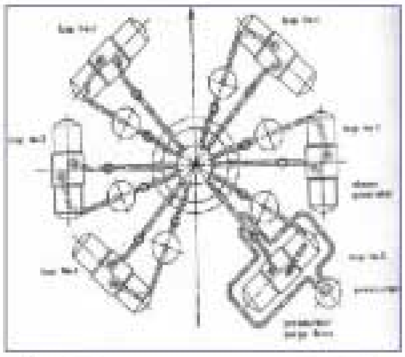
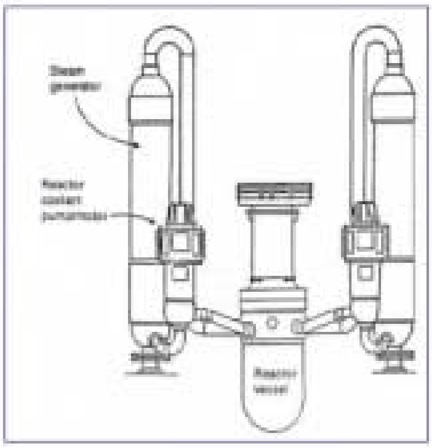


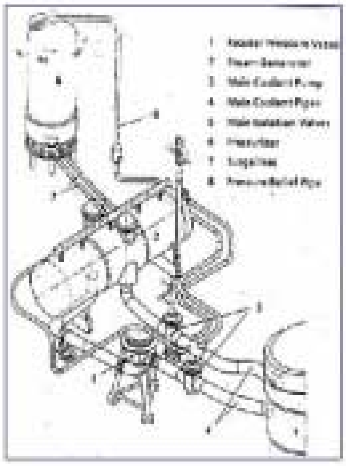
Figure 2: Westinghouse RCS piping systems
French, German, Japanese PWR's are similar





B&W





VVER

Figure 3: B&W and VVER RCS piping systems

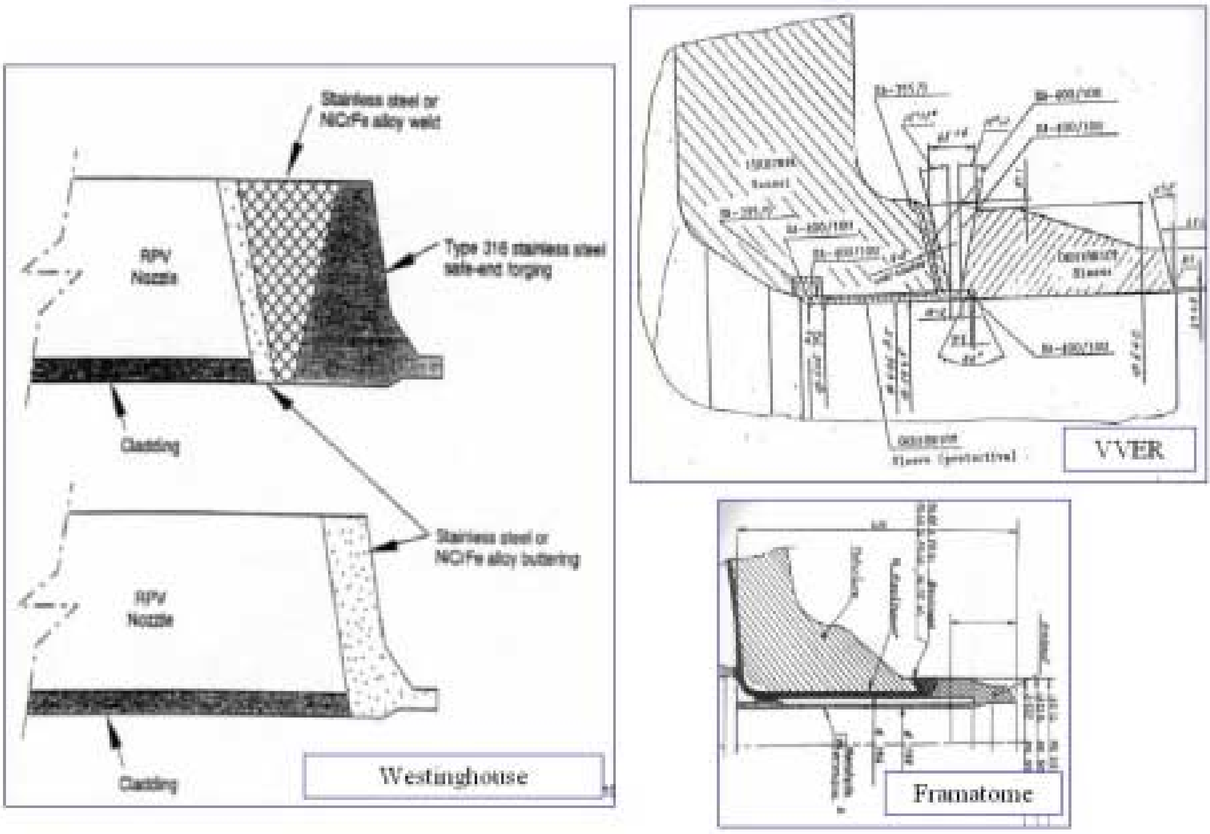


Figure 4: Examples of nozzle and DMW designs

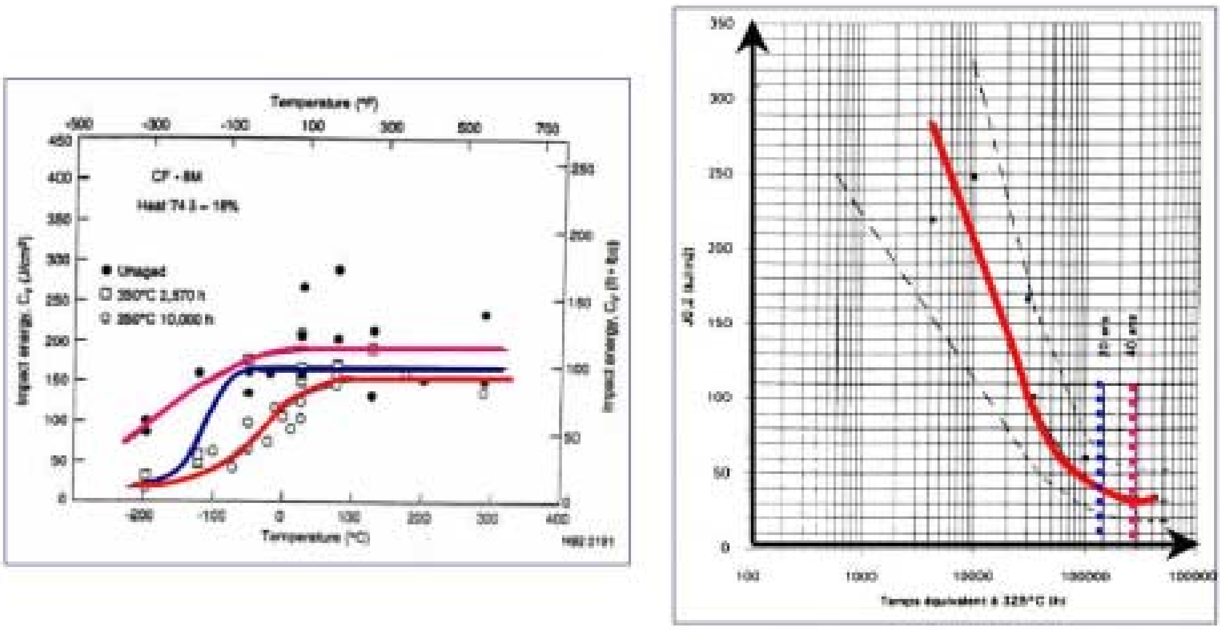


Figure 5 : Loss of toughness versus time and temperature for cast duplex stainless steel

Redefining the Large Break LOCA: Technical Basis and its implications
"US NRC Comments on Break Size Re-Definition"

Dr. S. Bajorek – Dr. N. Chokshi (US NRC, USA)
Dr. R. Tregoning (US NRC, USA)

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Technical Basis and its Implications

"USNRC Comments on Break Size Re-Definition

S. M. Bajorek
N. Chokshi
R. Tregoning
U.S. Nuclear Regulatory Commission

Zurich, Switzerland — June 23-24, 2003

REDEFINING THE LARGE BREAK LOCA

- USNRC Position: Safety Enhanced By Focusing Attention on Break Sizes of Greater Overall Risk Significance
- DEG LBLOCA Break Will Not Be Ignored – Treatment as a Severe Accident Will Put It in Proper Risk Perspective
- LBLOCA Remains a DBA: Size=?
- Changes to Plant Operation Must Demonstrate Overall Safety Benefit or Small Increases in Risk

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LBLOCA "Tentacles"

Large Break LOCA Affects Many Regulations –

Objective of NRC Work is to Remove Excess Conservatisms from 10CFR50.46

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Plant System Response

- **Intent is Not To Remove or Degrade Safety Systems.**
  - Eliminating Accumulators Not an Option
  - Safety Flow Rates Are Not to Be Degraded
  - No Reduction in Containment Capability
- **Key Point: Best-Estimate T/H Calculations Will Be Used to Determine LBLOCA Transient.**
- **Expectation: New LBLOCA Break Size Must Meet Acceptance Criteria**

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REGULATORY REQUIREMENTS

- **Basic Intent of 50.46 requires to be maintained:**
  - Postulated LOCAs for range of break sizes and locations
  - Performance evaluated using acceptable models.
  - Satisfies specific acceptance criteria
  - Ensures continued effective core cooling

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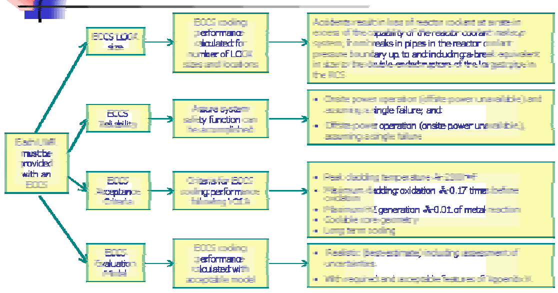
CURRENT REGULATORY REQUIREMENTS



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SESSION 3

What are possible new definitions for the LB-LOCA?
What are their implications on current and future reactors?

Chairman: Dr. A. Thadani, (RES Director, US NRC)
Moderators: Dr. J. Hyvarinen (STUK, Finland) – Dr. V. Snell, (AECL, CAN)

Slovak approach during the gradual upgrading of Bohunice V1 NPP

Mr. T. Kliment (VUJE, Slovakia)
Mr. A. Tkáč (VUJE, Slovakia)

Joint CSNI/CNRA Workshop on
"Redefining the Large Break LOCA
Technical basis and its implications",
Zurich, Switzerland, June 23-24, 2003

Abstract

The process of Bohunice V1 NPP nuclear safety and operational reliability level increase has been performed since units commissioning (1st unit in 1978, 2nd unit in 1980), continued "Small Reconstruction" (1991 - 1993) and finished "Gradual upgrading" (1994 - 2000). The attention is given to these steps realized in framework of Bohunice V1 safety increase, which direct relate to redefining design basic accident (DBA) and beyond design basic accident (BDBA). The paper describes the steps of emergency core cooling system (ECCS) and hermetic zone spray system reconstruction, which make possible to increase DBA from original value of LOCA 32 mm to LOCA 200 mm and to define LOCA 2 x 500 mm as a BDBA. In conclusion of the document, the results of V1 NPP Safety Analysis Report are listed and they confirm fulfilment of the V1 NPP Gradual upgrading goals.

1. Introduction

Jaslovske Bohunice V1 NPP consists of two units of VVER-440 type 230 series. Unit 1 was commissioned in 1978 and the second one in 1980. The original Russian design of units VVER-440/230 series originates from the end of sixties and the beginning of seventies. Rupture of primary coolant circuit with equivalent diameter of 32 mm was stated as maximal DBA, for managing of which the emergency systems were dimensioned.

Process of Bohunice V1 NPP nuclear safety and operational reliability improvements began immediately after commissioning. There were performed more than 1300 design modifications, which resulted from operational status evaluation, operational experiences and from various international recommendations and from regulations.

Further more significant step in safety improvements of Bohunice V1 NPP units was made after 1990, after realization of more national and international expert missions. Based on recommendations of individual expert missions the Czechoslovak Nuclear Regulatory Authority issued the decision No. 5/91 (dated January 11, 1991) and No. 213/92 (dated Jun 23, 1992), where 95 measures concerning further Bohunice V1 NPP safety and reliability improvement have been defined. These measures were performed during the period 1991 – 1993 years and they are known as "Bohunice V1 NPP Small Reconstruction".

Then "Bohunice V1 NPP Gradual upgrading", performed during the period 1996 – 200 years, presents most significant step in safety improvements of Bohunice V1 NPP. The main goal of

Bohunice V1 NPP Gradual upgrading was to reach internationally acceptable level of nuclear safety and operational reliability, conditions for which were defined in the Slovak NRA decision No. 1/94.

This paper is especially focused to these steps, which direct relate to redefining DBA and BDBA.

2. Original design of Bohunice V1 NPP

2.1. Original ECCS and spray system description

The original ECCS and spray system were design for a DBA represented by LOCA 32 mm. The capacities of ECCS and spray system were significantly oversized and they were able to deal with LOCA 100 mm accounting neither Loss Of Off-Site Power (LOOP) nor single failure criterion. The scheme of original ECCS and spray system is listed in Fig.1.

The original ECCS was characterized:

- Two groups each with three HPI pumps. Suction of HPI pumps is joined to Emergency Water Supply Tank (EWST).
- The first HPI pumps triplet was connected to the common collector that was connected to suction of Main Circulation Pumps (MCP) in each loop. The second HPI pumps triplet was connected to the common collector that was connected to discharge of MCPs in each loop. The HPI pump flow rate was 18 kg/s at the pressure of 13.5 MPa and 40 kg/s at the pressure of 4.5 MPa.
- Only two HPI pumps (one of each triplet) had power supply provided in the case of LOOP.
- LPI pumps were not included in the ECCS.
- ECCS was qualified neither for seismic effects nor for accident environmental conditions.
- The separation of both equipment and cabling was inadequate.
- The I&C for system actuation was not qualified.

The original spray system was design for the same DBA as the ECCS system and was characterized:

- One group with three pumps with shared suction from EWST
- The discharge of all three spray pumps were connected to the common collector and after that divided to 3 sprinkler sub-collectors
- The components of spray system were not qualified for seismic effects
- Neither electric nor physical separation was provided for spray system.

2.2 Possible reconstruction of ECCS and spray system study

At the beginning of nineties a number of studies were performed for purpose to evaluate the possibility of ECCS reconstruction. In these analyses, the new DBA was considered as LOCA 2 x 500, what represent guillotine break of primary main circulation line. In this manner DBA definition would be identical with DBA of the modern VVER-440/213 units, that are up to world-wide standard.

From the point of view of ECCS reconstruction installation of hydroaccumulators (HA) and LPI pumps were assumed. Although the results of these studies indicated the possibility of ECCS reconstruction in that manner to ensure core cooling after LOCA 2 x 500 mm, existing spatial problem in the hermetic zone didn't allow ECCS reconstruction by HA installation.

Regarding importance and expected scope of V1 NPP reconstruction works more foreign companies were addressed for submitting their tenders also. Selection of tenders finished in 1994, as winner Siemens KWU was contracted for Basic design of Bohunice V1 NPP Gradual upgrading. Therefore the Siemens approach, presented in Basic Engineering of Gradual upgrading, was applied for ECCS and spray system reconstruction.

In the phase of Basic Engineering preparation two possible cases of ECCS reconstruction were analysed. In the first case the installation of new LPI pumps was assumed to increase the efficiency of core cooling in the accident low-pressure phase. In the second case the installation of ejectors was assumed, which would supply the increase of HPI pumps flow rate in conditions of low primary pressure. The both cases of ECCS reconstruction led to ability of core cooling for larger break size (according to Slovak NRA Decision No. 1/94, see chapter 3). The results of these analyses showed that in case 1 (installation of new LPI pumps) the HA are dispensable for LOCA of the new defined break size scope. In case 2 (installation of ejectors) the installation of HA would be necessary.

3. Modifications of ECCS, spray system and hermetic zone in framework of Bohunice V1 NPP Gradual upgrading

3.1 Requirements

The requirements on the reconstruction of the ECCS, spray system and hermetic zone were specified in such way, that they meet the following criteria from Slovak NRA Decision No. 1/94:

For ECCS system:

- The system will be capable to cope with initiating event with a break of pressurizer surge line with the diameter of 209 mm with double-end coolant discharge and with a partial break of RCS line with the equivalent diameter of 200 mm at the most adverse point, meeting the following conditions:

  a) Maximum cladding temperature ≤ 1200 °C,

  b) No fuel melting occurs,

  c) Total cladding oxidation ≤ 1% from the total amount,

  d) Maximum local cladding oxidation ≤ 18% from the initial cladding thickness.

  The meeting of the mentioned criteria will be demonstrated by means of conservative methods.

- Periodic system performance surveillance during unit operation will be possible,

- The diameter of new connecting lines will be ≤ DN200,
- For BDBA with a break of RCS main circulation line with the diameter of 500 mm with double-end coolant discharge, the compliance with the conditions specified in items a - d will be demonstrated by means of realistic methods.

  For spray system and hermetic zone:

- To reconstruct the confinement, the accident localization system, and the spray system in such a way that the limiting values of over-pressure and sub-atmospheric pressure will not be exceeded, and also the authorized values of dose equivalents at the border of exclusion area (50 mSv whole body, 500 mSv thyroid) will not be exceeded, by means of conservative methods for the following initiating events:

  a) Break of pressurizer surge line DN 209 with double-end coolant discharge,

  b) Partial break of RCS line with the equivalent diameter of 200 mm,

  c) Break of main steam line or main feedwater line with double-end coolant discharge into SG box.

- Concerning Beyond DBA accident to evaluate a double ended pipe DN500 mm break, using realistic approach of calculations, the and to demonstrate intact of confinement in such a way, that safety function "cooling the core" will remain and the public dose equivalents at the boundary of the exclusion area (250 mSv whole body, 1500 mSv thyroid) are not exceeded.

As a result of new precise structural analyses using European standards it was stated that limiting values for hermetic zone structures (4.1 of NRA SR Decision 1/94) are the following:

DBA	- maximal overpressure		60 kPa
	- sub-pressure		15 kPa
	- EWST ceiling	- downstream pressure difference	33 kPa
	- EWST bottom	- overpressure above water level	31 kPa
BDBA	- maximal overpressure		120 kPa
	- sub-pressure		15 kPa
	- EWST ceiling	- downstream pressure difference	90 kPa for a few seconds
	- EWST bottom	- overpressure above water level	75 kPa

In framework of Bohunice V1 NPP Gradual upgrading the ECCS and spray system were modified to meet the following requirements:

- Application of single failure criterion,
- Minimum redundancy of 2 x 100%
- Physical and electrical separation between the redundancies and separation from operating systems,
- Seismic resistance up to 8° MSK64,
- Automatic actuation of systems without the need for operator intervention during the initial 30 minutes,
- Resistance against fire and flood,

- Equipment qualification for environmental conditions,
- Adequate inventories of fluids for 72 hours following initiating events,
- System testability in the course of power operation,
- Failure probability on demand less than 10^{-3}.

3.2 Description of the modified systems

3.2.1 ECCS system

The new ECCS system consists of 2 independent, seismically resistant redundancies. A single redundancy of ECCS consists of 2 HPI pumps and 1 LPI pump. The ECCS scheme is shown in Fig. 2

The first redundancy comprises of 2 HPI pumps connected to the cold leg of loop 2 and 5 and 1 LPI pump connected to the hot leg of loop 3. The second redundancy comprises of 2 HPI pumps connected to the cold leg of loop 2 and 5 and 1 LPI pump connected to the hot leg of loop 6.

The mean HPI pump characteristic is listed in Tab. 1.

Table 1 Mean HPI pump characteristic

Primary pressure [MPa]	0,10	9,04	9,46	9,72	10,56	12,28	12,72	12,73	12,74
Flow rate [kg/s]	30,0	30,0	29,1	28,5	26,4	19,0	9,2	4,6	0,0

The maximum of HPI pump flow rate is 30 kg/s at the pressure of 9.0 MPa. For safe operation as well as for tests of HPSI pumps, a 3-way valve is installed at their discharges that controls the recirculation flow rate depending on the flow rate of water injected to primary circuit.

The mean LPI pump characteristic is listed in Tab. 2.

Table 2 Mean LPI pump characteristic

Primary pressure [MPa]	0,1	1,61	1,74	2,22	2,69	3,02	3,14	3,24	3,35
Flow rate [kg/s]	206,7	206,7	198,3	164,3	119,0	76,5	48,7	22,6	0,0

The installation of 2 LPI pumps was the main sign of ECCS reconstruction. LPI pump start to inject coolant in to primary circuit at the pressure of 3.3 MPa and the flow rate maximum is 206 kg/s (in comparison with LPI pump of VVER-440/213 start to inject coolant in to primary circuit at the pressure of 0.82 MPa and the flow rate maximum is 100 kg/s).

3.2.2 Spray system

The new spray system consists of 2 independent, seismically resistant redundancies. Each redundancy consists of 1 pump with the flow rate of 110 kg/s at the pressure of 0.42 MPa, of spray cooler and of separate discharge line into the spray header in the confinement.

Based on a number of thermal-hydraulic and radiological consequences analyses and on necessary experimental results obtained during designing, the following design modifications of hermetic zone were implemented (Fig. 1) meeting requirements of Nuclear Regulatory Authority SR:

- Installation of confinement large relief flaps discharging into atmosphere: 4 x DN1130 per unit. All flaps (12) have the set-point of opening overpressure decreased down to $\Delta P \approx 50$ kPa.
- Installation of a blow-off pipe with the diameter of DN1200 from the EWST into the reactor hall atmosphere (ensuring that the limiting value of pressure in EWST will not be exceeded) with a relief flap ($\Delta P \geq 27$ kPa) at pipe discharge. The pipe is equipped with shut-off valve, which close the discharge line in 20^{th} minute since accident initiation (Fig. 3).
- Installation of six condensation pipes – Circulation Bubbler Condensers (CBC) with the diameter of DN510, four of which designated for the original function of draining the discharged coolant from the SG box floor into the EWST, with a protective grating around the collection sump. CBC are used as a protection of EWST inlet, bottom and walls against jet and condensation loads. Condensation in the EWST enables to reach earlier confinement pressure suppression and passively speeding up the finish of the direct steam-air mixture release to the environment (safety relief flaps earlier closing). The protective grating of the inlet openings ensures flow passage to the EWST in such a way that larger pieces of torn-off insulation material are not able to block the weakest cross-section of CBC.
- The existing venting pipes of EWST 2 x DN150 are equipped with check valves to avoid an inflow of steam-air mixture from SG box, and thus a potential direct release of the mixture from HZ into the reactor hall - environment, and also to equalise pressures in EWST and SG box during subsequent phases of accident.
- Installation of vacuum-breaker for HZ protection against very low sub-atmospheric pressure (maximal value of 15 kPa for HZ liner).
- With regard to the need of an effective use of HZ space for reducing pressure peaks mainly in case of BDBA, installation of burst membranes 4x DN800 in flanges (being able to be disassembled during outages) of the existing connection line between SG box (R002) with MCP platform (R102), with the burst pressure set at the value of $\Delta P \approx 30$ kPa.

4. T-H analyses performed within Bohunice V1 NPP SAR

The ECCS and spray reconstruction was analytical evaluated in Basic Engineering of Gradual upgrading, elaborated by Siemes. In the frame of PHARE project NS 09/01 another analyses were done in VUJE that confirmed the propriety of ECCS and spray system reconstruction from point of view of core cooling and HZ loads for DBA as well as BDBA.

In 1999 the SAR of BOHUNICE V1 NPP after Gradual upgrading was issued that reflected all modifications to the original systems since the commissioning of V1 also all modifications realized in the frame of the Gradual upgrading. Safety analyses were performed in accordance with all requirements and recommendation in the IAEA document “Guidelines for Accident Aanalysis of WWER NPPs [5]” regarding the spectrum of design basic analyses, usage of codes and methodology issues. Additionally to standard scope of SAR, the BDBA analyses were included into SAR.

4.1 Description of selected analyses

The DBA analyses (double ended rupture of one of the pressurizer surge lines 2x209 mm and rupture of the main coolant pipe with equivalent diameter 200 mm) as well as BDBA analyses (double ended rupture of main coolant pipe) were performed from the point of view of core cooling as well as of mass and energy release into the HZ. The following method and codes were used to evaluate acceptance criteria:

DBA and BDBA analyses from point of view of core cooling and/or mass and energy release into HZ

HZ loads calculation
TRACO 5

TH calculation ⇒ + ⇒ **Radiological consequences**
RELAP5/Mod3.2.2β

Fuel rods leaked
DEFOS-1A, FEMBUL-2

Methodology of DBA analysis

The 200 mm LOCA in cold leg of the loop (where two HPI pumps are connected) was defined as the bounding case for core cooling analyses. The 200 mm LOCA analysis was performed with the conservative assumptions from the point of view of core cooling. Some essentials assumptions used are listed below:

- Assumption of LOOP at the time o turbine trip, it means 10 s after scram signal.
- Assumption of single failure (SF) - non–availability of one emergency power bus bar.

  The consequence of LOOP and SF is:

  - 2 HPI pumps failing (1 connected to the loop 2, the second to the loop 5)
  - 1 LPI pump failing (connected to the loop 6)

  In addition, 1 HPI is connected directly to the broken loop (No. 2) consequently only 1 HPI and 1 LPI pumps are effective for the core cooling. SF assumption causes that only 1 spray pump is assumed in HZ loads calculation.

- Conservative boundary and initial conditions (104 % reactor power, conservative scram and ESFAS signal setting, minimum HPI and LPI pump characteristics, conservative reactivity coefficients, etc.).
- The most adverse break location (most unfavourable location through the primary, taking into account connections to ECCS).

The conservative assumptions from point of view of mass and energy release to the hermetic zone were assumed as follows:

- LOOP is not taken into account, therefore all ECCS pumps are available for the core cooling.

- SF is assumed in HZ loads calculation only. It means that only 1 spray pump is available.
- Conservative boundary and initial conditions.
- The most adverse break location.

Methodology of BDBA analysis

A realistic methodology was applied for evaluation of BDBA. Based on Slovak NRA requirements the relevant acceptance criteria for BDBA (not severe accident) are the same as for DBA, with exception of criterion for radiation consequences. The reason for inclusion of 2x500 mm LOCA into BDBA follows from the fact, that this initiating event was not included into list of original design basis accidents, but - as it is standard DBA for most nuclear units of PWR – while very low probability of occurrence there is a necessity to prove that even this accident do not challenge excessively the required safety margins.

Realistic approach used for evaluation of selected BDBA within safety analysis report (SAR) is based on application of expected values of parameters (initial and boundary conditions) and availability of systems. Against the conservative approach for DBA, the important difference is also in availability of each control system required, that both loss of off-site power and single failure in safety systems were not assumed.

In comparison with the above described DBA sequence, following differences were assumed:

- LOOP and SF are not assumed and whereupon all 4 HPI, 2 LPI and 2 spray pumps.
- Nominal boundary and initial conditions (100 % reactor power, nominal scram and ESFAS signal setting, mean HPI and LPI pump characteristics, mean reactivity coefficients, etc.).

In spite of the fact that the realistic approach is accepted for BDBA analysis within SAR, due to the safety importance and status of the analysis, some conservatism had to be included. The conservative assumptions were selected to describe the sequence from safety point of view. For the 2x500 mm LOCA initiating event the conservative assumptions encompass mainly:

- The most adverse break location (if the break is located in cold leg No.2 than 2 HPI is connected directly to the broken loop and consequently only 2 HPI and 2 LPI pumps are effective for the core cooling).
- Standard conservative approach to the hot channel (core nodalisation, axial power distribution,)
- Interface to confinement facilitating lower pressure out of the break,

4.2 Results of DBA and BDBA analyses

The purpose of the LOCA analyses was to show whether relevant acceptance criteria are met. The values with minimum margin reached for both LOCA sequences calculations are listed in Table 3.

Table 3 Results of analyses

Acceptance criteria	Maximum calculated value	
	DBA analyses	BDBA analyses
Maximum cladding temperature < 1200 °C	800 °C	1106 °C
The total local oxidation of the cladding < 17 % of the initial thickness before oxidation	0.5 %	3.97 %
The total amount of hydrogen generated from the chemical reaction of cladding with water or steam <1 % of the hypothetical amount that would be generated if all of the cladding in the core were to react	0.046 %	0.14 %
The radially averaged fuel pellet enthalpy < 963 J/g at any axial location of a rod	less than initial value of 499.5 J/g	less than initial value of 499.5 J/g
Maximum fuel temperature < 2639 °C for fresh fuel or 2605 °C for burned fuel	less than initial value of 1958 °C	less than initial value of 1958 °C
Calculated changes in geometry are such that the core remains amenable to cooling: - number of leaked fuel rods equals	0.423 %	6.037 %
Effective dose at the boundary of the protective zone (3 km from NPP) < 50 mSv for whole body / 500 mSv for the thyroid – DBA 250 mSv for whole body / 1500 mSv for the thyroid – BDBA	2.91/4.80 mSv	21.94/28.54 mSv
Maximum overpressure in HZ < 60.0 kPa for DBA < 120.0 kPa for BDBA	53.0 kPa	118.1 kPa
Maximum temperature in HZ < 120.0 °C for DBA < 130.0 °C for BDBA	111.1 °C	119.0 °C
Maximum sub-pressure in HZ < 15.0 kPa	6.3 kPa	6.3 kPa

5. Conclusion

By the implementation of the ECCS, spray system and hermetic zone reconstruction, together with reconstruction of additional systems in the frame of Bohunice V1 NPP Gradual Upgrading, the Slovak NRA requirements on significant upgrading of nuclear safety at the plant with VVER 440/V230 reactors was met.

Original DBA LOCA 32 mm was redefined after Gradual reconstruction to LOCA 200 mm, proved by means of conservative methods applied. After Gradual upgrading, the safety systems are able to cope BDBA LOCA 2x 500 mm, proved by means of realistic methods applied.

Carrying out the Gradual upgrading the internationally acceptable nuclear safety and operational reliability of Bohunice V1 NPP was achieved and confirmed. The technical assumption for reliable, safe economical and ecologic operation of Bohunice V1 units was established.

References

[1] Specialised seminar “Results of Bohunice V1 NPP Gradual Upgrading“, March 15, 2001, Piešťany, Slovak Republic.

[2] Mišák J.: Analyses of emergency core cooling system reconstruction for VVER-440/230 units in connection with DBA increasing, VUJE Trnava report No. 19/92, May 1992.

[3] Tkáč A.: Preliminary safety report of Accident Localisation System for Bohunice NPP V1 Gradual upgrading (in Slovak), VUJE Trnava report No. 104/98, June 1998.

[4] A-01: Safety analyses report of NPP-V1 after Gradual upgrading, chapter 15 (in Slovak), revision 2, April 2001.

[5] IAEA-EBO-WWER-01: Guidelines for Accident Analysis of WWER Nuclear Power Plants, December 1995.

[6] Report of the “Review of Results of the Gradual Upgrading at Bohunice WWER-440/230 Units 1 and 2, TC Project RER/9/046, IAEA, February 2001.

Figure 1 Schema of original ECCS a spray system of Bohunice V1 NPP

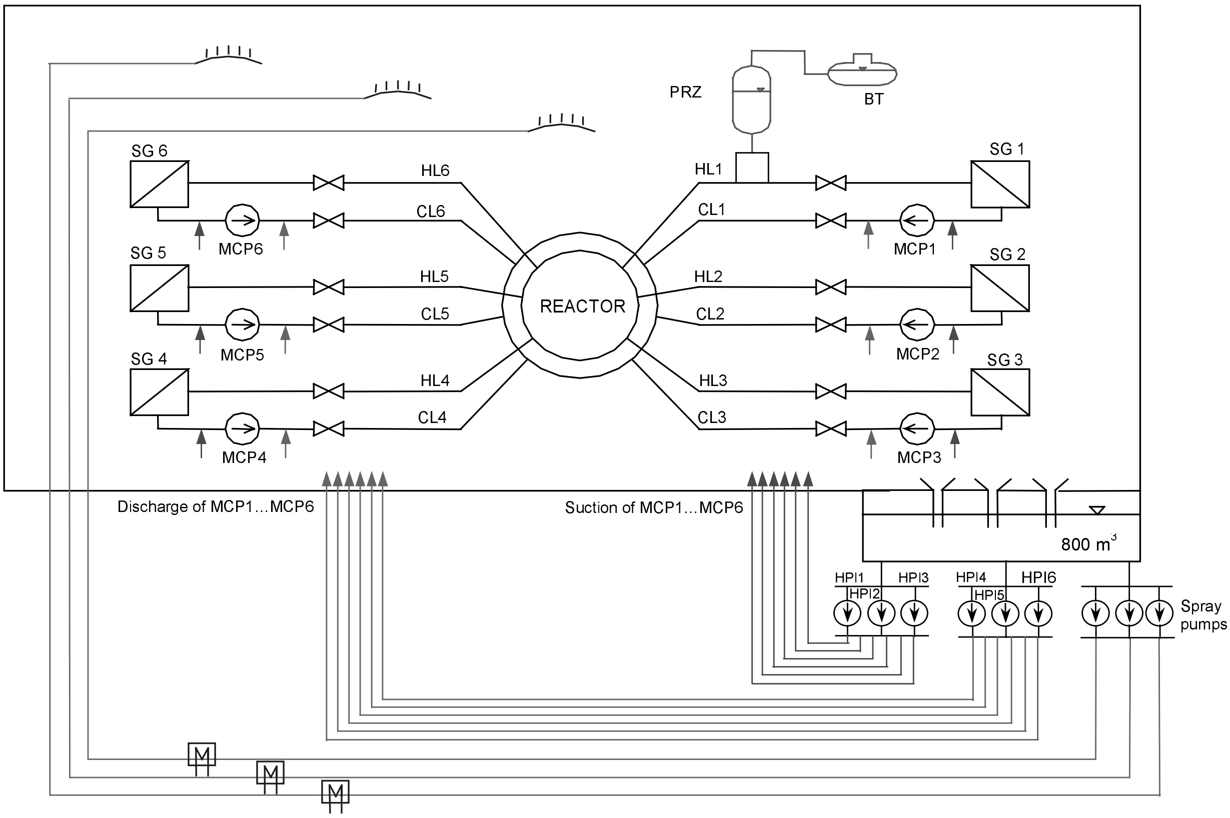


Figure 2 Schema of ECCS a spray system after Bohunice V1 NPP Gradual upgrading

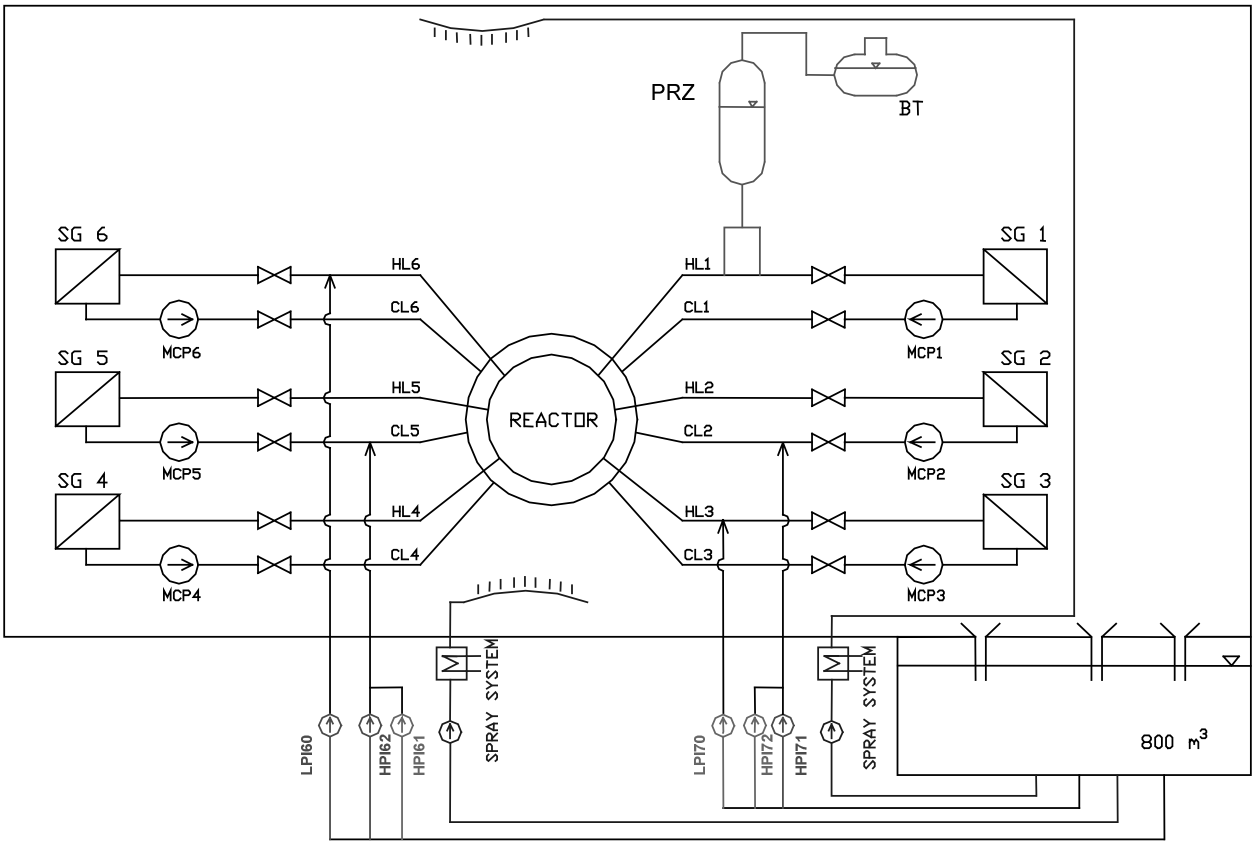
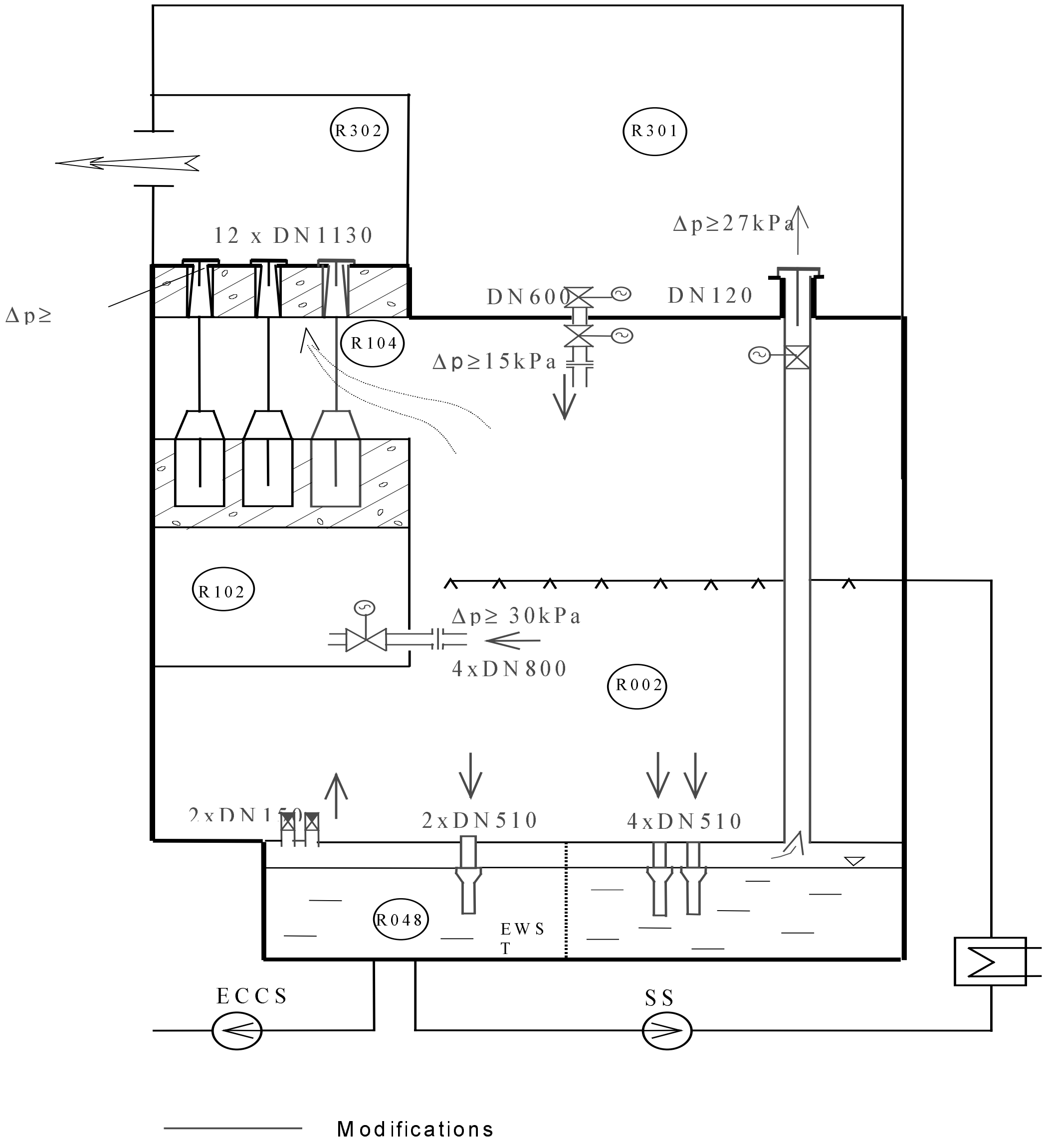


Figure 3 Accident Localisation System

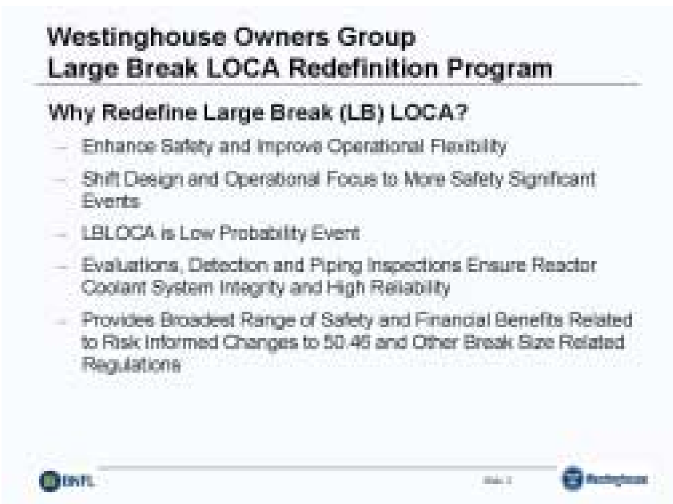


Acccident Localisation System (ALS)

Westinghouse Owners Group Large Break LOCA Redefinition Program

Mr. R. Bastien (Westinghouse, BE)

Joint CSNI/CNRA Workshop on
"Redefining the Large Break LOCA:
Technical basis and its implications",
Zurich, Switzerland, June 23-24, 2003

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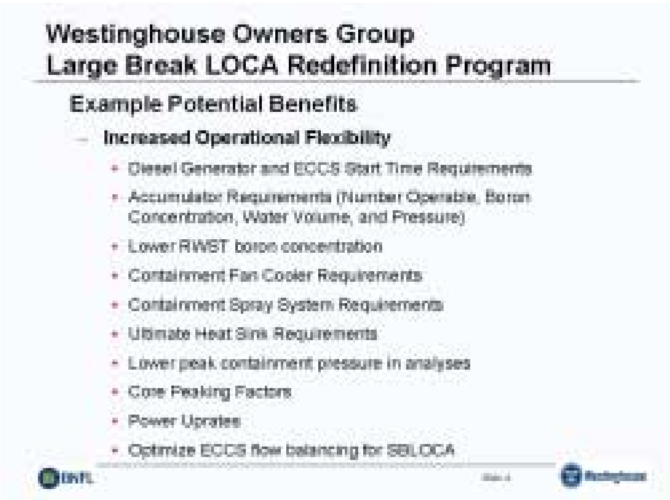
Large Break LOCA Redefinition Program
Safety Benefits
– Resources Focused on Activities of Greater Risk Significance
– Realistically Based Equipment Testing Requirements
– Consistency within the Regulations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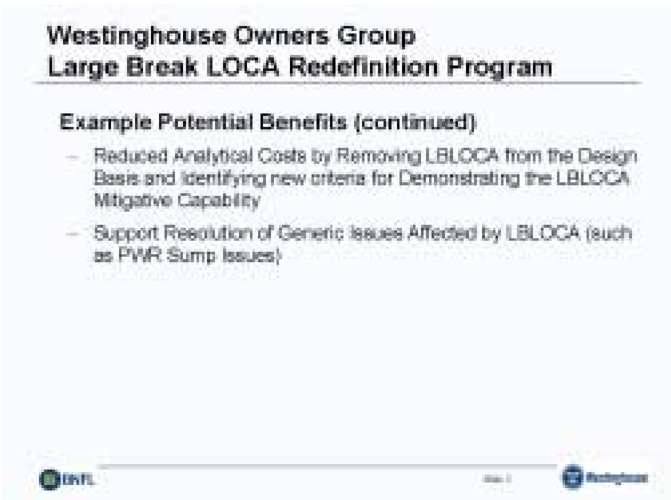
Large Break LOCA Redefinition Program
Example Potential Benefits
– Increased Operational Flexibility
• Diesel Generator and ECCS Start Time Requirements
• Accumulator Requirements (Number Operable, Boron Concentration, Water Volume, and Pressure)
• Lower RWST boron concentration
• Containment Fan Cooler Requirements
• Containment Spray System Requirements
• Ultimate Heat Sink Requirements
• Lower peak containment pressure in analyses
• Core Peaking Factors
• Power Uprates
• Optimize ECCS flow balancing for SBLOCA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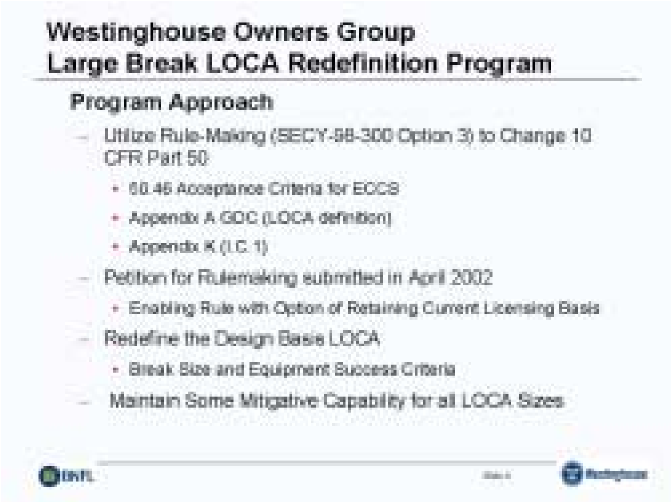
Large Break LOCA Redefinition Program
Example Potential Benefits (continued)
– Reduced Analytical Costs by Removing LBLOCA from the Design Basis and Identifying new criteria for Demonstrating the LBLOCA Mitigative Capability
– Support Resolution of Generic Issues Affected by LBLOCA (such as PWR Sump Issues)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Large Break LOCA Redefinition Program
Program Approach
– Utilize Rule-Making (SECY-98-300 Option 3) to Change 10 CFR Part 50
• 50.46 Acceptance Criteria for ECCS
• Appendix A GDC (LOCA definition)
• Appendix K (I.C.1)
– Petition for Rulemaking submitted in April 2002
• Enabling Rule with Option of Retaining Current Licensing Basis
– Redefine the Design Basis LOCA
• Break Size and Equipment Success Criteria
– Maintain Some Mitigative Capability for all LOCA Sizes


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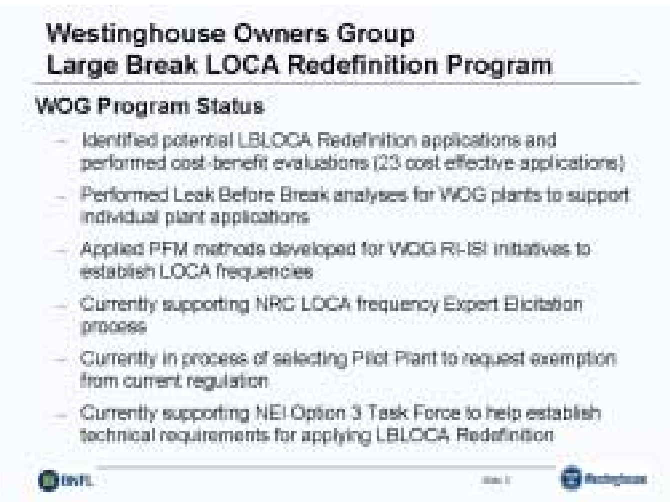
Large Break LOCA Redefinition Program
Program Approach (continued)
– Technical Justification
• Use Risk-Informed Technology to Show Low Risk of LBLOCA
– Utilize the Framework Contained in Regulatory Guide 1.174
• Use Leak Before Break Analysis to Support Choice of Maximum Size
• Define New Maximum Break Size & Reanalyze, as Necessary, to Obtain Benefits
– Submit Topical Report documenting results for pilot plant
– Pilot Plant will submit exemption request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Large Break LOCA Redefinition Program
WOG Program Status
– Identified potential LBLOCA Redefinition applications and performed cost-benefit evaluations (23 cost effective applications)
– Performed Leak Before Break analyses for WOG plants to support individual plant applications
– Applied PFM methods developed for WOG RI-ISI initiatives to establish LOCA frequencies
– Currently supporting NRC LOCA frequency Expert Elicitation process
– Currently in process of selecting Pilot Plant to request exemption from current regulation
– Currently supporting NEI Option 3 Task Force to help establish technical requirements for applying LBLOCA Redefinition

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Large Break LOCA Redefinition Program

Interactions with NRC Staff and ACRS

- The ACRS noted "The staff should continue to develop the technical bases and requirements for redefining the LBLOCA."
- In December 2002 meeting NRC Staff presented their two year plan.
- NRC Staff is undertaking an effort to determine appropriate LOCA frequencies for going forward.
  - Research has developed an "interim" set of "best estimate" LOCA frequencies, based in part on information provided by WOG and BWROG
  - NRC has formed an Expert Elicitation Panel made up of Industry & Staff experts
  - NRC plans to complete the Expert Elicitation process and have new estimates of LOCA frequencies by July of 2004.

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Large Break LOCA Redefinition Program

PRA Quality and Completeness

- This is currently an issue for most Risk-Informed Applications
- We believe that for LBLOCA Redefinition, PRA Quality and Completeness should be consistent with DG-1122
  - Demonstrate with appropriate documentation that those parts of a PRA used in a regulatory application are of sufficient quality to support the analysis, and
  - Determine the technical adequacy of the PRA results via, for example, consensus PRA standards and peer review of PRAs
- Together with the NEI Option 3 Task Force, WOG will support the NRC in the resolution of these issues

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Large Break LOCA Redefinition Program

Summary

- A Rule Change to facilitate Large Break LOCA Redefinition has substantial benefits both in terms of safety and Operational flexibility
- The expert elicitation process to update LOCA frequencies should be completed expeditiously
- Any other regulatory issues that would impede a pilot application of LBLOCA Redefinition should be identified and resolved expeditiously
- The enabling rule change should be implemented

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SESSION 4:
General Discussion – Conclusions and recommendations

Chairman: Dr. J. Laaksonen, (Director General STUK, Finland,)

F. List of Participants

Joint CSNI/CNRA Workshop on
"Redefining the Large Break LOCA:
Technical basis and its implications"
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