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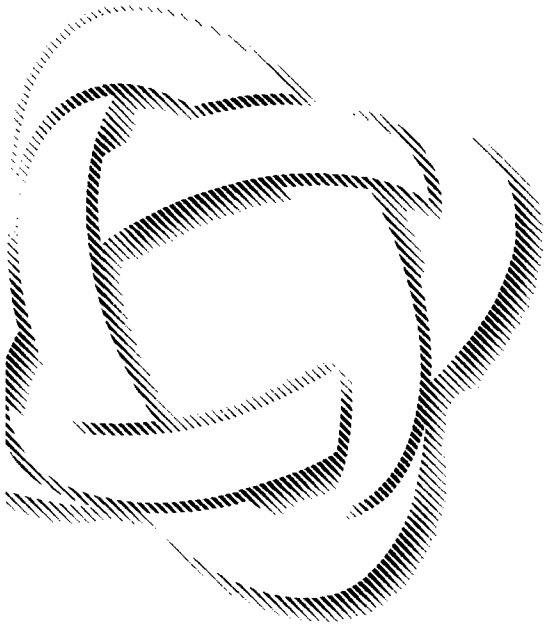
Redefining the Large Break LOCA: Technical Basis and its Implications



Workshop Proceedings
23-24 June 2003
Zurich, Switzerland

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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

PROCEEDINGS OF THE 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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- 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;
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NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of 28 OECD Member countries: Australia, Austria, Belgium, Canada, 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.

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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.

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Dr. J. Dusek	SUJB	CZ
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Mr. T. Kliment	VUJE	SK
Mr. M. Zimmermann	PSI	CH
Dr. N. Chokshi	USNRC	USA
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)
Dr. J. Laaksonen,	Director General (STUK, Finland)
Session 1:	Mr. L. Skanberg (SKI Sweden)
	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

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D. Summaries

Workshop Summary

Chairman:
Dr. A. Thadani (RES Director, USNRC)
Dr. J. Laaksonen (Director General STUK, Finland)

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

Workshop had three sessions for answering three different questions. The answers were discussed and summarized in the concluding session. Summaries are as follows:

1. What drives the need to redefine the LB-LOCA?

There was a consensus on well founded drivers for the redefinition and on observations that, if the change is made right, it can enhance the safety of nuclear power plants and also reduce the costs of power production. The following points were noted.

- A direct positive contribution to safety comes from improved reliability of response to a number of transients and accidents that may occur with much higher probability than a LB LOCA.

- If the requirements on very fast response of safety equipment can be relaxed, the equipment can be designed simpler and thus more reliable. Of special importance is the reliable function of emergency diesel generators and containment isolation valves that are needed in many events having significant risk contribution.

- Relieved requirements on equipment tests reduce the risk of failure in test conditions and the rate of aging.

- The ECCS flows can be better optimized for such leaks that have shown to make the highest risk contribution.

- Other identified safety benefits were the reduced fatigue loads to main structures, if integral ECCS tests with high flow rate can be avoided, and a possibility to optimize core reload if a higher limit can be used for local peaking factor; this could reduce the embrittlement caused by neutron fluence on reactor vessel wall and core support structures.

- Consuming less time and resources to demonstrate compliance with LB LOCA requirements permits more attention to all other risks. Change of culture towards more risk-informed approach can be expected to motivate the licensees and industry to take an integral look to risk, and to increase the efforts to find out and eliminate plant specific safety hazards.

- It was further concluded that even if the LB LOCA would remain as the design basis, there is a need to change the obsolete rules that require reactor reloads to be analyzed with certain unphysical assumptions. The currently required analysis consumes many

resources, but the possibility to influence the results with rather arbitrary choice of parameters hardly results in information that is relevant for safety.

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

None of the participants suggested that the probability of LB LOCA could be so high that it represents a significant contribution to the overall risk. There was a general confidence that the probability of a fast occurring large leak from the main reactor coolant circuit can be made insignificant with the right corrective measures. This is true at least in the new plants where lessons learned during the last 30 years have been implemented.

The following factors that can contribute to the break preclusion were noted:

- Improved design, improved material properties, and improved fabrication technology that together permit omission of welds from the nozzles and minimization of the total number of welds as well as optimized stress distribution.

- Complying with the Leak before Break criteria, or using another concept with similar principles.

- Improved reliability of in-service inspections by means of systematic qualification of the procedures, equipment, and inspectors; and

- Lessons learned from material research and operating experience.

  - In spite of the generally positive answer to the question on adequate technical basis, there was a consensus that the potential unknown threats to the primary circuit integrity cannot be completely neglected. It was noted that the several serious accidents or accident precursors that have taken place have surprised the entire nuclear community. Potential initiators of a large break LOCA could be various dynamic phenomena, including seismic events, if not properly accounted for in the design. Therefore, there was a consensus on a strong incentive to take all reasonable measures to retain the primary circuit integrity with high reliability, no matter what the possible redefinition of LB LOCA size will be. At the older plants it is not always evident that the preclusion of a large break is adequately small, and it is important to make detailed investigations of all relevant factors.

  - In addressing the question of break probability, it was noted that a significant amount of work has been done to improve the quality and accuracy of thermal-hydraulic codes, and extensive studies have provided a good understanding of the relative importance of the different risk contributors. The results of this research permit improved means to analyze the consequences of a LB LOCA and put the LB LOCA risk in a right perspective, if compared with the situation 30 years ago.

  - NRC presentations and discussions emphasized that the risk-informed approach, in addition to the results of a PRA, includes consideration of other safety principles, such as defense-in-depth and safety margins. Considerable discussion took place regarding the PRA quality, scope, and the ability to model all of the phenomena necessary to evaluate implications of the large breaks. In the current PRA studies the assumptions on LB LOCA frequency, as an initiating event, vary with a wide spectrum, and there is no clear technical reason for using different numbers. It was recognized that a good quality PRA that accounts for uncertainties is an important tool for assessing implications of LB LOCA redefinition. Clearly, the establishment of spectrum of break sizes (pipe and non-pipe breaks due to internal and external events) and frequencies is the critical step in this process.

- Due to the small remaining residual risk, it was considered important to preserve the capability to cope with the consequences of a LB LOCA, with the main aim to avoid consequent severe core damage. The analysis to demonstrate this capability could be realistic using qualified tools, consideration of uncertainties, and the possibly of less conservative assumptions of failures in safety equipment than what is the current practice.

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

- A generally supported principle was that in consideration of a possible redefinition of the LB LOCA it is important not to think only about the contribution of the LB LOCA to the estimated core damage frequency. Instead it is necessary to think about the connection between the redefinition and the overall risk. Balance of safety has to be evaluated in order to ensure that safety is not decreased.

- A starting point for any redefinition of the LB LOCA must be that something else comes in the place of LB LOCA as a design basis for systems and structures.

- The definition of LB LOCA has a connection with many design and operational aspects, and if requirements are changed in those aspects, the total impact to the safety margins has to be analyzed in a comprehensive manner. The following issues were mentioned among those that (today) are influenced by the use of LB LOCA as a design basis:

  - ECCS as it protects from a spectrum of LOCA's
  - Containment design requirements
  - Reactor core power and power distribution
  - Nuclear fuel design features and high burnup limits
  - Reactor internals (with special emphasis on mechanical parts of the reactor shutdown system)
  - Shielding from dynamic loads
  - Environmental qualification of equipment inside the containment
  - Breaks tolerated in the AC supply
  - Requirements on operability and testing in Technical Specifications
  - Emergency operating procedures

- A common view of the regulators attending the workshop was that current functional capabilities for various plant systems should not be changed even though the redefinition of LB LOCA would influence their original design basis. Instead, any such change should be considered separately after the modification of the rules on LB LOCA analysis, and the respective acceptance criteria. The views presented by some industry representatives were clearly different. They suggested consequent modification of many system specific design requirements.

- The planned rule making process in the U.S., for changing 10 CFR 50.46, and the Appendices A and K to 10 CFR 50, was not seriously questioned. This positive opinion was based on taking into account the regulatory views that it should not result in any significant reduction in the current safety margins of the overall plant design or increased risk. The workshop emphasized the importance of explicit safety arguments to support the proposed rule change, rather than a general reference to resource re-direction. Also a thorough analysis of the implications to any other design requirements, to operation, testing, and maintenance practices was seen as a necessary prerequisite of the rule change by many participants.

SESSION 1 Summary
"What drives the need to redefine the LB-LOCA?"

Moderators
Mr. L. Skanberg (SKI, Sweden)
Mr. J. Sugimoto (JAERI, Japan)

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

Three papers were presented. In the first paper, Mr Bajorek (USNRC) discussed the redefinition of the LB-LOCA in the context of risk informing their regulations. In the presentation and following discussion, it was clearly stated that the NRC:

- will not give up the defence in depth
- will not reduce the containment capacity
- will not take away the ECCS

Mr. Bajorek emphasised that redefining LB-LOCA is being considered by US NRC from a risk perspective to improve the safety focus and that regulators should better focus on safety and risk contributors and thereby formulate the regulations to better use available resources.

He added that the present LOCA definition has not only a great impact on the plant design but also on operating limits according to the Technical Specifications as well as on testing conditions.

More considerations should also be given to small and medium size LOCA: since these could have relatively large risk impacts in terms of CDF according to performed analysis.

In the second paper, Mr Pietrangelo (NEI, USA) presented a paper on the need to redefine the large break LOCA from the industrial viewpoints. He emphasised that the strong leadership commitments by both NRC and industry are necessary, and that redefining LB-LOCA is central to risk-informing technical requirement. He stressed the need to improve safety focus by reducing unnecessary and unrealistic requirements, which could result from current LOCA-definition. NEI also stressed the need for a culture change and to move from a deterministic approach to a more risk informed and performance based approach.

Prof Alonso raised the question on the possible paradoxes between redefining LB-LOCA and maintaining DBE, between redefining LB-LOCA and many parameters to be fixed, and between safety issues to focus on and analysis including PPM.

Dr. Thadani commented that LB-LOCA is defined as rupture greater than 6 inches break and the NRC is moving in the right direction. He also commented that the reversibility concept has not changed and that the current safety margins, and capability of ECCS will be maintained.

Chairman Diaz commented that there is no contradiction, since DBE will be maintained, LB-LOCA will not be a central part, and that redefining LB-LOCA will enhance the safety.

In the third paper, the European reactor vendor gave its view on LB-LOCA definition for new reactors (EPR). The proposed concept take into account the LB-LOCA (of the main coolant line) for designing the ECCS and the containment but not for mechanical design of the main coolant lines itself. An important prerequisite for LBB and break exclusion in the EPR is also reliable monitoring and inspection.

Anticipated implications of a redefined LB-LOCA discussed during the session were:

- extent of analysis and plant changes needed for power uprates can be reduced;
- peaking factors levels for the fuel can be changed;
- less demanding requirements for some components testing;
- emergency diesel generators start times can be changed.

SESSION 2 Summary

Does Adequate Technical Basis Exist to Support a Redefinition of the LB-LOCA?

Moderators
Dr. N. Chokshi (US NRC, USA)
Prof. A. Alonso (Univ. Madrid, Spain)

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

The session had a well coordinated set of four complementary presentations aimed at measuring the technical basis to support a redefinition of the LB-LOCA, through the potential development of a spectrum of break sizes, their expected frequencies and the corresponding consequences. With that aim, the four papers presented, in a sequential manner: the critical issues and technical approaches to the subject from the risk requirements point of view, the known and potential aging mechanisms in primary pipes, the technical and administrative developments to prevent pressure boundary fractures through in service inspections and the new developments to detect such fractures through advanced leak detection technologies.

The NRC presentation identified issues related to materials engineering, risk considerations, and plant response analysis, and discussed NRC's ongoing technical approaches to address these issues and develop a technical basis for the risk-informed revision of the rule. Among the key issues, the following were highlighted in presentations or ensuing discussions.

- At present, there is no actual experience on large brakes, so that the analysis of precursor events and other data have to be used for estimation.
- The double-ended guillotine break (DEGB) has been considered as a surrogate for many considerations not explicitly addressed in designs. If the DEGB is eliminated, these considerations will have to be explicitly accounted for in developing the redefinition.
- LOCAs as a result of non-pipe failures need to be included in the development of the spectrum of break sizes and frequencies.
- The experience data does not always account for aging and environmental effects for known and future mechanisms. However, these effects need to be accounted for in the estimation through other approaches.
- An integrated approach involving operating experience, probabilistic fracture mechanics, and expert elicitation is necessary for establishment of a spectrum of LOCA break sizes and frequencies to account for all potential sources of LOCAs.
- Continuous monitoring and assessment are necessary to evaluate trends and mechanisms which have potential effects on LOCA frequencies.
- The PRA needs to be of sufficient scope and quality to establish risk metric and to evaluate all implications of redefinition and plant changes.

- The PRA improvements may be needed to model effects of a LBLOCA to evaluate impacts of potential changes because of the redefinition.

- Plant response analyses and success criteria development will be needed to establish technical basis and to evaluate impacts of changes.

- A "living" PRA will have to be maintained to track effects of operational changes, changes in equipment availability/reliability, and impact of new mechanisms and other factors that could change LBLOCA frequency.

In addition to the above specific issues, the overarching issues of uncertainties, development of acceptance criteria (including the risk metric), and defense-in-depth were extensively discussed.

The EDF presentation was very insightful as it reflected viewpoints and experiences of the largest utility. The presentation summarized the experience data for known degradation mechanisms. One of the conclusions of the presentation was that the RCS piping system generally has high quality standard for design, fabrication, and operational rules, and the field experience confirms that as very limited degradations are encountered in these systems. The different design codes used in different countries all assure a high degree of quality and reliability; nevertheless, not all the existing and potential degradation mechanisms are covered by the design codes; moreover, those codes consider mainly initial material properties and not necessarily the changes in these properties as plants age; all of which calls for the introduction of preventive mechanisms such as in service inspection and leak detection, which were considered in detail in the subsequent presentations. The EDF presentation also covered leak-before-break concept which is one of the integral elements (along with the in-service inspections and leak detection) in redefining the LBLOCA.

The SKI presentation discussed the experience with the degradation specific in-service inspection programs. The experience show that, these programs, coupled with the requirements for demonstration of the NDE performance, have been effective in detecting serious instances of degradations. The presentation also covered a number of efforts that are currently underway to develop risk-informed ISI program. It should be recognized that any risk informed approach to in-service inspection programs must be based on an extensive experience, which at present will probably not cover every possible degradation mechanism. Therefore, such inspection approach has to be balanced with other optimization efforts to cover any unexpected degradation which may occur in the future.

The Framatome presentation described the standardize criteria, methods, and procedures for assuring the design and operational adequacy of reactor coolant pressure boundary leak detection systems used in plants. The presentation also covered a new system based on the measurements of local humidity. The sensitivity of this system appears to far surpass that required by the standard.

In the ensuing discussions after the presentations, some of the broader points dealt with the evaluation of implications of changes. The overall impact on the risk as a function of break size, timing and shape will be also necessary to redefining the LB-LOCA.. It was also pointed out that the break exclusion/preclusion concepts are easily applied to the relatively newer plants as the basis (e.g., leak-before-break concepts) can easily be established and demonstrated. For older plants, it may be much more difficult to establish the basis.

Overall, there was a general consensus that a wealth of knowledge has already been obtained to define sound technical basis in support of a LB-LOCA redefinition. Such knowledge is coming from theoretical exercises such as risk insights, expert elicitation processes, probabilistic fracture mechanics analysis and probabilistic in-service inspection

programs, as well as from the analysis of operating experiences, the construction of data banks on piping behaviour and the development of new and improved non-destructive testing examinations and leak detection of systems. All this knowledge can already be put into new regulations for present and future plants.

SESSION 3 Summary
New definitions for LB-LOCA and implications

Moderators
Dr. J Hyvarinen (STUK, Finland)
Dr. V. Snell, (AECL, Canada)

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

Two papers were presented which highlighted different approaches to incorporating a change in the LB-LOCA definition into a plant.

The first paper, "Slovak approach during the gradual upgrading of Bohunice V1", by Mr. Kliment, described a programme of backfits to an operating plant, to increase the DBA LB-LOCA from 32 mm equivalent to 200 mm. This derives from largest connecting line and has been analysed with typical DBA methods and assumptions with very satisfactory results. In addition, a best–estimate assessment showed that the plant ECCS and confinement were capable of handling a beyond-design-basis LB LOCA of up to 2 X 500 mm, the largest cooling pipe size.

All necessary analyses and extensive plant hardware modifications to implement the new Safety Case have been completed a few years ago.

The second paper, "Westinghouse Owner's Group Large Break Loca Redefinition Program", presented by Mr. Bastien, described the simplifications of design and operation that would be achieved if the DBA LB-LOCA were (significantly) reduced in size. They were extensive, including relaxed requirements on testing and on system performance. A pilot plant would be selected in the US and a formal request for a rule change was being prepared for submition to the US NRC.

The discussions and observations covered the following points:

1. On the backfits to Bohunice: the original design was already robust in terms of pipe failure prevention. Increasing ECC capability could have, however, also adverse safety effects: it may increase reactor pressure vessel PTS loads (These were carefully and thoroughly analysed when the new DBA was established). Likewise a much more concrete picture of neutral and adverse effects of "redefining LB-LOCA" need to be elaborated to avoid inducing safety detriments while reaping the well known benefits of such redefinition.
2. Hanging the size of the LB-LOCA did not automatically reduce requirements on sump clogging, which is relatively independent of break size. Changes to the design basis could be made only with a thorough understanding of such dependent effects
3. It could be challenging to establish confidently the frequency of a large pipe break in an existing plant, due to the difficulty of inspection in some areas and developments in fracture mechanics
4. Other sources of LOCA should be looked at, such as open manway covers.
5. There must be a balance between the safety benefits of redefining LB LOCA with the decrease in margins from relaxing the associated criteria (i.e. neutral & adverse effects clearly identified, addressed and settled).

E. Papers

Introductory session
Welcome

Dr. U. Schmocker, Chairman (HSK Director)

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

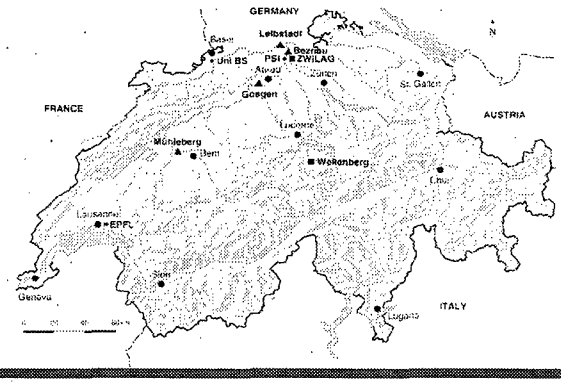
Joint CSNI/CNRA Workshop on:

Redefining the Large Break LOCA:
Technical basis and its implications

Welcome address by
U. Schmocker
Swiss Federal Nuclear Safety Inspectorate (HSK)

HSK Redefining the LLOCA – CSNI/CNRA-Workshop, Zurich June 23/24, '03 Folie 1

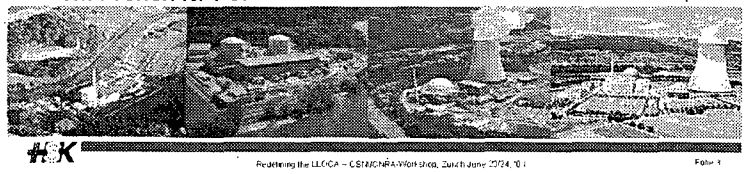
Nuclear Installations in Switzerland



HSK Redefining the LLOCA – CSNI/CNRA-Workshop, Zurich June 23/24, '03 Folie 2

The Swiss Energy Situation

Electricity generated:	60%	Hydroelectric power
	38%	Nuclear power (winter 45%)
	2%	Others
Demand for Electricity:	Basic Load:	5000 MWe, 3200 MWe from the 5 NPPs
	Peak Load:	Storage power station, imports
Procurement rights from French NPPs:	2400 MWe	



Public vote on "MoratoriumPlus" and "Power without Atom"

MoratoriumPlus

Moratorium for another 10 years:

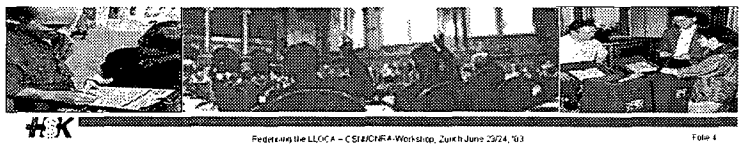
- no approval for new nuclear power stations, no capacity increase of existing stations and research reactors
- after 40 years of operation, possibility to extend operation for 10 years (optional referendum)

Result: voted down with 58% of all votes

Power without Atom

- 30 year limit on Gösgen and Leibstadt NPPs, shutdown of the older plants (Beznau I and II, Mühleberg) within 2 years; instant ban on reprocessing of spent fuel

Result: voted down with 66% of all votes



New Nuclear Energy Act

The new law, to replace the one of 1959, provides for:

- **Long-term option for new NPP technologies**
- **No limits for the operational lifetime of NPP**
- **Competences for nuclear facilities lie with the Federal Government only**
- **Regulations on decommissioning and radioactive waste management**
- **Financing of decommissioning and waste management**
- **10 yrs moratorium on the reprocessing of spent fuel; existing contracts to be fulfilled**
- **General licence is subject to an optional popular referendum**
- **Possibility to appeal against nuclear licences (exeption: general licence)**

New act has been published on May 27, 2003 – put into force 1.1.05 earliest

The Swiss Federal Nuclear Safety Inspectorate (HSK)

HSK organisation and their duties

- ~ 90 employees (physicists, engineers, geologists, biologists, computer experts, psycholigists, etc.)
- supervise the nuclear installation on aspects of nuclear safety and radiological protection;
- supervise the emergency preparedness onsite and offsite
- supervise the activities related to the transport of nuclear materials
- supervise the activities related to handling, storage and final dispoal of nuclear waste

Integrated oversight approach

Increase the Efficiency

- Process-oriented, integral management system, which Complies with ISO 9001: 2000

Increase the Effectiveness

- Oversight has to be balanced: it takes into account deterministic and probabilistic factors, but also operational aspects, maintenance and organisation issues to define the review depth, inspections aspects and frequency, and for decision making.

Traceability

- Apply a consistent and comprehensive oversight plan and set of regulations. Follow standardised decision making procedure. Decisions are based on indicators and on the associated decision criteria.

HSK Redefining the LLOCA – CSNI/OECD-Workshop, Zurich June 23/24, '03 Folie 6

Redefining Large LOCA?

Performance:

- No LLOCA worldwide
- Risk contribution small (some %)
- ISI is the tool to detect early signs of cracks – good performance shown over the years
- Pipes with cracks have been replaced (e.g. NPP Mühleberg)
- Likelihood for a complete double ended break is extremely small – maybe close to zero!

Oberservations:

- LLOCA analysis complicated and resource intensive (T/H programmes!)
- The structural loads are difficult to analyze correctly and completely
- Are the investigations balanced in the light of the overall contribution to risk?

Conclusions:

- A complete double-ended break should be ruled out as a design basis accidents, because:
  - The pipe material quality is high
  - The ISI is qualified and reliable
  - The operational experience clearly shows that a LLOCA is a hypothetical accident
- Redefining Large LOCA is a MUST!
- Worldwide harmonisation is also a MUST!

HSK Redefining the LLOCA – CSNI/OECD-Workshop, Zurich June 23/24, '03 Folie 7

Introductory Speech

Mr. B. Kaufer (OECD/NEA)

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

It is my pleasure to open this workshop and to welcome each of you on behalf of the NEA.

I would like to thank Dr. Schmocker and the staff of HSK and PSI for hosting this meeting and for providing the local organization. A successful workshop requires the co-coordinated effort of a number of people. The organizing Committee and its secretary Eric Mathet have put together a good programme and I'd like to thank them for that.

One measure of success for the performance of an organizing committee is to attract the right attendance. Therefore I am delighted to have Chairman Diaz as the keynote speaker. It is no secret that he has a strong personal interest in the workshop topic and I am sure he will not keep secret what should be done about large break LOCA.

I am also delighted to have attendance which is a good mix of regulators, researchers, operators and reactor designers representing LWR's, CANDUs and VVERs.

This is intended to be a workshop to discuss rather than to present. With an attendance of about 70 experts we should be able to have a good discussions.

Why this workshop and why now?

Many of you have spent a good part of their professional life analyzing or researching the consequences of large break LOCA. It has always been considered a stylised or enveloping design basis event which sets the requirements for a number of reactor systems. I think we can say that using it as a DBA has generally led to a robust design with good safety margins. On the other hand, since the original design philosophy for water reactors was determined, considerable additional technical knowledge has been accumulated through operating experience and research programmes. We now have much better analytical tools to calculate more precisely the available margins. Furthermore, there is a strong push in most NEA countries to move towards a risk informed regulatory process. Because of this, a number of countries are considering regulation that may redefine the LOCA which is considered design basis.

The CSNI and CNRA decided therefore to be proactive and bring together now the experts to discuss the technical basis and the implications of a LOCA redefinition.

What is the structure of the workshop?

The workshop programme is focused around three basic questions which correspond to Sessions 1, 2 and 3. We hope to put in clear focus and get some conclusions on what drives the need to redefine the LB-LOCA, do we know enough to do it, and if we do it what is the new definition and what are implications on current and future water reactors?

To round out the programme we will start on historical perspective which will tell us how we got to this point and we will conclude with a general debate and hopefully consensus conclusions.

To complement the technical discussion which will take place here, we have collected, through a questionnaire, information on the relevant regulatory framework in our member countries. The results of this questionnaire will be summarized by Eric Mathet and are available in a room document.

What are we hoping to achieve?

As is the tradition of the CSNI and the CNRA, we will seek to review and achieve technical consensus on the three basic questions. How to translate this technical consensus into regulatory policy is a question that is better dealt by each regulatory body. Clearly it is not possible to address in detail all technical issues in two days. Therefore the workshop should also come to some conclusion on what other issues may require additional research discussion by the CSNI and its working groups.

In terms of tangible output we plan to issue the proceedings, a compilation of the responses for the questionnaire, a one to two page answer to each of the three fundamental questions. We are counting on moderators to draft these, and a set of conclusions which will come from the discussion chaired by Jukka Laaksonen.

To conclude let me just go through some house keeping items.

- Coffee breaks outside the room
- Lunch in the Exchange Center – you should have paid for it at the registration desk
- Dinner tonight hosted by HSK-PSI (details will be given tonight)
- We intend to finish by 4 p.m. tomorrow

I'd like now to turn to Dr. Schmocker and ask him to introduce our keynote speaker Chairman Diaz.

Keynote Speech at redefining the Large Break LOCA

"... an idea Whose Time has come"

US NRC Chairman, Dr. N. Diaz

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

I am very pleased to be here with so many technical experts from around the world to discuss a subject which I feel very strongly about. I would like to express my thanks to the NEA committees, the CNRA and the CSNI, and to our Swiss colleagues for organizing and hosting the meeting, and for the opportunity to share my views with you.

Let me say from the beginning that redefining the Large Break LOCA is for me, and I hope for all of us, a significant safety initiative. I cannot stress that fact enough ... a safety initiative. We in the US experienced our most serious reactor accident at Three Mile Island (TMI) in 1979 -- twenty-four years ago, yet still fresh in our memories. The TMI accident was not a Large Break LOCA, it was not the event that we had invested so much of our time and technical resources in. The TMI accident was a small LOCA, an event given significantly less attention because of the overwhelming amount of attention on the Large Break LOCA concern. During the four decades since nuclear power plants began operation, each of our nations has experienced important reactor safety events, yet none were Large Break LOCAs. The only power or production reactor accident – Chernobyl – that resulted in loss of life on-site and massive radioactivity releases was many things but not a Large Break LOCA. All the other reactor safety events include occurrences such as small LOCAs, or loss of decay heat removal or fires or reactivity events. With today's improved know-how, shouldn't we be focused on the right safety issues? Shouldn't we assure the public, whom we are protecting, that our attention and the attention of our licensees is focused on the most important issues and activities for preserving their health and safety? I believe the record shows that we do a good job, but we can do a better job by using what we now know is more safety-focused, cognizant of the past and of present and future needs, and dedicated to the task at hand: protection of public health and safety and the environment.

I believe the nuclear regulatory agencies, cognizant of the present safety experiences and assessment capabilities, need to take the next step. The licensees and reactor vendors cannot change their focus until we change ours. That's a fact. Regulation and technology need to progress in parallel, in phase. And in this particular case, the regulators are currently lagging the technological capabilities. We also need to recognize, consider, and address the technical, legal, and political impediments to change, so whatever we do has to be right, scrutinized and well communicated.

Let me remind you of a quote from the well known 19th century author Victor Hugo, who said,

"Nothing else in the world is so powerful as an idea whose time has come."

Well, I believe that redefining the Large Break LOCA through a risk-informed and performance-based approach is an idea whose time has come. The double-ended rupture of the largest pipe in the RCS should be moved from the design basis to severe accident management space. This change will not create a void, it will create the opportunity for safety

improvements per se, and will establish the due process and requirements to eventually replace design bases with a better, living and dynamic safety basis.

We have a good reason for a change; we need to have the technical basis to support that change. Therefore our first expectation for this meeting should be to identify, clarify, and, if possible, agree upon the current state of knowledge on the probability and consequences of various LOCAs.

As a second expectation, and as I alluded to above, we should also explore a related question (and answer it as best we can); that is, "If we change the Large Break LOCA, what should replace it?"

There is no doubt that, we will need to consider all of the design and operational implications of redefining the Large Break LOCA, and do it better than well. These include issues such as fuel and core design; containment design basis; ECCS design; RCS supports; emergency diesel generator start time; maximum hypothetical accident for dose assessment, emergency preparedness and control room habitability. These sets of issues need to be reduced for holistic system and probabilistic analysis.

Before discussing possible changes to the large Break LOCA, let me first speak about the current NRC regulations in this area, that is, 10 CFR 50.46 and Appendix K, which establish the requirements for Emergency Core Cooling Systems. I will also mention some of the history of these requirements.

50.46 requires that "... ECCS cooling performance must be calculated ... for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated..". In this context, "loss-of-coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system." In Appendix K, "ECCS Evaluation Models," the word "instantaneous" is added to the phrase "double-ended breaks" making the traditional maximum LOCA (but not necessarily the worst LOCA) the instantaneous double-ended break of the largest pipe in the reactor coolant system (usually the hot and occasionally the cold leg of the RCS).

50.46 analyses are all about consequences. And understanding consequences without understanding the associated probabilities is particularly meaningless for this case. We know that now very well, but the US Atomic Energy Commission (AEC) also knew that back in the 1970's. Qualitative judgements were made about the probability of a LOCA. That's why pipe failures are included in 50.46 but reactor vessel failures are not. A judgement was made that vessel failures were so unlikely that protection was not necessary. That was a qualitative judgement about probability.

The approach to classifying events as "anticipated operational occurrences" and "postulated accident," is more than three decades old. It is a qualitative (or at best semi-quantitative) approach to event probability.

As operating experience and research data become available over time, those qualitative judgements are first validated and later replaced with quantitative information. It is a normal technical progression to go from qualitative judgements to quantitative estimates over time. That's expected progress.

In the December 28, 1973, "Opinion of the Commission," on the rulemaking hearing on 50.46, the Commission stated:

"in adopting this course [the 50.46 rule], we are not blinding ourselves to new knowledge acquired as a result of ongoing research. On the contrary, we believe that it is important that research programs – both analytic and experimental – continue, in order that we may increase knowledge relevant to ECCS performance.... As new knowledge is acquired, the Commission will analyze it, and at the appropriate time consider the possibility of amending the rule we announce today."

The Commission expected the regulatory requirements to change and progress along with the technology. However, they probably didn't think it would take 30 years!

In developing WASH-1 400, the original "Reactor Safety Study," the AEC used the best information at that time to estimate the probability of various LOCA's --including Large Break LOCAs and even vessel failures -- that was 1974.

Following the TMI accident, the NRC undertook a deep and serious look into its regulations and regulatory practices in the "NRC Special Inquiry" often referred to as the "Rogovin Report." In that report, a number of recommendations call for the increased use of risk analysis and risk insights. These recommendations include the following:

"The best way to improve the existing design review process is by relying in a major way upon quantitative risk analysis" and

"What we [the NRC Special Inquiry] are suggesting is that [the existing review process] be augmented and that quantitative methods be used as the best available guide to which accidents are the important ones, and which approaches are the best for reducing their probability and consequences," and again,

"We strongly urge that NRC begin the long and perhaps painful process of converting as much as is feasible of the present review process to a more accident-sequence-oriented approach."

I agree with their recommendations and with their predictions that the transition would be "long" and "painful." It should not have been that long and that painful, but it has been. The wheels of "nuclear" progress turn slow because predictability became equated to success. I do not disagree with that; I just disagree with the interpretation of predictability and success. Predictability must be rooted in today's know-how and success (in our case safety success) has to be meaningful for 2003 and beyond.

In 1995, eight years ago, the Commission issued a formal Commission Policy Statement supporting the increased use of PRA. We have made significant progress in the use of PRA since then, but we are far from done. That's our history and we cannot change it. But we have the opportunity to change the future, and I submit to you that we have the obligation to do so.

Now, in 2003, LOCA probabilities are routinely included in Probabilistic Risk Assessments (PRAs or PSAs). They are calculated every day and all around the world and are used in operational safety decisions ... why not in the basic design requirements too? We have a sound understanding of the probabilities and consequences sufficient to progress to the next rational level of regulation to improve reactor safety.

The changes being considered by the NRC are headed in this direction. The situation is as follows:

The Commission has recently agreed to consider redefining the design basis large-break loss-of-coolant accident (LOCA) in view of the low risk associated with such events. The NRC staff was directed to provide the Commission a comprehensive "LOCA failure analysis and frequency estimation" that is realistically conservative and amenable to decision-

making and to consider use of a 10-year period for the estimation of LOCA frequency distributions, with a rigorous re-estimation conducted every 10 years and a review for new types of failures every five years.

In that effort, the staff was directed to use Service-Data, Probabilistic Fracture Mechanics (PFM), and Expert Elicitation in a process that is risk-informed and consistent with the principles of RG 1.174.

The staff was also directed to credit leak-before-break where a licensee establishes a reliable and comprehensive means of detecting primary system leaks of the relevant size.

The staff was further directed to establish an appropriate risk "cutoff" for defining the maximum LOCA size and to require strict configuration controls during plant operation and a high quality PRA, including low power and shutdown operations.

These directives from the Commission to the NRC staff, highlight the two key technical issues involved with re-defining the LBLOCA; namely, LOCA frequencies and "PRA Quality". "PRA Quality" means having the appropriate scope level of detail reliability data and realism in accident progression and success criteria to support the regulatory decisions to be made. Since the risk assessment will play a significant role in this important change (i.e., redefining the LBLOCA), we expect the PRA to be of high quality so that the results are both reliable and convincing. The PRA does not need to be perfect, but it does need to be "good enough". How good is "good enough" is an issue that we face for each risk-informed activity. And, as with previous activities, we will work with experts in the field to develop guidelines on "PRA quality" for this issue, and will probably use a NRC Regulatory Guide. The "PRA quality" issue will be difficult but it is well within our technical capabilities, and will be resolved in a prudent manner.

I am convinced that, as a matter of improving safety, the consideration of very low probability Large Break LOCAs should be addressed as severe accident scenarios, in the severe accident management program, rather than as the design basis accident. Effectively, the current LBLOCA would not be a design basis accident when utilizing a risk-informed approach. With an alternative definition of the LOCA, the really important, risk-significant, accident scenarios would remain within the design basis; in fact, their consideration would be enhanced by a new focus on their risk-importance.

All of these activities are in the formative stage and the NRC staff will develop proposed rule changes and associated guidance for public review and comment over the next several months. In addition, we expect one or more pilot applications which would request risk-informed changes to the Large-break LOCA requirements through the NRC exemption process. This will provide a way of getting direct and practical experience with some of the important decisions to be made. We have found this approach very useful in the past.

I have no doubt that some, perhaps many, of the details of the rules and guidance will change, will mature and will become clearer as the staff discusses alternatives with interested parties ... and that is good. Some new alternative approaches may even be developed. Information from this meeting may also influence the NRC's plans -- and that would be good too.

What I believe will not change is our commitment to improving safety and modernizing the treatment of the Large Break LOCA through the use of the best available information on the likelihood and potential consequences of these events and the best available approaches. And beyond the Large Break LOCA? 10 CFR 50 Appendix A and all it touches.

Remember:
"Nothing else in the world ... is so powerful as an idea whose time has come"
I look forward to your contributions to make it happen.

Historic Perspective on Design Basis Accident based on experience in Canada, Russia and the United States

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

Synthesis

J. A. Murphy, (Consultant, USA)
N. S. Fil (EDO Gidropress, Russia)
J. C. Luxat (Nuclear Safety Solutions Limited, Canada)

Early Developments

United States

The regulatory process in the United States was developed shortly after the Congress passed the Atomic Energy Act of 1954. In 1959, the Advisory Committee on Reactor Safeguards. A committee was formed which made several recommendations, which shaped the direction and form of the regulatory process in the USA over the next several years. The more important of these are the following:

- Rules should be established by which sites would be considered acceptable for locations of reactors.
- The AEC should not attempt to standardize the technical design and construction specifications and procedures for reactors or for the various components of reactors
- There should be a continuing effort on the collection and organization of safety guides, or state-of-the-art practices; and a systematic tabulation of safety performance objectives for reactors. These should be made available to the industry, but not as regulations.
- A guide should be developed on the preparation of hazards summary reports, the term then used for what is now called safety analysis reports.

In the US, this recommendation on development of siting rules led directly to the development of the AEC's siting criteria contained in 10 CFR Part 100 issued in 1962 and of the accompanying technical guidance, TID-14844, Calculation of Distance Factors for Power and test reactor Sites.

In 1962, the AEC Licensing Guide was issued for comment. This document incorporated all the applicable regulatory requirements and provided guidance on the contents of Hazards Summary Reports in terms of high-level performance objectives.

Examples of the level of detail in the AEC Licensing Guide follow:

" . . . the Hazards Summary Report should provide information appropriate to a determination of the adequacy of auxiliary systems to provide those required functions

necessary to maintain the facility in a safe condition during normal operation and in the event of accident."

"(3) Emergency core water injection system, in sufficient detail to permit an evaluation of the adequacy and limitations of the system to perform its intended function in the event of a loss of coolant accident."

It is interesting to note, that as written, the concept of the double-ended large LOCA was primarily a tool used for analyzing containment performance in this guide. This led to early designs that focused on containment performance and on meeting the dose requirements of the siting regulations. Examples here would include the containment spray system at one plant employing a steel sphere containment where the sprays were on the outside of the containment, and the several sub-atmospheric containment, where the plant was maintained below atmospheric pressure so that leakage was calculated to cease within a short period of time of the occurrence of a loss of coolant accident. Because of the emphasis on containment performance, many of the early plants (which were of low power levels and low power densities) did not require operation of emergency core cooling systems if off-site electrical power were to be lost as a part of the initial plant design philosophy.

Canada

The safety philosophy for CANDU reactors was very strongly influenced by lessons learned from the loss-of-regulation accident that occurred in 1952 in the NRX (National Research Experimental) research reactor at Chalk River Nuclear Laboratories. Out of the NRX accident grew the requirements for functional and physical separation of special safety systems from process systems, the requirement for fast-acting shutdown systems, the requirements for demonstrating high unavailability and testing of passive safety systems, and the incorporation of an elementary risk-based licensing framework. All of these requirements were encapsulated at a high level in a licensing requirement referred to as the Siting Guide and issued by the Atomic Energy Control Board. Central to the framework were three guiding concepts.

First, the "*safety systems shall be physically and functionally separate from the process systems and from each other.*" The safety systems included the reactor shutdown system (SDS), the emergency core cooling system (ECCS) and containment. The process systems included all systems necessary for control of the plant during normal power operation and shutdown conditions, such as the reactor regulating system, boiler pressure and feedwater control, shutdown cooling, moderator and end-shield cooling, service water, electrical and instrument air, etc.

Second, the process systems should be of high quality to limit the frequency of failures that could lead to accidents and the special safety systems should be highly reliable such that their unavailability is a low probability condition. Specifically, the principles were articulated as follows.

" The quality and nature of the process systems essential to the reactor shall be such that the total of all serious process failures shall not exceed 1 per 3 years. A serious process failure is one that in the absence of protective action would lead to serious fuel failures."

And,

"Each safety system shall be readily testable, as a system, and shall be tested at a frequency to demonstrate that its (time) unreliability is less than 10^{-3}."

Third, the concept of risk was introduced in an elementary manner by requiring lower dose consequences for more probable failures (single failure of a process system) and

allowing higher dose consequences for less probable events (dual failures involving failure of a process system and coincident unavailability of a special safety system). Dose consequences were specified for an individual and for the population at risk, according to:

> *"The effectiveness of the safety systems shall be such that for any serious process failure the exposure of any individual of the population shall not exceed 500 mrem and of the population at risk, 10^4 man-rem."*

> *"For any postulated combination of a (single) process failure and failure of a safety system, the predicted dose to any individual shall not exceed (I) 25 rem, whole body (ii) 250 rem, thyroid, and to the population, 10^6 man-rem."*

This represents one of the earliest uses of risk information in reactor regulation.

Russia

In Russia, the safety systems of these power units were designed at the time when no special domestic standards for nuclear safety are available, proceeding from the industrial standards and regulations. Nevertheless, the most important equipment of reactor plants was designed according to the specially developed normative-technical documentation. It provided for a high level of reliability and safety of these plants that was confirmed by their operational experience and by licensing of their operation beyond the design service life with account for the domestic and international standards of nuclear safety being in force nowadays.

The principle of safety assurance of the first generation WWER-440 (reactor plants V-179, V-230, V-270) was based on the measures directed to prevent the dangerous initiating events, including LOCA. First of all, the measures were taken for quality assurance of design, manufacturing, installation and operation of the equipment. The main equipment of primary circuit (reactor, steam generator, coolant pump, components of reactor coolant circuit) was of high reliability and had significant structural margins, that excluded their possible break with formation of large leak. Therefore, only breaks in the pipelines attached to the primary circuit were considered as the design basis for the first generation WWER-440. All these pipelines were provided with the insertions of 32 mm diameter to ensure scale of the maximum design basis accident with LOCA as a leak from the hole of diameter 32 mm within the reactor coolant pressure boundary. Such accident was included into the design bases for emergency core cooling system and localization system.

The coolant outflowing in this accident from the primary circuit is localised in the confinement designed for pressure 0,1 MPa. When the pressure increases to 0,02 MPa, the spray system is actuated. This system consists of the storage tank for boron solution of 800 m^3 and three pumps with capacity of 400 m^3/h each. With further pressure increase in the confinement, the relief valves from SG boxes open. The first valve has capacity 25 kg/s and it opens at pressure 0,06 MPa, others 8 valves of capacity 180 kg/s each open at pressure 0,08 MPa.

The same scale of maximum design LOCA - leak of 32 mm diameter – is included into the design basis for emergency core cooling system of the first generation WWER-440 (safety boron injection system). This system makes use of the same storage tank for boron solution of 800 m^3. The system consists of two groups of pumps (in all 6 pumps) of total capacity equal to 300 m^3/h at primary pressure up to 13 MPa. The system is started when the pressurizer level decreases and by some other signals.

Advent of Higher Power Facilities

United States

With the advent of the next generation of power reactors, the importance of the emergency core cooling function became obvious. In the United States, the basic requirements evolved with time and were expressed primarily in the conditions of licenses that had been litigated before AEC hearing boards. For example, as initially proposed by applicants, not all reactors being proposed for licenses or construction permits could operate the ECCS systems in the absence of off-site power. One three-loop PWR employed the steam generator main feed pumps as ECCS pumps. They could deliver a large quantity of water at pressures approaching 70 atmospheres, but could not accommodate a smaller break where the pressure remained high. They also required the presence of off-site power. Late in the design, a single pump capable of handling a small break while operating on emergency ac power was added.

As case law was developed through the hearing process it became standard practice to insert a condition into the license similar to the following from the St. Lucie Operating License in 1970:

> *The proposed facility has separate systems designed to provide adequate core cooling and pressure reduction within the containment structure even if a loss-of-coolant accident should occur. For immediate short-term cooling, an emergency core cooling system (ECCS) will inject cool borated water into each of the primary coolant loops and directly into the reactor vessel, thereby limiting energy and fission product releases into the containment. For cooling containment air and to reduce containment vessel internal pressure in the event of an accident, the design includes a spray system for delivering cool borated water into the containment atmosphere, and also a containment fan cooling system with four cooling units and a centrifugal exhaust fan. Either of these systems is designed to meet post accident containment cooling requirements. (Specifics of this condition varied with plant design.)*

Canada

In Canada, one unique aspect of the Siting Guide was the concept of dual failures involving a process failure and failure of one of the special safety systems. Since there was no overall lower frequency limit imposed on the process failure event frequency, this forced analysis of low probability - high consequence events, including a large break LOCA with Loss (failure) of the Emergency Core Cooling system (LOECC). This event, when carried to the extreme and with conservative treatment in the safety anlysis, effectively becomes a severe fuel damage event, thereby placing some core damage events directly in the design basis.

A second consequence was that logical application of the dual failure resulted in process failures with failure to shutdown were also to be included in the design basis for reactors with only one independent shutdown system. Again, an event with the potential for significant core damage was included in the design basis. However, this was a difficult analytical issue since the state of knowledge regarding the consequences of such events was limited and any analysis was considered speculative. The inability to satisfactorily resolve this issue led to development of a requirement for new designs to be fitted with a second independent shutdown system. The requirement was initially enforced in the course of licensing the next nuclear station, Ontario Hydro's Bruce A station on the shores of Lake Huron.

Russia

In Russia in 1971, the safety requirements for nuclear power plants were formulated for the first time. On the basis of these requirements, the regulatory document "Basic regulations on NPP safety assurance" (OPB-73) was compiled in 1971 and published in 1973. Item 1.2.5 of this document contains the following requirement:

> *Guillotine break of the pipeline being the most dangerous with regard to radiation consequences shall be considered as the maximum accident with loss of circulation circuit integrity, if it is not excluded by the implemented technical measures.*

This requirement was interpreted in the design practice so that break of the primary pipeline with maximum diameter was included into the design bases of all NPP systems and components.

In 1974, the designs of 24 power units with WWER-440/V-213 reactor plant were developed for Finland, USSR, DDR, Hungary and Czechoslovakia. In these designs the technical measures are stipulated to ensure NPP safety under the accident with guillotine break of the main coolant pipeline with 500 mm internal diameter. Such accident was included into the design basis of the emergency core cooling system and localization system. In particular, equipment, pipelines and primary circuit systems of WWER-440/V-213 are arranged in the protective boxes, and condensation of steam generated during the accident is performed by the bubbling-vacuum system (for Loviisa NPP – by the ice condenser).

The accident with guillotine break of pipeline with diameter 500 mm is also included into design bases of other systems and equipment of the WWER-440/V-213 reactor plant (support and restraints on the main equipment and pipelines, core, reactor internals, etc.).

Further Evolution of Design Criteria

United States

In the United States, concerns were raised in the mid-1960s regarding the ability of the containment to maintain its function in the event of a loss of coolant accident. An extensive experimental and analytical research program was begun to investigate ECCS performance. Some scientists perceived that inadequate attention was being given to research programs to determine the effectiveness of systems to mitigate the consequences of a loss of coolant accident, if it were to start. The Atomic Energy Commission convened a public hearing to explore the safety question. Ultimately, after extensive public hearings, in 1974, the regulations were modified to provide a set of specific requirements for computer codes for ECCS analyses in Appendix K and a new section, 50.46, requiring ECCS meet established standards. This included a definition that loss-of-coolant accidents (LOCAs) are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. The Commission's opinion also promised a safety research program to firm up the basis for acceptable features of the computer codes, and to permit relaxation of the requirements where that became possible.

In parallel with the research activity, the regulatory staff of the Atomic Energy Commission began developing a set of general design criteria to codify existing staff practices. A draft version of General Design Criteria for Nuclear Power Plant Construction Permits was circulated for comment in 1967 and received extensive public comment. A final version of the General Design Criteria for Nuclear Power Plants was issued in 1971. This first approved version of the General Design Criteria to be formally incorporated in the regulations acknowledged that it was still incomplete *because "... specific requirements had not yet been*

sufficiently developed and uniformly applied in the licensing process to warrant their inclusion ..." These matters included

> *Consideration of the type, size, and orientation of possible breaks in the components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss of coolant accidents.*

However, Criterion 35 – Emergency Core Cooling did require

> *The system safety function shall be to transfer heat from the reactor core following any loss of coolant accident at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.*

Similar language is used in Criterion 50, Containment Design Basis, but this criterion also requires provision of sufficient margin considering " ... the limited experience and experimental data available for defining accident phenomena and containment response, and ... the conservatism of the calculational model and input parameters."

Canada

In Canada, Given the importance of the special safety systems in mitigating the progression of events in CANDU accidents, the AECB developed a formal set of requirements for these systems in Regulatory Documents that are applicable to reactors licensed for construction after either 1977 or 1981. The first Regulatory Document R-10, a Regulatory Policy Statement entitled *The Use of Two Shutdown Systems in Reactors* stated that, amongst other requirements:

> "*All nuclear power reactors licensed for construction in Canada after January 1, 1977 shall incorporate two independent protective shutdown systems unless otherwise approved by the Board.*"

And,

> "*The protective shutdown systems shall be of diverse designs and each shall be physically and functionally separate from the other, from process systems, and from other special safety systems.*"

The other Regulatory Documents that followed were:

- *R-7, Requirements for Containment Systems for CANDU Nuclear Power Plants,*
- *R-8, Requirements for Shutdown Systems for CANDU Nuclear Power Plants, and*
- *R-9, Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants.*

These documents articulated design requirements, operating requirements for both normal and accident conditions, and testing requirements that covered commissioning, in-service testing and inspection and testing of system availability.

Although simple in concept and relatively effective in application, Canada's Siting Guide did not provide adequate guidance in a number of problem areas. For example;

- There was no distinction between single failures occurring at differing frequencies and having different consequences.

- There was no distinction between dual failures occurring at differing frequencies and having different consequences.

- There was no guidance with respect to the frequency of events below which consequences need not be considered (i.e. no cut-off frequency).

- Safety system impairments (failures) were defined without consideration of the fact that systems, such as containment, have a number of separate subsystems, each of which should be separately considered.

These problems contributed to difficulties in addressing issues at the boundary between design basis and beyond design basis events. The AECB proposed a revised framework, defined in the Consultative Document C-6, Requirements for Safety Analysis of CANDU Nuclear Power Plants to address, in part, these problems. Intended to be applied to all nuclear power plants without a construction license as of July1, 1988, C-6 was applied on a trial basis in the licensing of the Darlington nuclear generating station, the last station constructed in Canada. This proposed requirements document extended the elementary risk-based concepts by defining five classes of events with reference dose limits for each class. The classes of events were broadly related to decreasing event frequency with increasing class number. However, the framework was basically deterministic in nature. It was not fully risk-based in that initiating events were pre-specified in the event classes together with combinations of initiating events and failures of special safety systems and safety support systems, irrespective of the specific features of a design which can influence the estimates of event frequency. Finally, there was no guidance regarding an event frequency cut-off.

These issues were subsequently addressed in an updated Consultative Document, C-6 Rev. 1, in which specific requirements for a probabilistic safety assessment were introduced, event frequencies were assigned, and a cut-off frequency (10^{-6} per annum) was specified, below which, with regulator approval, analysis of event consequences was not required.

Russia

In Russia, The requirement as regards LOCA in new version of OPB issued in 1982 (OPB-82) was even more defined. Item 4.1.1 of this document runs as follows:

> *Guillotine break of the maximum-diameter pipeline with free double-ended discharge of coolant during reactor operation at nominal power with account for its possible exceeding due to errors and tolerances of I&C shall be considered as the maximum design basis accident with loss of primary circuit integrity.*

The quantitative requirements for emergency cooling systems in terms of maximum permissible temperature (1200^0C), local oxidation (18 %) and general oxidation (1 %) of fuel rod claddings were also established for the first time in OPB-82 (item 4.1.3)

The similar requirement is in the USSR State standard *"Nuclear power reactors of vessel type with pressurized water" (GOST 24722-81)* being also put down in the register of Russian State standards. Item 1.5.1 of this document runs:

> *Guillotine double-ended break of the main coolant pipeline in combination with condition of complete loss of NPP power supply shall be considered as the maximum accident in the reactor design. For the reactors to be delivered to seismic-active NPP site the above accident together with SSE shall be considered as the maximum accident.*

The subsequent versions of OPB as the basic safety standard (OPB-88 issued in 1989, and OPB-88/97 valid nowadays) no longer contain the indication on the size of design LOCA, nor do the regulatory documents intended specially to define the design bases for NPP localization system and emergency core cooling system do not contain such instructions as well:

Rules for design and operation of the localizing safety systems of nuclear stations (НП-010-98).

Rules for design and operation of the systems for emergency cooling and heat removal from nuclear reactor to ultimate sink (ПНАЭ Г-5-020-90).

Regulatory document PNAE G-01-036-95 intended to specify format and contents of the safety analysis report contains the "provisional and minimum" list of design basis initiating events for the accident analysis. In this list, the "break in the main pipelines containing the primary circuit medium" is mentioned but nothing is said about the size and the characteristics of such break.

Thus, the current documents of RF Gosatomnadzor give no specific instructions concerning the scale of LOCA, which shall be assumed as design basis for the corresponding safety systems. Nevertheless, the current design practice of WWER actually makes use of OPB-82 wording as the design basis to select the characteristics of emergency core cooling system and containment depressurization system. With this, some components of the reactor plant and NPP, including those effecting the functioning of these safety systems, are designed without consideration, for example, of the dynamic loads arising during guillotine break of the primary pipeline with maximum diameter. It creates some formal discrepancy in the design because many regulatory documents require the designer *to take into account all the effects arising during the design basis accidents.*

Leak Before Break (LBB) Considerations

United States

In the United States in 1975, the NRC staff was informed that improved analytical techniques had revealed newly calculated asymmetric loads that result by postulating rapid-opening double-ended ruptures of PWR primary system piping. Analyses performed in response to this request indicated that some plants would require extensive modifications in the form of pipe whip restraints if the rapid-opening double-ended break were to remain a design basis postulation.

In addition to the requested analyses, the industry prepared several topical reports to demonstrate that an assumed double-ended rupture was not a credible design basis event for primary system piping. The NRC staff evaluated these reports and other available research results and issued a Safety Evaluation in 1984 (Letter to All Operating PWR Licensees, Construction Permit Holders, and Applicants for Construction Permits from Darrell G. Eisenhut, Director, Division of Licensing, Office of Nuclear Reactor Regulation.) The attached Safety Evaluation found

> *. . . the probability of a full double-ended rupture of tough piping in a typical PWR primary coolant system is vanishingly small.* (Note this Safety Evaluation addressed only the pressurized water rectors.)

This safety Evaluation concluded with certain exceptions that

> *... the safety margins indicate that the potential for failure is low enough so that full doubleended breaks need not be postulated as a design basis for defining structural loads.*

The overall conclusion was that asymmetric blowdown loads resulting from double-ended pipe breaks in main coolant loop piping need not be considered provided leak detection systems were sufficient to provide adequate margin to detect the leakage from a postulated circumferential throughwall flaw before unstable crack extension occurred.

This finding had a significant effect on design and on operational practices. With appropriate analyses, it permitted removal of pipe whip restraints and jet impingement devices. This, in turn, had a very positive effect on the radiation exposure to plant personnel since equipment became easier to access for inspection or maintenance. This letter also indicated that *the "staff intends to proceed with rulemaking changes to GDC-4 to permit the use of fracture mechanics to justify not postulating pipe ruptures."* for defining structural loads. This criterion has since been modified and now states:

> *However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.*

The value-impact analysis which accompanied the safety evaluation estimated an overall savings of 11,000 person-rem by eliminating the need to install, inspect and maintain the pipe restraints and making it easier to inspect piping and components nearby.

There is a comprehensive summary of more recent research and regulatory activity devoted to leak before break considerations in NUREG/CR-6765, Development of Technical Basis for Leak-Before-Break Evaluation Procedures.

Canada

In Canada, the concept of leak-before-break (LBB) was introduced in Consultative Document C-6 Rev. 0 but it was restricted to pressure vessels with a specific exclusion that the reactor headers not be considered as vessels. For heat transport piping and header failures it was required that:

"Pipe failure shall consider both circumferential, guillotine, and longitudinal failures at any location in a system.

(a) For circumferential pipe failures a discharge area up to and including, twice the cross-sectional area of the pipe shall be analyzed.

(b) Failures resulting from longitudinal cracks shall also be considered and justification given for the maximum crack size postulated."

The licensee did develop, and the regulator accepted, an LBB case for large diameter (NPS greater than 6 inches) heat transport system piping to support removal of pipewhip restraints from the Darlington design. These considerations notwithstanding, all locations and sizes of postulated breaks were required to be analyzed to demonstrate that the special safety systems were effective in assuring the integrity of design barriers to the release of fission products.

Russia

In Russia, the applicability of LBB concept for the existing and future Nuclear Power Plants (NPPs) is defined in the following documents approved by the national safety authority:

Technical requirements for application of "Leak before break" concept for the pipelines of operating NPPs (1995).

Guide on application of "Leak before break" safety concept to the pipelines of nuclear power installations (P-ТПР-01-99, 1999).

It is stated in document P-ТПР-01-99 that comprehensive application of the guillotine break of the pipeline with maximum diameter as the design basis LOCA results in elaboration of special measures on protection of all components and structures important to safety against the corresponding emergency loads. To satisfy these purposes, the applicable protective measures include physical separation of the equipment, supports and restraints for equipment and pipelines, protective shields, etc. These measures make NPP construction and operation more expensive and can exert negative influence upon safety because they significantly complicate the operational inspections of equipment and pipelines.

The methodology of the LBB concept is accepted as the technically justified approach to not considering the dynamic impacts from postulated double-ended guillotine rupture in the pipeline systems with high stored energy. The LBB concept can be used both for newly designed plants and for operating ones if the following conditions are met:

- The analyzed components and pipelines have been defined potentially suitable for LBB concept application from the viewpoint of design bases, meeting the requirements of the main technical principles and assessment of the gained operational experience;
- The leak monitoring systems have been developed and passed the calibration tests, and subsequent improvements of these systems for the early-generation plants are ensured;
- The sufficient margins obtained using the calculation procedure on substantiation of LBB concept have been demonstrated;
- The structural tests of actual large-scale models or pieces of full-scale tubes have shown viscous character of their failure.

Successful application of the LBB concept giving technical substantiation of impossibility for sudden guillotine failure (or its longitudinal equivalent) in the pipeline systems allows:

1. Excluding specific consideration of the static and dynamic impacts related to large-scale breaks and leaks, for example:
    - jet forces, force of reactions, pipe whipping and impingement, missiles;
    - impacts upon the component supports and upon the civil-engineering structures (for example, pressure differentials on the in-containment walls);
    - shock pressure waves inside the equipment and pipelines.
2. Excluding the necessity for development and installation of additional supports, fasteners, pendants, restraints of pipe whipping, protective shields and other massive and expensive devices required to mitigate the consequences of guillotine failure, and thereby to facilitate in-service inspection and maintenance of the pipeline systems.

For example, in the application of LBB concept to an operating NPP the dynamic impact of coolant to the core elements and reactor internals are to be assumed for the postulated leak size equal to 10% of flow area of the main coolant pipeline. For WWER-1000, it is equivalent to one-ended break of the pipeline with diameter about 270 mm. At the same time, application of LBB concept does not cancel the current requirements for safety systems and does not mean that it is possible to remove such protection means as the containment, emergency core cooling system, accident localization system, etc.

Summary

The design bases have evolved with time in each country, reflecting the lessons learned from operational experience and ongoing research efforts. Credit has been given for leak before break analyses to some extent in each country, primarily to permit removing pipe whip restraints that protect against very fast opening large breaks in the primary circuit. These have been done based on fracture mechanics analyses, and the availability of leakage detection devices. There are several concerns, explored in the detailed papers in this session, as well as the rest of this workshop, that an over-emphasis on an accident initiator with exceedingly low frequency of occurrence may divert from a n optimal safety design.

Historical Perspective on the Design Basis Accident Based on Experience in Canada, Russia, and the United States

Joseph A. Murphy, Consultant, USA
N. S. Fil, Edo Gidropress, Russia
John C. Luxat, Nuclear Safety Solutions Limited, Canada

Early Developments – U. S.

- 1959 study recommended:
  - Siting regulations be developed.
  - No attempt should be made to standardize technical design.
  - Continuing effort to develop safety guides and safety performance objectives
- 1962 – siting regulations issued; draft AEC Licensing Guide issued.
  - Safety guidance quite general expressing top-level principles
  - Double-ended large LOCA primarily for analyzing containment performance.

2

Early Developments - Canada

- Early safety philosophy strongly influenced by 1952 NRX accident.
- Requirements published in Siting Guide by AECB:
  - Safety systems should be physically and functionally separate.
  - The quality and nature of process systems shall be such that the total of all serious process failures should not exceed 1 per 3 years.
  - Each safety system should be testable ... to demonstrate unreliability is less than 10^{-3}.
- One of the early examples of the use of quantitative risk in regulation.

3

Early Developments - Russia

- Safety assurance based on measures to prevent dangerous initiating events.
  - High quality and reliability, large safety margins.
  - LOCA size based on size of lines attached to primary circuit.
  - Confinement system designed for 32 mm LOCA.

4

Advent of Higher Power Facilities United States

- Importance of the emergency core cooling function became obvious.
- Requirements varied with time as expressed in individual licensing cases, while generic guidance was still under development.
- Large double-ended pipe breaks considered, but in some cases, limited consideration of small breaks or process availability.
- General Design Criteria when first issued in 1971 did not specify size of LOCA required to be considered.

5

Advent of Higher Power Facilities Canada

- Siting Guide required consideration of dual failures (process failure + failure of special safety system).
  - Forced analysis of low probability, high consequence events, including large break LOCA with loss of ECCS, placing some core damage events in design basis.
- Dual failure principle also led to failure to shutdown being included in design basis.
  - Led to requirement of second independent shutdown system.

6

Advent of Higher Power Facilities Russia

- Safety requirements formulated in 1971.
  - Required consideration of guillotine break of primary pipeline of maximum diameter (500 mm)
  - This size break also used for design of protective boxes, and condensation of steam by bubbling vacuum system (or ice condenser).
  - Also used in design of supports and restraints and core internals, etc.

7

Further Evolution of Design Criteria United States

- Concerns raised in mid-1960s regarding containment performance in event of LOCA, and effectiveness of ECCS to mitigate consequences.
- Extensive analytical and experimental research program initiated.
- Lengthy public hearing held.
- AEC took following actions:
  - Modified regulations to provide specific requirements for computer codes.
  - Defined large break LOCA
  - Confirmed continuation of research program to firm up basis for acceptable features of computer codes, and to permit relaxation of requirements.

8

Further Evolution of Design Criteria Canada

- In late 1970's AECB issued formal set of requirements in Regulatory Documents.
- Covered
  - Use of two shutdown systems
  - Requirements for containment systems
  - Requirements for shutdown systems
  - Requirements for Emergency Core Cooling Systems.

9

Further Evolution of Design Criteria Russia

- In 1982, issued regulation further defining large LOCA and conditions for the analysis
- Established quantitative requirements for maximum permissible temperature, degree of local oxidation of clad, and degree of total oxidation of clad.
- Later versions of regulation do not contain definition of size of LOCA to be considered.
- Current practice makes use of 1982 definition.

10

Leak Before Break Considerations United States

- 1975 -- Improved analytical techniques revealed that under the assumption of a rapid opening double-ended rupture, large asymmetric loads were calculated which would require extensive plant modifications.
- After extensive study by industry and NRC, in 1984 NRC issued Safety Evaluation finding:
  - ... the probability of a full double-ended rupture ... is vanishingly small.
  - ... safety margins indicate that the potential for failure is low enough so that full double-ended breaks need not be postulated as a design basis for defining structural loads.
- Summary of recent history and regulatory actions in NUREG/CR-6765.

11

Leak Before Break Considerations Canada

- Regulations restrict LBB to pressure vessels with specific exclusion of reactor headers.
- Licensee developed and regulator accepted an LBB case to support removal of pipe whip restraints at one plant.

12

Leak Before Break Considerations Russia

- LBB accepted as the technically justified approach to not considering dynamic impacts provided:
  - The analyzed components and piping are adequately designed and have acceptable operational history.
  - Leak monitoring systems have been designed, tested and meet current requirements
  - Structural tests have demonstrated the nature of the failure.

13

Summary

- Design bases have evolved with time in each country, reflecting lessons from research and operational experience.
- LBB credit has been given to some extent in each country, primarily in designing against dynamic loads from a rapid opening break, based on fracture mechanics and leak detection.
- There are concerns that over-emphasis on an initiator with an exceedingly low frequency of occurrence may divert the designer from an optimal safety design.

14

Historical Perspective on Design Basis Development in the United States

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

Dr. A. Murphy (Consultant, USA)

Early Developments

The regulatory process in the United States was developed shortly after the Congress passed the Atomic Energy Act of 1954. In 1959, the ACRS recommended to the Commission that the AEC study the need to make more articulate and definitive the safety standards which were applied to nuclear reactors in the regulatory process. The Chairman of the AEC, in turn, established an ad-hoc committee made up of members from the government, the national laboratories in the U. S., and academia, chaired by Dr. Clifford K. Beck. This committee in their report made several recommendations, which shaped the direction and form of the regulatory process over the next several years. The more important of these are the following:

- Rules should be established by which sites would be considered acceptable for locations of reactors.
- The AEC should not attempt to standardize the technical design and construction specifications and procedures for reactors or for the various components of reactors
- There should be a continuing effort on the collection and organization of safety guides, or state-of-the-art practices; and a systematic tabulation of safety performance objectives for reactors. These should be made available to the industry, but not as regulations.
- A guide should be developed on the preparation of hazards summary reports, the term then used for what is now called safety analysis reports.

The recommendation on development of siting rules led directly to the development of the AEC's siting criteria contained in 10 CFR Part 100 issued in 1962 and of the accompanying technical guidance, TID-14844, Calculation of Distance Factors for Power and test reactor Sites. The nature of these documents was influenced not only by the committee report , but also by the separate comments of committee member Dr Theos J. Thompson.

The remaining recommendations indicated emphasis should be on developing high-level performance objectives coupled with development of safety guides. The safety guides would not be considered as regulatory requirements, but as acceptable approaches which could be modified. The report also recommended preparation of an overall guidance document to assist the public and industry to understand how the regulatory Staff of the Atomic Energy Commission performed its function.

AEC Licensing Guide

In August 1962, the Atomic Energy Commission's Division of Licensing and Regulation issued for public comments the "AEC Licensing Guide, Purpose, Organization, and Contents of Hazards Summary Reports for Power Reactors. This document gathered in one place the applicable regulations in existence at the time (Parts 20, 50, 100, and 115 title 10 of the Code of Federal Regulations), and provided the requisite guidance on Hazards

Summary Reports called for by the Beck Committee. It made clear that *"ritualistic adherence to the suggested format and coverage should not be substituted in any case for systematic analysis of individual safety aspects peculiar to a given plant"*. This document, supplemented by decisions made in individual licensing cases, became the basis for much of the regulatory activity by both the Regulatory Staff and the industry until the development of Safety Guides (now more accurately termed Regulatory Guides) in 1970.

The AEC Licensing Guide identified items, which must be discussed at a high level in keeping with the recommendation of the committee chaired by C. K. Beck to avoid prescriptive requirements at this stage of development of nuclear technology. An example will make this clear:

> *... the Hazards Summary Report should provide information appropriate to a determination of the adequacy of auxiliary systems to provide those required functions necessary to maintain the facility in a safe condition during normal operation and in the event of accident.*
>
> *... The following guides are ... illustrative of the types of information that should be considered:*
>
> *(3) Emergency core water injection system, in sufficient detail to permit an evaluation of the adequacy and limitations of the system to perform its intended function in the event of a loss of coolant accident.*

The remainder of the ECCS system is discussed in similar depth.

In contrast, considering the situation in 1962 when power levels and power densities were significantly lower than today, the AEC Licensing Guide requested substantial information about containment design and layout. Here, as a basis of evaluating containment design the guide requested:

> *Sources and amounts of energy... as the result of credible ruptures of various sized pipes in the primary coolant system ... and a comparison of the effects ... with those which would result from a release due to a circumferential rupture of the largest pipe in the primary coolant system with both ends of this pipe, at the point of rupture, free and open.*

The accident analysis section of the AEC Licensing Guide has more definitive information needs identified. For example:

> *The types of accidents considered should result from a systematic analysis of the facility and should include, for example:*
>
> *(3) Loss of coolant accidents resulting from rupture of a pipe or failure of pressure vessel, pressurizer, rupture disc, valve or pump.*

This guidance was somewhat modified by the concept of the Maximum Credible Accident. Here the Licensing Guide requested the applicant include specification of an accident, or type of accident, considered to be capable of causing more serious consequences than any other *considered credible* (emphasis added.).

It is interesting to note, that as written, the concept of the double-ended large LOCA was primarily a tool used for analyzing containment performance in this guide. This led to early designs that focused on containment performance and on meeting the dose requirements of the siting regulations. Examples here would include the containment spray system at one plant employing a steel sphere for a containment where the sprays were on the outside of the

containment, and the several sub-atmospheric containments, where the plant was maintained below atmospheric pressure so that leakage was calculated to cease within a short period of time of the occurrence of a loss of coolant accident. Because of the emphasis on containment performance, many of the early plants (which were of low power levels and low power densities) did not require operation of emergency core cooling systems if off-site electrical power were to be lost as a part of the plant design philosophy.

Advent of Higher Power Facilities

With the advent of higher power reactors, the importance of the emergency core cooling function became obvious. The basic requirements evolved with time and were expressed primarily in the conditions of licenses that had been litigated before AEC hearing boards. For example, as initially proposed by applicants, not all reactors being proposed for licenses or construction permits could operate the ECCS systems in the absence of off-site power. One three-loop PWR employed the steam generator main feed pumps as ECCS pumps. They could deliver a large quantity of water at pressures approaching 70 atmospheres, but could not accommodate a smaller break where the pressure remained high. They also required the presence of off-site power. Late in the design, a single pump capable of handling a small break while operating on emergency ac power was added.

As case law was developed through the hearing process it became standard practice to insert a condition into the license similar to the following from the St. Lucie Operating License in 1970:

The proposed facility has separate systems designed to provide adequate core cooling and pressure reduction within the containment structure even if a loss-of-coolant accident should occur. For immediate short-term cooling, an emergency core cooling system (ECCS) will inject cool borated water into each of the primary coolant loops and directly into the reactor vessel, thereby limiting energy and fission product releases into the containment. For cooling containment air and to reduce containment vessel internal pressure in the event of an accident, the design includes a spray system for delivering cool borated water into the containment atmosphere, and also a containment fan cooling system with four cooling units and a centrifugal exhaust fan. Either of these systems is designed to meet post accident containment cooling requirements.

Detailed bases for the staff conclusion were also included in the AEC staff's safety analysis report such as the following from Oyster Creek in 1968:

The ECCS is provided to mitigate the consequences of loss-of-coolant accident resulting from any size rupture of the primary system piping or equipment. The break spectrum considered included breaks equivalent to that resulting from pump and valve seals leakage as well as double-ended pipe failures. The largest rupture considered during our evaluation was the double-ended rupture of a 26-inch recirculation line which is equivalent to a break area of 6.22 ft2.

The applicant stated that the Oyster Creek ECCS design criterion was that no clad melt would result for any postulated primary system rupture up to and including the double-ended rupture of a recirculation pipe. We did not accept this as the sole criterion because in our view the peak fuel rod cladding temperature should be limited to a temperature such that reasonable assurance is provided that the ECCS would terminate the temperature transient and assure an intact core geometry for effective long-term cooling. Based on our review of the available data in this regard, we concluded that peak fuel rod cladding temperatures should not exceed about 2000F. Furthermore, the functional aspects of the core spray cooling are sufficiently well determined by tests and analysis to give reasonable assurance of its efficacy when clad temperatures are held to less than 2000F. The results of the applicant's analysis

indicate that the maximum predicted temperatures for the entire spectrum of break sizes and locations that could occur in the design bases accidents do not exceed 2000F. ...Consequently, we conclude that there is reasonable assurance that the core spray system would be effective in the unlikely event of a loss-of-coolant accident..

General Design Criteria

The early committee chaired by C. K. Beck recommended a systematic tabulation of safety performance objectives for reactors. The Regulatory Staff of the former Atomic Energy Committee began developing a set of general design criteria in response to that recommendation. A draft version of General Design Criteria for Nuclear Power Plant Construction Permits was circulated for comment in 1967 and received extensive public comment. A final version of the General Design Criteria for Nuclear Power Plants was issued in 1971. For the purposes of this paper, the 1967 draft is important, since it reflects the regulatory philosophy that was being applied in case-by-case licensing actions, even though it had not yet been codified into the formal regulations of the Atomic Energy Commission and was initially deemed applicable to construction permits only. The 1967 draft was characterized by the Commission as " ... interim guidance until such time as the Commission takes further action on them."

The following are taken from the 1967 draft:

Criterion 9 Reactor Coolant System Boundary (Category A)
The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

Criterion 10 Containment
Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

Criterion 37 Engineered Safety Features Basis for Design
... As a minimum, each engineered safety feature shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

Criterion 44 Emergency Core Cooling Systems Capability
... Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe.

Other proposed General Design Criteria in this draft of interest to this meeting include Criterion 33, reactor Coolant Pressure Boundary Capability; Criterion 34, Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention; and Criterion 35, Reactor Coolant Pressure Boundary Brittle Fracture Prevention.

In this draft version of the General Design Criterion note how they comply with the early guidance that they provide general performance objectives, and not standards.

The General Design Criteria for Nuclear Power Plants were codified as Appendix A to Part 50 of the AEC's regulations in 1971. Note they no longer applied only to construction

permits. This first approved version of the General Design Criteria to be formally incorporated in the regulations acknowledged that it was still incomplete *because "... specific requirements had not yet been sufficiently developed and uniformly applied in the licensing process to warrant their inclusion ..."* These matters included

> *Consideration of the type, size, and orientation of possible breaks in the components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss of coolant accidents.*

However, Criterion 35 – Emergency Core Cooling did require

> *The system safety function shall be to transfer heat from the reactor core following any loss of coolant accident at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.*

Similar language is used in Criterion 50, Containment Design Basis, but this criterion also requires provision of sufficient margin *considering "...the limited experience and experimental data available for defining accident phenomena and containment response, and ... the conservatism of the calculational model and input parameters."*

ECCS and Containment

In the mid-1960s, concerns were raised regarding the ability of the containment to maintain its function in the event of a loss of coolant accident. The Director of Regulation formed a special committee, headed by William Ergen of Oak Ridge to investigate the situation. The Ergen Committee stressed the absolute importance of prevention and mitigation of loss of coolant accidents through what are now called engineered safety features which had already begun to be added to the design of plants. An extensive experimental and analytical research program was begun to investigate ECCS performance. Some scientists perceived that inadequate attention was being given to research programs to determine the effectiveness of systems to mitigate the consequences of a loss of coolant accident, if it were to start. The Atomic Energy Commission convened a public hearing to explore the safety question. Ultimately, after extensive public hearings, in 1974, the regulations were modified to provide a set of specific requirements for computer codes for ECCS analyses in Appendix K and a new section, 50.46, requiring ECCS meet established standards. This included a definition that loss-of-coolant accidents (LOCAs) are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. The Commission's opinion also promised a safety research program to firm up the basis for acceptable features of the computer codes, and to permit relaxation of the requirements where that became possible*.

Pipe Break Frequency and Leak Before Break

The Reactor Safety System, issued as WASH-1400 in 1975, had estimated the frequency of a large pipe break as 10^{-4} per reactor-year based on limited operational data and inferences drawn from experiential data on the performance of large bore piping in other industries. Also, in 1975, the NRC staff was informed that improved analytical techniques had revealed newly calculated asymmetric loads that result by postulating rapid-opening double-ended ruptures of PWR primary system piping. These asymmetric loads were a consequence of the rapidly-opening break at the most adverse location in the piping system. In June 1976, the NRC staff requested licensees to evaluate their primary systems for these loads. Analyses performed in response to this request indicated that some plants would require extensive

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*. Herbert J. C. Kouts, History On Safety Research Programs and Some Lessons To Be Drawn From It, 26th Water Reactor Safety Information Meeting, October 1998.

modifications in the form of pipe whip restraints if the rapid-opening double-ended break were to remain a design basis postulation.

In addition to the requested analyses, the industry prepared several topical reports to demonstrate that an assumed double-ended rupture was not a credible design basis event for primary system piping. The NRC staff evaluated these reports and other available research results and issued a Safety Evaluation in 1984 (Letter to All Operating PWR Licensees, Construction Permit Holders, and Applicants for Construction Permits from Darrell G. Eisenhut, Director, Division of Licensing, Office of Nuclear Reactor Regulation. The attached safety Evaluation found

> *... the probability of a full double-ended rupture of tough piping in a typical PWR primary coolant system is vanishingly small.* (Note this Safety Evaluation addressed only the pressurized water rectors.)

This safety Evaluation concluded with certain exceptions that

> *... the safety margins indicate that the potential for failure is low enough so that full doubleended breaks need not be postulated as a design basis for defining structural loads.*

The overall conclusion was that asymmetric blowdown loads resulting from double-ended pipe breaks in main coolant loop piping need not be considered provided leak detection systems were sufficient to provide adequate margin to detect the leakage from a postulated circumferential throughwall flaw before unstable crack extension occurred.

This finding had a significant effect on design and on operational practices. With appropriate analyses, it permitted removal of pipe whip restraints and jet impingement devices. This, in turn, had a very positive effect on the radiation exposure to plant personnel since equipment became easier to access for inspection or maintenance. This letter also indicated that *the "staff intends to proceed with rulemaking changes to GDC-4 to permit the use of fracture mechanics to justify not postulating pipe ruptures."* for defining structural loads. This criterion has since been modified and now states:

However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

The value-impact analysis which accompanied the safety evaluation estimated an overall savings of 11,000 person-rem by eliminating the need to install, inspect and maintain the pipe restraints and making it easier to inspect piping and components nearby.

There is a comprehensive summary of more recent research and regulatory activity devoted to leak before break considerations in NUREG/CR-6765, Development of Technical Basis for Leak-Before-Break Evaluation Procedures.

Summary

In the early days of reactor safety design and regulation in the United States, the concept of the double-ended break in the largest pipe in the reactor coolant system was applied primarily to set criteria for the design of the reactor containment and in performing consequence analysis for siting considerations. On a case-by-case basis this concept was employed as a requirement for ECCS design as power levels and power densities increased. It was codified into the regulations in the US in 1974. Following extensive analyses and research, it was found that it was not necessary to consider a rapid-opening double-ended break in structural design in 1984. This was based on the ability to detect throughwall flaws (1 gpm in four hours) before they would lead to large pipe breaks.

Historic Perspective on Design Basis Development in the United States

Joseph A. Murphy
Consultant

1

Early Developments

- 1959 – ACRS recommended AEC study to make safety standards more articulate and definitive.
- In response, AEC formed committee chaired by C. K. Beck which recommends the following:

2

Beck Committee Recommendations

- Siting rules should be established.
- AEC should not attempt to standardize technical design and construction specifications and procedures.
- Systematic development of safety performance objectives, and development of safety guides on state-of-the-art practices.
- Develop guide on content of "Hazard Summary Reports.

3

AEC Response

- Siting regulations issued in 1962 (10CFR100), with companion technical guidance on calculation of distance factors for siting.
- Also, AEC Licensing Guide issued.
  - Guide at high level, e.g., ECCS should be described in sufficient detail to permit evaluation of adequacy and limitations in the event of LOCA.

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AEC Licensing Guide (cont.)

- Recommends containment design be based on circumferential rupture of the largest pipe in the primary coolant system with both ends of the pipe … free and open.
- Emphasis is on the use of double-ended large LOCA as tool for analyzing containment performance.

5

General Design Criteria

- Draft issued in 1967 and finalized in 1971
  - 1967 draft reflected "case law."
  - Required ESFs to be designed to cope with any size break up to and including circumferential rupture of any pipe.
- Issued as Appendix A to 10 CFR50 in 1971
  - Final version does NOT specify break size to be considered.

6

1971 General Design Criteria

- "...specific requirements had not yet been sufficiently developed and uniformly applied in the licensing process to warrant their inclusion
  - Consideration of the type, size, and orientation of possible breaks in the ... reactor coolant pressure boundary.

7

ECCS and Containment

- In mid 1960s, concerns raised regarding ability to maintain containment function in a LOCA, and ability of ECCS to cool core.
- Ergen Committee stressed absolute importance of prevention and mitigation of LOCAs.
- Extensive experimental and analytical research program begun.
- Comprehensive public hearing held.

8

Conclusions reached by Atomic Energy Commission

- Regulations modified to provide a set of specific requirements for computer codes.
- Formalized definition of large break LOCA into regulations.
- Promised continuation of a safety research program to
  - Firm up the basis for acceptable features of the computer codes.
  - Permit relaxation of requirements where that became possible.

9

Early Leak Before Break Considerations

- 1975 – Industry informed NRC staff that potential existed for large asymmetric loads in rapid opening large break LOCA.
- Substantial analysis by industry and NRC
- Safety Evaluation issued in 1984 found;
  - ... probability of a full double-ended rupture of tough piping in a typical PWR primary coolant system is vanishingly small ...
  - ...potential for failure is low enough so that full doubleended breaks need not be postulated as a design basis for defining structural loads.

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Historical Perspective on Design Basis Development for CANDU reactors

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

Background

The nuclear power program in Canada evolved out of research and development efforts to provide an indigenous reactor design based on natural uranium fuelled and heavy water moderated reactor core. The prototype reactors in Ontario, the Nuclear Power Demonstration (NPD) reactor near Chalk River Laboratories and the Douglas Point reactor on the shores of Lake Huron were the predecessors to the first commercial nuclear power generating facility, the four unit Pickering A station outside Toronto, Ontario. The first unit of this multi-unit station came into service in 1972.

The nuclear safety philosophy and the regulatory licensing process in Canada developed relatively independently of other jurisdictions, although it did benefit from developments in these jurisdictions. In part, this independence was driven by the unique aspects of the CANDU design.

The safety philosophy for CANDU reactors was very strongly influenced by lessons learned from the loss-of-regulation accident that occurred in 1952 in the NRX (National Research Experimental) research reactor at Chalk River Nuclear Laboratories. Out of the NRX accident grew the requirements for functional and physical separation of special safety systems from process systems, the requirement for fast-acting shutdown systems, the requirements for demonstrating high unavailability and testing of passive safety systems, and the incorporation of an elementary risk-based licensing framework. All of these requirements were encapsulated at a high level in a licensing requirement referred to as the Siting Guide, which, after nearly 50 years of use, remains in effect to the current time. The reasoning underlying the development of the Siting Guide is articulated below.

Over the years modifications to the Siting Guide have been proposed, driven in large measure by perceived deficiencies and limited, in equal measure, by the simplicity and relative effectiveness of the licensing framework.

In the period from the late 1960's to the current time the large break Loss of Coolant Accident (LOCA) has played a major role in CANDU licensing issues and has been transformed from an accident that provides challenges to a design to an accident that imposes significant limitations on the safe operating envelope. The role of this accident is at the early stages of being reconsidered in Canada. The history of this evolution will be summarized in this paper and the factors influencing the desire to redefine the role of large break LOCA events will be described.

The Siting Guide

At the time of licensing Ontario Hydro's Pickering A station, staff of the Atomic Energy Control Board, the nuclear regulator, developed a basic set of high level safety criteria within an elementary risk-based framework (Reference 1). The safety criteria included dose limits applicable to accident conditions. These dose limits became known as the Siting Guide and became de-facto top-level licensing requirements. Central to the framework were three guiding concepts.

First, the "*safety systems shall be physically and functionally separate from the process systems and from each other.*" The safety systems included the reactor shutdown system (SDS), the emergency core cooling system (ECCS) and containment. The process systems included all systems necessary for control of the plant during normal power operation and shutdown conditions, such as the reactor regulating system, boiler pressure and feedwater control, shutdown cooling, moderator and end-shield cooling, service water, eletrical and instrument air, etc. The need for separation between the special safety systems and the process systems was driven directly from lessons learned from the accident in the NRX research reactor. Not specifically articulated was another requirement for the independent shutdown system to be fast acting – also a lesson learned from the NRX accident.

Second, the process systems should be of high quality to limit the frequency of failures that could lead to accidents and the special safety systems should be highly reliable such that their unavailability is a low probability condition. Specifically, the principles were articulated as follows.

> " *The quality and nature of the process systems essential to the reactor shall be such that the total of all serious process failures shall not exceed 1 per 3 years. A serious process failure is one that in the absence of protective action would lead to serious fuel failures.*"

And,

> "*Each safety system shall be readily testable, as a system, and shall be tested at a frequency to demonstrate that its (time) unreliability is less than* 10^{-3}."

At the time it was felt that there was insufficient operating experience with the design and components of the special safety systems in commercial CANDU power reactors. The requirement for demonstrating high system reliability through on-line testing was developed to mitigate against this lack of operating experience.

Third, the concept of risk was introduced in an elementary manner by requiring lower dose consequences for more probable failures (single failure of a process system) and allowing higher dose consequences for less probable events (dual failures involving failure of a process system and coincident unavailability of a special safety system). Dose consequences were specified for an individual and for the population at risk, according to:

> "*The effectiveness of the safety systems shall be such that for any serious process failure the exposure of any individual of the population shall not exceed 500 mrem and of the population at risk,* 10^4 *man-rem.*"

> "*For any postulated combination of a (single) process failure and failure of a safety system, the predicted dose to any individual shall not exceed (I) 25 rem, whole body (ii) 250 rem, thyroid, and to the population,* 10^6 *man-rem.*"

Additionally, in computing the doses if was required that conservative weather conditions equivalent to Pasquill category F be used if adequate local meteorological data is not available.

These requirements were summarized concisely in a table that formed the basis of the siting guide licensing requirements, which is presented as Table 1.

TABLE 1

OPERATING DOSE LIMITS AND REFERENCE DOSE LIMITS FOR ACCIDENT CONDITIONS

(THE SITING GUIDE REQUIREMENTS)

From Reference 1

Situation	Assumed maximum frequency	Meteorolgy to be used in calculation	Maximum individual dose limits	Maximum total population dose limits
Normal Operation		Weighted according to effect, i.e. frequency times dose for unit release	0.5 rem/yr whole body 3 rem/yr to thyroid	10^4 man-rem/yr 10^4 thyroid rem/yr
Serious Process Equipment Failure	1 per 3 years	Either worst weather existing at most 10% of the time or Pasquill F condition if local data incomplete	0.5 rem/yr whole body 3 rem/yr to thyroid	10^4 man-rem/yr 10^4 thyroid rem/yr
Process Equipment Failure plus Failure of any Special Safety System	1 per $3x10^{-3}$ years	Either worst weather existing at most 10% of the time or Pasquill F condition if local data incomplete	25 rem whole body 250 rem to thyroid	10^6 man-rem 10^6thyroid rem

Implications of the Siting Guide

One unique aspect of the Siting Guide was the concept of dual failures involving a process failure and failure of one of the special safety systems. Since there was no overall lower frequency limit imposed on the process failure event frequency, this forced analysis of low probability - high consequence events, including a large break LOCA with Loss (failure) of the Emergency Core Cooling system (LOECC). This event, when carried to the extreme and with conservative treatment in the safety anlysis, effectively becomes a severe fuel damage event, thereby placing some core damage events directly in the design basis.

A second consequence was that logical application of the dual failure resulted in process failures with failure to shutdown were also to be included in the design basis for reactors with only one independent shutdown system. Again, an event with the potential for significant core damage was included in the design basis. However, this was a difficult analytical issue since the state of knowledge regarding the consequences of such events was limited and any analysis was considered speculative. The inability to satisfactorily resolve this issue led to development of a requirement for new designs to be fitted with a second independent shutdown system. The requirement was initially enforced in the course of licensing the next nuclear station, Ontario Hydro's Bruce A station on the shores of Lake Huron.

Given the importance of the special safety systems in mitigating the progression of events in CANDU accidents, the AECB developed a formal set of requirements for these systems in Regulatory Documents that are applicable to reactors licensed for construction after either 1977 or 1981. The first Regulatory Document R-10, a Regulatory Policy Statement entitled *The Use of Two Shutdown Systems in Reactors* stated that, amongst other requirements:

> "*All nuclear power reactors licensed for construction in Canada after January 1, 1977 shall incorporate two independent protective shutdown systems unless otherwise approved by the Board.*"
>
> And,
>
> "*The protective shutdown systems shall be of diverse designs and each shall be physically and functionally separate from the other, from process systems, and from other special safety systems.*"

The other Regulatory Documents that followed were:

- *R-7, Requirements for Containment Systems for CANDU Nuclear Power Plants,*
- *R-8, Requirements for Shutdown Systems for CANDU Nuclear Power Plants, and*
- *R-9, Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants.*

These documents articulated design requirements, operating requirements for both normal and accident conditions, and testing requirements that covered commissioning, in-service testing and inspection and testing of system availability.

Difficulties with the Siting Guide

Although simple in concept and relatively effective in application, the Siting Guide did not provide adequate guidance in a number of problem areas. For example,

- There was no distinction between single failures occurring at differing frequencies and having different consequences.

- There was no distinction between dual failures occurring at differing frequencies and having different consequences.

- There was no guidance with respect to the frequency of events below which consequences need not be considered (i.e. no cut-off frequency).

- Safety system impairments (failures) were defined without consideration of the fact that systems, such as containment, have a number of separate subsystems, each of which should be separately considered.

These problems contributed to difficulties in addressing issues at the boundary between design basis and beyond design basis events. As mentioned previously, the boundary was further blurred by the fact that application of the dual failure definition, accompanied by conservative safety analysis methods and assumptions, resulted in analyzed consequences of events in the design basis that exhibited many features of a severe accident – severe fuel damage, high temperature deformation of core components and significant production of hydrogen from the zircaloy-steam reaction. All of these events involved large break LOCA as the initiating event.

The AECB proposed a revised framework, defined in the Consultative Document C-6, Requirements for Safety Analysis of CANDU Nuclear Power Plants to address, in part, these problems. Intended to be applied to all nuclear power plants without a construction license as of July1, 1908, C-6 was applied on a trial basis in the licensing of the Darlington nuclear generating station, the last station constructed in Canada. This proposed requirements document extended the elementary risk-based concepts by defining five classes of events with reference dose limits for each class, as shown in Table 2. The classes of events were broadly related to decreasing event frequency with increasing class number. However, the framework was basically deterministic in nature. It was not fully risk-based in that initiating events were pre-specified in the event classes together with combinations of initiating events and failures of special safety systems and safety support systems, irrespective of the specific features of a design which can influence the estimates of event frequency. Finally, there was no guidance regarding an event frequency cut-off.

These issues were subsequently addressed in an updated Consultative Document, C-6 Rev. 1, in which specific requirements for a probabilistic safety assessment were introduced, event frequencies were assigned to the classes as shown in Table 2, and a cut-off frequency (10^{-6} per annum) was specified, below which, with regulator approval, analysis of event consequences was not required.

TABLE 2
REFERENCE DOSE LIMITS SPECIFIED IN CONSULTATIVE DOCUMENT C-6, Rev 0 AND EVENT FREQUENCIES FROM C-6, Rev.1

EVENT CLASS	REFERENCE DOSE LIMITS (C-6, Rev. 0)		TOTAL FREQUENCY (C-6 Rev.1) [per annum]
	WHOLE BODY	THYROID	
1	0.0005 Sv (50 mrem)	0.005 Sv (500 mrem)	$>10^{-2}$
2	0.005 Sv (500mrem)	0.05 Sv (5 rem)	10^{-2} to 10^{-3}
3	0.03 Sv (3 rem)	0.3 Sv (30 rem)	10^{-3} to 10^{-4}
4	0.1 Sv (10 rem)	1.0 Sv (100 rem)	10^{-4} to 10^{-5}
5	0.25 Sv (25 rem)	2.5 Sv (250 rem)	$<10^{-5}$

Leak Before Break Considerations

Notably, the concept of leak-before-break (LBB) was introduced in C-6 Rev. 0 but it was restricted to pressure vessels with a specific exclusion that the reactor headers not be considered as vessels. For heat transport piping and header failures it was required that:

> *"Pipe failure shall consider both circumferential, guillotine, and longitudinal failures at any location in a system.*
>
> *(a) For circumferential pipe failures a discharge area up to and including, twice the cross-sectional area of the pipe shall be analyzed.*
>
> (b) Failures resulting from longitudinal cracks shall also be considered and justification given for the maximum crack size postulated."

The licensee did develop, and the regulator accepted, an LBB case for large diameter (NPS greater than 6 inches) heat transport system piping to support removal of pipewhip restraints from the Darlington design. These considerations notwithstanding, all locations and sizes of postulated breaks were required to be analyzed to demonstrate that the special safety systems were effective in assuring the integrity of design barriers to the release of fission products.

LOCA Issues

Ever since the initial licensing of the four unit Pickering A station in the early 1970's large break LOCA events have been a constant focus of safety analysis. This was driven by a series of discoveries that affected analysis outcomes which, in conjunction with the limitations of the safety analysis methodology of the day, exerted significant pressure on the ability of the licensees to demonstrate that derived safety criteria were met. Not only did this highlight the derived safety criteria, but it also initiated large-scale research and development programs aimed at improving their technical basis and reducing uncertainty in knowledge of governing phenomena. Major large break LOCA discovery issues were associated with identification of critical break behaviour and pressure tube deformation potential[1], fuel-string relocation reactivity effects[2], and uncertainty in the coolant void reactivity[3]. The net result of this focus on large break LOCA was that these events increasingly became limiting with regard to definition of many operating parameters – i.e. definition of the safe operating envelope.

Safety analysis methodology was adapted to respond to these discovery issues by applying progressively more conservative and bounding assumptions and methods to accommodate unresolved uncertainties in phenomena and modeling. The "limit-consequence" approach developed in the late 1970's essentially took no credit for the ECCS functioning in order to bound the uncertainties associated with phase separation and two-phase flow and provided conservative prediction of the consequences of critical breaks. These consequences were clearly in the realm of severe fuel damage, although the analysis did demonstrate that the moderator system provided CANDU reactors with a passive ultimate

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1 A critical break is one in which fuel channels in one pass of a loop experience low flows for a period of time until the pump head degrades. In this interval some pressure tubes can heat up and deform through creep strain into contact with their calandria tube – referred to as pressure tube ballooning.

2 Coherent relocation of the fuel in all channels in a pass due to reverse hydraulic forces associated with the rupture. This can give rise to a bounded positive or negative reactivity insertion depending upon the direction of fuelling in the channels.

3 Coolant voiding in fuel channels gives rise to a positive reactivity feedback which induces a power increase that is terminated by fast-acting shutdown systems.

heatsink capability. With the development of two-fluid thermalhydraulic codes and best effort analysis of ECCS performance, some benefit was gained for the LOCA events but these gains were eroded as further limiting assumptions became necessary to accommodate the changes in reactivity associated with fuel-string relocation and coolant void reactivity uncertainty. However, no gains were obtained for the dual failure LOCA+LOECCS event in the design basis. Further effort to improve safety analysis methods has occurred in the past five years involving the development of best estimate methods with integrated quantification of uncertainties.

Redefinition of LOCA Events

The initial efforts to achieve redefinition have addressed the LOCA+LOECCS event which, from both a frequency perspective and a consequence perspective, is of the nature of a severe accident. Because of the dual failure definition in the Siting Guide and the bounding analysis methodology applied to address phenomena associated with severe accident behaviour, this accident results in a design basis event with the potential for high fission product release and hydrogen production source terms. This leads, in turn, to problems in such areas as equipment qualification where it is essentially impossible to execute a rigorous program to qualify equipment to severe accident conditions. For these reasons, a position has been advanced to consider LOCA with more probable impairments of ECCS function (as opposed to stylized gross failures of the ECCS system) as constituting design basis events. The more extreme events would be part of the beyond design basis set and would be handled through severe accident management. This position is currently under discussion in Canada.

More fundamentally, the issue of large LOCA as a design basis initiating event is under reexamination by the industry in Canada. Under the "propose-dispose" regulatory environment the industry can propose changes which, through formal communication, then initiates discussion with the regulator. Some of the necessary prerequisites include moving toward a risk-informed regulatory regime and successfully aligning deterministic defence-in-depth concepts within a probabilistic framework. Ultimately, this could be achieved by managing the risk associated with low frequency, high potential consequence accidents by means of the controlling the event frequency component of the risk equation, while defining a reference acceptable consequence (or range of consequences). One means of controlling the LOCA frequency would be through application of LBB and ensuring that the supporting elements involving fracture mechanics evaluation, leak detection capability and inspections are in place. It is believed that CANDU designs are well positioned to support such an approach because:

a) fracture mechanics capability has been established in the industry,
b) heavy water management has resulted in designs with good leakage detection capability at very low leak-rate thresholds, and
c) in-service inspection programs have been part of the basic CANDU requirements from the very start of the power program – the only issue is establishing an agreed level of inspection to support LBB.

Reference

1. D.G.Hurst and F.C.Boyd, "*Reactor Licensing and Safety Requirements*", Paper No. 72-CNA-102, 12th Annual CAN Conference of the Canadian Nuclear Association, June 11-14, 1972, Ottawa.

Historical Perspective on Design Basis Development for WWER reactors

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Technical basis and its implications",
Zurich, Switzerland, June 23-24, 2003

1. Introduction

Large-scale implementation of WWER technology into nuclear power refers to the early 1970s, when the first power units with WWER-440 were put into operation. In 1971 and 1972, WWER-440 (V-179) were put into operation at NV NPP Units 3&4 , in 1973 and 1974 operation of WWER-440 (V-230) was started at Kola NPP Units 1&2, in 1974 and 1975 - WWER-440 (V-230) at NPP "Kozloduy" Units 1&2 in Bulgaria. Construction of WWER-440 (V-230) plants in Germany (NPP "Nord" Units 1&2), in Czechoslovakia (NPP "Bohunice" Units 1&2), in Bulgaria (NPP "Kozloduy" Units 3&4), and also NPP with WWER-440 (V-270) in Armenia (2 Units) was being conducted.

The above power units have completed the first stage of development of WWER plants. The safety systems of these power units were designed at the time when no special domestic standards for nuclear safety are available, proceeding from the industrial standards and regulations. Nevertheless, the most important equipment of reactor plants was designed according to the specially developed normative-technical documentation. It allowed to provide a high level of reliability and safety of these plants that was confirmed by their operational experience and by licensing of their operation beyond the design service life with account for the domestic and international standards of nuclear safety being in force nowadays.

The second stage of WWER technology implementation started with the construction of the second phases of plants with WWER-440 (V-213) in Germany (NPP "Nord"), Czechoslovakia (NPP "Bohunice"), Russia (Kola NPP), and also construction of new NPPs with these reactors in Finland, Hungary, Ukraine, Czechoslovakia. This stage includes also construction of the power units with WWER-1000 reactor plants of various models (V-187, V-302, V-338, V-320).

At the second stage, the WWER reactor plants were provided with the safety systems intended to ensure core protection and to meet the norms on radioactivity release under any accident taken into account in the western safety standards and PWR design practice (including break of the largest pipeline of the reactor coolant system).

This paper briefly describes development and current status of the normative requirements and corresponding design bases of WWER reactor plants for loss of coolant accidents (LOCA). Consideration of these accidents as the design ones to the great extent predetermines both the configuration of NPP as a whole, and nomenclature and characteristics of many systems and equipment of the plant.

2. Evolution of LOCA scale in WWER designs

WWER-440 of the first generation

The principle of safety assurance of the first generation WWER-440 (reactor plants V-179, V-230, V-270) was based on the measures directed to prevent the dangerous initiating events, including LOCA. First of all, the measures were taken for quality assurance of design, manufacturing, installation and operation of the equipment. The main equipment of primary circuit (reactor, steam generator, coolant pump, components of reactor coolant circuit) was of high reliability and had significant structural margins, that excluded their possible break with formation of large leak. Therefore, only breaks in the pipelines attached to the primary circuit were considered as the design basis for the first generation WWER-440. All these pipelines were provided with the insertions of 32 mm diameter to ensure scale of the maximum design basis accident with LOCA as a leak from the hole of diameter 32 mm within the reactor coolant pressure boundary. Such accident was included into the design bases for emergency core cooling system and localization system.

The coolant outflowing in this accident from the primary circuit is localized in the confinement designed for pressure 0,1 MPa. When the pressure increases to 0,02 MPa, the spray system is actuated. This system consists of the storage tank for boron solution of 800 m^3 and three pumps with capacity of 400 m^3/h each. With further pressure increase in the confinement, the relief valves from SG boxes open. The first valve has capacity 25 kg/s and it opens at pressure 0,06 MPa, others 8 valves of capacity 180 kg/s each open at pressure 0,08 MPa.

The same scale of maximum design LOCA - leak of 32 mm diameter - is included into the design basis for emergency core cooling system of the first generation WWER-440 (safety boron injection system). This system makes use of the same storage tank for boron solution of 800 m^3. The system consists of two groups of pumps (in all 6 pumps) of total capacity equal to 300 m^3/h at primary pressure up to 13 MPa. The system is started when the pressurizer level decreases and by some other signals.

WWER-440 of the second generation

In 1974, the designs of 24 power units with WWER-440/V-213 reactor plant were developed for Finland, USSR, DDR, Hungary and Czechoslovakia. In these designs the technical measures are stipulated to ensure NPP safety under the accident with guillotine break of the main coolant pipeline with 500 mm internal diameter. Such accident was included into the design basis of the emergency core cooling system and localization system. In particular, equipment, pipelines and primary circuit systems of WWER-440/V-213 are arranged in the protective boxes, and condensation of steam generated during the accident is performed by the bubbling-vacuum system (for Loviisa NPP – by the ice condenser).

The accident with guillotine break of pipeline with diameter 500 mm is also included into design bases of other systems and equipment of the WWER-440/V-213 reactor plant (support and restraints on the main equipment and pipelines, core, reactor internals, etc.).

WWER-1000 reactor plants

All designs of WWER-1000 plants account for the accidents related to double-ended guillotine break of the primary pipeline with maximum diameter (850 mm) with free coolant discharge from both ends. With this it is postulated that the accident is accompanied by complete loss of NPP power and coincides with safe shutdown earthquake (SSE).

The accident with guillotine break of the pipeline with diameter 850 mm is also included into the design bases of other systems and equipment of power units with WWER-

1000/V-187, V-302, V-338 reactor plants. In particular, all the static and dynamic loads arising during this accident were considered in development of the core, reactor internals, supports and restraints for the main equipment and pipelines. The designs of standardized power units of WWER-1000/V-320 as a whole follow this concept as well, but the implementation of the elements of approach considered now as "Leak before break" (LBB) concept has started as regards some components of the reactor plant and containment.

3. Regulatory requirements concerning LOCA

Designs of the first-generation WWER-440 were developed with no special domestic safety standards available using the industrial standards and regulations. The most important reactor plant equipment was designed and manufactured according to the specially elaborated normative-technical documentation. The safety concept of these reactors was based on impossibility of large loss of the primary circuit integrity which could result in significant decrease of core heat removal.

In 1970, in connection with the development of NPP design for Finland, the safety requirements for nuclear power plants were formulated for the first time in our country. On the basis of these requirements, the regulatory document "Basic regulations on NPP safety assurance" (OPB-73) was compiled in 1971 and published in 1973. Item 1.2.5 of this document contains the following requirement:

> *Guillotine break of the pipeline being the most dangerous with regard to radiation consequences shall be considered as the maximum accident with loss of circulation circuit integrity, if it is not excluded by the implemented technical measures.*

This requirement was interpreted in the design practice so that break of the primary pipeline with maximum diameter was included into the design bases of all NPP systems and components.

The requirement as regards LOCA in new version of OPB issued in 1982 (OPB-82) was even more defined. Item 4.1.1 of this document runs as follows:

> *Guillotine break of the maximum-diameter pipeline with free double-ended discharge of coolant during reactor operation at nominal power with account for its possible exceeding due to errors and tolerances of I&C shall be considered as the maximum design basis accident with loss of primary circuit integrity.*

The quantitative requirements for emergency cooling systems in terms of maximum permissible temperature (1200^{0}C), local oxidation (18 %) and general oxidation (1 %) of fuel rod claddings were also established for the first time in OPB-82 (item 4.1.3)

The similar requirement is in the USSR State standard *"Nuclear power reactors of vessel type with pressurized water" (GOST 24722-81)* being also put down in the register of Russian State standards. Item 1.5.1 of this document runs:

> *Guillotine double-ended break of the main coolant pipeline in combination with condition of complete loss of NPP power supply shall be considered as the maximum accident in the reactor design. For the reactors to be delivered to seismic-active NPP site the above accident together with SSE shall be considered as the maximum accident.*

The subsequent versions of OPB as the basic safety standard (OPB-88 issued in 1989, and OPB-88/97 valid nowadays) no longer contain the indication on the size of design LOCA.

The below regulatory documents intended specially to define the design bases for NPP localization system and emergency core cooling system do not contain such instructions as well:

Rules for design and operation of the localizing safety systems of nuclear stations (НП-010-98).
Rules for design and operation of the systems for emergency cooling and heat removal from nuclear reactor to ultimate sink (ПНАЭ Г-5-020-90).

Regulatory document PNAE G-01-036-95 intended to specify format and contents of safety analysis report contains the "provisional and minimum" list of design basis initiating events for the accident analysis. In this list, the "break in the main pipelines containing the primary circuit medium" is mentioned but nothing is said about the size and the characteristics of such break.

Thus, the current documents of RF Gosatomnadzor give no specific instructions concerning the scale of LOCA, which shall be assumed as design basis for the corresponding safety systems. Nevertheless, the current design practice of WWER actually makes use of OPB-82 wording as the design basis to select the characteristics of emergency core cooling system and containment depressurization system. With this, some components of the reactor plant and NPP, including those effecting the functioning of these safety systems, are designed without consideration, for example, of the dynamic loads arising during guillotine break of the primary pipeline with maximum diameter. It creates some formal discrepancy in the design because many regulatory documents require *to take into account all the effects arising during the design basis accidents.*

4. Role of LBB concept

Applicability of LBB concept for the existing and future NPPs is defined in the following documents approved by the national safety authority:

Technical requirements for application of "Leak before break" concept for the pipelines of operating NPPs (1995).
Guide on application of "Leak before break" safety concept to the pipelines of nuclear power installations (Р-ТПР-01-99, 1999).

It is stated in document Р-ТПР-01-99 that comprehensive application of guillotine break of the pipeline with maximum diameter as the design basis LOCA results in elaboration of special measures on protection of all components and structures important to safety against the corresponding emergency loads. To satisfy these purposes, the applicable protective measures include physical separation of the equipment, supports and restraints for equipment and pipelines, protective shields, etc. These measures make NPP construction and operation more expensive and can exert negative influence upon safety because they significantly complicate the operational inspections of equipment and pipelines.

The methodology of LBB concept is accepted as the technically justified approach to not consider the dynamic impacts from postulated double-ended guillotine rupture in the pipeline systems with high stored energy. The LBB concept can be used both for newly designed plants and for operating ones if the following conditions are met

- The analyzed components and pipelines have been defined potentially suitable for LBB concept application from the viewpoint of design bases, meeting the requirements of the main technical principles and assessment of the gained operational experience;

- The leak monitoring systems have been developed and passed the calibration tests, and subsequent improvements of these systems for the early-generation plants are ensured;

- The sufficient margins obtained using the calculation procedure on substantiation of LBB concept have been demonstrated;

- The structural tests of actual large-scale models or pieces of full-scale tubes have shown viscous character of their failure.

Successful application of the LBB concept giving technical substantiation of impossibility for sudden guillotine failure (or its longitudinal equivalent) in the pipeline systems allows:

1. To exclude specific consideration of the static and dynamic impacts related to large-scale breaks and leaks, for example:
   - jet forces, force of reactions, pipe whipping and impingement, missiles;
   - impacts upon the component supports and upon the civil-engineering structures (for example, pressure differentials on the in-containment walls);
   - shock pressure waves inside the equipment and pipelines.
2. To exclude necessity for development and installation of additional supports, fasteners, pendants, restraints of pipe whipping, protective shields and other massive and expensive devices required to mitigate the consequences of guillotine failure, and thereby to facilitate in-service inspection and maintenance of the pipeline systems.

For example, in application of LBB concept to operating NPP the dynamic impact of coolant to the core elements and reactor internals are to be assumed for the postulated leak size equal to 10% of flow area of the main coolant pipeline. For WWER-1000, it is equivalent to one-ended break of the pipeline with diameter about 270 mm. At the same time, application of LBB concept does not cancel the current requirements for safety systems and does not mean that it is possible to refuse such protection means as the containment, emergency core cooling system, accident localization system, etc.

5. Effect of LOCA scale to ECCS configuration

Calculations and experimental studies show that a stable sub-critical crack growth always precedes to the catastrophic failure of pipelines made of ductile materials and having high resistance to unstable crack growth. It allows either to find out a crack during periodic inspection long before it becomes the through-wall crack, or to detect a leak by the leak monitoring system before the through-wall crack can reach the critical size (this is just the essence of LBB concept). Thus, with full-scale implementation of LBB concept the probability of sudden occurrence of coolant leak becomes the less the more is size of such leak. In particular, this probability already with rather moderate size of leak can appear to be less than the threshold being characteristic of the range for design basis accidents (for example, 10^{-4} per reactor-year according to IAEA recommendations in Safety Report Series No. 23).

This leak size can be further accepted as design basis for the emergency core cooling system, if it will not contradict "letter and spirit" of the valid regulatory documents. With this, the formal contradiction mentioned at the end of section 3 of this paper will be completely eliminated: *all loads accompanying the LOCA can be taken into account in the same manner for all NPP components important to safety*. Guillotine break of the maximum-diameter pipeline can also be included into a list of the postulated initiating events, but now as beyond

design-basis accident because probability of this event is less than the value being the lower threshold for design basis accidents.

Decrease in scale of the design LOCA can essentially influence the ECCS configuration and characteristics required for meeting the acceptance criteria established for LOCA. To demonstrate possible influence, the LOCA calculations were made for the following WWER-1000 scenarios:

- (1) Guillotine break of pipeline at the reactor inlet (diameter 850 mm) for ECCS design configuration and at design approach to the initial and boundary conditions (reference scenario).

- (2) Pipeline leak at the reactor inlet with equivalent diameter 200 mm, ECCS without hydraulic accumulators and low-pressure pumps, at design approach to the initial and boundary conditions (simplified configuration of ECCS).

- (3) Guillotine break of pipeline at the reactor inlet (diameter 850 mm), ECCS without hydraulic accumulators and low-pressure pumps, with realistic initial and boundary conditions (beyond design-basis accident).

In reference scenario, the maximum clad temperature obtained by RELAP5/MOD3.2 code, amounts to about 900°C in the first peak and 1050 °C in the second peak (Fig. 1). About the same values are obtained in the design safety analyses performed in EDO Gidropress using our own computer code TECH-M-97 certified by the national regulatory body.

For 200-mm leak with simplified configuration of ECCS (only high-pressure pumps are available) the code RELAP5/MOD3.2 shows a short-term increase of the maximum clad temperature to 780°C (Fig. 2) that is also in agreement with the results obtained for this scenario using the computer code TECH-M-97. It means, that if 200-mm leak is accepted as maximum design LOCA, the necessary configuration of ECCS becomes much simpler with corresponding reduction of expenses for construction and operation of the power unit.

For scenario (3), it is obtained by the code RELAP5/MOD3.2 that maximum clad temperature exceeds the value permissible for design LOCA (Fig. 3). However, period of time during which such exceeding takes place (about 200 s) and the value of exceeding (about 200°C) show that the coolable configuration of the core will be kept and no core melting will occur (although it is formally allowed for beyond design-basis accidents). Thus, considered simplification of ECCS configuration will not result in increase of total frequency of severe core damage. Therefore one should not expect the violation of the relevant criterion established in OPB-88/97 for total probability of severe beyond design-basis accidents (less than 10^{-5} per reactor-year).

Comparative efficiency of the design and simplified configurations of ECCS for guillotine break of the cold pipeline with diameter 850 mm can be also shown in terms of water inventory in the reactor pressure chamber. This parameter determines to a great extent the core cooling conditions soon after the blowdown stage. The Fig. 4 shows variation of water inventory in the reactor pressure chamber obtained by the TECH-M-97 code. This result demonstrates that at realistic consideration the effect of ECCS configuration disappears already in 3 min after accident initiation. Considering the TMI-2 accident experience, it is difficult to expect that for so short interval of time the core configuration could become so uncoolable that the process of its degradation would occur despite the water supply into the reactor in amount quite sufficient for long-term core cooling.

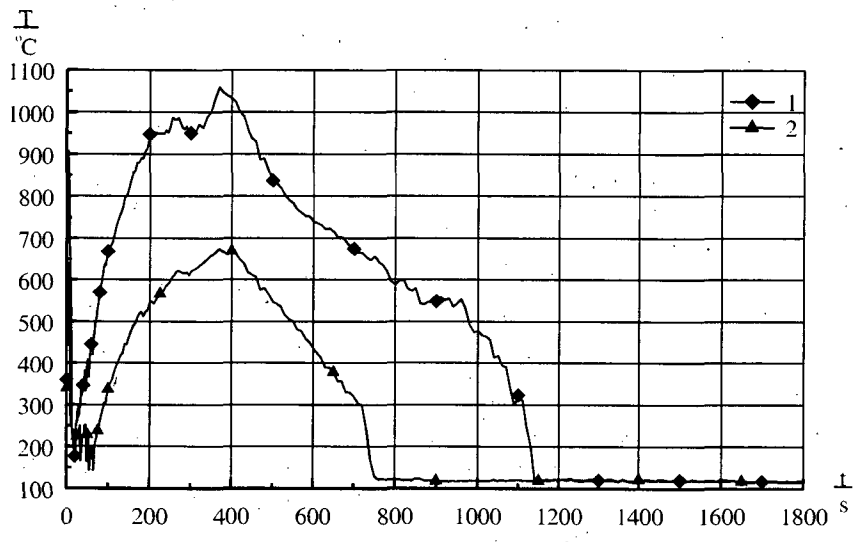
Of course, the above data should not be considered as the quantitative characteristics of LOCA consequences at various ECCS configurations. These data show only a degree of possible influence of scale of the maximum design LOCA upon the required number and capacity of the ECCS means. Analysis of a wide spectrum of postulated accidents with account for all functions to be fulfilled by the ECCS components and considering efficiency

of means stipulated by LBB concept is required to determine the minimum required ECCS configuration and characteristics in each specific design.

6. Conclusion

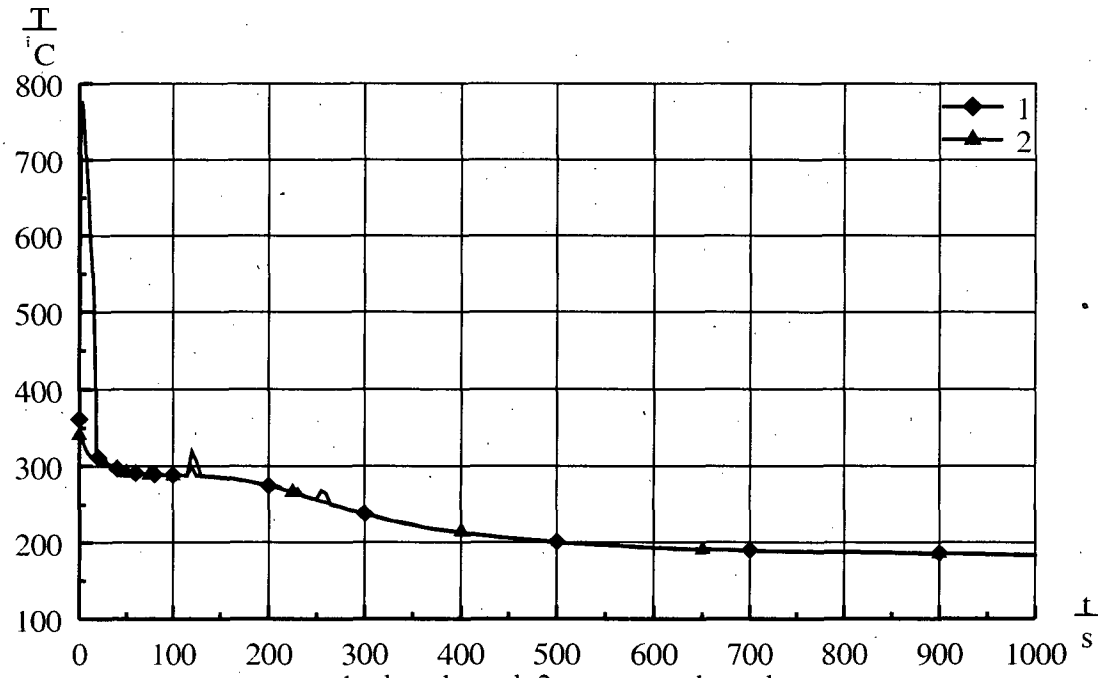
The principle of safety assurance of the first-generation WWER-440 is based on prevention of the dangerous initiating events, including LOCA. The main equipment of primary circuit had high reliability and significant structural margins that excluded possibility for their break with formation of the large leak. Thus, only the break of a pipeline connected to the primary circuit through insertion of diameter 32 mm was considered as the design LOCA. The second-generation WWER-440 and the WWER-1000 designs include a complete scope of technical measures necessary for safety assurance in case of guillotine break of the main coolant pipeline.

In 1973, the first regulatory document (OPB-73) containing the requirement on consideration of guillotine break of the primary pipeline was published. Since then this requirement has been taken into account in the WWER design practice, though the current documents of RF Gosatomnadzor do not prescribe the size of the maximum design LOCA. The full-scale implementation and application of LBB concept would allow, in principle, to reduce a scale of the design LOCA and thereby to simplify significantly the configuration of ECCS and other plant systems.



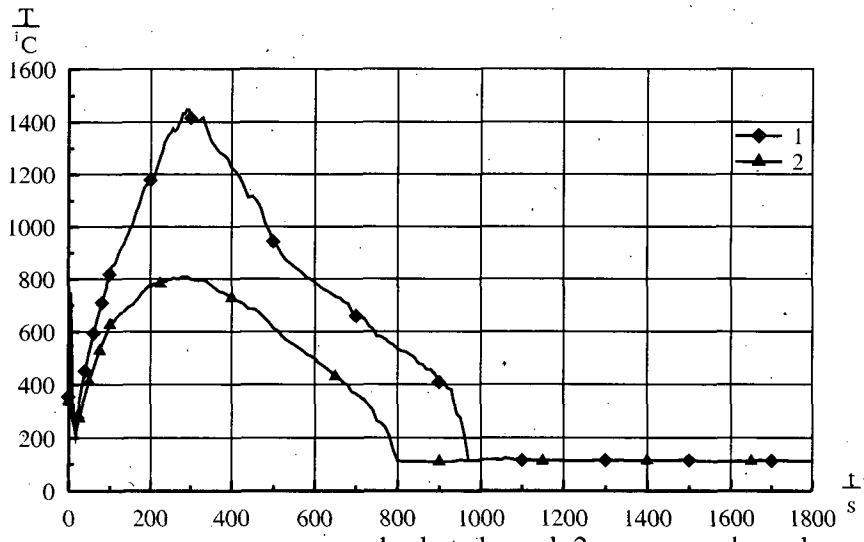
1 – hot channel; 2 – average channel

Fig. 1 Maximum clad temperature (break 850 mm, reference scenario)



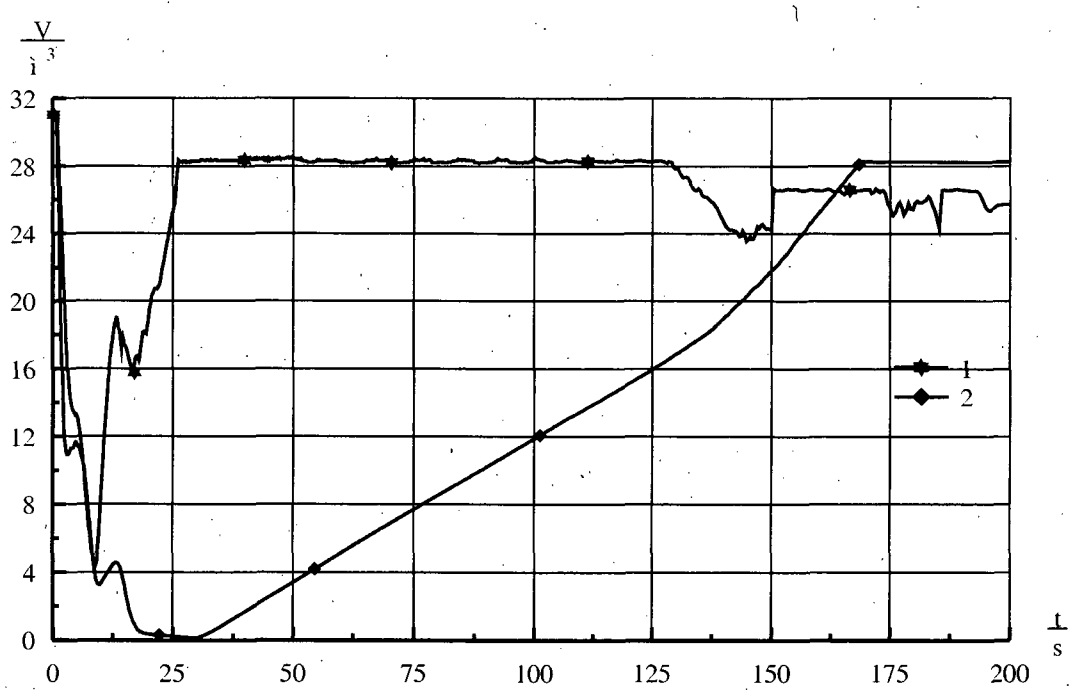
1 – hot channel; 2 – average channel

Fig. 2 Maximum clad temperature
(leak 200 mm, simplified ECCS configuration)



1 – hot channel; 2 – average channel

Fig. 3 Maximum clad temperature
(break 850 mm, beyond design-basis accident)

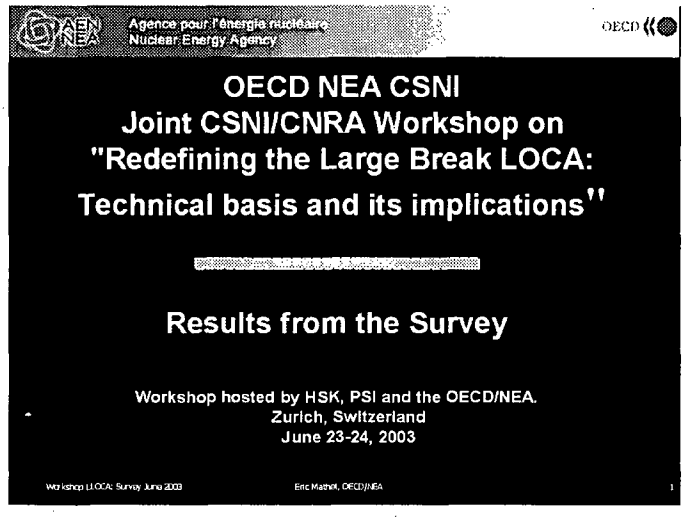


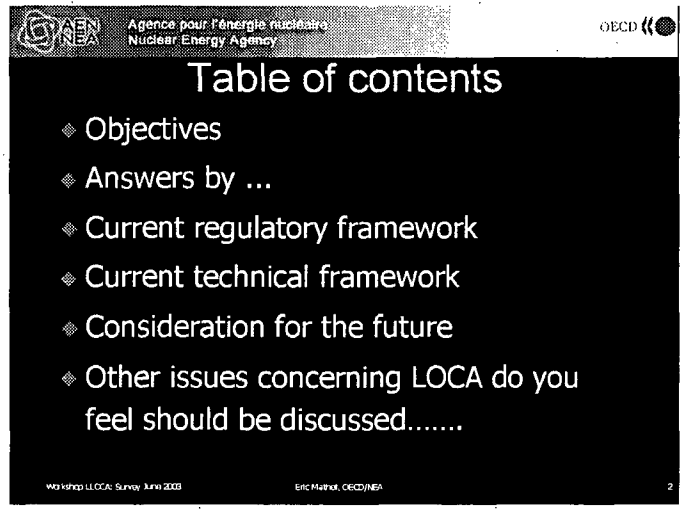
1 – reference scenario; 2 - beyond design-basis accident
Fig. 4. Water inventory in reactor pressure chamber (break 850 mm)

Presentation by the Secretariat on results of the Survey

Mr. E. Mathet (OECD/NEA)

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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Objectives of the survey:

- to provide material for discussion
- to clearly summarize current national regulations
- to understand rationales and incentives for changing or not
- to list technical issues to be resolved before implementing a new regulation, if any
- to focus the workshop on technical issues

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Answers by

BELGIUM	AVN
CANADA	CNSC
CZECH REPUBLIC	NRI
	SUJB
FINLAND	STUK
FRANCE	IRSN
GERMANY	GRS
JAPAN	NUPEC-JAERI
MEXICO	CNSNS
SLOVAK REPUBLIC	Nuclear Regulatory Authority
	VUJE Trnava, Inc.
SPAIN	CSN
SWITZERLAND	HSK
UNITED KINGDOM	NII
USA	USNRC
	Westinghouse Owners Group (WOG)
RUSSIA (Observer)	EDO Gidropress
	GAN
Slovenia (Observer)	Slovenian Nuclear Safety Administration

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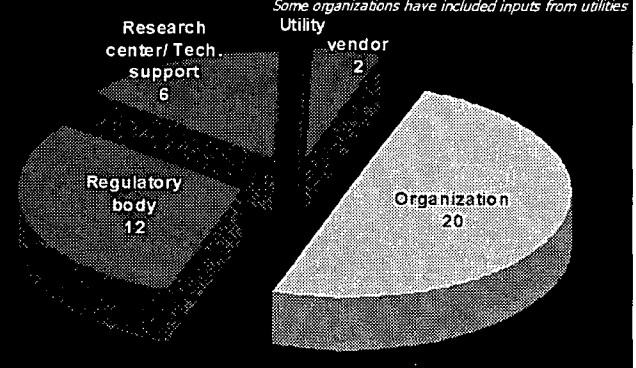
Some organizations have included inputs from utilities



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Current regulatory framework

Largest or limiting break size assumed as design basis (LOCA)

DEGB of the largest pipe in the reactor coolant system (except for earliest versions of VVER)

"Risk" considerations in the current LOCA break size definition

NO
actually implicitly

1 and 2

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Current regulatory framework

Regulatory implications on design, operational procedures, testing, inspection program ....

- *Obviously a broad impact on structural design, testing, qualification of equipments.*
- *Severe accident procedures: no changes*
- *Operating procedures: changes possible*

Is Leak-Before-Break accepted (or being considered) in your regulation ?

- *Yes for a vast majority*
- *ONLY to remove (or not installing) whip restraints, snubbers, modify supports or justify internals behavior*

3 and 4

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Current technical framework

Technical issues currently of concerns for SSC....

- *sump debris generation and sump blockage*
- *fuel behavior*
- *containment leak tightness*
- *.../...*

To be addressed by Researchers ..... to change the current LOCA definition

- *LOCA initiating event frequency*
- *Best estimate analysis methods and uncertainties*
- *Inspection programs*
- *Fracture mechanics and probabilistic fracture mechanics,*
- *Leak detection systems,*
- *Integration of the deterministic, defense in depth principle within a risk-informed framework*

1 and 2

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Consideration for the future

Are you considering changes in your regulation ?

- *Regulators are not considering changes to the LLOCA except the US.*
- *a more balance approach to LLOCA possibly in Japan and Canada in the future.*

What would be the incentives ? ....

- *For both the Regulator and the Industry:*
  - *Focus research, review effort, maintenance in areas of greater risk significance*
- *For the Industry*
  - *Relaxing operating restrictions*
  - *Remove power restrictions*

1 and 2

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Consideration for the future

If you are considering replacing large break LOCA .....

Performance requirements for the emergency cooling systems ..... following a large break LOCAs

No definite answers but several approaches

- *LLOCA as a beyond design basis accident*
- *LLOCA probability considered as low as vessel rupture probability and thus excluded*
- *LLOCA still uses for design (hardware) but Tech. Specs. modified*

Computer codes and models adequate ?

- *Available codes seem to be mature enough*
- *Reexamine applicability range for codes*
- *Structure reliability models*
- *PRA models and probabilistic fracture mechanics*
- *Core reflooding and vessel-corium interaction.*

3a and 3b

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Other issues concerning LOCA do you feel should be discussed.....

- *Operation/maintenance consequences*
- *Sump clogging*
- *PSA quality and completeness*
- *Frequency of the LLOCA and of LOCAs*
- *High burn-up fuel under LOCAs conditions*
- *"Realistic" operator response assumptions*
- *Pilot submittal to validate*

Last

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SESSION 1
What drives the need to redefine the LB-LOCA?

Chairman: Dr. A. Thadani, (RES Director, USNRC)
Moderators: Mr. L. Skanberg, (SKI Sweden),
Mr. J. Sugimoto (JAERI, Japan)

What drives the need to redefine the LB-LOCA?

Why Redefine Large Break LOCA?

Dr. S. Bajorek, (USNRC, USA)

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

Joint CSNI/CNRA Workshop on

Redefining the Large Break LOCA:
Technical Basis and its Implications

"Why Redefine Large Break LOCA"

Stephen M. Bajorek
U.S. Nuclear Regulatory Commission

Zurich, Switzerland June 23-24, 2003

WHY REDEFINE LARGE BREAK LOCA?

- Background
- Current requirements and regulatory impacts
- Risk perspective
- Improved safety focus

June 23-24, 2003 Joint CSNI/CRNA Meeting Page 2 of 16

BACKGROUND

- **In 1995, the Commission published a Policy Statement on the Use of Probabilistic Risk Assessment (PRA)**
  - The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports NRC's traditional defense-in-depth philosophy
- **In 1998, the Commission published a White Paper on Risk-Informed and Performance-Based Regulation**
  - "A risk-informed approach to regulatory decision-making represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety."

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BACKGROUND (cont'd)

Greater knowledge and understanding based on

- Over 40 years of regulatory experience and several thousand years of collective operating experience
- Results from extensive research on fuel behavior and severe accident
- Improved tools allowing more realistic analyses and evaluation of margins

Improved Safety Focus

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CURRENT REGULATORY REQUIREMENTS

- 50.46 requires licensees to provide an Emergency Core Cooling System (ECCS)
  - Postulated LOCAs for range of break sizes and locations
  - Performance evaluated using acceptable models.
  - Satisfies specific acceptance criteria
  - Ensures continued effective core cooling
- Spectrum of LOCA sizes is defined as breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS.

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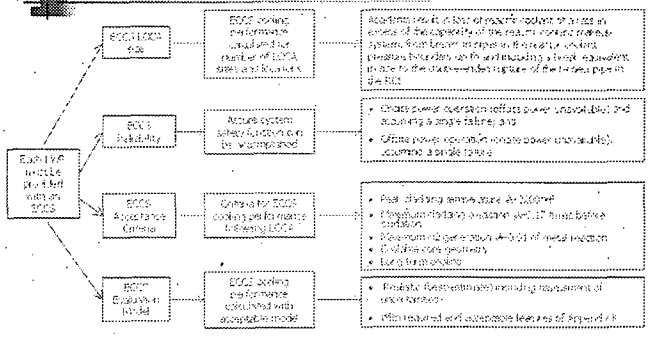
CURRENT REGULATORY REQUIREMENTS



LBLOCA IMPACT GOES BEYOND 10 CFR 50.46

- Regulations themselves are not independent of each other
- Understanding of their interrelationship is essential
- LBLOCA impacts numerous other regulations and general design criteria (GDC)

LBLOCA IMPACT GOES BEYOND 10 CFR 50.46 (cont'd)

Examples:

- 50.36: Technical Specifications
- 50.49: Environmental Qualification of Electric Equipment
- 50.65: Effectiveness of Maintenance
- GDC 1: Quality standards and records
- GDC 4: Environmental and dynamic effects design basis
- GDC 14/15, 30/31/32/33: Reactor coolant pressure boundary, system design
- GDC 17 Electric power systems
- GDC 19: Control room
- GDC 35/36/37: Emergency core cooling, Inspection/Testing of ECCS
- GDC 38: Containment heat removal
- GDC 41: Containment atmospheric cleanup system
- GDC 50: Containment design basis

CURRENT REGULATORY IMPACTS

- Evaluation requirements for plant ECC system LBLOCA response has had a large influence not only on plant design and Tech Spec limits, but also on analysis methods and ECC research.
- LBLOCA frequently restricts total core power and associated peaking factors in PWRs. Many PWRs are LBLOCA limited.
- LBLOCA analysis results are often sensitive to plant operating conditions and parameters (accumulator water level, thermal-design conditions, steam generator tube plugging level, accumulator pressure, etc.).
- LBLOCA mass and energy release often challenge containment analysis.

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IMPROVING SAFETY FOCUS

- **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.**
  - Best estimate methods developed for LBLOCA, but not SBLOCA.
  - Experimental data range of conditions may not be adequate for new scenarios.
- **Increased emphasis on smaller break sizes and risk significant accidents.**

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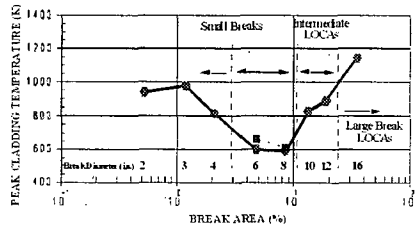
PLANT SYSTEM RESPONSE

- **Need to understand sensitivity of PCT vs. break size changes.**



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ANTICIPATED IMPLICATIONS

- Power uprates
- Emergency diesel generator (DG) start times
- Peaking factor levels
- Analysis and re-analysis costs
- Testing (e.g., integrated safety injection/loss of offsite power test, DG load shed and logic test, motor operated valve (MOV) stroke tests)
- Maintenance (e.g., DG overhauls, MOV actuator set-up and retests)
- Design (e.g., DG load sequencing, MOV actuator replacements/seismic concerns)
- Operation (e.g., extension of MOV allowed outage times)

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RISK PERSPECTIVES: CURRENT PRA LOCA MODELING

- Each LOCA size is not individually modeled
- LOCA sizes are grouped and defined based on plant response and required mitigating equipment
  - Large break LOCA: reactor depressurizes quickly enough so that low pressure injection occurs automatically in time to provide sufficient core cooling
  - Medium break LOCA: reactor does not depressurize quickly enough for low pressure injection to automatically occur in time to provide sufficient core cooling; however, the high capacity systems are needed to provide sufficient core cooling
  - Small break LOCA: reactor does not depressurize quickly enough for low pressure injection to automatically occur in time to provide sufficient core cooling; however, the low capacity systems are sufficient for core cooling
- Typical PWR break sizes include:
  - Large: greater than 6 inches (150 mm)
  - Medium: 2 to 6 inches (50 to 150 mm)
  - Small: ½ to 2 inches (10 to 50 mm)

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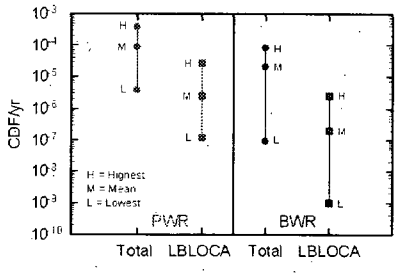
RISK PERSPECTIVES

Total & LBLOCA CDF Values for US PWRs and BWRs Obtained from Individual Plant Examinations



- Updated PRA models indicate a reduced LBLOCA contribution.

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RISK PERSPECTIVES (cont'd)

- Safety focus should be on risk contributors
- Accident initiators contributing to CDF typically involve:
  - Transients, LOOP/station blackout, small LOCAs (~less than 2 inch break size)
- Failures contributing to CDF typically involve:
  - RCP seals
  - Operator actions
  - Loss of power
  - Support system failures
- Rapid containment failure risk dominated by energetic high pressure core damage, not LBLOCAs
- Delayed containment failure risk is dominated by insufficient containment heat removal caused by factors unrelated to the LOCA size

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SUMMARY & CONCLUSIONS

- From a "risk" perspective and to improve the safety focus, redefining the large break LOCA is being considered
- Staff performing the necessary technical work to support a redefinition of the large break LOCA
- Technical issues in three general areas need to be addressed:
  - LOCA Frequency Reevaluation
  - PRA considerations
  - Plant system response

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The need to redefine the large break LOCA

Mr. A. Pietrangelo (NEI, USA)

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

CSNI/CNRA Workshop
Zurich, Switzerland
June 23, 2003

The Need to Redefine the Large Break LOCA

Tony Pietrangelo
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NEI

Overview

- Motivation for Redefinition
- Technical Considerations
- Policy Issues
- Other Considerations
- Necessary Elements
- Outlook/Conclusions

NEI

Motivation for Redefinition

- Existing Plants
  - Improve safety focus
  - Allocate resources accordingly
  - Improve coherence of regulatory framework
    - New Regulatory Oversight Process is more risk-informed than the regulations
    - Vast majority of inspection findings are of low safety significance

NEI

Motivation for Redefinition

- New Plants
  - Use risk-informed and performance-based principles at the outset of the design and licensing process
  - Maintain standards for protection of public health and safety
  - Provide a more cohesive, effective and efficient regulatory process

NEI

Technical Considerations

- LBLOCA frequency distributions
  - Service data
  - Probabilistic fracture mechanics
  - Expert elicitation
- Level of precision/rigor in defining an alternative bounding break size
- Defining mitigation capability for beyond design basis events

NEI

Policy Issues

- Scope and quality of PRA
  - Minimum/maximum standard vs. PRA adequacy commensurate with application
- Percent of LBLOCA contribution to CDF captured in design basis
- Need(?) to assess potential impacts from any application in advance

NEI

6

Other Considerations

- Public communications/acceptance
- Culture change
- Timeliness of the rulemaking process
- Use of "backstops"

NEI

7

Necessary Elements

- Strong leadership commitment
  - Commission/NRC senior management
  - Industry executives/plant management
- Sound technical bases
  - Clear and understandable
- Reasonable implementation requirements
  - Option must make cost/benefit case to be viable

NEI

8

Outlook/Conclusions

- Commission memo has provided needed direction to proceed with development of revised LBLOCA
- A pilot activity would help to resolve technical and policy issues
- Redefining LBLOCA is central to risk-informing technical requirements

NEI

THE LB LOCA in the design of the EPR

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Joint CSNI/CNRA Workshop on
"Redefining the Large Break LOCA:
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Summary

The large break LOCA taken into account in the design of the European Pressurized Water Reactor (EPR) is defined in consistency with the application of the break preclusion (BP) concept. For the EPR, the BP concept is applied to the design of the main coolant lines (MCL) and the main steam lines.

According to the BP concept, a guillotine break of a MCL is ruled out as a design basis event for the mechanical design of the structures and components located inside the reactor building. Only connected line breaks are assumed to occur. Nevertheless, the containment, the safety systems and components are still designed/qualified against the thermal-hydraulic consequences of a "double-ended guillotine break" (2A break) in the MCL.

This paper presents in the frame of the EPR design, in the first part, the general consistency of the safety approach with respect to large break LOCA and, in the second part, the practical application of the BP concept with respect to plant design and operation.

1 Background

Historically, the USNRC General Design Criterion 4 was the first to request that the total severance of a MCL be assumed as a Design Basis Accident (DBA) for which the damage of the core and the pressure challenge of the containment had to be assessed. For these thermal-hydraulic studies, a "double ended guillotine break" (2A break) is assumed. This postulate was used to cover a failure anywhere on the pressure boundary except the pressure vessel. Indeed, on the basis of mechanical evaluations, failure of the vessel was declared as sufficiently improbable, i.e. out of the design basis.

As revised in the early 70's the GDC4 introduced an additional requirement. In addition to the initial requirement, the assessment of the mechanical consequences on the structures and components had to be performed. Only a limited number of locations, those with the highest usage factor and/or stresses, were selected as break locations. The break area to be assumed in these mechanical studies can take into account the specific MCL supports.

The design of the French plants is based on this practice, up to the last plant series (N4 plants). Specific supports of the MCL are provided (restrains and anti-whipping devices) which limit the break area in case of a total severance of a MCL. The mechanical design of the components is based on such limited breaks, with conservative assumptions such as the cumulative effects of the break and of the Design Earthquake (DE). However, complete 2A breaks are assumed in the DBA analysis with respect to core design. This provides margins with respect to the emergency core cooling function, containment design and qualification of the components within containment.

In the early 80's, in Germany, a different approach, designated as the Break Preclusion Concept, was accepted by the Safety Authorities. It stated that the catastrophic failure of any given piping system may be excluded if a set of mechanistic criteria is satisfied. It is a purely deterministic approach relying essentially on the quality of the design and of the construction as considered in the "Basic Safety Concept". Nevertheless, a large break of the MCL is still considered as a DBA which gives margins for the design of the safety systems and of the containment.

For the EPR, it was decided to promote the Break Preclusion (BP) concept and to present this approach to the French and German Safety Authorities. The statements of these Safety Authorities (GPR and RSK) on the EPR BP for the MCL can be summarized as follows:

- Considering the state of technology, it appears feasible to design and to operate future PWR plants so as to "exclude" the complete guillotine break of the main coolant line.
- The loads to be considered for the mechanical design of the internal structures of the reactor vessel and for the design of the structures in the containment buildings can be limited to those resulting from a break equivalent to the complete guillotine rupture of the largest pipe connected to the main coolant lines (namely, the pressurizer surge line).
- These loads must take into account the combination of the DE with the LOCA (largest pipe connected to a main coolant line).
- The mass flow equivalent to a 2A break of the MCL has to be assumed for the design of the emergency core cooling function (safety injection system), the containment pressure boundary, the supports of the components and for the qualification of equipment.

2 Mechanical Design of the Heavy Components

The BP concept is based on the threefold requirements:

- Preventive measures relative to the component design, material and manufacturing,
- Leak Before Break (LBB) demonstration,
- Surveillance measures.

These requirements are described in details in section 4. A general overview of the BP concept is given in table 1.

According to this concept, a catastrophic failure of the MCL is deterministically ruled out. One advantage of this approach is that there is no need of anti-whipping devices to limit the MCL piping displacement in case of 2A break, as provided at the current operating plants.

The mechanical design of the structures and components is based on breaks of the lines connected to the MCL, the largest one being the pressurizer surge line break.

From the engineering point of view this means:

- the initiating break shall not induce a failure of the MCL or a rupture of another connected line,
- the internals of the Reactor Pressure Vessel (RPV) shall remain in a mechanical state
- consistent with core coolability,
- the geometry of the Rod Control Cluster Assemblies (RCCA's) guide tubes shall permit the RCCA's drop and core shutdown,
- the component supports shall remain integer,
- the Reactor Coolant Pumps (RCPs) and the Steam Generators (SGs) shall remain integer as well as their internals.

These mechanical studies are performed with conservative margins. In particular, the dynamic loads of the postulated breaks are combined with the loads of the DE (0.25 g zero period). The loads are combined according to the square root of the sum of the squares.This provides margins with respect to the mechanical design of the components.

As an additional margin, the stability of the heavy components (RPV, RCPs, SGs) must be ensured when considering a 2A break. A static load is taken into account: 2 PA, P being the operating pressure (155 bars) and A being the MCL section.

3 Thermal-hydraulic Analyses

3.1 Design Basis Accidents

The safety objective of these analyses is to demonstrate:

- subcriticality of the core by the scram system,
- a long-term safe shutdown condition,
- core coolability,
- non-aggravation of the event.

The largest break assumed as a DBA is the complete rupture of the pressurizer surge line. Its analysis is presented in the Safety Analysis Report (SAR) with the usual conservative rules and assumptions used in the DBA analyses:

- the residual heat is considered with conservative margins,
- the main parameters are the most penalizing, including uncertainties, e.g. the initial power is 102% nominal power,
- the Single Failure Criterion (SFC) is applied to the systems necessary to reach the safe shutdown,
- unavailability of systems for preventive maintenance is assumed,
- the loss of off-site power and the DE are superposed to the initiating event.

The safety injection system (SIS) of the EPR is a 4 train system without headers. In particular, it is assumed in the analysis that:

- one SIS train is lost due to the initiating event (it flows directly to the break),
- one SIS train is unavailable due to application of the SFC,
- one SIS train is unavailable due to preventive maintenance during power operation.

The 4th train is sufficient to meet the safety criteria:

- local peak clad temperature < 1200°C,
- local fuel cladding oxidation < 17% of cladding thickness,
- the Zirconium-water reaction is limited to 1% of the total cladding material,
- the containment pressure and temperature are lower than the design pressure and temperature, and lower than the qualification pressure and temperature of the accidentproof equipment inside containment,
- the radiological releases are lower than the safety limits.

3.2 Analysis of the 2A break: Realistic Approach

As already mentioned above, the 2A break is not analyzed as a DBA. However, for reason of defense in depth, its analysis is provided in the SAR, using realistic rules and assumptions.

In particular :
- the SFC is not applied,
- however, the injection train which feeds the break is considered to be lost,
- unavailability of systems for preventive maintenance is not assumed,
- the loss of off-site power is not to be superposed,
- it is assumed that the RPV internals and fuel elements geometry is kept, so that core coolability is ensured,
- the acceptance criteria are the same as for the DBA,
- the containment pressure and temperature criteria are the design criteria.

This approach was presented to the French and German Safety Authorities and agreed by them.
An overview of these additional measures is given in table 2.

4 Practical application of the Break Preclusion concept

The BP concept is based on the threefold requirements as follows:
- Preventive measures relative to the component design, material and manufacturing
- LBB demonstration
- Surveillance measures

4.1 Preventive measures relative to the component design, material and manufacturing

At first, the materials and the manufacturing processes are selected with the objectives to yield a high quality product with high toughness properties, to improve inspectability and to significantly reduce the number of welds.

The main coolant lines (see figure 1) consist of integrally austenitic forged legs. The selectedgrade of steel is nitrogen-strengthned steel grade Z2CN19.10. The straight parts are made from drilled forgings or form expanded tube blanks. The curved parts of the hot and cold legs are welded, hot stamped and forged bends, except the 6° angle bend at the RPV outlet which is machined in an extra thickness. The cross over leg is made of three parts, the elbows being manufactured by induction bending. Consequently, the homogeneous girth weld number is significantly reduced, ten girth welds per loop, eight of which being manufactured on site. An alternative solution consisting of one piece hot and cold leg with integrally bent elbows could be selected with the view to reduce the number of girth welds in the workshop.

In addition, connecting nozzles for auxiliary piping whose diameter is greater than 150 mm are forged with the leg and machined out. This is the case of the pressurizer surge line and of the Safety Injection System and Residual Heat Removal System nozzles.

For the operating French plants, the principle of "one piece" technology was applied first at Civaux 1. For this unit, the cold leg of the loop n°4 was made by forging, boring and then bending, with the 45° inclined nozzle integrated in the piping. This principle was applied once more at Civaux 2 where the entire loop n°1 was made using this "one piece" technology. Such a "one piece" technology was also applied for the German Konvoi plants, with ferritic loops. Our feedback experience guarantees to yield a high quality product.

The dissimilar weld joining the safe end to the ferritic nozzle (RPV and SG nozzles) is made by direct automatic TIG welding of Inconel 52 in a narrow groove. This implementation

is intended to be performed without any preliminary buttering of the nozzle bevel before welding. The ferritic side is isolated from the primary water by 6 to 8 mm of a stainless steel cladding. The weld process needs preheating and post weld heat treatment. The number of beads is reduced to one bead per layer and the risk of lack of fusion on the bevel is reduced to zero. Additionally, Inconel 52 exhibits very high toughness properties much higher than 300 kJ/m2.

Secondly, the materials and the manufacturing processes are selected with the objective to match the environmental conditions, ensuring the absence of indirect failure over the entire life of the plant (erosion, erosion-corrosion, intergranular stress corrosion cracking, cleavage embrittlement, vibrations, water hammer, fatigue damage). For the MCL, those situations are either irrelevant or have been shown to be acceptable on operating plants:

- Water flow and chemistry conditions cannot lead to degradation mechanisms by erosion, erosion-corrosion or intergranular stress corrosion-cracking. Experimental tests performed on dissimilar welds strengthen these conclusions.
- Materials are not subject to cleavage embrittlement over the full range of temperatures encountered in operating conditions.
- Vibrations are known to be very low mainly because of the high stiffness of the MCL.
- Water hammer is irrelevant to the MCL.
- Operating thermal transients are reasonably well known on the MCL at design stage and monitored during operation. They are rather mild and ensure the absence of fatigue over the entire life of the plant.

Finally, plant life crack growth and stability of any reasonable defect which might survive theinspection have to be examined. The crack propagation analysis assumes an initial defect and makes it propagate. Its dimensions are inferred from the manufacturing process and the sensitivity of the Non Destructive Tests (NDT) capabilities. One has to demonstrate that the defect does not grow significantly during the plant life (less than 1/4 of the thickness of the pipe is the acceptance criterion for the EOL crack) and the stability of the EOL defect is ensured even under the worst accidental loading conditions cumulating the full power normal operation. In addition, a conventional defect, arbitrarily twice as long as the reference defect is postulated with the aim to show the fundamental tendency of the crack growth process. Its propagation has to be such that it does not go through the wall before the end of life of the plant.

Given the manufacturing process and the examination capabilities in the forged steel piping, the demonstration has been conducted assuming as an upper allowable bound a defect (5x50 mm) in the most sensitive area (e.g RPV outlet homogeneous and bimetallic welds). The major conclusions are given herafter.

At first, such an envelope defect does not grow during the plant life. It only propagates 1.4 mm in depth direction whereas the side point propagation is limited to 0.2 mm. The crack propagation follows Paris'law, neglecting initiation phase. The computations were simultaneously performed at the tip of the crack and at the surface points, using the technique of influence functions for a semi-elliptical surface crack located on the inner surface of a cylinder. Finally, in a defence in depth approach, it is needed to submit this defect to condition equivalent of 10 times the operating transients for one plant life to propagate through the wall thickness and even the conventional defect 5x100 mm twice as long as the reference defect does not grow through the wall before the end of life.

The stability of the EOL reference defect when subject to the most severe loading conditions cumulating the full power operation and design earthquake is ensured, using J integral approach.

4.2 Leak Before Break demonstration

One has to demonstrate for all potentially sensitive girth welds of the main coolant lines that there is a sufficient margin between the size of a so-called detectable through-wall crack derived from the minimum leak detection capabilities under the full power normal operating conditions (NOC) and the size of the maximum allowable through-wall crack subject to Design Earthquake load cumulated with the full power normal operating load. The investigations have been launched concerning the different inlet/outlet nozzles of heavy components (e.g. RPV, SG, RCP inlet and outlet)

The first step aimed at predicting a reliable fluid leak rate through the crack under NOC, given a flaw size, pipe dimensions, material stress strain properties and loads. The leak rate computations were derived from the Henry thermal hydraulic model [3] which assumes two phase critical flow accounting for non equilibrium effects between the liquid and vapour phases. In addition, friction factors were introduced depending on the roughness, the length of flow path and the hydraulic diameter of the discharge opening. Finally, the cross section flow area was assumed to be elliptical, with opening displacements derived from Kumar et al [1] relative to circumferential crack in a pipe. The detectable crack lengths are plotted in the next table versus leakage flow rate for the different inlet/outlet nozzles at normal operating conditions.

The half crack length aq (mm) at normal operating conditions					
Leakage flow rate	1 gpm (3.8l/mn	5 gpm (19l/mn)	10 gpm (38l/mn)	15 gpm (57l/mn)	20 gpm (76l/mn)
RCP INLET	56	97	122	139	152
RCP OUTLET	44	78	99	113	125
SG INLET	50	87	111	118	140
SG OUTLET	51	89	113	130	142
RPV INLET	43	76	97	111	123
RPV OUTLET	34	61	78	91	101

The critical through-wall crack evaluation was based on predictive simplified J approach developed by Kumar et al [1]. Such an approach provides an estimation procedure which enables the construction of elastic-plastic J solutions. The approach is applicable to a wide range of flaws, pipe sizes and materials through the combination of results from EPRI handbook for fully plastic solutions and existing elastic solutions. A conservatism engineering procedure was introduced, using the lowest tensile and toughness properties of weld areas (e.g. tensile properties of base metal and toughness properties of welds). Nevertheless, in order to avoid over-conservatism, the most loaded zone in the primary circuit (RPV outlet), has been assessed including mismatch effects. Within this framework, the ARAMIS method developed by Framatome-ANP and EDF [2] was used.

These preliminary investigations have shown the following results. Concerning the crossover leg and the cold leg, critical crack sizes are greater than 700 mm even under the most severe loading conditions (NOC+DE) and the leakage flow rate induced at NOC is higher than 800 l/mn. That means the critical crack size is greater than 7 times the detectable crack size providing a leak flow rate of 4 l/mn. In addition, the leakage flow rate induced by the critical size at NOC is greater than 200 times the leak detection capacities assumed to be 4 l/mn.

Concerning SG inlet, the critical crack size is greater than 530 mm under the most severe loading conditions, which is greater than 5 times the detectable crack size providing a

leak flow rate of 4 l/mn. In addition, the leakage flow rate induced by the critical size at NOC is greater than 125 times the leak detection capacities assumed to be 4 l/mn.

At the most loaded zone in the primary circuit (RPV outlet), even under the most severe loading conditions (NOC+DE), the critical crack size is greater than 250 mm and the leakage flow rate induced at NOC is greater than 140 l/mn (which is greater than 35 times the leak detection capacities assumed to be 4 l/mn).

As a consequence, in spite of conservatism relative to predictive methods and loading conditions, very important safety margins exist in term of leakage flow rate and through-wall crack size regarding the critical conditions. Keeping in mind that these margins could be increased taking into account more sophisticated finite element model, this study clearly establishes the capability for the MCLs to fulfil the LBB requirements.

4.3 Surveillance measures

First of all, surveillance is ensured by leak detection systems. Leakage inside the containment may be detected and localized by various means appropriately located in the loop compartments, in the HVAC system and in the sump. A combination of several of the following means is considered for implementation: (1) Humidity sensors based on the dew point, (2) Temperature sensors, (3) Condensate flow sensors located on the coolers of the air recirculation, (4) Activity detectors located on the containment sampling system, (5) Water level measurements in the sumps.

The leak detection sensitivity depends quite obviously of the layout and of the venting rate. The feedback of experience for the French and German plants shows that detecting a leak rate in the range of 4 l/mn is achievable.

The main design objectives and criteria of EPR leak detection systems (LDS) are expressed hereafter. First, the LDS must be able to detect a leak on the MCL long before it becomes critical. Second, the LDS consists in 2 diverse and independent subsystems, each one able to detect the minimum specified leak rate. Finally, independent alarms are located in the control room. They are trigged when the minimum specified leak rate is exceeded. Orderly manual shutdown of the plant is required if the leak cannot be localized and isolated.

Furthermore, the LBB concept improves the practical conditions for In Service Inspection (ISI). It permits to remove some restraints compared to the previous practice where large LOCA was assumed. Access to the MCL becomes easier which permits a more efficient ISI and reduces the doses of the operating personal. On the other hand, ISI program must not be influenced by the LBB. ISI is a surveillance method by sampling, based on risk analysis, not a deterministic insurance against damage. The locations for and the extensiveness of ISI should be decided from a mechanistic logic, and must not be influenced by the removal of restraints.

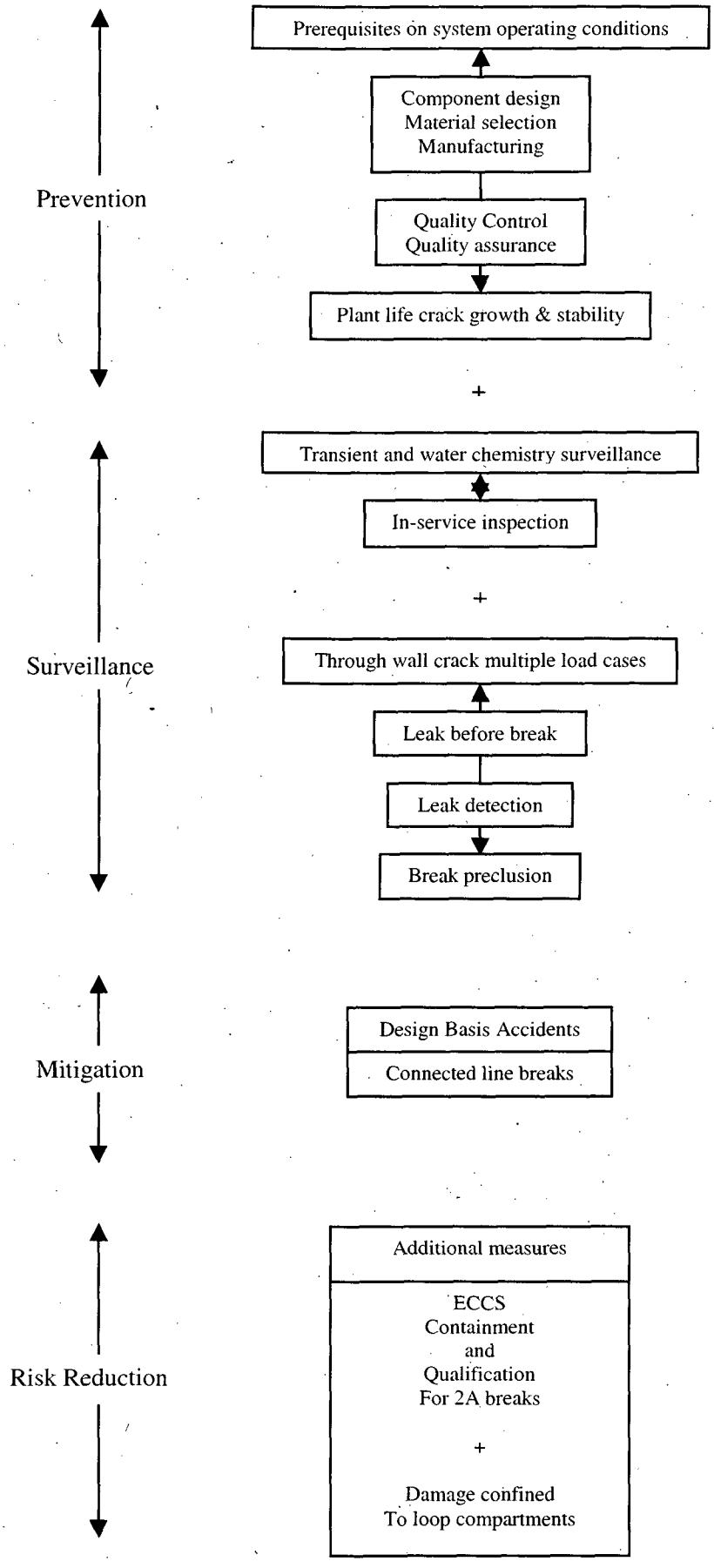
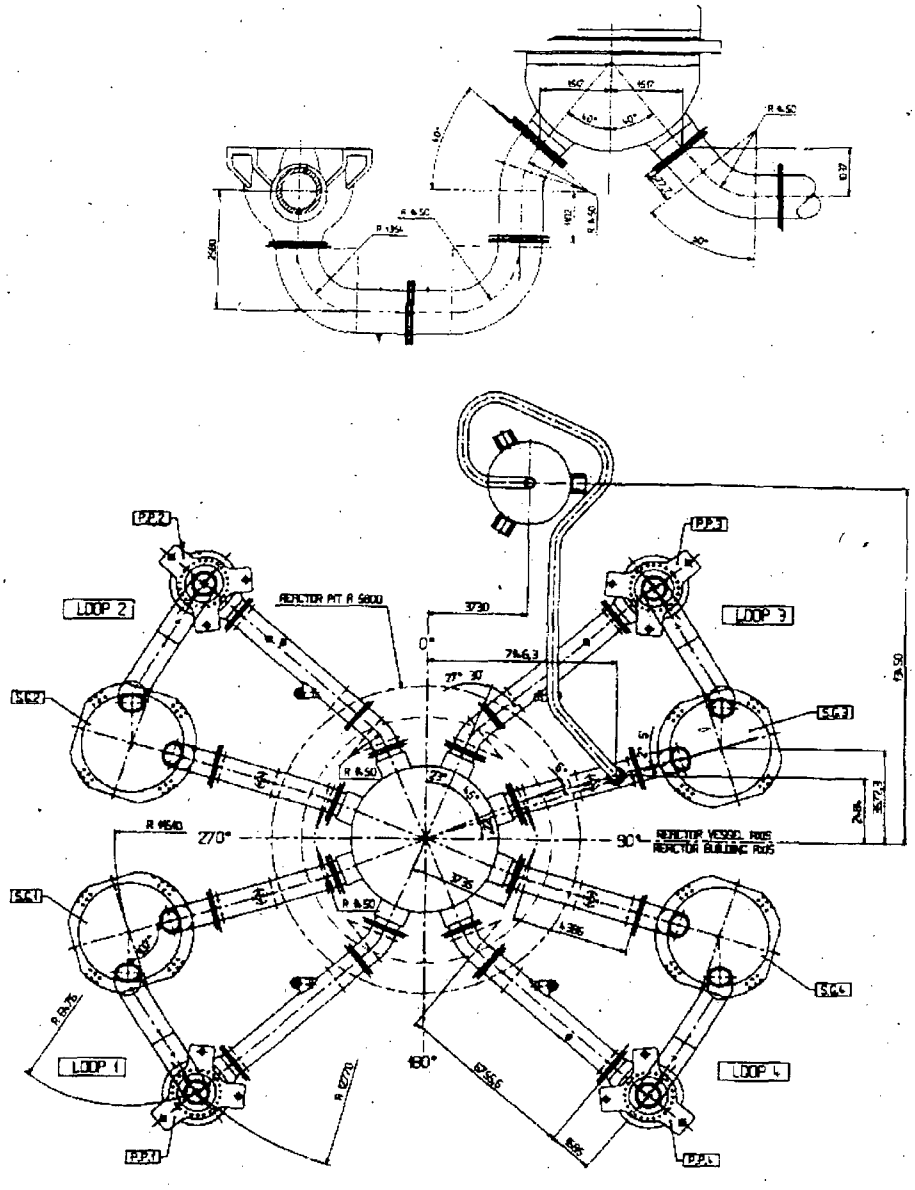


Table 1 : Consistency of the Break Preclusion Concept

Effects		**Postulated pipe break**
on	**of**	
ECCS performance	loss of coolant	leak/break on MCL up to 2A opening
containment	pressure build-up temperature	2A opening of MCL
environmental qualification of equipment	flooding pressure temperature humidity radiation	2A opening of MCL
primary components (incl. internal and supports)	dynamic effects of decompression	guillotine break of any connected line at nozzle weld
containment internal structures	differential pressure temperature flooding	guillotine break of any connected line at nozzle weld
main components supports	2pA force colinear with nozzle	equivalent to a 2A opening of a MCL

Table 2 : Additional Measures despite Break Preclusion of the MCL



Workshop circumferential welds	8
Site circumferential welds	32

Figure 1: Main Coolant lines and girth weld location

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