Unclassified **NEA/CSNI/R(2003)16**

Organisation de Coopération et de Développement Economiques
Organisation for Economic Co-operation and Development **30-Sep-2003**

English text only

NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

NEA/CSNI/R(2003)16
Unclassified

Cancels & replaces the same document of 10 September 2003

RESPONSES TO THE SURVEY ON "REDEFINING THE LARGE BREAK LOCA: TECHNICAL BASIS AND ITS IMPLICATIONS"

Zurich, Switzerland
June 23-24 2003

JT00150408

Document complet disponible sur OLIS dans son format d'origine
Complete document available on OLIS in its original format

English text only

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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

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

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

NUCLEAR ENERGY AGENCY

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

The mission of the NEA is:

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

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

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

COMMITTEE ON NUCLEAR REGULATORY ACTIVITIES

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

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

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

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

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA) is an international committee made up of senior scientists and engineers. It was set up in 1973 to develop, and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety among the OECD Member countries.

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

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

In implementing its programme, the CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

A. Foreword

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

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

In December 2002, the CNRA and the CSNI jointly requested the NEA to organize a workshop on “Redefining the Large Break LOCA: Technical basis and its implications".

The Workshop was held on June 23-24, 2003 in Zurich, Switzerland hosted by HSK (Swiss Federal Nuclear Safety Inspectorate), PSI (Paul Scherrer Institut) and the OECD/NEA.

While the Workshop addressed technical aspects, the 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 was designed

- To provide material for discussion;
- To clearly summarize current national regulations;
- To understand rationales and incentives for changing or not the regulation with regard to the Large LOCA;
- To list technical issues to be resolved before implementing a new regulation, if any.

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?

The Workshop proceedings have been divided into two separate volumes under the references 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.

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

Synthesis and compilation of
Responses to the survey

A. Foreword	5
B. Table of Contents	9
C. Synthesis and compilation of responses	11
• Current regulatory framework	17
• Current technical framework	39
• Consideration for the future	47
• Other issues	63
Appendix 1	65
Appendix 2	71
Appendix 3	77

B. Table of Contents

Current regulatory framework

1. What is the largest or limiting break size assumed as design basis (LOCA) in your regulation? 17

2. Are there any "risk" considerations in the current LOCA break size definition? 21

3. Describe regulatory implications on design, operational procedures, testing, inspection program associated with the current LOCA break size definition? 23

4. Is Leak-Before-Break accepted (or being considered) in your regulation ? If so, what are the consequences on component or piping supports, system analysis, fuel assembly, containment… 33

Current technical framework

1. What technical issues are currently of concerns for Structures, Systems and Components (associated with the current LOCA definition)? (please list) 39

2. What technical issues would be needed to be addressed by Researchers to support the regulatory decision making process to change the current LOCA definition? 43

Consideration for the future

1. Are you considering changes in your regulation? For operating plants? For future plants? 47

2 What would be the incentives? From the regulatory viewpoint? From the Industry viewpoint? 51

3. If you are considering replacing large break LOCA by a smaller break size within the design basis, some degree of core damage, short of core melt resulting in vessel failure, may be expected if a large break LOCA actually occurred.

   3.a How would you establish performance requirements for the emergency cooling systems in order to provide some assurance that damage following a large break LOCAs can still be mitigated before vessel failure? What is the technical basis? 55

   3.b Are currently available computer codes and models adequate for the required analyses or new tools will have to be developed? 59

Other issues

What other issues concerning LOCA do you feel should be discussed during the workshop? 63

Appendices

I. Information provided by USNRC to complement the answer on question future 2 65

II. Information provided by GRS to complement the answer to question "Current Regulatory Framework 4" 71

III. Information provided by STUK *on the LBB and failure frequency requirements in the Finnish guideline YVL 3.5* 77

C. Synthesis and Compilation of Responses

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

Current regulatory framework

1. What is the largest or limiting break size assumed as design basis (LOCA) in your regulation?

Most of the countries consider the Double Ended Break Guillotine of the largest pipe in the reactor coolant system. The Slovak Republic assumes instantaneous guillotine break of pressurizer surge line with the diameter of 200 mm for VVER-440/230 reactors. For these reactors, the 2x500 mm LLOCA is considered as a beyond design basis events.

2. Are there any "risk" considerations in the current LOCA break size definition ?

The LLOCA is in the design basis. There are no explicit risk consideration in the likelihood of this initiating event and in analysis.

3. Describe regulatory implications on design, operational procedures, testing, inspection program associated with the current LOCA break size definition ?

LLOCA has obviously a broad impact on structural design (containments, pipe restraints, capacity of emergency systems(PORVs, ECCS), redundancy), testing (containment, piping), environmental qualification of equipments, operating procedures. ISI programs are sometimes governed more by structural integrity considerations (i.e. degradation) than by LLOCA considerations.

It is interesting to note that some countries mentioned that no changes would be made to procedures to mitigate a LLOCA as procedures are similar for the SLOCA (Small LOCA).

One organization indicates that LLOCA requirements influence every level of plant operations as well as design and that some restrictions on plant operations might be removed if the LOCA definition is changed.

4. Is Leak-Before-Break accepted (or being considered) in your regulation ? If so, what are the consequences on component or piping supports, system analysis, fuel assembly, containment…

A vast majority of countries is using LBB. It is used in all cases ***only*** *to remove (or not installing) whip restraints, snubbers, modify supports or justify internals behavior. One country approved LBB only for analysis and did no physical changes in the plants.*

Current technical framework

1. What technical issues are currently of concerns for Structures, Systems and Components (associated with the current LOCA definition)? (please list)

The most frequently listed are:

- *sump debris generation and sump blockage;*
- *fuel behavior;*
- *containment leak tightness;*
- *Diesel (EDG);*
- *Bimetallic welds failure mode.*

2. What technical issues would be needed to be addressed by Researchers to support the regulatory decision making process to change the current LOCA definition?

LOCA initiating event frequency and best estimate analysis methods and uncertainties are the most frequent answers*. Inspection programs, fracture mechanics and probabilistic fracture mechanics, leak detection systems and the integration of the* *deterministic, defense in depth principle within a risk-informed framework are also mentioned.*

Consideration for the future

1. Are you considering changes in your regulation? For operating plants? For future plants?

Except the USA, regulators are not considering changes in the regulation. Nevertheless Canadian regulators would be ready to discuss a more balance approach to LLOCA for both existing and future plants. As to the Industry, only the USA has done some detailed work.

2. What would be the incentives? From the regulatory viewpoint? From the Industry viewpoint?

- *Regulatory viewpoint*

As said before, very few countries have already considered the topic. Nevertheless, the Canadian answers give a good overview. It has to be noted that ***an important incentive mentioned is to focus resources in areas of greater risk significance*** *It is also mentioned by one country that unnecessary conservative could be removed and safety margins better used. The latter is consistent with what is said about best-estimate analysis methods and uncertainties.*

- *Industry viewpoint*

power uprates, EDG start times, testing, economical benefits, ….

3. If you are considering replacing large break LOCA by a smaller break size within the design basis, some degree of core damage, short of core melt resulting in vessel failure, may be expected if a large break LOCA actually occurred.

3.a How would you establish performance requirements for the emergency cooling systems in order to provide some assurance that damage following a large break LOCAs can still be mitigated before vessel failure? What is the technical basis?

No clear answers but several approaches

- *LLOCA would not totally disappear. It would be considered as a beyond design basis event and analyzed using state of the art and best estimate methods.*
- *LLOCA probability considered as low as vessel rupture probability and thus excluded*
- *ECCS would be based on the break of the largest pipe but ancillary requirements (e.g., technical specifications for safety injection flow rates and inspection frequencies for accumulator condition) will be subject to risk-informed modification.*

3.b Are currently available computer codes and models adequate for the required analyses or new tools will have to be developed?

Available codes seem to be mature enough although applicability range for codes may need to be reexamined. Struture reliability models, PRA models and probabilistic fracture mechanics may need to be further developed Also core reflooding and vessel-corium interaction could be further developed.

What other issues concerning LOCA do you feel should be discussed during the workshop?

Appendices

I Information provided by USNRC to complement the answer on question future 2

II. Information provided by GRS to complement the answer to question "Current Regulatory Framework 4"

III. Information provided by STUK on the LBB and failure frequency requirements in the Finnish guideline YVL 3.5

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

Answers received from	
BELGIUM	AVN (Association Vinçotte Nuclear)
CANADA	CNSC (Canadian Nuclear Safety Commission)
CZECH REPUBLIC	SUJB (State Office for Nuclear Safety)
	NRI Rez (Nuclear Research Institute)
FINLAND	STUK (Radiation and Nuclear Safety Authority)
FRANCE	IRSN (*Institut de Radioprotection et de Sûreté Nucléaire*)
	EDF SEPTEN (*Service Études et Projets Thermiques & Nucléaires*)
GERMANY	GRS (*Gesellschaft für Anlagen- und Reaktorsicherheit mbH*)
JAPAN	NUPEC (Nuclear Power Engineering Corporation) JAERI (Japan Atomic Energy Research Institute)
MEXICO	CNSNS (*Comision Nacional de Seguridad Nuclear y Salvaguardias*)
SLOVAK REPUBLIC	Nuclear Regulatory Authority
	VUJE (Nuclear Power Plant Research Institute) Trnava, Inc. (This is the view of VUJE and it was discussed with Slovak regulator as well as with Bohunice NPP)
SPAIN	CSN (*Consejo de Seguridad Nuclear*)
SWEDEN	SKI (Swedish Nuclear Power Inspectorate)
SWITZERLAND	HSK (Swiss Federal Nuclear Safety Inspectorate)
UNITED KINGDOM	NII (Nuclear Installations Inspectorate)
USA	USNRC (US Nuclear Regulatory Commission)
	WOG (Westinghouse Owners Group)
RUSSIA Observer	EDO Gidropress
	GAN (Science and Engineering Center for Nuclear and Radiation Safety)
Slovenia Observer	Slovenian Nuclear Safety Administration

Current regulatory framework

		1. What is the largest or limiting break size assumed as design basis (LOCA) in your regulation?
BELGIUM		The Belgian regulation does not specify a limiting break size for DBA (LOCA), but all operating plants (all PWRs) were designed and licensed on the basis of the double ended break of the largest pipe in the primary circuit.
CANADA		Largest – instantaneous guillotine failure of the largest diameter pipe in the primary system (reactor header). This is normally modelled as a break of 200% pipe area. Limiting – "critical" break that leads to flow split in core is limiting for some derived acceptance criteria. Licensing basis analysis is deterministic, with conservative assumptions, such as reactor initial conditions at limit of operating envelope (LOE) and minimum safety system performance (e.g. reactor trip on second trip parameter of second shutdown system, two rods unavailable, minimum ECC flow). Analysis is normally with best estimate physics and thermal-hydraulics codes.
CZECH REPUBLIC	**SUJB Safety Authority**	Double Ended Guillotine LB LOCA (2x500 for VVER 440/213 or 2x850 for VVER 1000)
	NRI Rez	DN 500 IDN 850
FINLAND		It is 2A, a double-ended guillotine break of any pipeline.
FRANCE	**IRSN**	For the existing plants the frames of large break studies concern break diameters, the largest break size assumed being the double-ended break of any line of the Primary Circuit; the most limiting break size is determined through sensitivities studies between 14 inches and 2A. Concerning the EPR project, the possibility to exclude the 2A LLOCA on the basis of LBB demonstration was accepted. However the methodology used for this demonstration (design, manufacturing and control rules, calculation of critical defects, breaks area, leak flow, safety coefficient, nature and sensitivity of leak detection devices) was not presented to the Safety Authority by the designers and consequently was not approved. Anyway the break size considered as design basis accident will be at least the rupture of the surge line.

	EDF SEPTEN	DEGB in 11 locations in 1 ms.
GERMANY		For the light water reactors the guillotine type break of the primary coolant line is the design basis in principal. The break assumptions to be used for the safety concept are laid down in the RSK guidelines, chapter 2.1.1. In the years 1979/1981 a change in the requirements took place. Regarding the capacity of the ECCS as well as the design pressure of the containment the instantaneous break of the largest line has been kept as the design requirement. For piping systems which have been qualified according to the basic safety concept the design requirements regarding pipe whip, jet impingement and stability of internals have been based on a opening of a pipe with a 10 % of the cross section of the largest pipe.
JAPAN		Double ended guillotine（200%）
MEXICO		The instantaneous double-ended break of one of recirculation pipes.
SLOVAK REPUBLIC	**Safety Authority**	According to a current legislation (Regulation No. 2/1978), the limiting break size is considered as an accident with the biggest radiological impact to the envinronment. For NPPs of WWER 440/213 type the design basis LOCA is Double Ended Guillotine Break (DEGB) of inner diameter (ID) of 500 mm. For NPP of WWER 440/230 type (two units of Bohunice V-1 NPP) the original design basis LOCA has been DEGB of ID of 32 mm. After the “small” and “gradual” reconstruction the design basis LOCA has been increased up to ID of 200 mm for all, normal, upset and emergency operational regimes using a conservative approach and ID of 500 mm using the best estimate assumptions. The requirements and conditions for gradual reconstruction have been issued in the regulatory decision No. 1/94 including definition of criteria for improvement of ECCS system to be able manage this size of break.

	VUJE	Limiting break size: VVER-440/213 – instantaneous guillotine break of Reactor Coolant System main circulation line with the largest diameter (diameter of 500 mm, double-end coolant discharge). VVER-440/230 – instantaneous guillotine break of pressurizer surge line with the diameter of 200 mm (double-end coolant discharge), resp. with a partial break of RCS line with equivalent diameter of 200 mm. For this type of reactor, LBLOCA 2x500 mm is assumed as Beyond DBA. Break location: Break location is assumed at the most adverse point. It means that break location can differ for evaluation acceptance criteria from point of view of core cooling (maximum cladding temperature) and from point of view of mass and energy release into the confinement (confinement loads calculation, radiological consequences). Design basic analyses are performed using the best estimate thermal-hydraulics code and conservative assumptions (initial and boundary conditions) in term of acceptance criterion evaluation. For VVER-440/230, where LB LOCA 2x500 mm is assumed as BDBA, acceptance criteria fulfillment has to be demonstrated using the best estimate thermal-hydraulics code and best estimate (or realistic) assumptions.
SPAIN		We follow the regulations of our vendors: USA and Germany
SWEDEN		All Swedish reactors have been analysed for a LOCA equivalent to the dimension of the largest pipe connected to the reactor pressure vessel. This has been a prerequisite in the PSAR and FSAR as a basis for the licence. For the BWR with internal re-circulation pumps a hypothetical break size of 80 cm^2 has been basis for the licence and the design of emergency core cooling system. (See also the answer to the questions of “Current technical framework”) SKI has not yet issued any general regulations that specifically address these aspects. However, SKI’s general design regulations that now are being prepared will require that the reactor core must be cooled by sprinkling or water covering in event of a loss of coolant that can follow a break of any pipe connected to the reactor pressure vessel. It must also be possible to reach stable conditions with a water-covered core or core melt and established residual heat removal.
SWITZERLAND		A complete double-ended break of a main coolant pipe (200% break) according to HSK-Guideline R-101 "Design Criteria for Safety Systems of Light-Water Reactors" (May 1987).
UNITED KINGDOM		For PWR, full double ended cold leg guillotine primary loop pipework break.

USA	**US NRC**	10 CFR 50.46 Acceptance criteria for emergency core cooling systems (ECCS) for light-water cooled nuclear power reactors, Section c and Appendix A to Part 50 -- General Design Criteria for Nuclear Power Plants, Definition of Loss of Coolant accidents. Both include the following text defining the design base LOCA: *“..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.”* Appendix K to Part 50 -- ECCS Evaluation Models Includes the following text defining the spectrum of possible break sizes that shall be considered in analysis of LOCAs: *”This spectrum shall include instantaneous double-ended breaks ranging in cross sectional area up to and including that of the largest pipe in the primary coolant system”*
	WOG	A double-ended rupture of the largest pipe in the reactor coolant system.
RUSSIA Observer	**Gidropress**	There is no direct indication of the largest or limiting break size in the current safety standards approved by the Russian regulatory body. In fact, current design practice is still based on old regulations where the break of the largest primary pipeline was implied (OPB-73) or directly prescribed (OPB-82). In particular, the OPB-82 item 4.1.1 sounds: “Instantaneous rupture of the largest pipeline with unimpeded coolant discharge … must be considered in design as the maximum design accident with the primary circuit depressurization”. Also, the USSR standard called “Nuclear power pressurized water reactors. General technical requirements” (GOST 24722-81) is still in force in Russia. The item 1.5.1 of this standard sounds: “Instantaneous transverse rupture of the main circulation pipeline with double-ended coolant discharge … must be considered in design as the maximum accident…”.
	GAN	Instantaneous guillotine break of primary pipe of maximum diameter on hot or cold leg for WWER reactors and of main circulation pump pipe or main circulation pump header for RBMK reactors.
Slovenia Observer		Limiting break is DECLGB (double ended cold leg guillotine break) for the case of our only NPP.

Current regulatory framework

		2. Are there any "risk" considerations in the current LOCA break size definition ?
BELGIUM		No.
CANADA		Not in current licensing basis. Break size up to full guillotine break is in the design basis. There is no consideration of the likelihood of the initiating event. Analysis is “deterministic” safety analysis as described above. Licensees are proposing Best Estimate Analysis + Uncertainty (BEAU) methodology (similar to CSAU) as a basis for consequence analysis. This is under consideration Incredibility of failure arguments are accepted for large vessels that were appropriately designed, fabricated and are appropriately operated and inspected (e.g. steam generators).
CZECH REPUBLIC	**SUJB Safety Authority**	No.
	NRI Rez	No.
FINLAND		Not quantitatively. Qualitatively seams welded in the field may be more likely to crack, as operating experience from Summer and Ringhals plants implies.
FRANCE	**IRSN**	If "risk" means probability/consequences considerations: No.
	EDF SEPTEN	No.
GERMANY		In the current LOCA break size definitions there are no explicit risk considerations but implicit judgements on component reliability have played a major role in the change of the regulatory requirements.
JAPAN		No.

MEXICO		In general, all breaks are assumed to occur in the recirculation piping because they are the most severe from the cladding heatup viewpoint. Because the steam quality of the fluid leaving break in a recirculation line is zero initially, these are generally know as liquid breaks, however breaks on other localizations, including feedwater, core spray and steam lines are also analyzed. The large break LOCA is an event with low probability, however the small LOCA and small-small LOCA have a higher probability and therefore this kind of LOCA's should be considered seriously in the definition of the design basis LOCA.
SLOVAK REPUBLIC	**Safety Authority**	At present there are no risk considerations in the LOCA break size definition.
	VUJE	Not in current licensing basis.
SPAIN		No.
SWEDEN		No risk considerations have been used in present LOCA break size definition.
SWITZERLAND		No.
UNITED KINGDOM		No, large LOCAs are infrequent events, very small & small LOCAs are frequent events.
USA	**USNRC**	Risk is considered implicitly insofar as very low frequency events are not included in the spectrum of LOCA sizes that must be included in the design basis. For example, the largest loss of coolant accident, reactor vessel rupture, is not included in the design basis. It was also assumed (when the definition was established) that the design for double-ended rupture would bound all potential risk-significant scenarios. The 1984 revision to General Design Criteria (GDC) 4, based in part on the low probability of pipe rupture, allows the use of leak-before-break (LBB) to exclude the dynamic effects of postulated pipe rupture subject to certain conditions. (See additional discussion in item 4).
	WOG	No, it is solely deterministic.
RUSSIA Observer	**Gidropress**	The "risk" aspect does not exist explicitly. However, larger frequencies for smaller breaks are assumed in PSA.
	GAN	Ruptures of equipment casings and vessels, whose manufacture and operation shall be carried out in accordance with the most stringent requirements of federal regulations and rules in the field of use of nuclear energy, are not included into the list of initiating events. In this case it should be demonstrated that the probability rate of reactor vessel destruction does not exceed 10-7 per reactor per year.
Slovenia Observer		No.

Current regulatory framework		
		3. Describe regulatory implications on design, operational procedures, testing, inspection program associated with the current LOCA break size definition?
BELGIUM		Implications of the current LOCA break size definition are multiple. Indeed, at the design, the capacity of many systems (containment structures, core support structures, ECCS, containment heat removal, support systems, ...) has been determined by requirements defined by the Large Break LOCA (and/or steam line break). Concerning testing, the current definition of LBLOCA has still an impact, for example, on the ten year periodic test pressure of the containment. Also for the qualification of components for post-accidental conditions, the current definition of LBLOCA still has an impact.
CANADA		**Design** – LLOCA has always been part of the design basis for Canadian reactors and it sets performance requirements for shutdown, emergency core cooling and containment systems (the Special Safety Systems). Reliability requirements are set independent of LLOCA. **Operational procedures** – procedures include response to LLOCA and also response to accident precursors such as shutdown limits for primary leakage. The procedures for response to LOCA would be largely unchanged if LLOCA was not in design basis since the same procedures cover small and large LOCA. **Testing** – extensive on-line testing of special safety systems ensures that these systems satisfy minimum performance standards and that systems meet unavailability/reliability targets. This would be largely unchanged if LLOCA was not in design basis as all systems are still required for SLOCA. It may be possible to reduce some performance targets, e.g. standby generator start time, shutdown system operation times. **Inspection –** in-service inspection programs require inspection of class 1 piping at predefined intervals. Inspection program is based on initial weld inspection results and pipe stress / fatigue characteristics. This would be largely unchanged if LLOCA was not in design basis. More extensive inspection program would be required to support a LBB approach.

CZECH REPUBLIC	**SUJB Safety Authority**	LBLOCA has been always assessed in the Safety Analysis Reports as the design basis for both NPP in the Czech Republic (4xWWER-440, 2xWWER-1000). The current LOCA break down definition has been and must be considered in the design of the emergency core cooling systems and the containment systems (including bubble condenser). Primary circuit: – Safety analyses : Use the validated computer code. Safety margin definition. Definition of safety limits (also for high bur up fuel). Long term Cool ability of the core – Follow up control fuel assembly integrity (for WWER 440/213) – Coolant – in vessel structure interaction **Containment integrity.** Bubble condenser integrity (WWER 440/213)
	NRI Rez	Design and maintenance of pipe whip restraints, higher radiation exposure of maintenance personal, worse access to pipe welds for ISI, cost saving
FINLAND		LOCA serves as a convenient “envelope” definition for an extreme condition for e.g. ECCS (flooding capacity vs. pressure, recirculation filters, etc.) / Containment overpressure, -temperature and leaktightness / Reactor core - total power and power distribution/Nuclear fuel (burnup; pellet cracking, internal gas pressure, etc).. / ..Reactor internals, including mechanical part of scram system / Protection and shielding from dynamic loads (pipe whip restraints etc.) / Environmental qualification of equipment inside the containment.

FRANCE	**IRSN**	For existing plants, concerning operational limitations, LLOCA accident designs the Fq max ensuring the respect of the accident criteria. LLOCA is also used for the design of piping supports, of whip restraints, of reactor vessel internals , of Safety Injection System and Sump recirculation, for qualification of mechanical and electrical equipment important for safety. LLOCA participates to containment design, together with SLB. The LLOCA consequences for maintenance are on the one hand the periodic calibration of the gap between pipes and whip restraint devices, and on the other hand the decennial containment pressure test. Concerning EPR, the maximum break size considered as design basis accident is at least the rupture of the surge line. But, the mass flow equivalent to a 2A-opening of a main coolant line has to be assumed for the design of the emergency core cooling function (using realistic assumptions and models and appropriate criteria to be proposed by the designer) and of the containment pressure boundary, so as to implement safety margins concerning the cooling of the core to prevent core melt and concerning the containment function; the 2A-opening is also to be assumed for the supports of the components and for the qualification of equipment.

	EDF SEPTEN	– large break LOCAs are in the design transient list to design all concerned components – LBLOCA are considered for containment design, safety systems design, component qualification ,fuel design and safety analysis
GERMANY		The LOCA break size definition determines to a major part the overall safety concept of light water reactors. This means that there are strong implications regarding the overall design. There are detailed testing requirements regarding the reliability of the related safety systems. The inspection program for the reactor pressure boundary is not really governed by LOCA considerations. It is more directed by the goal to detect any degradations which could potentially challenge the integrity of the component is picked up at a very early stage.
JAPAN		In the current regulation for a design, operation, and the examination, the assumption of LOCA break size specifies or constrains such as the capacity, integrity, the performance etc. of an ECCS system, a containment, an inflammable gas concentration control system, start-up time of EDG, plugging number of SG tube, radiation measurement etc.
MEXICO		The 10 CFR part 50 Appendix A “General Design Criteria for Nuclear Power Plant” and the ASME code establish that that the structures, systems and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. The component Class 1, that are important to the safety which are part of the coolant pressure boundary, are tested and subject to inspection program according with the ASME code, but there is not an explicit referent to LOCA definition even many of the safety systems are designed in case of LOCA’s.
SLOVAK REPUBLIC	**Safety Authority**	Regulatory authority issued its requirements for gradual reconstruction of V-1 NPP in the decision No. 1/94. One of the group of requirements deals with core cooling at the LOCA conditions, including analyses of ID 200 mm and ID 500 mm LOCA, elaboration of emergency operational procedures, etc. Similarly, for V-2 NPP, in the regulatory decision No. 4/96 (based on assessment of Safety Analysis Report after 10 years of operation) is direct requirements “To submit new analysis of LB LOCA using better modeling of thermal-hydraulic processes in reactor and primary circuit and in bubble condenser compartements of confinement”. It was also recommended to use the IAEA document No. IAEA--EBP-WWER-01 – Guidelines for Accident Analysis of WWER Nuclear Power Plants for these analysis.

	VUJE	Design – LB LOCA has always been part of the design basis for VVER-440 reactors and it sets performance and reliability requirements for shutdown, emergency core cooling and containment systems. The LB LOCA response procedures would be unchanged if LB LOCA was not in design basis since the same procedures cover small and large LOCA. At the present there is no reason to change LB LOCA definition in Slovakia.
SPAIN		See answer to question 1
SWEDEN		There are no specific regulatory implications on design (in addition to the requirements of core cooling), operational procedures, testing, inspection program associated with the presently used LOCA break size definition. The design requirements for piping systems in Swedish reactors correspond to the requirements in ASME Section III. Components in the cooling systems must be tested periodically. For in-service inspection a qualitative risk informed approach is used which take into account both the safety significance as well as the susceptibility of different degradation mechanisms.
SWITZERLAND		The current LOCA break size definition must be considered in the design of the emergency core cooling systems and in the evaluation of the primary containment behaviour.
UNITED KINGDOM		UK PWR design and safety case assumes full double ended guillotine break. Given that the case has been made, no known reason for changing.

USA	**US NRC**	The LBLOCA requirements broadly affect design, operation, testing, inspection, and maintenance aspects of structures, systems, and components (SSCs) of a nuclear power plant. In addition to the implications for the Emergency Core Cooling System which is discussed in more detail below, the LBLOCA affects some of the key requirements as listed below. Environmental and dynamic effects design basis (GDC 4) – SSCs are required to be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. These SSCs are to be protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids. Piping layout, restraints, jet impingement shields, and interior compartment design are directly affected by this requirement. Containment Design Basis (GDC 50) – The reactor containment structure, including access openings, penetration, and the containment heat removal system are designed so that containment structure and its internal components can accommodate, without exceeding the design leakage rate and with sufficient margins, the calculated pressure-temperature conditions resulting from any loss-of coolant accident. The containment leakage rate test pressure is based on the LBLOCA pressure. Environmental qualification of electrical equipment (10 CFR 50.49) – This regulation, in part, requires that the electrical equipment qualification program include and be based on the temperature-pressure, humidity, radiation, and other effects associated with most severe design basis accident. Fuel (10 CFR Part 50, Appendix K)– The LBLOCA affects fuel design limits such as peak cladding temperature, and maximum cladding oxidation. LBLOCA affect containment sump design and performance requirements. Ultimate heat sink requirements are associated with the LBLOCA. For the Emergency Core Cooling System (ECCS) design, 10 CFR 50, Appendix A, General Design Criterion 35 requires that, “suitable redundancy in components and features, and suitable interconnections, leak detection, isolation and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.” To achieve this safety function over the spectrum of LOCA break sizes, ECCS systems are designed with redundant trains of high and low pressure ECCS piping, pumps, valves and other necessary equipment.

		The ECCS is designed to provide core cooling and negative reactivity insertion (Pressurized Water Reactors) following pipe breaks in the Reactor Coolant System (RCS) which cause a discharge larger than that which can be made up by the normal makeup system, up to and including the instantaneous circumferential rupture of the largest pipe in the RCS. 10 CFR 50.46 and 10 CFR 50, Appendix K provide the acceptance criteria and evaluation model requirements for ECCS design. 10 CFR 50.46 allows use of either an evaluation model containing many conservative or bounding assumptions, or a less restrictive Best-Estimate model for evaluating ECCS performance. Regulatory Guide 1.157 describes a Best-Estimate approach which defines the full break spectrum and accompanying uncertainty analyses. In addition to the break sizes considered, the methodology, particularly the Appendix K approach, will influence design due to the conservatism in the analysis. Some examples of ECCS requirements based on an analysis of the full spectrum of LOCA break sizes in a Pressurized Water Reactors (PWR) include: • Large breaks up to a double ended guillotine break require 3 or 4 accumulators which can rapidly refill and reflood the core • Break size will influence operator actions to control boric acid precipitation and the need for operator action to switch to hot/cold leg injection • Peak linear heat generation rate (PLHGR) is typically limited by the LBLOCA and can be quite restrictive for earlier generation nuclear plants • Limits on Axial Shape Index (ASI) or axial offset to preclude highly skewed power shapes in the top of the core are important and established by the small break LOCA analyses, often resulting in a loss of operating margin • Small Break LOCA's influence the capacity for high pressure safety injection pumps and the need for loading charging pumps on the diesel generators to meet 10 CFR 50.46 criteria for power uprates • LOCA break size will influence sizing of the atmospheric dump valves and PORVs • The most positive moderator temperature coefficients (MTC's) are sometimes set by the LOCA analyses because a positive MTC will produce moderator-density feedback which can result in an initial over-power following opening of the break • The realignment to establish long term cooling is determined by the need to control boric acid precipitation after a LOCA. The Boiling water Reactor (BWR) ECCS systems are typically more diverse as compared to PWR designs, and are not impacted as significantly by LOCA break size. For BWRs, an important factor is diesel generator start times and load sequencing delays which can burden diesel generator reliability. The ECCS and ECCS components are tested during the initial plant startup test program to ensure that the system can perform its intended design function. Surveillance testing is also performed periodically to confirm the assumptions of the safety analyses. Technical Specifications include limiting conditions for operation, required actions and completion times, and surveillance

		requirements for the ECCS and various ECCS components. These specifications identify the number of ECCS trains needed to be operable, and the requirements for specific ECCS components including pumps, valves, water storage tanks and other equipment. For PWRs, boron concentration requirements are also included. Plant specific Emergency Operating Procedures, which are symptom based, not event based, also provide guidance to operators for responding to LOCA's of various break sizes. The timing of operator actions is governed by the analyses required by 10 CFR 50.46 and Appendix K, and sometimes provide limited margin for operator error and equipment failures. General implications to the inspection programs are summarized as follows. To check the functionality of whole systems, the NRC assembles a team of inspectors and implements baseline inspection procedure, "Safety System Design and performance Capability." This procedure requires that a particular mitigation system be selected and reviewed in several engineering disciplines like electrical, mechanical, instrumentation, etc. The basis for selecting a system is whether it mitigates some type of accident. If the accident was a LOCA then ECCS systems would be concentrated on especially. For small break LOCAs, this would typically be the high head injection system and interfacing systems. For large break LOCAs, the systems reviewed would be the low head injection systems including any passive ECCS systems(e.g., for PWRs the accumulators). In addition, the inspectors may review the design and operational capability of support systems such as Diesel systems, cooling water, and DC systems. The systems to be reviewed are determined by looking at the results of Individual Plant Examinations (IPEs) or some equivalent means for the plants in question and determining for a particular plant which accidents are the most risk significant. Once that is determined then accident sequences are reviewed to ascertain which accident sequence is most damaging. Then the inspection will concentrate on those systems which comprise that accident sequences.
	WOG	The large break LOCA (LBLOCA) is the main driver of many of the current analytical assumptions such as no control rod insertion following a LBLOCA, ECCS hot leg switchover recirculation, break opening times, and containment sump debris generation, as well as other issues such as the ultimate heat sink temperature limitations. The margins contained in the Large Break LOCA analysis establish the testing and surveillance requirements for equipment of safety significance (e.g., fast start diesel generator testing). The LBLOCA also affects fuel design limits The Technical Specifications impose limiting conditions for operation based on the operability of components required to mitigate the design basis large break LOCA. These LCOs are often unnecessarily restrictive with respect to shutdown actions and required action times. In general, the design basis LBLOCA requirements influence plant design and operation at every level.

RUSSIA Observer	**Gidropress**	In fact, all the plant configuration (i.e., design features, EOPs, inspection and maintenance programs, etc) takes into account the current design break size (largest pipeline). However, many limitations are made with the LBB concept implementation.
	GAN	For the design basis accidents including the limiting LOCA the following design limits of fuel rod damage should not be exceeded: • ˜ fuel cladding temperature not higher then 1200 0C; • ˜ local cladding oxidation not higher then 18 % of original wall thickness; • ˜ total amount of oxidized zirconium not higher then 1 % of its initial mass in claddings. During design basis accidents the core should retain its mechanical stability and absence of deformations able to impair normal functioning of reactivity control and reactor scram devices or to impede fuel elements cooling. All equipment and pipelines of the reactor coolant circuit should withstand nondestructively dynamic loads and temperature effects arising during all DBA considered. Primary pipes should be equipped with displacement limiters to prevent unacceptable deformations due to reactive force resulting from LOCA. Limiting LOCA boundary conditions should be considered also in the design of ECCS and Containment system considering the independent failure principle. The limiting LOCA along with other DBAs should be covered by EOPs.
Slovenia Observer		Consistent with the US regulations, guides and standards .

Current regulatory framework		
		4. Is Leak-Before-Break accepted (or being considered) in your regulation? If so, what are the consequences on component or piping supports, system analysis, fuel assembly, containment…
BELGIUM		Application of Leak-Before-Break has been accepted, but limited to the reactor coolant piping. The Leak-Before-Break concept was used: - to justify the reactor pressure vessel internals under stretch-out conditions; - to remove some snubbers attached to heavy components of the reactor coolant loops (primary pumps, steam generators); - to remove some restraints on the reactor coolant loops; - to modify the supports of some heavy components (steam generators)
CANADA		LBB is not normally considered for LLOCA in Canada. LBB was accepted as a basis for not installing pipe-whip restraints for large diameter pipes in the newest CANDU station. In this case, deterministic LLOCA analysis was still required to show adequate fuel cooling and integrity of containment boundary. LBB has been accepted for certain main steam line breaks (as part of a cost-benefit argument) where there is an extensive periodic inspection program, fracture mechanics analysis and leak detection.
CZECH REPUBLIC	**SUJB Safety Authority**	It will be accepted in future
	NRI Rez	Yes. Main consequences are as follows: SSE and LLOCA are not concurrent events, lower stress state of all reactor internals and heavy component supports (RPV, MCP, SG, PRZ)

FINLAND		LBB is covered by Guide YVL 3.5: if LBB is demonstrated, pipe whip restraints may be eliminated, but no reduction in ECCS nor containment cooling capacity is allowed.
FRANCE	**IRSN**	- for existing PWR, no consideration of Leak-Before-Break is assumed. - concerning EPR, LBB could be accepted if the designers provide an acceptable demonstration methodology (see points 1 and 3 above)
	EDF SEPTEN	Not for the moment, it's under re-evaluation process inside the company is included in the design basis of new reactors
GERMANY		Leak-Before-Break is accepted on a case by case decision if certain requirements regarding the integrity of the pressure boundary are fulfilled. The different aspects are dealt with in the RSK guideline in chapter 21 which is enclosed as addenda (See attachment).
JAPAN		Yes. Introduction of an LBB concept is accepted by regulation for the austenite stainless steel pipes on the consideration for the design to the internal generated missile resulting from a piping break. Consequently, the design of piping support structure has been simplified for the newly constructed plants; Genkai-4 and Onagawa-3, and for the existing plants with SG replacing; Mihama-1, 2 and 3, Takahama-1and 2, Oi-1 and 2, Genkai-1 and 2, and Ikata-1 and 2. However, application of the LBB concept to containment design conditions, ECCS performance evaluation, exposure evaluation etc. is not accepted yet.
MEXICO		The leak before break is not considered in the present regulation. The technical specification made reference to leakage in some components or system and establish a timing to repair it.
SLOVAK REPUBLIC	**Safety Authority**	Leak-Before-Break concept is implemented at all Slovakian NPPs. In specific cases the implementation of LBB concept has been required by regulatory decision. For V-1 NPP in the decision No. 5/91 there were e.g. following requirements: - based on regulatory guidelines to declare the acceptibility of Leak-Before-Break concept, - to prove that the occurrence of break of pipeline of ID higher than 100 mm is less than 10-5 per year - to introduce leak detection systems, etc. LBB concept is applied for main coolant lines, surge lines and ECCS-to-MCL lines. For main steam lines the integrity re-assessment has been done.
	VUJE	LBB is considered for LB LOCA in Slovak NPPs. LBB concept is implemented for RCS piping with the diameter > 200 mm (RCS main circulation line, PRZ surge line, steam line inside confinement).

SPAIN		See answer to question 1
SWEDEN		The Leak-Before-Break (LBB) concept has not yet been formally accepted by SKI. However, one of the licensees has applied for the use of LBB in two of their reactors. LBB will also be addressed in SKI's general design regulations that now are being prepared. The proposed rule in these regulations state that <u>dynamic effects</u>, such as pipe whipping, missiles, associated with postulated pipe ruptures may be excluded from the design basis if - the piping system is designed so that the conditions for degradation as a result from any identifiable degradation mechanism have been reduced as far as possible, and - that any flaws (defect, crack) that despite this measures are initiated permit timely detection before a pipe rupture occur. - dynamic effects should not completely fail a safety function or the containment leak tightness. The considered procedure for assessing LBB applications will include consideration aspects such as used material, environmental conditions in all operational states and design bases accident conditions, procedures for testing and in-service inspection, equipment and systems for leak detection, etc. Global effects on containment and core cooling capacity will however still be analysed with the assumption of a failure of the largest pipe connected to the reactor pressure vessel.
SWITZERLAND		The Leak-Before-Break concept is not mentioned in the regulations, but it is being considered upon request and has been accepted for one PWR plant. The consequences were that a 10% break of a main coolant pipe break was considered for the analysis of components (e.g. the reactor pressure vessel sup-port structure) and piping supports and in the evaluation of the mechanical consequences of a pipe break on safety relevant systems. Also, Leak-Before-Break has been taken into account in the probabilistic safety analysis (PSA) of the plant. Leak-Before-Break is under evaluation for a second PWR.
UNITED KINGDOM		No, Leak Before-Break is not considered as a primary safety argument.

USA	**US NRC**	The use of leak-before-break (LBB) analysis in the regulation is stipulated in General Design Criteria (GDC) 4 of Appendix A to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50). GDC 4 states, "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." So far, NRC has approved applications of the LBB methodology to primary coolant loop, reactor coolant bypass piping, pressurizer surge, accumulator, safety injection, and residual heat removal (RHR) lines. As a consequence of the NRC approvals, licensees have removed pipe whip restraints and jet impingement barriers from a number of systems. On the system-analysis side, licensees have also been permitted to modify the licensing basis for their facilities by excluding from consideration, asymmetric LOCA blowdown loads on reactor primary coolant systems and dynamic loads on steam generator internals. However, the NRC has not permitted the exclusion of such dynamic effects in determining the design requirements for emergency core cooling systems (ECCS), environmental qualification (EQ) of safety related electrical and mechanical equipment, and containment.

	WOG	General Design Criterion (GDC) 4 of Appendix A to 10CFR50 allows the use leak-before-break (LBB) to exclude the dynamic effects of missiles, pipe whipping and discharging fluids associated with postulated piping failures (LOCAs) on structures, systems and components important to safety. The 1984 GDC 4 rulemaking explicitly excluded the application of LBB to the emergency core cooling system (ECCS) and containment. Rule changes proposed by the U.S. nuclear power industry would add a provision for an alternate design basis break size. No specific alternate break size is prescribed, nor is there a prescribed analytical method specified to analyze the event. The WOG is planning to apply probabilistic fracture mechanics - LBB methods to support the selection of an alternate maximum break size.
RUSSIA Observer	**Gidropress**	The LBB concept is not introduced in the high-level safety standards (like OPB) but accepted by the document RD 95 10547-99 “Guidelines on LBB safety concept application to NPP pipelines”. This document is approved by Minatom RF and agreed by regulatory body. According to this document, loads induced by main pipeline break are not being taken into account, the pipeline restraints are not installed, reactor internals, fuel assembly and in-containment structures are designed to withstand to break of those pipes where the LBB concept is not applicable, etc. However, design basis for ECCS and containment itself are kept the same (i.e., instantaneous break of largest primary pipe).
	GAN	LBB is accepted in regulation and is under implementation at NPPs. However it does not cause changes of the regulatory requirements applied to the design of NPP components. At the same time for NPPs of the first generation LBB is considered as a measure able to compensate the inconsistency between current LOCA definition and limiting LOCA size accepted in the original design.
Slovenia Observer		Yes. It has been accepted only for the analysis in connection with S/G replacement and power uprate of the NPP. We didn't allow any proposed physical modification such as snubber reduction or pipe whip restrains (shims) removal. The number of piping supports and pipe whip restrains remained the same as before the acceptance of LBB. LBB has been applied to the Reactor Coolant Loops, Surge Line, RHR Lines and SI Accumulator Lines. New mechanical analysis eliminating the 11 large primary loop breaks (> 6 inch) from the original analysis (Hydraulic Forcing Functions).

Current technical framework		
		1. What technical issues are currently of concerns for Structures, Systems and Components (associated with the current LOCA definition)? (please list)
BELGIUM		Containment leak-tightness (in relation with the source term definition Recirculation (sump clogging issue) Dynamic response requested for pressure sensors in the containment
CANADA		Canada has a number of unresolved technical concerns associated with LLOCA. Many are related to the positive void coefficient of CANDU reactors: magnitude of void reactivity error allowance, shutdown system performance, fuel behaviour in power pulse. Some relate to CANDU specific features: pressure tube ballooning, moderator as a heat sink. Others relate to basic thermal-hydraulics: uncertainty in core voiding rate, CHF, post dry-out heat transfer.
CZECH REPUBLIC	**SUJB Safety Authority**	Fuel integrity (also high burn up fuel integrity), PTS, contaiment integrity, bubble condenser integrity (VVER440/213)
	NRI Rez	Stability of heavy components, jet impingement forces, design and maintenence of ECCS, environmental gualification of I&c, ECCS sump screen blocking risk
FINLAND		There are no major concerns now that the sump recirculation issue has been resolved.

FRANCE	**IRSN**	Sump clogging is an important issue, although it does not concern only large LOCA, but also intermediate and in some cases small LOCA. See the point 3 response above
	EDF SEPTEN	- structures and components concerned: RPV, SG divided plate, RCP flywheel, support of components and piping systems, and containment pressure - all safety systems - fuel
GERMANY		There are presently no open technical issues associated with the current LOCA definition, although uncertainties related to the fracture mode of bimetallic welds are difficult to treat.
JAPAN		An extensive argument on the LBLOCA related subjects has not been started, while resolution and modernization of maintenance rules for SSC are pursued actively in Japan.
MEXICO		Please see question 3 of Current Regulatory Framework
SLOVAK REPUBLIC	**Safety Authority**	In Slovakia, there is under way a programme for the modernisation and safety upgrading of Bohunice V-2 NPP. The main aim of this programme, expect of modernisation is also to fully implement all recommendations from the IAEA document No. IAEA-EBP-WWER-03 – Safety Issues and Their Ranking for WWER-440 Model 213 Nuclear Power Plants. Within this programme, following tasks are under way: - analyses of all groups of initiating events according to the IAEA-EBP-WWER-01 document including of all LOCAs up to 2x500 mm with respect the modifications of safety systems during reconstruction - qualification of NDT systems For V-1 NPP there are under way accident analyses following from the aim to use modified fuel assemblies.

	VUJE	Technical concerns associated with LB LOCA in VUJE Trnava are related to fuel behaviour, vessel integrity and hermetic compartment integrity during the LOCA accident. a) Analyses of the fuel rod cladding stress-strain behaviour (with minimal and maximal thermal power), determination of the internal pressure time histories and location of the cladding axial node with maximal deformation, b) Statistical determination of the number of ruptured rods and fuel channel blockage in the analysed fuel rods, c) validation of modelling of thermal mechanical fuel behaviour and its simulation in the system codes (e.g. RELAP5), d) vessel integrity analyses, e) Hermetic compartment integrity analyses.

SPAIN		(No answer provided)
SWEDEN		For the BWR with internal re-circulation pumps a hypothetical break size of 80 cm^2 has been basis for the licence and the design of emergency core cooling system. However, one of the licensees with this type of reactor are planning to remove the core sprinkling part of the system.
SWITZERLAND		-The behaviour of highly burnt fuel under LOCA conditions is not well known. By means of experiments the validity of the present LOCA fuel safety criteria should be confirmed or new criteria should be established. -One plant has requested that a 10% main coolant pipe break can be considered for the analysis of the dynamic effects of a LOCA on the fuel assemblies and reactor internals, if Leak-Before-Break is accepted. This request is still under review.
UNITED KINGDOM		None.

USA	**US NRC**	Some of the technical issues associated with SSCs include: - The emergency diesel generator start time for LBLOCA may impose unnecessary stress and operational limitations. - Adverse impact on reliability/availability of EDGs - Adverse impact on reliability of the ECCS MOVs - Increased probability of inadvertent injection from accumulators - Focus on reliability of SSCs that are risk significant (including mitigation of LOCAs at shutdown) **Containment sump debris generation and sump blockage.** **Equipment qualification requirements.**
	WOG	No control rod insertion following a LBLOCA resulting in higher boron concentrations / Switchover from ECCS cold leg to hot leg injection to address boron precipitation / Containment sump debris generation and clogging of the sump / Fast diesel generator starts (10 sec.) to mitigate the LBLOCA with a coincident loss of off-site power / Containment design changes to the existing containment structural design and limits are not envisioned, although benefits in operational margin are expected. / Equipment qualification - other breaks that would still be in the design basis will still require EQ limits similar to the existing limits / Ultimate heat sink requirements and associated temperature limits are associated with the LBLOCA / The containment leakage rate test pressure Pa is currently based on the LBLOCA resultant pressure. Reducing the break size would reduce Pa for the containment leakage rate testing.
RUSSIA Observer	**Gidropress**	Even applying LBB concept (i.e., for 200-300 mm breaks as the design basis LOCA), a significant number of support-constraints for pipes, additional supports for equipment, essential enforcement of in-containment constructions are still needed. It is difficult to substantiate the equipment strength due to absence of the relevant normative basis, and therefore the costly experimental works are often needed.
	GAN	• Insufficiency of primary pipe supports and displacement limiters at operating NPP units • Clogging of filtering grids in the containment sump • inconsistency between current LOCA definition and limiting LOCA size accepted in the original design of old NPP units concerning ECCS capacity
Slovenia Observer		None really. The question of thermal stratification (on the surge line, normal and alternate charging check valves, auxiliary spray line, the RHR isolation valve and on the pressurizer nozzle and pressurizer spray nozzle) and thermal variations in general is followed closely, for this purpose also additional sensors have been installed at these lines. Besides that, the sensitivity and reliability of leak monitoring system could be improved.

Current technical framework

		2. What technical issues would be needed to be addressed by Researchers to support the regulatory decision process to change the current LOCA definition?
BELGIUM		Not applicable
CANADA		This would depend on the proposed changes to the LLOCA definition. To allow the use of the Best Estimate Analysis + Uncertainty, more information would be needed on sensitivity to important variables and the range of key parameter uncertainties. To accept LBB redefinition, then considerable work would be needed to develop the inspection programs, fracture mechanics and leak detection parts of a LBB methodology and there would be an ongoing program of augmented periodic inspections. To move to a risk-informed decision-making regime, clear ground rules are needed for integrating traditional deterministic, defence-in-depth design concepts within a risk-informed framework.
CZECH REPUBLIC	**SUJB Safety Authority**	Evaluation of LBLOCA frequency, LBB, use best estimate computer code+uncertanties (BE method)
	NRI Rez	Recent operational experiences with large diameter piping, qualified ISI, progress in fracture mechanics, progress in monitoring systems
FINLAND		All aspects of nuclear plant design which were based or involved assumptions derived from the LOCA envelope. See "Current Regulatory Framework" for a short list of most obvious factors and issues.
FRANCE	**IRSN**	Realistic LOCA frequencies assessment is an important issue.
	EDF SEPTEN	– refreshing all the probabilistic analysis of leak and break of RCS piping, included all the existing knowledge of degradation mechanism . Pre-workshop survey – redefining the Large-break Loss-of-coolant Accident
GERMANY		There are no plants to change current LOCA definitions. Some research work has been performed in the past regarding the requalification of older primary pressure components to the basic safety standard.

JAPAN		We address the following major three technical issues important to consider. (1) Evaluation of LBLOCA frequency a. The improvement of precision for evaluation by Probabilistic Failure Mechanics (PFM) with operating experience data is indispensable. b. In connection with LBB, evaluation of the reliability of inspection and detection is the biggest problem and it also serves as an important factor for the evaluation by PFM. Besides, this depends greatly on human and an organization factor as an example of Davis-Besse event. c. For the evaluation of occurrence frequency of LB-LOCA, we think it is required that the check of taking into consideration all important LOCA generating mechanisms, such as an earthquake and plant aging, by synthetic analysis of the actual event about the leakage from pressure vessel or piping, and modeling for each mechanism of LOCA generation. (2) Evaluation of maximum allowable pipe break size The technical issue is whether or not we can calculate the risk profile with sufficient precision along with the break size of LB-LOCA which is one of the DBEs. When the break size is altered, the plant design changes and then the risk may largely change. There might exist many difficulties to do it. First, we might need to establish a new DBE for the containment design which is largely dependent on 200% LB-LOCA at present. (3) Best estimate evaluation method and uncertainty In the evaluation analysis of PFM, since the greatest load to piping is considered to be an earthquake, it is essentially thought that uncertainty is very large. If the uncertainty of frequency evaluation is taken into consideration, in parallel to the evaluation of frequency, it might be important to confirm that the risk contribution of LOCA exceeding DBE is small. In that case, we think that the accident progression analysis which combined the best estimate code with uncertainty analysis and the knowledge about the coolability for damaged reactor core derived from the severe accident research is important.
MEXICO		The Probabilistic Safety Assessment , ECCS cooling performance are some issues to be considered in case of change in the definition of LOCA.
SLOVAK REPUBLIC	**Safety Authority**	The precondition for a safe operation of NPP is the long term assurance of structural integrity of components. Therefore we consider following technical issues as important for the regulatory decision making: - precising and validation of procedures for the integrity and lifetime assessment (at present, e.g. EC Project VERLIFE) - precising of monitoring methods for the ageing of structural material of primary circuit piping

	VUJE	First of all, for the small “nuclear” countries (Slovakia, Czech Republic, Hungary, etc.), the changes of current LOCA definition depend on international cooperation. When the process of LB LOCA redefining is to be supported by OECD Nuclear Energy Agency as well as International Atomic Energy Agency the Slovak Nuclear Regulatory Authority would like attend this process.
SPAIN		Nuclear fuel performance in a LOCA (including high burnup effects)
SWEDEN		No answers provided
SWITZERLAND		LOCA analyses must be performed on a conservative basis, for which the application of the USNRC requirements (10 CFR §50.46 and Appendix K) is accepted although not compulsory. Best-estimate calculations with conservative initial and boundary conditions or comprehensive uncertainty analysis would support the regulatory decision making process for plants with small safety margins.
UNITED KINGDOM		Elimination of full guillotine break for the primary coolant loop pipework would require a demonstrable level of structural integrity analysis, or a physical limitation on pipe displacement which would limit the discharge flow area, i.e. plant specific. No “research” is required.
USA	**US NRC**	The following are some of the technical issues which need to be addressed by researchers to support the regulatory decision making process to change the current LOCA definition. • Determination of the small-break, medium-break, and large-break LOCA frequency distributions and failure mechanisms which reflect potential changes in the expected LOCA frequencies and consequences • Determination of the relationship between break size and expected event frequency for large primary system pipes (> 150 mm diameter) to assess the feasibility of redefining the design break size. • Development of the break size frequency spectrum for LBLOCAs considering PRA quality and results to develop suitable generic and/or plant specific approaches which should be the basis for any rule change. • Determination that the plant can deal with certain events that were not addressed previously in the design basis because they were bounded by the LBLOCA. Examples of such events include sources of LOCAs other than pipe breaks (e.g.,

		steam manway failure). • See Item 3b of Consideration for the future for discussions of analytical tools and models which may require development
	WOG	LBLOCA initiating event frequencies Probabilistic fracture mechanics (PFM) to support both frequency and break size determination.
RUSSIA Observer	**Gidropress**	One of the most important problems is to prove very low probability of large pipeline break. Although the LBB is substantiated by designer and accepted by regulatory body, the probabilistic analyses have displayed small but not negligible probability of large-scale rupture of a large size pipe. As the nearest task, it would be useful to develop and agree with the regulator the method that would allow to limit the number of assumed locations for pipe breaks (to define the points where the protective measures are really needed). As the long-term task, the substantiation and implementation of LBB concept for smaller pipes (at least, up to 150 mm) would be extremely useful.
	GAN	No answers provided
Slovenia Observer		The relevant ones. E.g.: New ECCS design criteria,…

Consideration for the future

		1. Are you considering changes in your regulation? For operating plants? For future plants?
BELGIUM		No.
CANADA		The Canadian regulator has no current plans to remove LLOCA from the design basis set of accidents, either for current or future reactors. However, the regulator is investigating ways to develop a more balanced approach to LLOCA, including using risk insights. Licensees are proposing best estimate analysis plus uncertainty methodology. Regulator does not dictate an analysis methodology but will evaluate the licensee's proposals, accepting them if they demonstrate safety targets are met with sufficient confidence. The position is the same for operating and future plants. The Canadian industry is considering proposing changes which would be aligned with movement towards a risk-informed decision making regime. Such proposed changes will then initiate discussions with the regulator.
CZECH REPUBLIC	**SUJB Safety Authority**	No.
	NRI Rez	No answers provided
FINLAND		No.
FRANCE	**IRSN**	- For operating plants: in the framework of periodic safety reassessment, EDF has recently presented a proposal for changing the LLOCA definition, but due to a lack of methodology which could be used for the corresponding demonstration, the proposal was not accepted by the Safety Authority. - Status of EPR: See the point 1 response above
	EDF SEPTEN	- for operating plant, internal review is under progress - for future plants included as design basis and no LBLOCA in the design transient list
GERMANY		There are no considerations to change the regulations.
JAPAN		Since we are not regulators, we cannot answer this question. However, the demand for rationalizing safety evaluation is increasing in Japan: such as an application of RIR, safety goal, and performance indicators.

MEXICO		Up to date there is not any consideration to change the actual regulation
SLOVAK REPUBLIC	**Safety Authority**	At present we do not consider any changes in regulation related to LB LOCA definition. But in case that there are enough evidences for the LB LOCA re-definition, the definition of limiting break size in the current legislation could be changed for both, operating and future plants.
	VUJE	No.
SPAIN		We currently follow the regulation of our vendors (USA and Germany)
SWEDEN		As mentioned in the answer to question no. 1 and 3 SKI is preparing general design regulations, which will applicable to all presently operation nuclear power plants in Sweden. The main purpose of issuing these new regulations is to strengthening the defence-in depth, particular for the older reactors by require more use of redundancy and diversity of safety systems. No future plants are presently considered in Sweden.
SWITZERLAND		No change to the LOCA break size definition is foreseen at the present time and no corresponding application has been submitted by the owners of the nuclear power plants.
UNITED KINGDOM		No.
USA	**US NRC**	Changes are currently being considered in the regulation (10 CFR 50.46) for operating plants. It is intended that any rulemaking associated with the proposed changes will be a voluntary alternative to the current rules. The NRC staff is developing a “framework” for the future plants. The staff expects to follow the similar approach as that taken for this initiative.
	WOG	The petition for rule making submitted by the NEI could be applied to either operating plants or future plants. The proposed change is simply worded and provides for an optional alternate break size that would have to be approved by the NRC. The proposed rule would not prescribe the break size. Additional risk-informed criteria are being considered for inclusion in the proposed rule that would give the NRC a standard for acceptance based on LOCA contribution to risk. There is no technical reason that the same set of criteria cannot be applied to existing and new plants.
RUSSIA Observer	**Gidropress**	Since the current safety standards contain no direct indication on the maximum size of the design basis LOCA, the changes (to decrease the design basis LOCA size) are not expected.

	GAN	At the moment there is no plans to change the regulation in view of limiting LOCA definition.
Slovenia Observer		Not regarding design basis. For operating plants? Not at the moment. For future plants? N/A.

Consideration for the future

		2. What would be the incentives? From the regulatory viewpoint? From the Industry viewpoint?
BELGIUM		Not applicable
CANADA		Canadian regulator sees some incentive to remove some LLOCA breaks from design basis: - concentrate review effort on more risk-significant events - refocus research effort - reduce doses to workers (if gain from reduced testing is greater than increase from inspection). Industry could gain from the above and also: - relaxation of operating restrictions - removal of power restrictions - better utilization of resources on higher importance safety issues.
CZECH REPUBLIC	**SUJB Safety Authority**	N/A
	NRI Rez	From the Industry viewpoint cost saving
FINLAND		There are none from either side.
FRANCE	**IRSN**	- for the regulatory : LBB could lead to an improvement of leak detection systems. -for the industry : less constraints for maintenance of whip restraints and for operation evolutions (fuel cycle, power changes….).
	EDF SEPTEN	No answers provided
GERMANY		No answers provided
JAPAN		We think that it is a common recognition of the industry and regulator to remove unnecessary conservatives. The best use of safety margin will be also another incentive.

MEXICO		N/A
SLOVAK REPUBLIC	**Safety Authority**	As the main precondition for the LB LOCA re-definition from the regulatory point of view we consider to perform the re-classification of initiating events according to the probability of theirs occurrence.
	VUJE	To change the current LB LOCA definition should be based on risk analyses for new generation of NPPs.
SPAIN		Better knowledge of the LOCA concerns. In Industry: economic reasons, drawing more resources to safety. Regulatory body: more resources to other safety issues.
SWEDEN		See answer to question no. 1 above
SWITZERLAND		Not applicable.
UNITED KINGDOM		None
USA	**US NRC**	Regulatory viewpoint: The Commission, in its policy statement on PRA (Federal Register, Vol. 60, No. 158, August 16, 1995, p. 42622) has stated that: “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 traditional defense-in-depth philosophy.” The Commission, in order to move forward to implement the policy statement, in a White Paper on Risk-Informed and Performance-Based Regulation (Staff Requirements - SECY-98-144, March 1, 1999), has defined, in part, “Risk-Informed Approach” as: “A “risk-informed” approach to regulatory decision-making represents a philosophy whereby risk insights are considered together with other factors to establish requirements <u>that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety.”</u> With the above over-arching principles, consideration of the following factors, and input from the stakeholders provide incentive for considering requirements for the analysis of design basis LOCAs, and some of the associated requirements as candidates for risk-informed regulation. The factors include: • The design basis LOCA and requirements were established in the nineteen-sixties with very little operating experience. Since then, there has been several thousand years of collective operational experience, with no occurrence of a large-break LOCA.

		• Risk assessments show that LBLOCAs are not major contributors to risk. • There has also been extensive research including research in fuel behavior and severe accidents which allow us to conduct more realistic analysis and evaluation of margins. • Both the utility and regulators devote significant resources to assure that all of the requirements resulting from the DEGB are met. The other specific factors (e.g., improved reliability of EDGs and fast-operating valves whose operating times are derived from LBLOCA analysis) which result in safety enhancements, operational flexibility, and other benefits are discussed in various parts of this survey. In summary, the regulatory incentive is better focus of both the licensee and the regulator on more safety significant initiators with reduction of unnecessary burden. Industry viewpoint The incentives from the industry viewpoint are numerous. The Westinghouse Owners Group (WOG) anticipates that economic benefits will accrue from potential power uprates, relaxation of DG start times, increases in peaking factors, reduced analysis costs, etc. The Boiling Water Reactor Owners Group (BWROG) expects to reduce costs associated with testing (e.g., integrated safety injection/loss of offsite power test, DG load shed and logic test, 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), and operation (e.g., extension of DG and MOV allowed outage times). See additional tabular information in Item 4.

	WOG	The safety benefit is in terms of focusing resources in areas of greater risk significance, such as SBLOCA, and the potential for increased reliability and availability due to more realistic surveillance requirements and fewer operational restrictions. There is also a safety benefit in the area of operator training and post accident operator actions, for example eliminating the requirement for hot leg switchover. There are additional safety benefits from plant changes that could result from the revised regulation. For example, ECCS flow could be balanced to better mitigate the more probable small break LOCA. The safety margin is maintained by the Risk-Informed approach, which utilizes the Defense-In-Depth philosophy that underlies the safety regulations (SECY-98-300). By utilizing risk insights, resources can be directed to areas of greater risk significance, providing an increase in overall plant safety. The fundamental design basis concept is not eliminated by redefining the LBLOCA. The definition of LOCA is being revised consistent with operating experience and current technologies. There would be a net increase in plant safety due to realistic equipment testing requirements and increased reliability, and the focusing of resources in more risk significant areas, such as SBLOCA. The existing design basis LBLOCA regulation also imposes an artificial restraint on future development. The current state of knowledge, technology, analytical techniques, and operating experience provide a basis for regulatory change.
RUSSIA Observer	**Gidropress**	Probably, the only incentive would be the plant economics. Therefore it is difficult to expect that regulatory body will initiate the changes with respect to design basis LOCA decreasing. As for the industry, VVER NPP designers try to decrease the costs incurred by current size of design LOCA by applying new configurations of the relevant safety systems (implementation of passive features, assignment of normal and safety functions to one system, combination of low- and high pressure safety injection, etc). Probably, the utility initiatives would be the most effective for re-consideration of design LOCA size (since the inspection and maintenance of safety systems may essentially effect to the capital investments and generation costs). However, there are no such indications even in the recent relevant documents (like EURD).
	GAN	N/A
Slovenia Observer		From the regulatory viewpoint? None From the Industry viewpoint? Nothing at the moment. Dose and cost savings after possible snubber reduction programme.

Consideration for the future		
		3. If you are considering replacing large break LOCA by a smaller break size within the design basis, some degree of core damage, short of core melt resulting in vessel failure, may be expected if a large break LOCA actually occurred.
		3.a How would you establish performance requirements for the emergency cooling systems in order to provide some assurance that damage following a large break LOCAs can still be mitigated before vessel failure? What is the technical basis?
BELGIUM		N/A
CANADA		Canada is not considering replacing LLOCA.
CZECH REPUBLIC	**SUJB Safety Authority**	Czech Republic is not considering replacing LBLOCA. The largest or limiting break size assumed as design basis (LOCA) is double Ended Guillotine LB LOCA (2x500 for VVER 440/213 or 2x850 for VVER 1000) . All spectrum of LOCA Analyses are required for Safety Analysis.
	NRI Rez	Optimize passive ECCS
FINLAND		Cannot answer; too sensitive dependence on desired confidence level for the “some assurance” and also meaning of “mitigation”. Performance requirements in terms of release limits etc can be set largely independent of technical basis. Technical basis is poorer than “generic” severe accident knowledge base (which has focused on later phases where state-of-the-art is now much more mature than the early transient phases, between beyond-1200°C PCTs and core collapse).
FRANCE	**IRSN**	For the existing plants, there is no decision for replacing LLOCA by a smaller break size. For the EPR project, if large break LOCA is replaced by a smaller break size within the design basis, no safety objectives have been yet discussed concerning large break.
	EDF SEPTEN	- Can we assure safety injection in RPV under LOCA condition without LOCA in the design transient list? Not sure, it has to be analyzed
GERMANY		At the level of PSA a large break LOCA as an initiating event is treated in the same way as a failure of large components. This is based on integrity analysis being performed according to the state of the art. As far as structural reliability models are applied the validation of such codes are not satisfactory up to now for both types of components, pipings as well as vessels.

JAPAN		(1) We think that the LB-LOCA should be regarded as residual risk if the occurrence frequency confirmed to be sufficiently low with technical basis. This might be the same reason as why RV bottom failure is not considered in the ECCS design basis. (2) It is not rational to equip with too much safety system for the event with very low occurrence frequency. If a further rationalization is sought for mitigation potential against an LB LOCA, application of the sate-of -the-art knowledge about IVR and best use of mitigation measures such as mitigation accident management will be of value.
MEXICO		The emergency core cooling system (ECCS) must be designed in accordance with and acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accident of different sizes, locations and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accident are calculated. The model must to included sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during loss-of-coolant accident. The uncertainty in the calculated results must be estimated.
SLOVAK REPUBLIC	**Safety Authority**	At present, thermal-hydraulic analysis done for all LOCA sizes (up to 2x500 mm) showed that even in the case of a single failure of ECCS these transients do not lead to the core damage or to the reactor pressure vessel failure.
	VUJE	N/A
SPAIN		The treatment should follow the same path as other scenarios like this, mainly those of level 2 PSA. The acceptance criteria of the frequency of exceedance of damage to the vessel and/or containment should be more precise. The enveloping character of SBLOCA relative to LBLOCA should be explored in detail, reviewing all the aspects of design including containment and radiological protection concerns.
SWEDEN		No answers provided
SWITZERLAND		No corresponding considerations have been performed at HSK.
UNITED KINGDOM		N/A
USA	**US NRC**	The design criterion for the ECCS (i.e., hardware) is expected to still be based on the break of the largest pipe, but the ancillary requirements (**e.g**., technical specifications for safety injection flow rates and inspection frequencies for accumulator condition) will be subject to risk-informed modification. That is, the ECCS will maintain its mitigation function for all break-size LOCAs. The redefinition would also provide more operational flexibility by allowing power uprates, higher peaking factors and assembly discharge burn-up. In addition, the 10 CFR 50.46 acceptance criteria are being re-examined along with a re-definition of the maximum break size. A voluntary alternative to the existing criteria is under consideration that would maintain the requirement for coolable geometry while

		making specific requirements for fuel and fuel cladding performance based.
	WOG	There may be a break size threshold that can be acknowledged as comparable to vessel failure in probability. The double-ended break of the largest RCS pipe is likely to fall in this category. This spectrum of essentially incredible breaks could be addressed in severe accident management as a beyond-design-basis event. Revised PSA success criteria would be established and the capability to mitigate the LBLOCA as a severe accident would be retained.
RUSSIA Observer	**Gidropress**	The TMI case has proved that the in-vessel retention of the essentially damaged core is possible by injection of water. Even if the ECCS will be designed for smaller LOCA size (e.g., for 200 mm break), it will ensure the (already damaged) core coolability in the mid- and long-term periods of LBLOCA. The calculated core damage during short-term is expected to be limited (since LBLOCA becomes beyond-DBA, we will apply the realistic approach instead of conservative deterministic analysis). So, the existing technical basis would be sufficient to prevent vessel failure (if such a requirement will accompany the re-definition of design LOCA).
	GAN	N/A
Slovenia Observer		N/A

Consideration for the future		
		3. If you are considering replacing large break LOCA by a smaller break size within the design basis, some degree of core damage, short of core melt resulting in vessel failure, may be expected if a large break LOCA actually occurred.
		3.b Are currently available computer codes and models adequate for the required analyses or new tools will have to be developed?
BELGIUM		N/A
CANADA		Canada is not considering replacing LLOCA.
CZECH REPUBLIC	**SUJB Safety Authority**	N/A
	NRI Rez	No answers provided
FINLAND		Cannot answer – depends too much on the ambition level based on previous question.
FRANCE	**IRSN**	See 3a
	EDF SEPTEN	- Partially yes for mechanical aspect; I don't know for TH aspects
GERMANY		At the level of PSA a large break LOCA as an initiating event is treated in the same way as a failure of large components. This is based on integrity analysis being performed according to the state of the art. As far as structural reliability models are applied the validation of such codes are not satisfactory up to now for both types of components, pipings as well as vessels.

JAPAN		(1) Yes. We think that the current tool is adequate for the analysis of the process at the initial phase of the core damage, however, not sufficient for the process in late phase core damage. (2) However, we do not think it important to develop further the code with detailed modelling. Rather we should consider the approach of re-defining the conditions that can maintain reactor core cooling by the use of existing codes and experimental data. (3) In structure relation, calculation codes which evaluate piping (also including RPV) reliability should be improved. For example, we regard the failure modes (ratcheting, etc.) which cannot be treated by the simple fracture mechanics, the latest aging phenomenon (not only IGSCC but PWSCC), wall thinning due to flaw assisted corrosion, the failure criteria by the earthquake, etc. should be taken into consideration in the code development.
MEXICO		The currently computer codes available in the nuclear industry are considered adequate to simulated different breaks LOCA. However in case of evolution of the accident to core damage, the experiments in this field have been show that the models of the computer codes need to be improve. The properties of the material under high temperatures condition, hydrogen production, core degradation, coolability of the debris, are some topics that should be investigated.
SLOVAK REPUBLIC	**Safety Authority**	At present, for the thermal-hydraulic analysis the RELAP 5 code is used and for the reactor pressure vessel integrity assessment the ADINA code is used.
	VUJE	N/A
SPAIN		Codes and models seem to be evolving in the right direction
SWEDEN		No answers provided
SWITZERLAND		Not applicable. HSK has not performed any corresponding code evaluations.
UNITED KINGDOM		N/A

USA	**US NRC**	Currently available codes should be capable of handling new LBLOCA definition. However, new assessment and review will be necessary as applicants modify plant operating conditions to take advantage of the proposed change. Codes may be used outside of their approved range of applicability. Because the large break LOCA analysis has had such a dominant impact on plant operating conditions, much of the prior experimental research has been directed at understanding large break LOCA phenomena and in demonstrating safety margins for reactor systems subject to a large LOCA. Test results obtained from major integral test facilities under small break LOCA conditions may not be representative of the expected transients in reactor systems that have been significantly uprated and assume more realistic safety injection flow rates. Thus, the existing experimental database for smaller break sizes at new plant operating conditions will need to be re-examined. Improvement in PRA models may be needed to incorporate passive system failure modes. Improvements to probabilistic fracture mechanics codes are needed to model recent degradation mechanisms to estimate failure frequencies and break sizes for piping systems. Similarly, models are needed to estimate frequencies and break sizes for non-pipe failure LOCAs.
	WOG	It is envisioned that currently available codes and models would be used to analyze the LBLOCA as a severe accident.
RUSSIA Observer	**Gidropress**	Yes, available codes (like RELAP/SCDAP, MELCOR, similar Russian codes) are as whole applicable for such analyses. Probably, limited changes would be needed (e.g., development and validation of more accurate models for core debris reflooding or corium-vessel interaction).
	GAN	N/A
Slovenia Observer		N/A

Other issues		
		What other issues concerning LOCA do you feel should be discussed during the workshop?
BELGIUM		Redefining the LOCA is not just a matter of some probabilistic calculations together with some break preclusion considerations. A broad discussion on all safety implications, with a defense-in-depth approach, is needed. We expect that the workshop will allow the relevant topics to be identified and put in perspective.
CANADA		Canada does not wish to add other issues to the agenda.
CZECH REPUBLIC	**SUJB Safety Authority**	N/A
	NRI Rez	No Break Zones
FINLAND		Adequate reference to sump / suction strainer performance issues (and related ongoing work such as the forthcoming NEA Workshop on the issues) should be made. This is an example where much more (also regulatory) attention is needed to attain a consistent set of safety requirements (consistency between requirements on design, demonstration, operational surveillance, maintenance, etc).
FRANCE	**IRSN**	- issues concerning LOCA studies methodology (assumptions, boundary conditions, codes models, criteria…). - generally speaking: what could be the consequences for safety (negative and/or positive).
	EDF SEPTEN	- justification of LBB with different type of possible degradation mechanisms for 40 or 60 years of operation - leak detection capability requirements - RI-ISI results and realistic ISI program
GERMANY		No answers provided
JAPAN		Approaches practicable to resolve the issues; roles of regulator, researchers and the industry. Such issues as availability of plant performance data are of interfacing concerns. Also possible adverse effects on operation or maintenance originated from current LB-LOCA assumptions, or adequate resource distribution to enhance safety from the risk perspectives.

MEXICO		N/A
SLOVAK REPUBLIC	**Safety Authority**	Long term structural integrity of primary circuit piping.
	VUJE	Slovakia does not wish to add other issues to the agenda.
SPAIN		Risk-informed regulation / Consequences or removing LBLOCA from design basis envelope / Best-estimate LOCA analyses / Advanced designs concerns, and feedback to our current technology.
SWEDEN		No answers provided
SWITZERLAND		Evaluation of the frequency of a large break LOCA and even of an excessive LOCA (break of the reactor pressure vessel).
UNITED KINGDOM		If appropriate, High burn up fuel behaviour under LOCAs conditions.
USA	**US NRC**	“Realistic” operator response assumptions. Contribution of non-pipe-break LOCAs. PRA Quality and completeness Elements of the new requirement (a risk-informed alternative to the maximum LOCA break size) including configuration control during all modes of operations.
	WOG	Possible pilot submittals to demonstrate the feasibility of the rule change
RUSSIA Observer	**Gidropress**	What would be the “mechanistical” basis to assign the scale (size) of design LOCA (i.e., what processes/phenomena in the RCS pressure boundary might lead to sudden leak of essential size and how to calculate those processes/phenomena)? Or the design LOCA size is solely an administrative decision?
	GAN	N/A
Slovenia Observer		No Suggestions.

Appendix 1

Information provided by the USNRC to complement the answer on question future 2

Consideration for the future		
	2. What would be the incentives? From the regulatory viewpoint? From the Industry viewpoint?	
	Industry viewpoint The following list is from an industry letter to NRC	
	Item	***Safety Benefit***
	a) Accumulator 1) Decrease the number of accumulators required to be operable 2) Parameters (boron concentration, water volume, cover pressure) - relax acceptable parameter range. 3) Increase AOTs	Reduced chance for inadvertent injection from accumulator. (Not very likely, but more probable than a LBLOCA) Revision of Tech. Specs. shutdown requirements associated with accumulators would reduce likelihood of forced shutdown and resulting thermal cycle on plant. More realistic Tech. Specs eases operational burdens enabling operators to better focus on safety significant activities Reduces the potential for unnecessary plant shutdowns and reduces the number of operational and thermal transients Wider accumulator parameter bands would reduce periodic adjustments and thus the chances for ECCS valve misalignment

	b) Diesel Generator Start Time Increase (Expand to all ECCS response times.) requirements.	Reduced wear and tear on diesel from more reasonable testing Increased diesel reliability - less need for invasive troubleshooting. Reduces the potential for maintenance errors which could result in challenges to the plant's safety systems by reducing the frequency of maintenance and inspection activities.
	c) Diesel Generator Loading times	Relaxed diesel loading times during an accident response would enhance diesel reliability.
	d) Core Peaking Factor Increases (FQ or FdeltaH)	Wider peaking factor bands would result in less operator reactivity manipulations and potentially less adverse excursions

	e) Containment Design Calculations 1) Lower Peak Pressure in Analysis Many plants are limited by SLB, but Pa could be lowered since it is driven by LOCA only. 2) Evaluate elimination of sub-compartment analyses with the smaller credible maximum LOCA size.	Worker safety benefit from performing leak rate tests at lower pressure.
	f) Modify Spray System 1) Reduce the required flow rate of the sprays, and/or relax surveillance requirements. Due to defense in depth concerns, reducing the number of sprays will not be pursued.	Elimination of TS shutdown requirement associated with CS would reduce likelihood of forced shutdown and associated thermal cycle on plant. (Also see item (a) above)

	g) Modify Fan Cooler requirements. Reduce the number or increase the AOT. Consider relocating to the Technical Requirements Manual.	See items (a) and (b) above
	h) Ultimate Heat Sink - Relax Requirements. Maximum post-LOCA heat loads occur during the injection phase for plants with safety-related containment air recirculation coolers, or at the time of sump recirculation switchover for plants without containment air recirculation coolers.	Increase in operational margins reduce the likelihood of unnecessary plant shutdowns. (see (a) and (b) above)
	i) Power Uprates	
	j) ECCS Flow Issues 1) Change Flow balancing requirements 2) Decrease system resistance if LBLOCA runout is no longer a credible concern 3) For three train systems, eliminate the need for some ECCS pumps 4) Reduces sensitivity to pump degradation	Increases ECCS effectiveness for more probable events. Reduces operator burden by enhancing focus on more probable events Simplifies configuration management

	k) Operator Action Time for RWST Switchover Review the operator-training program to determine if too much emphasis is placed on the LBLOCA.	Reduced operator burden. Operator actions can be improved by better sequencing of operator actions consistent with safety-significant operational needs and scenario progressions NOTE: does not apply to all designs, some designs and plants have automatic switchover to recirculation
	l) Resolution of Sump Debris Issues (show that less (or no) debris is created with the revised LOCA break size to be analyzed)	Reduce worker exposure. ALARA: Potential for avoiding occupational dose from modifications that could result from the sump debris resolution.
	m) Resolution of GL 89-10 MOV Issues (change MOV test requirements including closure times and motive forces I valve delta pressure) May be a special treatment requirement. Note that this rule change is not required to risk inform MOV testing.	Reduce worker exposure. ALARA benefits from reduced testing scope. More reliable valves if set for more realistic requirements.
	n) Resolution of Containment Purge Valve Issues Relax mini purge valve closure times and leakage rates.	
	o) Reduce RWST Boron concentration	Improved material reliability and reduces operator action requirements
	p) Improved Fuel Design Issues Consider reducing boron and burnable poison requirements, and lowering enrichments.	

	q) Containment EQ Temperature Profile Relaxation	Increased operational margins reduce the potential for unnecessary outages.

Appendix II

Information provided by GRS (Germany) to complement the answer to question "Current Regulatory Framework 4"

RSK Guidelines will be added later

TRANSLATIONS-SAFETY CODES AND GUIDES

Amendments to **Edition 5/82**

Contents

Revised Sec. 21 of the RSK Guidelines for Pressurized Water Reactors, 3rd Edition, October 14, 1981

Uberarbeitetes Kap. 21 der RSK Leitlinien fiir Druckwasserreaktoren 3. Ausgabe, 14 Oktober 1981

Sec. 21 of the RSK Guidelines for Pressurized Water Reactors has been revised as published by the German Federal Minister of the Interior in the Bundesanzeiger
No. 106 on June l0, 1983 and
No. 104 on June 5, 1984
If there are any interpretation difficulties, the German original as published is definite.

21 Postulated Leaks and Breaks

21.1 Postulated Leak Cross Sections in the Main Coolant Pipe Including Austenitic Connection Pipes (Steel 1.4550) of Diameter > 200 mm and in the Reactor Pressure Vessel.

(1) Reaction and jet forces acting on pipes, components, component internals, and buildings.

1* Concerning the load assumption for reaction and jet forces on pipes, components, and structures a leak with a cross section of 0,l F (F = open cross section) and static outflow shall be postulated for different break positions. As load assumption for the reactor pressure vessel internals a spontaneously opening leak (linear opening behaviour, opening time 15ms) with a cross section of 0,l F in the main coolant pipes shall be postulated for different break positions.

2. In order to cope with the consequences (pressure increase in the reactor pit, release-pressure-wave acting on the reactor pressure vessel internals) of a postulated leak with a cross section of 0,l F between the reactor pressure vessel and the biological shield, measures shall be taken, e.g. double pipes in the area of the main coolant pipe penetrations through the biological shield.

(2) Presumptions for the design and the safety demonstration of the emergency core cooling systems, the containment vessel and its internals as well as the supports of the reactor coolant system components.

For the design and examination by calculation the following postulates are relevant:

1. The analysis of the emergency core cooling efficiency (reference to Sec., 22.1.1) shall be based on leak cross sections in the main coolant pipes up to 2 F. The emergency core cooling systems shall be designed accordingly.

2. The determination of the containment vessel design pressure as well as the determination of pressure differences inside the containment vessel shall be based on leak cross section up to 2 F.

The determination of design pressure and design temperature for incident resistant electrical equipment shall be based on leak cross section of 2 F as well.

3. For the demonstration of stability of the components reactor pressure vessel, steam generators, main coolant pumps, and pressurizer the following assumptions shall be made:

The stability of the components shall be assured for a static force P_{ax}

magnitude: $P_{ax} = p \times F \times S$
P = nominal operating pressure
F = open cross section
S = 2 (safety margin)

* Note: This definition is relevant for the design requirements in Sec.
3.3 (1) Reactor pressure vessel internals
5.1 (5) Containment vessel internals
5.2 (l), (5) Electrical equipment inside the containment vessel

origin middle line of the pipe cross section in the area of the nozzle
of force: circumferential weld middle line of the nozzle acting
direction
of force: towards the component

This force acts only on one nozzle at a time. The stability shall be demonstrated for each nozzle separately.
Note: with respect to the steam generator the stability shall be assured for the connection to the secondary circuit in the same way.

(3) Deterministic postulated leak cross section in the reactor pressure vessel

1. In view of the restraints of the reactor pressure vessel, the stresses acting on the reactor pressure vessel internals and the design of the emergency core cooling system, a leak of about 20 cm^2 (geometric cross section: circular) shall also be postulated below, the reactor core upper edge. Prior defects of the reactor pressure vessel which might lead to a leak size of more than 20 cm^2 shall be detectable in time by means of suitable monitoring measures.

2. The design shall also be based on the consequences of the sudden break of a control assembly nozzle involving the maximum possible leak cross section as well as the postulated leaks in the reactor pressure vessel.

(4) Pressure barrier of the low-pressure system towards the high-pressure system.

Provisions shall be made against pressurizing of the low-pressure system as a result of a failure of the pressure barrier towards the high-pressure system. (pressure-retaining boundary) (e.g. recurrent tests of valve functions, measurements of the pressure between two successive valves and the indication of leaks in the control room).

21.2 Postulated Leaks and Breaks in the Main Steam and/ or Feedwater Pipe

(1) For the main steam and feedwater pipes between steam generator and valve station outside the containment vessel, leaks resulting from subcritical cracks are postulated. These can either be determined on the basis of fracture mechanics or are limited to 0,l F.

With regard to the load assumptions for the reaction and jet forces acting on the main steam and feedwater pipes in the area between steam generator and first isolating valve outside the containment vessel, an 0,l F leak opening ("F" = open cross-section of the pipe) and static outflow shall be postulated to cover all possibilities.

(2) With regard to dynamic loads, incoming release-pressure-waves either resulting from breaks in pipe areas located behind the first isolating valve outside the containment vessel, or postulated as a result of external impacts, shall be applied and used as a design basis. For this purpose, a circumferential rupture having a linear opening behavior and an opening time of 15 msec is postulated as input for the calculation. Using this assumption, analyses of dynamic loads resulting from subcritical cracks are not necessary.

(3) As far as the stability of the steam generator is concerned, the following formal assumptions shall be made with a view to the connection of the second ary circuit (cf. Sec. 21-l (2) 3.):

The stability of the steam generator has to be assured for the static equivalent force P_{ax} in addition to its dead weight.

Magnitude: P_{ax} =2 x p x F
P = nominal operating pressure
: F = open cross section

origin of force: middle of pipe cross section in the area of the first connecting weld
direction of force middle line of the nozzle acting towards
force the component

This force only acts on one nozzle at a time. Stability shall be demonstrated for each nozzle separately.

(4) The loads acting on the steam generator heating tubes due to the static and transient stresses (pressure waves, flow forces, static pressure differences over the steam generator heating tubes) in the case of a main steam or feedwater pipe break or the non-closure of a safety valve on the secondary side, shall be determined. It shall be demonstrated that the steam generator heating tubes cope with these stresses.

However, as a matter of principle, when carrying out an incident analysis for a main steam pipe break, the failure of a few steam generator heating tubes shall be postulated as an additional failure occurring accidentally and not as a result of the main steam pipe break. This failure shall be taken into account by postulating, as an envelope, the complete rupture (2F) of a steam generator heating tube in the steam generator concerned.

In such a case, a single failure at some other location need no longer be postulated in this incident analysis.

In the case of a main steam pipe break outside the outer isolating valve and accompanied by an additional single failure referred to as "non-closure of the isolating valve", a steam generator heating tube failure need not be postulated if the aforementioned load demonstration had a positive result.

In the case of a feedwater pipe break, a steam generator heating tube failure need not be postulated.

If subcritical cracks such as referred to in (1) above, or a rupture of a small pipe, are postulated, no additional steam generator heating tube failure is superimposed.

(5) The effects of a main steam pipe break and of a cold water transient on the reactivity behavior and on pressure and temperature changes inside the reactor, as well as the resulting stresses acting on the reactor pressure vessel and its internals, must be coped with.

Appendix III

Information provided by STUK (Finland)

LBB AND FAILURE FREQUENCY REQUIREMENTS IN THE FINNISH GUIDELINE YVL 3.5: "ASSURING THE STRENGTH OF NUCLEAR POWER PLANT PRESSURE EQUIPMENT

STUK has issued the guideline YVL 3.5 on 5.4.2002. It applies to new NPPs while the enforcement to existing plants is still pending.

Section 2.2 stipulates the strength-related documents to be submitted in conjunction with the application for a construction license of a NPP. Among them is a document entitled “Principles of assuring the strength” which shall clarify 1) the primary circuit and containment construction principles to eliminate the anticipated failure mechanisms; 2) the provision against pipe breaks. An unofficial English version of the latter requirements is given below.

Provision against Pipe Breaks (para. 2.2.2)

The design of a nuclear power plant shall make provision against complete, instantaneous breaks of large piping with regard to

- *loss of coolant and overpressurization of containment*
- *reactor pressure vessel and reactor core support loadings*
- *primary circuit pump loadings*
- *PWR steam generator support and tube bundle loadings and other global safety implications such as flooding, rise of humidity and temperature, and impurities entering the emergency coolant.*

Pipe whips, missiles and jet impingement following a pipe break shall not cause such damage and leakages of other components that would challenge the success of consequently needed safety functions such as reactor trip, emergency cooling, residual heat removal and containment isolation. The vital components shall be located at sufficient distance with respect to high-energy piping, and structural departmenting shall be arranged for mutual separation of safety systems assuring each other and of redundant parts of safety systems. Whip restraints and jet impingement shields, complying with the guidance of [2], shall be primarily provided to prevent impact loads arising from breaks of most stressed pipe portions.

In the event that primary circuit piping were not to be provided with whip restraints and jet impingement shields, an authorization for such a plan has to be received from STUK while applying the construction licence. The plan shall specify the affected systems and parts of systems, as well as the separation principle implementation for each.

*Presented in the plan shall also be the experimental results, validated analyses and comparable operating experiences providing the justification. Probabilistic assessments may be presented using the methodology prescribed in paragraph 2.3.3. This evidence shall demonstrate that the piping and their fittings, with regard to the dimensioning, materials, fabrication, quality assurance, loadings and environmental conditions, render development of crack sizes constituting a threat of break very unlikely. The scheduled in-service inspection and condition monitoring programmes, as well as leakage monitoring, shall facilitate crack detection and the necessary actions long before attaining a hazardous crack size (**leak-before-break principle, LBB**). The candidate piping may not be prone to unpredictable excessive loading situations and degradation mechanisms such as water hammer and corrosion phenomena.*

The analyses pertaining to the design-basis pipe breaks and their mechanical consequences shall be submitted as part of the strength analysis report of the particular piping component. As regards the systems and parts of systems not supplied with devices to prevent dynamic effects of pipe breaks, the LBB principle shall be verified by analysis. The analysis may follow the procedures presented in [3] and [4]. The fracture mechanics stability evaluation for the postulated break locations shall be based on the locally most stressing service conditions, including the design-basis earthquake addressed in the guideline YVL 2.6.

Section 2.3 stipulates the strength-related documents to be submitted in conjunction with the application for the operation licence of a NPP. Among them is a document addressing the “Leak and break probabilities” relevant to the assumed initiating events. An unofficial English version of these requirements is given below.

Leak and Break Probabilities (para. 2.3.3)

The nuclear power plant design and safety analyses shall account for the strength-related uncertainties of the main pressure boundary components. The risks due to failures and following accident sequences shall not exceed the probabilistic safety analysis goals laid down in the guideline YVL 2.8. The requirements relating to probabilistic nonductile failure analysis of the reactor pressure vessel are given in paragraph 3.3.7.

The submitted evaluation of the initiating event frequencies shall categorise the pressure equipment leaks and breaks according to their location, type and cross-sectional leak area. A complete loss of pressure bearing capability of the vessel or part of it, where the leak is accompanied with the dynamic effects discussed in paragraph 2.2.2, shall be treated as a break. Failures of single passive or active parts like heat exchanger tubes, flanged connections and gaskets as well as leaks and breaks due to malfunctions, operating errors and maintenance errors shall be taken into account.

The frequency estimates shall make to an adequate extent use of statistics from comparable facilities, correlations between various degrees of leaks and breaks as well as probabilistic fracture mechanics analyses. The fracture mechanics analyses shall be based on physical models of the degradation mechanism evolution (fatigue, corrosion and ageing phenomena). Other factors to be considered are:

- *loading and defect size variability*
- *crack growth rate in relation to the inspection interval*
- *in-service inspection and leak monitoring effectiveness*
- *the failure mode and the governing strength and toughness properties.*

During the operation, a component reliability database, maintained in compliance with the guideline YVL 2.8, shall be updated with observed leaks and breaks and defect indications, as well as with their causes and means of detection.

The references used in these sections are:

2. *Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, Standard Review Plan 3.6.2, Rev. 1, U.S. Nuclear Regulatory Commission, 1981.*

3. Leak-Before-Break Evaluation Procedures, Standard Review Plan 3.6.3, U.S. Nuclear Regulatory Commission, Federal Register, Vol. 52 No. 167, Aug. 28, 1987.

4. Leak-Before-Break Evaluation Procedures for Piping Components, K. Ikonen et al., STUK-YTO-TR 83, Helsinki, 1995.