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Future Directions for Using the Leak-Before-Break Concept in Regulatory Assessments

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ABSTRACT

This report describes efforts conducted by Engineering Mechanics Corporation of Columbus (Emc^2) for Canadian Nuclear Safety Commission (CNSC) to explore possible future directions for Leak-Before-Break (LBB) analyses for nuclear power plant piping systems. This is an objective assessment that considers alternative approaches that might be deterministic, probabilistic, or a hybrid deterministic approach, and summarizes input from responses to a questionnaire to knowledgeable people in the field in 17 different countries. To explore these possibilities, we also included a significant amount of background material on LBB so that the CNSC staff and readers of this report can better understand the recommendations made in this report. The background information includes the following:

- The history of leak-before-break prior to application to the nuclear industry and different technical definitions of LBB,
- The first applications of LBB and developments in the US, including definitions of US documents like Standard Review Plans, Regulatory Guides, and key reports,
- On-going efforts in the US relative to LBB including the Transition Break Size (TBS) efforts, and new probabilistic efforts being initiated by the United States Nuclear Regulatory Commission (US NRC) and EPRI for a probabilistic code called xLPR, and
- International uses of LBB, including a summary of international LBB procedures prior to the year 2000 for eight countries other than the US, and responses from 17 countries other than the US to a questionnaire created and sent out for this program to briefly assess past, current, and future LBB procedures.

The final section of this report provides an overview of potential options for deterministic, probabilistic, or hybrid deterministic-probabilistic LBB approaches. The main application of these approaches was for primary pipe systems in new nuclear power plants. Interestingly, the general opinion of the LBB international questionnaire was that probabilistic analyses are *not* desired for LBB analyses of *new* plants. Probabilistic analyses may be of value for piping with active degradation mechanisms, but such analyses are really fitness-for-service analyses with inspections beyond leakage detection to ensure LBB behavior.

One of the main suggestions for optional new LBB procedures was to include additional considerations on protection against new degradation mechanisms that may develop. Mechanisms that allow long circumferential surface flaws to develop are the most threatening to leak-before-break behavior. Of these more threatening mechanisms, stress corrosion cracking (SCC) is the most prevalent degradation mechanism in nuclear power plant piping, and unfortunately *SCC is not directly addressed by any nuclear pipe system design code*. SCC can occur due to the combination of material susceptibility, environment (water chemistry and temperature), and high tensile stresses. Historically, the industry has learned how to make better materials and adjust water chemistries to avoid or minimize SCC in service, but there has not been much consideration given to reducing weld residual stresses during plant construction. Since the expected life of nuclear plants is no longer considered 40 years, but is now proposed for 60 years or longer, it is difficult to know if the current SCC measures will be effective over these long time periods.

Consequently, one key suggestion from the surveys and review was to include an incentive in the LBB procedure so that plant fabricators will prepare welds in a manner that produces compressive longitudinal stresses (or significantly reduced tensile stresses) on the internal surface (or ID) of girth welds through the use of “Fabrication Enhanced SCC Resistance Welds”. Some weld sequencing aspects to produce “Fabrication Enhanced SCC Resistance Welds” are discussed, and could be adopted in existing weld procedures without much additional cost impact. If the plant uses “Fabrication Enhanced SCC Resistance Weld Procedures” during construction, then the deterministic and probabilistic approaches could be much simpler and easier to satisfy LBB considerations. If “Fabrication Enhanced SCC Resistance Weld Procedures” are not used, then the LBB application needs to consider all aspects of SCC in the deterministic or probabilistic LBB approach, which can be much more penalizing.

A few of the respondents from the different countries were interested in probabilistic analyses, but would still require deterministic analyses. A hybrid deterministic/probabilistic approach may be a more realistic compromise, where more elaborate analyses not possible in a probabilistic code could be conducted for key aspects of the assessment. One such hybrid approach for LBB was presented in this report, where the probabilistic nature of seismic loading was incorporated by conducting analyses at SSE loads (with comparable current safety factors) and then at 10^{-6} seismic event loads with reduced safety factors. Rather than assuming an idealized flaw type, the flaw size was determined from detailed crack growth analyses, such as the SCC analyses in used PWSCC cracking evaluations in the US, and was termed a “Robust LBB Approach”. Of course, reasonable bounding material properties also need to be used, and some suggestions were given on improved selection of ferritic steels to eliminate detrimental effects of dynamic strain aging or accounting for thermal aging in all materials (not just cast stainless steels). This type of hybrid analysis is somewhat comparable to the approach used for “Seismic Considerations to the Transition Break Size” in NUREG-1903.

In summary, the two main recommendations from this project are;

1. Develop fabrication procedures that can be used to prevent high tensile stresses on the ID surfaces of primary loop piping, which if used would allow LBB without having to consider SCC, and
2. Conduct sensitivity studies on the hybrid deterministic-probabilistic “Robust LBB Procedure” for flaw shape development from SCC and seismic loading effects. Guidelines may evolve to better improved deterministic as well as probabilistic analyses.

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EXECUTIVE SUMMARY

This report describes efforts conducted by Engineering Mechanics Corporation of Columbus (Emc^2) for Canadian Nuclear Safety Commission (CNSC) to explore possible future directions for Leak-Before-Break (LBB) regulatory procedures for nuclear power plant piping systems. This was an objective assessment that considered alternative approaches that might be deterministic, probabilistic, or a hybrid deterministic approach, and also took into account the general consensus from a LBB questionnaire sent to knowledgeable people in the field in 17 countries. To explore the LBB procedure possibilities, we also included a significant amount of background material on LBB so that the CNSC staff and readers of this report can better understand the recommendations made in this report. The background information included the following:

- The history of leak-before-break prior to application to the nuclear industry and different technical definitions of LBB, i.e., under load-controlled stresses, displacement-controlled stresses, under combined load-controlled and displacement-controlled stresses, and time varying stresses like seismic.
- A summary of the first applications of LBB and developments in the US, including definitions of US documents like Standard Review Plans, Regulatory Guides, and key reports.
- A summary of on-going efforts in the US relative to LBB including the Transition Break Size (TBS) efforts, and new probabilistic efforts being initiated by the United States Nuclear Regulatory Commission (US NRC) and EPRI for a probabilistic code called xLPR (extremely low probability of rupture).
- International uses of LBB, including a summary of international LBB procedures prior to the year 2000 for eight countries other than the US, and responses from 17 countries other than the US to a questionnaire created and sent out for this program to briefly assess past, current, and future LBB procedures. Much more details of the international responses were included in a large number of appendices to this report.

The final section of this report provides an overview of potential options for future deterministic, probabilistic, and hybrid deterministic-probabilistic LBB approaches. The main application of these approaches was for primary pipe systems in new nuclear power plants, rather than dealing with existing piping with specific active degradation issues (which are fitness-for-service analyses with technical LBB considerations).

From the questionnaire on LBB sent out to 17 different countries, it was apparent that using probabilistic methods for LBB in the design of new plants was *not* a desired approach. LBB when applied to existing plants, particularly those with an active degradation mechanism that has significant cost impacts, may be worthwhile to undertake probabilistically, but there must be much care in that development *for each degradation mechanism of interest*.

A few countries were interested in probabilistic analyses, but would still require deterministic analyses. A hybrid deterministic/probabilistic approach may be a more realistic compromise, where some more elaborate analyses not possible in a probabilistic code could be conducted. One such hybrid approach for LBB was presented, where the probabilistic nature of seismic loading was incorporated by conducting analyses at SSE loads (with comparable current safety factors) and then at 10^{-6} seismic event loads with reduced safety factors. Rather than assuming an idealized flaw type in this hybrid analysis, the flaw size was determined from detailed crack growth analyses, such as the SCC analyses in used Primary Water Stress Corrosion Cracking (PWSCC) evaluations in the US, and was termed a "Robust LBB Approach". This type of hybrid analysis is somewhat comparable to the approach used for "Seismic Considerations to the Transition Break Size" in NUREG-1903. One of the main considerations for any new LBB procedure was to include additional considerations on protection against new degradation mechanisms that may develop. Mechanisms that allow long circumferential surface flaws to develop are the most threatening to leak-before-break behavior. Of these more threatening mechanisms, stress corrosion cracking is the most prevalent degradation mechanism in nuclear power plant piping, and unfortunately *SCC is not directly addressed by any nuclear pipe system design code*.

Since the life of nuclear plants is no longer considered to be 40 years, but is expected to reach 60 years or longer, it is difficult to know if the current SCC measures (i.e., substitute materials or water chemistry modifications) will be effective over these long time periods. Consequently, *one key suggestion was to include an incentive in the LBB procedure so that plant fabricators will prepare the welds in a manner that produces compressive longitudinal stresses on the internal surface (or ID) of girth welds through the use of "Fabrication Enhanced SCC Resistance Welds."* Some weld sequencing procedures to produce "Fabrication Enhanced SCC Resistance Welds" were discussed, although more refinement is needed for actual application. These weld sequencing procedures in many cases could be adapted in to existing weld procedures without much additional cost impact. If the plant uses "Fabrication Enhanced SCC Resistance Weld Procedures" during construction, then the deterministic and probabilistic approaches could be much simpler and easier to satisfy LBB considerations. If "Fabrication Enhanced SCC Resistance Weld Procedures" are not used, then the LBB application needs to consider all aspects of SCC in the deterministic or probabilistic LBB approach, which can be much more penalizing.

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We would also like to thank all of the international participants that responded to the questionnaire that contained information on the LBB efforts in their country. The US NRC staff also encouraged disseminating the information on their xLPR program to Canada through this effort, and we thank them for their willingness to do so.

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ACRONYMS

AECB	Atomic Energy Control Board
AECL	Atomic Energy of Canada Limited
AGS	annulus gas system
APR	Advanced power reactor
ARN	Argentine Regulatore Nucleare
ASTM	American Society for Testing and Materials
ASME	American Society of Mechanical Engineers
BARC	Bhabha Atomic Research Centre
BTP	Branch Technical Position
BWR	boiling water reactor
CANDU	Canadian Deuterium (reactor)
CAT	crack arrest temperature
CF	corrosion-fatigue
CFR	Code of Federal Regulations
CNEN	Comissão Nacional de Energia Nuclear (Brazil)
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
COD	crack-opening displacement
COL	Combined Operating Licenses
CRDM	control-rod drive mechanism
CSA	Canadian Standards Association
CT	compact tension (specimen)
DEGB	double-ended guillotine break
DMWs	dissimilar metal welds
ECCS	emergency core cooling system
EDF	Electicité de France
EDO	Executive Director for Operations (of the US NRC)
Emc^2	Engineering Mechanics Corporation of Columbus
EPR	European Power Reactor or Evolutionary Power Reactor
EPRI	Electric Power Research Institute
FAD	Pellini Failure Analysis Diagram also used for the Failure Assessment Diagram
FBR	Fast breeder reactor
FOAK	first-of-a-kind
FTE	fracture transition elastic
FTP	fracture transition plastic
GDC	General Design Criterion
gpm	gallon per minute
GRS	Gesellschaft für Reaktorsicherheit (Germany)
HAZ	heat-affected zone
ID	inside diameter
IGSCC	intergranular stress corrosion cracking
INER	Institute of Nuclear Energy Research (Taiwan)
IPIRG	International Piping Integrity Research Group
ISI	In-service inspection

ITTAC	Inspections, Tests, Analyses, and Acceptance Criteria
J/T	J-integral / tearing modulus
JAEA	Japan Atomic Energy Agency
JAERI	Japan Atomic Energy Research Institute (now JAEA)
JNES	Japan Nuclear Energy Safety (Authority)
JSME	Japan Society of Mechanical Engineers
KINS	Korea Institute of Nuclear Safety
KSNP	Korea standard nuclear plant
KTA	Kerntechnischer Ausschuss (German nuclear safety standards commission)
LBB	Leak-Before-Break
LB-LOCA	large-break loss-of-coolant accident
LEFM	Linear-elastic fracture mechanics
LMFBR	liquid metal fast breeder reactor
LNG	Liquid natural gas
LOCA	loss-of-coolant accident
lpm	liter per minute
LWR	Light water reactor
MCL	Main coolant line
NASA	National Aeronautical and Space Administration
NDE	non-destructive evaluation
NDT	nil ductility temperature
NII	Nuclear Installation Inspectorate (UK)
NPP	nuclear power plant
NRI	Nuclear Research Institute (Czech)
NRO	New Reactor Office (of the US NRC)
NRR	Office of Nuclear Reactor Regulation (of the US NRC)
ORNL	Oakridge National Lab
PBMR	Pebble Bed Modular Reactor
PBR	Pebble Bed Reactor
PGA	peak ground acceleration
PVP	Pressure Vessel and Piping (conference)
PWHT	post-weld heat treatment
PWR	pressurized water reactor
PWSCC	Primary Water Stress Corrosion Cracking
QA	quality assurance
RCC-M	French Nuclear Construction Code for mechanical design
RCL	Reactor coolant loop
RES	Office of Nuclear Reactor Research (of the US NRC)
RG	Regulatory Guide
RHR	Residual heat removal
RPV	reactor pressure vessel
RSK	Reaktor-Sicherheitskommission (German reactor safety commission)
SAR	safety analysis report
SAW	submerged arc weld
SCC	stress corrosion cracking
SG	steam generator

SMAW	shielded-metal arc weld
SRP	Standard Review Plan
SSE	safe shutdown earthquake
STUK	Säteilyturvakeskus (Finland Radiation and Nuclear Safety Authority)
TBS	Transition Break Size
TEPCO	Tokyo Electric Power Company
TSSD	terminal solid solubility for hydrogen dissolution
UK	United Kingdom
US NRC	United States Nuclear Regulatory Commission
USI	Unresolved Safety Issue
UT	ultrasonic testing
xLPR	Extremely Low Probability of Rupture (probabilistic computer code)

1 INTRODUCTION

The objective of this report is to take a fresh look and conduct an objective examination of possible directions in which Leak-Before-Break (LBB) analysis procedures could evolve. To do this and to put LBB in an objective perspective, the first section of this report reviews LBB applications in the general industry, prior to nuclear piping applications. Then, the historical basis of LBB in the nuclear industry is reviewed. This is followed by the current status of LBB in the US (the initial developer of LBB in the nuclear industry) and international applications. As part of this effort, the international LBB application review from 2000 in different countries was updated via a questionnaire that was sent to many more countries than those in the original 2000 summary. These responses are summarized in the main body of the report with further details provided in the appendices. The next section of this report summarizes some of the unique and relevant on-going LBB efforts of which the authors are currently aware. At this point, the stage is fully set to discuss the possible future directions for LBB analyses in the nuclear industry. Deterministic, probabilistic, and hybrid deterministic/probabilistic approaches are examined with some advantages and disadvantages of each approach discussed in detail.

2 HISTORICAL BASIS OF LBB – NON-NUCLEAR APPLICATIONS

The following is a summary of a paper by G. Wilkowski entitled "Leak-Before-Break What Does It Really Mean?" that was presented at the 1998 ASME PVP conference and published in the Journal of Pressure Vessel Technology[1].

Leak-before-break (LBB) is a term that has been used for decades in reference to a methodology that means that a leak will be discovered prior to a catastrophic break occurring in service. LBB has been applied to missile casings, gas and oil pipelines, pressure vessels, nuclear piping, etc. LBB also has several technical definitions. For instance, LBB can occur for an axial flaw in a pipe where the penetration of the wall thickness will result in a stable axial through-wall crack. This is LBB under load-controlled conditions. LBB could also occur for a circumferential crack in a pipe with high thermal expansion stresses. This might be LBB under compliant displacement-controlled conditions. Finally, LBB might occur when the flaw is stable under normal operating conditions and *remains stable* when there is a sudden dynamic event (i.e., seismic loading). This might be a time-dependent inertial LBB analysis. These analyses are deterministic, and could be extended to probabilistic evaluations as well. The following discussion describes some of the technical LBB approaches, applications, and significance of the methodology used in the applications.

2.1 Early Development of LBB

Perhaps one of the earliest technical approaches for leak-before-break (LBB) was one published by Irwin[2] in 1961. This was for the application of an axial flaw in a pressure vessel using linear elastic fracture mechanics for missile applications. LBB was postulated to occur if the length of the flaw was less than twice the thickness of the cylinder, see Figure 1.

$$K_{1c} \geqslant \sigma_{ys}\sqrt{\pi(B + r_p^*)}\,.$$

By taking $r_p^* = \sigma^2 a/2\pi\sigma_{ys}^2$ with $a = B$, and $\sigma = \sigma_{ys}$, it follows that:

$$K_{1c}^2 \geqslant (\pi + \tfrac{1}{2})B\sigma_{ys}^2 \quad \text{or} \quad \frac{K_{1c}^2}{B\sigma_{ys}^2} \geqslant \pi + \tfrac{1}{2}\,.$$

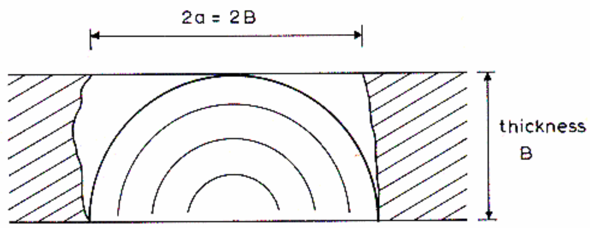
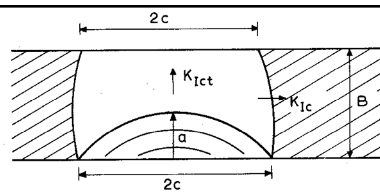


Figure 1 LEFM LBB approach by Irwin

From LEFM analysis, Irwin[2] showed that the crack-driving force would be greater in the radial direction than in the axial direction as long as the axial crack length was less than twice the cylinder thickness.

In 1965, Kobayashi[3] modified the Irwin LBB model by making an improvement to the surface flaw stress intensity factor expression, see Figure 2.



$$\frac{K_{Ic}}{K_{Ict}} > \frac{\frac{\pi}{8}\left(3 + \frac{a^2}{c^2}\right) M_F \cdot \frac{R}{B}}{1.12 M_K \left(1 + \frac{R}{B}\right)\sqrt{\frac{a}{c}}}$$

then

$$\frac{K_{Ic}}{K_{Ict}} > \frac{\pi}{9M_K}\left(3 + \frac{a^2}{c^2}\right)\left(\frac{a}{c}\right)^{-\frac{1}{2}}$$

Figure 2 LEFM LBB model with Kobayashi surface flaw improvement
(Assuming R/B>>1, and for short flaw lengths $M_F = 1$)

As shown in the equations in Figure 2, there are three factors not described in the sketch. These are R, M_K, and M_F. The M_K term is the Kobayashi stress-intensity-magnification factor accounting for the proximity of front free surface. The M_F term is a bulging stress magnification factor from Folias for axial through-wall flaws[4]. Finally, R is the pipe radius. The terms K_{Ic} and K_{Ict} refer to the toughness in the axial and radial directions, respectively. The upper equation in Figure 2 simplifies to the lower equation when R/B>>1 and $M_F \sim 1.0$. Hence, this expression incorporated free surface effects, bulging effects, and toughness differences in the through-wall crack versus surface-crack growth directions.

A different approach for LBB was also being pursued at that time. This was the Pellini Failure Analysis Diagram or FAD[4], see Figure 3. [Note, the *FAD* (failure assessment diagram) is a similar sounding term used in the R6 type analyses[5], but is quite different from the Pellini approach.] Pellini and co-workers developed many terms to describe fracture behavior as a function of a reference temperature. For instance, they coined the terms:

- NDT = nil ductility temperature,
- CAT = crack arrest temperature,
- FTE = fracture transition elastic, and
- FTP = fracture transition plastic.

The Pellini approach was primarily developed for ship applications, but he also suggested applying it to piping flaws as well. One of the difficulties in applying the Pellini approach to LBB was that there was no way to include bulging effects such as determined by Folias.

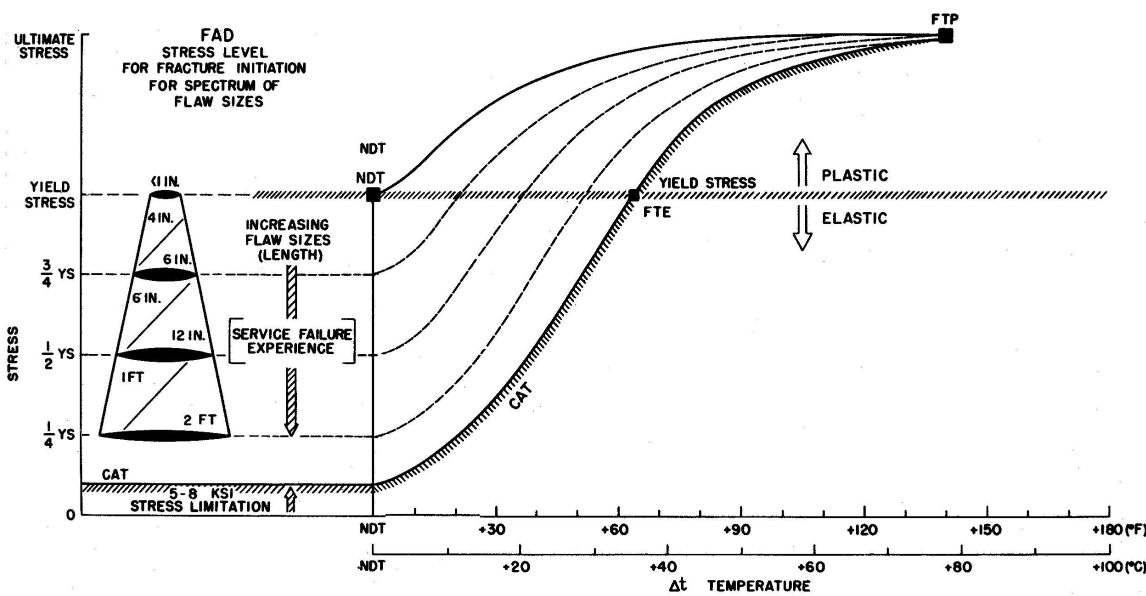
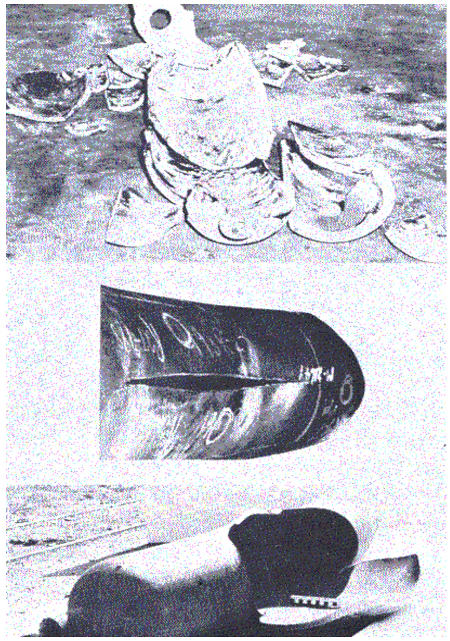


Figure 3 Pellini failure analysis diagram

The Pellini method, however, was quite useful in differentiating between catastrophic brittle fracture versus ductile tearing, see Figure 4.

In the 1960's, Battelle started to develop LBB methodologies for axial flaws in gas pipelines, that was also extended to nuclear piping, LNG piping, and chemical plant piping[6]. This methodology compared the failure pressures of axial through-wall flaws to axial surface flaws.

The Battelle methodology involved non-linear fracture mechanics analysis by including a Dugdale plastic zone, and defined plane stress toughness, K_c, empirically by relating it to the Charpy upper shelf energy. This methodology worked well, as long as the failure mode was ductile tearing from the surface or through-wall flaws. The relationship for an axial through-wall crack (TWC) and surface crack is shown in Figure 5 for the limit-load condition, but can also be expressed for toughness dependent behavior as well.



Top - Service failure at NDT
Middle - arrested hydrostatic burst (FTE + 20F)
Bottom - Pneumatic burst at FTP

Figure 4 Examples of failure modes at Pellini reference temperatures

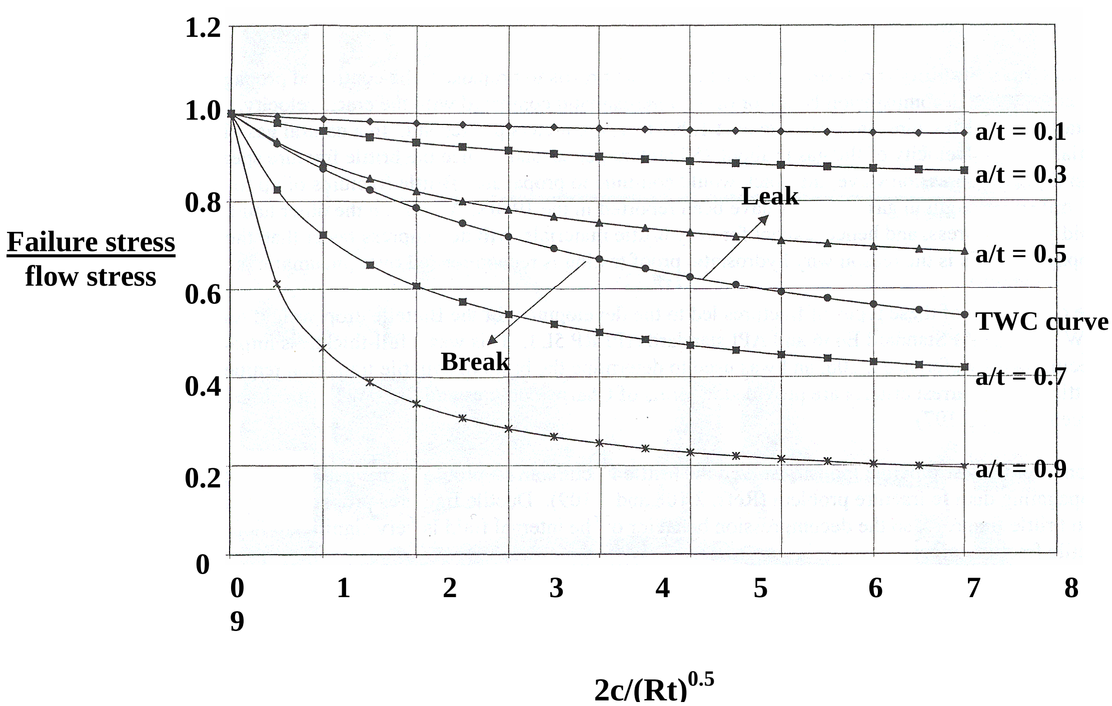


Figure 5 LBB analysis from Battelle relationship for axial cracks in pipes using limit-load equations

The Battelle axial flaw equations have been implemented into many standards, i.e., nuclear piping applications in Appendix C and Appendix H of Section XI of the ASME Boiler and Pressure Vessel code[7], and ANSI B31G for corrosion flaws in gas transmission piping[8].

Many other subsequent axial flaw LBB studies were conducted after the Battelle work in the 1960's to 1970's. Some of these were[a]:

- MPA-Stuttgart Phenomenological Burst Test Programs[9] – nuclear piping
- Reactor pressure vessels (ORNL)[10] - nuclear
- Steam generator tubes[11] – nuclear
- CANDU zirconium pressure tubes[12] - nuclear
- Zirconium cladding over uranium fuel tubes - nuclear
- NASA and aerospace applications to cabin chambers[13] – aerospace
- Cryogenic pipeline applications[14] - refinery & LNG
- Storage tanks - refinery & LNG
- Gas cylinder applications (steel and aluminum) [15] - chemical and automotive
- Composite pipe - natural gas and oil pipelines
- Offshore pipe laying[16] - natural gas and oil pipelines

2.2 Different Industries and Applications

LBB procedures and analyses can change from industry-to-industry based on different levels of risk and types of loading that might occur for the different industrial applications. Two very different approaches are summarized below.

In the natural gas pipeline industry, there are very long-distance transmission pipelines in remote areas. Typically, these are buried pipes where axial flaws are more of a concern since longitudinal stresses are frequently compressive. Leakage will occur at normal operating service loads (pressures) and there are seldom any pressure excursions above the operating pressures. For such remote pipelines, a large leakage could be tolerated from a risk viewpoint. Hence, a tolerable leakage might be up to 30-percent of the cross-sectional opening under normal operating pressures.

In the nuclear industry, LBB has been applied to piping for the purpose of eliminating equipment that is used for restraining pipe whipping during a postulated pipe rupture[17]. The concern in this application is with above ground plant piping systems where circumferential flaws are historically more prevalent than axial flaws. In this LBB approach, it is desirable to detect small amounts of leakage at normal operating conditions so that the leakage size flaw (with some safety factor) would be stable at transient (typically seismic) stresses. Hence, the flaw orientation in these analyses is circumferential, and pressure stresses as well as many other stress components contribute to the LBB analysis. The stresses that need to be considered are normal operating stresses for leakage detection, and transient stresses for crack stability analysis. It is also essential that there not be any subcritical crack growth mechanism that could cause a long surface flaw to occur. Such long surface flaws could lead to failure under the transient loads

(a) References are given for some cases, the author can be contacted for additional information.

without any leakage warning. If there were a mechanism that could cause long surface flaws, then one would have to invoke an augmented inspection process for LBB to work. For example, ultrasonic testing might be deemed adequate to confirm that flaw lengths would be less than a desired value.

2.3 LBB under Different Loading Conditions

In many structural applications, there can be a variety of sources of stresses or loads. Crack stability behavior can depend significantly on the type of loading. The major categories are:

- Load-controlled stresses,
- Pure displacement-controlled stresses,
- Displacement-controlled stresses in structures with significant compliance so there is still a large amount of stored elastic energy, and
- Time-dependent stresses.

These aspects are differentiated below.

2.3.1 Load-Controlled LBB

Examples of pure load-controlled stresses occur from pressure or dead-weight loads. In the case of pressure loads, if the fluid can decompress quickly (i.e., water at ambient temperature), then even the load-controlled pressure stresses may behave like a displacement-controlled stress. A gas-pressurized line might be a good example of a true load-controlled stress. Examples of failures from these cases are illustrated in the middle versus bottom pictures in Figure 4.

Dead-weight loads are typically considered as true load-controlled stresses. Pipe hangers or other supports, however, may physically limit dead-weight loads. In such cases, the pipe may experience load-controlled stress until the displacements reach a limiting value.

An axial flaw leak-before-break analysis is shown in Figure 5. Similarly, circumferential surface flaws may also behave in a LBB manner under load-controlled stresses. This is illustrated in Figure 6 and Figure 7, which were developed to assess the maximum girth weld repair that could be made on an offshore lay barge[16]. In this case, the girth weld may have to be repaired at a location on the barge where the dead-weight bending loads on the pipe are significant.

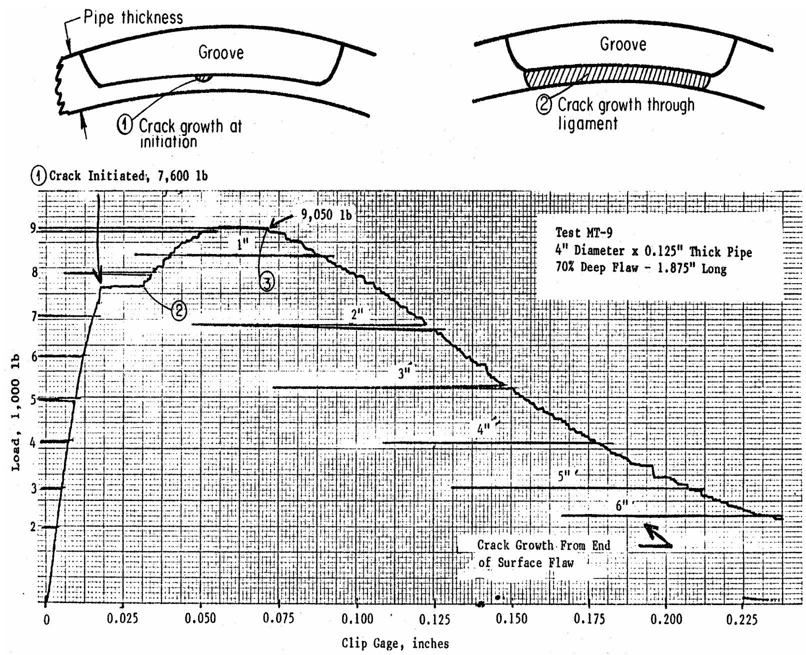


Figure 6 Experiment with 70-percent deep circumferential groove in a pipe under bending which exhibited load-controlled LBB behavior[16]

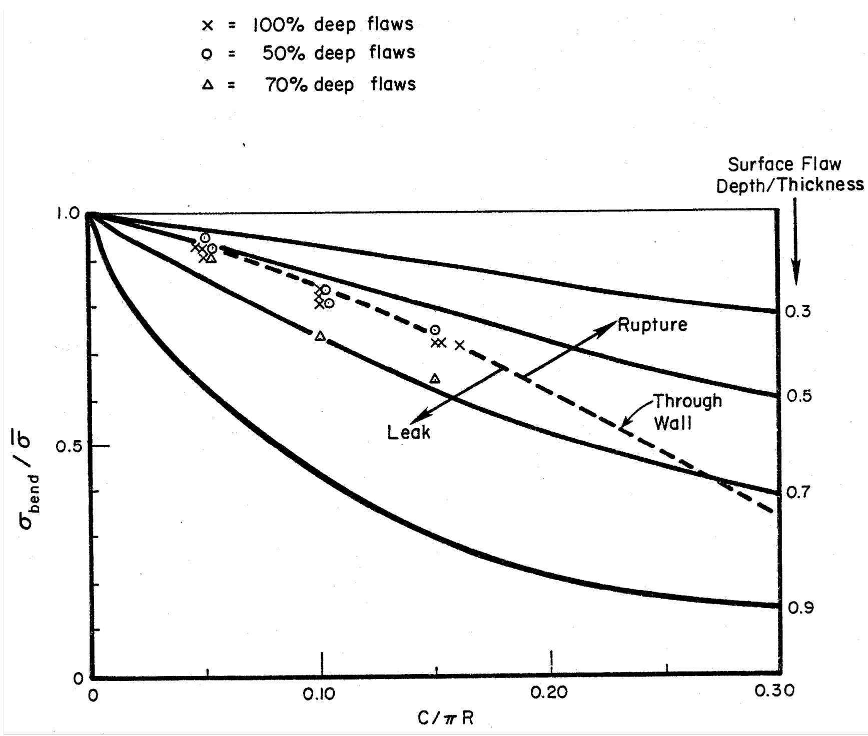


Figure 7 Circumferential blunt flaw LBB criterion[16]

2.3.2 Displacement-Controlled LBB (Local secondary stresses)

Displacement-controlled stresses could be classified as local displacement-controlled stresses (i.e., weld residual stresses or through-thickness temperature gradients), or global displacement controlled stresses where the compliance of the structure can add to unstable behavior (i.e., restraint of thermal expansion stresses in a pipe system). Local displacement-controlled stresses, which involve energy stored over a short gage section, seldom contribute to fracture unless the material is extremely brittle. Weld residual stresses may be important for subcritical cracking (i.e., stress-corrosion cracking) and for crack-opening displacement under elastic loading for leakage considerations, but seldom contribute to fracture for ductile materials used in most pressure vessel and piping applications.

2.3.3 Displacement-Controlled LBB (Global secondary stresses)

Global displacement-controlled stresses can contribute to ductile fracture even in extremely tough materials such as TP304 stainless steel. To address this aspect, the J-integral/tearing modulus methodology was developed. The J/T analysis approach is illustrated in Figure 8. Figure 9 shows a circumferentially through-wall cracked TP304 pipe experiment used to validate this analysis procedure. In this experiment, the helical spring represented the stored elastic energy from thermal expansion stresses for a pipe length of 28 feet[18].

The J/T stability analysis procedure predicts when instability might start. From experimental evidence, it is possible for the crack to jump only a small length in certain conditions. To assess such displacement-controlled instability conditions, an energy-balance was developed in Reference [19]. Figure 10 shows test results of two identical surface-flawed pipe experiments with different pipe lengths. The top case shows a limited instability (smaller load drop) than the bottom case.

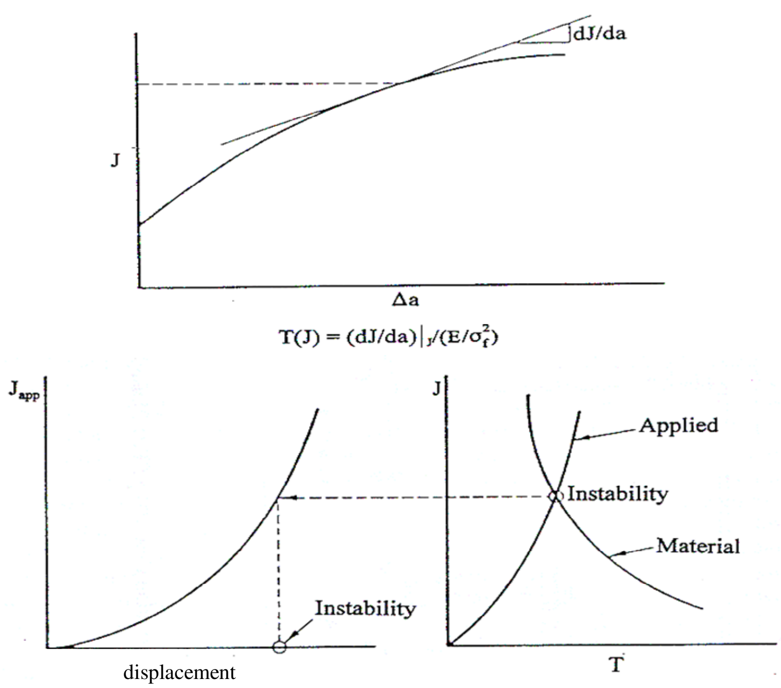


Figure 8 J/T stability analysis procedure

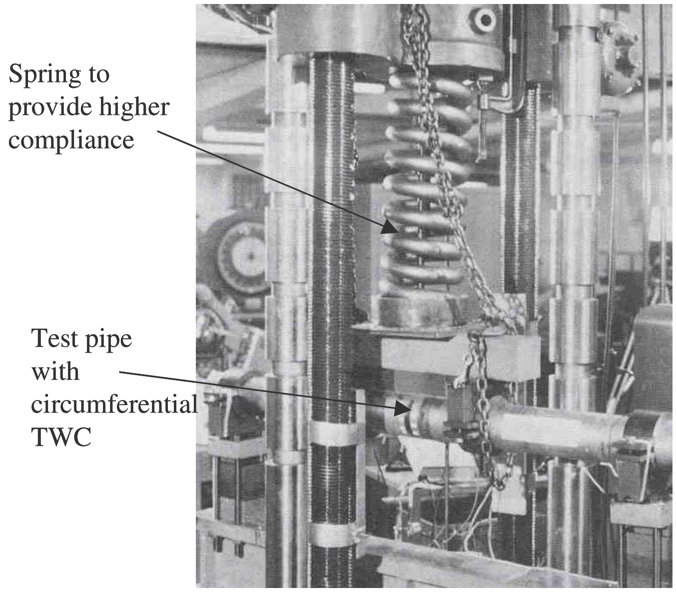


Figure 9 Circumferential through-wall-cracked TP304 pipe experiment under compliant displacement-controlled loading to validate J/T analysis[18]

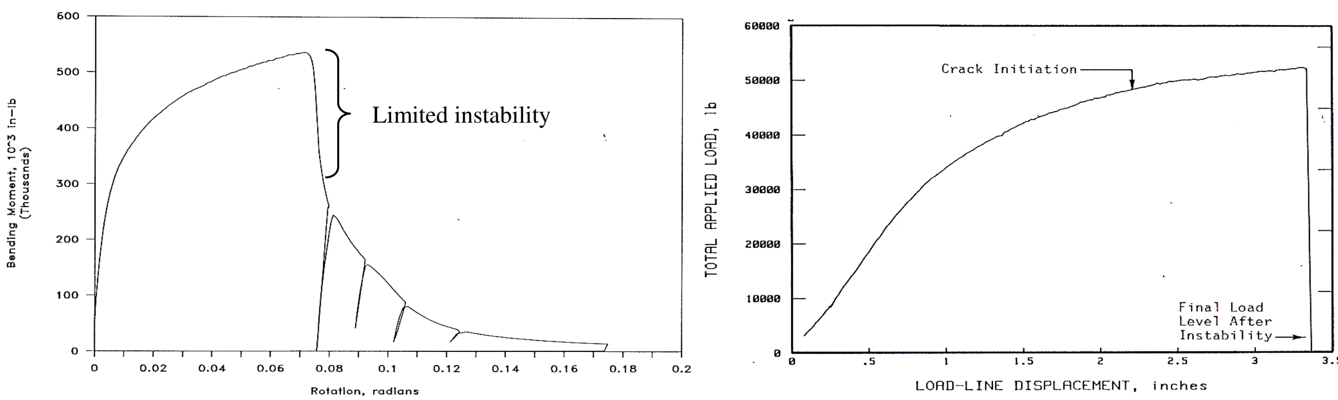


Figure 10 Experimental records of limited instability and total instability of circumferential surface flaws in TP304 pipe

2.3.4 Combined Load-Controlled and Displacement-Controlled LBB

In addition to allowing for the estimation of how far a crack jump might be, the energy-balance analysis method also allows for several other key aspects. The transition of a surface crack to a through-wall crack, and the magnitude of unstable through-wall crack growth can be assessed by the energy-balance approach, see Figure 11. Additionally, it is possible to assess the stability for combined load-controlled stresses and displacement-controlled stresses, see Figure 11. In Figure 11, the resulting crack would jump to point “I”. The crack would be completely unstable if the load-controlled stresses were equal to P_1, but would be stable if the load-controlled stress was P_2.

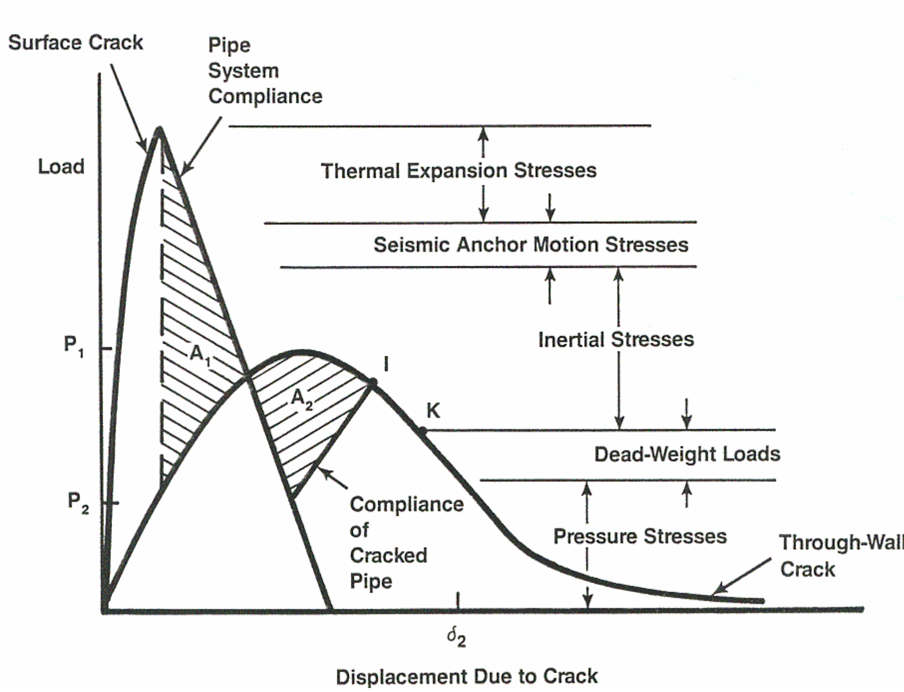


Figure 11 Energy-balance analysis[19] showing how surface to through-wall crack transition is predicted, as well as stability under combined load-controlled and displacement-controlled stresses

2.3.5 Time-Dependent Stress LBB

Although Figure 11 shows a schematic of conducting a combined load-controlled and displacement-controlled stresses energy balance analysis, real piping system stresses for a nuclear plant are more involved. Including the effects of time-dependent stress components like inertial and seismic anchor motion stresses in all the previously mentioned analyses is typically done by assuming that those stress components do not vary with time.

A more detailed analysis procedure to account for time-dependent stress variations was established in Reference [20]. In this analysis procedure, a special cracked-pipe element was used to represent the global moment-rotation behavior due to the crack, see Figure 12. This element can be adjusted to account for constant pressure axial forces, and then a dynamic pipe analysis can be conducted. A significant experimental effort was undertaken as part of a program called the International Piping Integrity Research Group (IPIRG) program to assess circumferentially cracked-pipe systems under seismic loading at LWR temperatures[21]. Figure 13 is a schematic of a 406-mm (16-inch) diameter pipe system. Figure 14 shows a comparison of experimental and predicted moment versus time behavior. From such dynamic analysis, the effects of the crack plasticity on damping and changing the system response can be determined. In typical nuclear pipe LBB analyses, a response-spectrum analysis is used to determine the seismic loads. This is an uncracked pipe elastic stress analysis. These elastic stresses are then used with elastic-plastic fracture mechanics, which gives a conservative estimation of the actual crack-driving force. This dynamic non-linear analysis procedure allows this inherent conservatism to be quantified.

Currently, there are two on-going programs at Emc^2 that involve similar analyses to that completed during the IPIRG program. One is for the USNRC in analyzing pipe combined component tests conducted by JNES in Japan[22], and the other is an assessment of the predicted rate of break opening for an international nuclear facility.

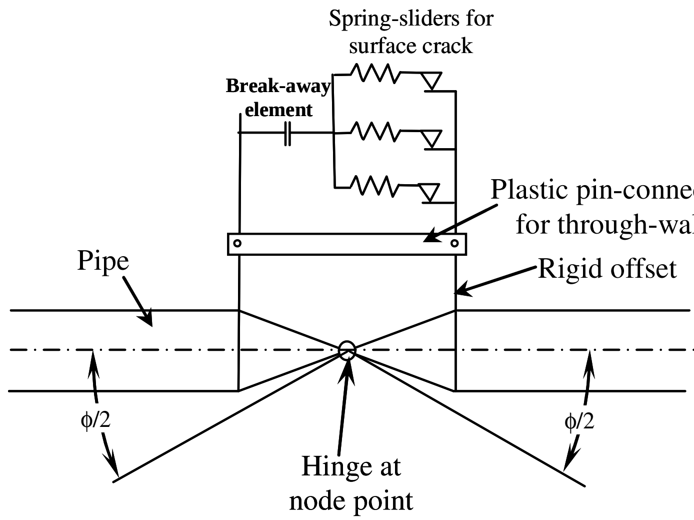


Figure 12 Schematic of how to create a cracked-pipe element for dynamic LBB analysis during IPIRG program efforts in 1990 using ANSYS

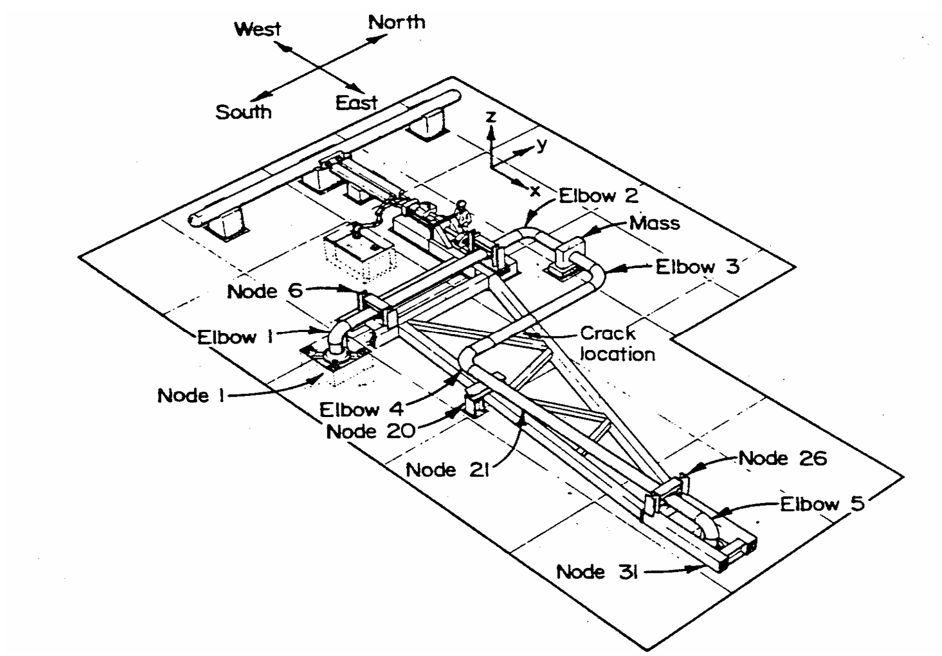


Figure 13 Illustration of pipe system used in IPIRG program to assess crack stability under seismic loading at LWR temperatures

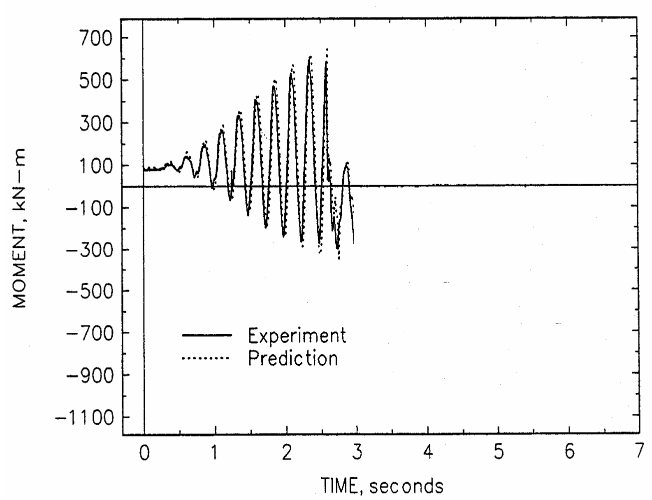


Figure 14 Comparison of cracked-pipe element analysis and pipe system experimental data from the IPIRG program[21]

3 LBB IN THE NUCLEAR INDUSTRY

The following section discusses specific efforts in the nuclear industry relative to LBB. LBB was first initiated for nuclear piping applications in the U.S, and is frequently used by other countries. Hence, the following section gives the historical basis for USNRC LBB procedures. Following that section is a section on International LBB applications in the nuclear industry.

3.1 US NRC LBB Development, Guidelines, and Regulations

3.1.1 GDC-4

In the US, the governing section of the regulations related to LBB is General Design Criterion 4 (GDC-4) on Environmental and Dynamic Effects Design Bases in Appendix A of Part 50 (Domestic Licensing of Production and Utilization Facilities) of Title 10 (Energy) of the US Code of Federal Regulations (10CFR50)[23]. GDC-4 states that:

> "Structures, systems, and components important to safety shall 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 loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

Of particular interest to the subject of LBB, is the stipulation in GDC-4 that allows the use of "analyses reviewed and approved by the Commission" to eliminate from the design basis the dynamic effects of pipe ruptures.

Another specific reference in Appendix A of 10CFR50 that is particularly pertinent to LBB is the definition of a loss-of-coolant accident (LOCA):

> "Loss of coolant accidents means those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system."

The footnote to the definition of a loss-of-coolant accident warrants further discussion. Criteria relating to the type, size, and orientation of postulated breaks have been developed by the NRC, although not specifically promulgated in the regulations. These criteria have been published in the form of Regulatory Guides and Standard Review Plan (SRP) sections.

3.1.2 Generic Issue A-2

Generic issues are issues or problems that are identified by the NRC that are common to a number of operating plants. One issue, or problem, of specific concern from an LBB perspective was due to the asymmetric blowdown loads on pressurized water reactor (PWR) primary systems. The problem of asymmetric blowdown loads on PWRs primary systems, initially identified to the NRC staff in 1975, was designated Unresolved Safety Issue (USI) A-2. This issue deals with safety concerns following a postulated major double-ended pipe break in the primary system. Previously unanalyzed loads on primary system components had the potential to alter primary system configurations or damage core-cooling equipment and contribute to core melt accidents. The resolution of this issue would have required some licensees for operating PWRs to add massive piping restraints to address the consequences of these postulated large-pipe ruptures. Instead of resorting to these measures, this issue was resolved by the industry and the NRC staff by the adoption of the LBB approach utilizing advanced fracture mechanics techniques.

3.1.3 Regulatory Guides

The US Regulatory Guide series[24] provides guidance to licensees and applicants on implementing specific parts of the NRC's regulations, techniques used by the USNRC staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits or licenses. With regard to LBB, one Regulatory Guide of specific interest, and referenced in SRP 3.6.3 on LBB Evaluation Procedures, is Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems[25].

3.1.3.1 <u>Regulatory Guide 1.45 (Reactor Coolant Pressure Boundary Leakage Detection Systems)</u>

General Design Criterion 30 (Quality of Reactor Coolant Pressure Boundary) of Appendix A to 10CFR50 requires means be provided for detecting, and to the extent practical, identifying the location of the source of reactor coolant leakage. Regulatory Guide 1.45 describes acceptable methods of implementing this requirement with regard to the selection of leakage detection systems for the reactor-coolant pressure boundary. The position of Regulatory Guide 1.45 is that at least three different detection methods should be employed. Two of these methods should be; (1) sump level and flow monitoring, and (2) airborne particulate radioactivity monitoring. The third method may involve either monitoring of condensate flow rate from air coolers or monitoring of airborne gaseous activity. The regulatory guide recommends that leak rates from identified and unidentified sources should be monitored separately, with the latter being monitored within an accuracy of 1 gallon per minute (gpm). Indicators and alarms for leak detection should be provided in the main control room. Other recommendations specified in Regulatory Guide 1.45 include:

- The sensitivity and response time of each leakage detection system should be adequate to detect an unidentified leakage of 3.8 lpm (1 gpm) in less than 1 hour.
- The leakage detection systems should be capable of performing their functions following a seismic event that does not require a plant shutdown.
- The leakage detection systems should be equipped with provisions to readily permit testing for operability and calibration during plant operations.

There is an update to Reg Guide 1.45 (Revision 1 issued in May 2008). Many of the additional provisions have to deal with monitoring and quantification of for small leakage quantification for purposes other than LBB, i.e., for boric acid control to avoid problems such as occurred at the Davis Besse plant due to reactor pressure vessel (RPV) head corrosion from a control rod drive mechanism (CRDM) nozzle leak. Those provisions are not mandatory for LBB.

As an additional note on leakage detection, although most US plants used 1 gpm from their Tech Spec limits for LBB analyses, the newer plants were able to use 0.5 gpm for LBB analyses.

3.1.3.2 Standard Review Plans

Standard Review Plan (SRP) sections are prepared for the guidance of the Office of Nuclear Reactor Regulation (NRR) staff responsible for the review of applications to construct and operate nuclear power plants. The various SRP sections are incorporated in NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants[26]. SRP sections are not substitutes for Regulatory Guides or the Commission's regulations, and compliance with them is not required. They were developed as guidance to the NRC staff for review of new nuclear power plant applications. Two Standard Review Plan sections of prime interest to LBB are SRP 3.6.2 on "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping"[27] , and draft SRP 3.6.3, "Leak-Before-Break Evaluation Procedures"[28].

3.1.3.2.1 SRP 3.6.2 (Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping)

The US GDC-4 requires that structures, systems, and components important to safety shall be designed to accommodate the effects of postulated accidents, including appropriate protection against the dynamic and environmental effects of postulated pipe ruptures.

Information concerning break and crack location criteria and methods of analysis for evaluating the dynamic effects associated with postulated breaks and cracks in high- and moderate-energy fluid system piping inside and outside of containment should be provided in the applicant's safety analysis report (SAR). This information is reviewed by the NRC's Mechanical Engineering Branch in accordance with SRP Section 3.6.2, to confirm that requirements for the protection of structures, systems, and components relied upon for safe reactor shutdown, or for the mitigation of the consequences of a postulated pipe rupture, are met.

SRP 3.6.2 was updated in March 2007 (Revision 2) for new US plants, with specific information on Combined Operating Licenses (COL) and ITTAC reviews. The break locations inside and outside containment are defined in Branch Technical Position (BTP) 3-4.

The original arbitrary intermediate break location requirement was eliminated, and typically the SRP 3.6.3 reviews show the high stress locations being close to nozzles at terminal ends of the pipe system.

3.1.3.2.2 SRP 3.6.3 (Leak-Before-Break Evaluation Procedures)

GDC-4 of Appendix A to 10CFR50 allows the use of analyses reviewed and approved by the Commission to eliminate from the design basis the dynamic effects of the pipe ruptures postulated, consistent with the guidance provided in SRP Section 3.6.2. The NRC reviews and approves each submittal to eliminate these dynamic effects. Approval of these LBB analyses by the NRC permits the case-by-case removal of protective hardware, such as pipe-whip restraints and jet-impingement shield barriers, the redesign of pipe-connected components, their supports, and their internals, and other related changes in operating plants.

This draft SRP section (3.6.3) was used by the NRC to evaluate all submittals from licensees and applicants dealing with the implementation of LBB technology for existing plants. This draft SRP section has as its genesis the USNRC Piping Review Committee Report, NUREG-1061, Vol. 3, dated November 1984[29].

SRP 3.6.3 was elevated from a Draft to a full SRP with Revision 1 in 2007. This revision was made for new plants to go through the COL rather than the 2-step licensing process. Modifications were made in particular to not allow LBB for piping containing materials susceptible to primary water stress corrosion cracking (PWSCC).

3.1.3.2.3 NUREG-1061 Volume 3

In the 1983/84 period, the Executive Director for Operations (EDO) of the USNRC requested that a comprehensive review be made of NRC requirements in the area of nuclear power plant piping. In response to this request, an NRC Piping Review Committee was formed. The activities of this review committee were divided into four tasks handled by appropriate task groups, namely:

- Pipe Crack Task Group (dealing with BWR IGSCC issues),
- Seismic Design Task Group (dealing with new design analyses),
- Pipe Break Task Group (dealing with LBB), and
- Dynamic Load/Load Combination Task Group.

As a result of this Piping Review Committee, a five-volume NUREG report (NUREG 1061) was published in 1984 and 1985. Volume 3 of this NUREG was the report prepared by the Pipe Break Task Group and dealt with the Evaluation of Potential for Pipe Breaks. Volume 3 summarizes a review of regulatory documents and contains the Task Group's recommendations for application of the leak-before-break (LBB) approach to the NRC's licensing process. Some of the key recommendations from NUREG-1061 Volume 3 that were later implemented into the Draft SRP 3.6.3 on LBB include:

- A caveat on the use of LBB instead of the double-ended guillotine break (DEGB) criteria is the absence of excessive loads or cracking mechanisms that could adversely affect the accurate evaluation of flaws and loads. Specific examples include water hammer and water slugging, other large dynamic loads, intergranular stress corrosion cracking (IGSCC) and fatigue.
- Examination of leak-detection systems in existing nuclear plants on a case-by-case basis to ensure that suitable detection margins exist so that the margin of detection for the largest postulated leakage size crack used in the fracture mechanics analyses is greater than a factor of ten on unidentified leakage.
- Postulate the existence of a circumferential through-wall flaw at the location(s) of the highest stresses coincident with the poorest material properties. The size of the flaw should be large enough so that the leakage is assured of detection with margin using the installed leak-detection capability when the pipes are subjected to normal operating loads.
- Assume that a safe shutdown earthquake (SSE) occurs prior to detection of the leak to demonstrate that the postulated leakage flaw is stable under normal operating and SSE loads.
- Determine the flaw size margin by comparing the postulated leakage size flaw to the critical crack size. For normal plus SSE loads, demonstrate that there is a margin of at least 2 between the leakage size flaw and the critical crack size to account for the uncertainties inherent in the analyses and leak detection capabilities.
- Determine the margin in terms of applied loads by a crack stability analysis. Demonstrate that the leakage-size crack will not experience unstable crack growth even if larger loads (at least $\sqrt{2}$ times the normal plus SSE loads) are applied.

3.1.3.2.4 Industry Standards

The industry standard of most interest to LBB in the US is the ASME Boiler and Pressure Vessel Code[30]. There are several sections of the ASME Code that are referenced by the Draft SRP 3.6.3 on LBB. Specific references to the ASME Code within Draft SRP 3.6.3 include:

- The stipulation that LBB should only be applied to ASME Code Class 1 and 2 high-energy piping or equivalent. [In practice, typically LBB has only been approved for Class 1 piping systems. On rare occasions, such as the CE System 80+ steam lines, it has been applied to Class 2 piping inside containment.]
- The stipulation that piping susceptible to intergranular stress corrosion cracking (IGSCC) with any planar flaws in excess of those allowed by Article IWB 3514.3 of Section XI of the ASME Code would not be permitted to use LBB analyses.
- The stipulation that when dynamic effects of pipe rupture are eliminated from the design basis, current NRC criteria, and industry codes, such as ASME, may be required for calculating the seismic loads in the heavy component support redesign.

- The use of the Z-factor approach for elastic-plastic fracture mechanics analyses came directly from Appendix C of Section XI of the code.

Note, that ASME Section XI Appendix C updated the Z-factors for stainless steel SAW and SMAW welds to have the same equation. That was based on statistical analyses of CT specimen data[31], and some pipe tests from the US NRC's Degraded Piping Program[32]. Rev 1 of SRP 3.6.3 did not use the new ASME Z-factors for SAW and SMAW welds.

3.1.4 Technical Basis for a Regulatory Guide on LBB – NUREG/CR-6765

The U.S. Nuclear Regulatory commission has had a Standard Review Plan (3.6.3) in place since 1986[17]. The analysis capabilities and knowledge of material property behavior under cyclic dynamic loading corresponding to seismic loads has increased since then[33]. Hence, the NRC plans to create a Regulatory Guide for LBB that would replace the Draft Standard Review Plan. The regulatory guide would be more specific than the standard review plan on how the LBB analysis should be conducted.

Draft Standard Review Plan 3.6.3 for LBB has the following features.

- The procedure uses a flaw tolerance-based analysis.
- It is applied to analyzable sections of Class 1 piping.
- It requires demonstration of adequate margin between a postulated leakage-size flaw under normal operational loads, and a critical flaw size under design-basis loads (load-controlled LBB).
- To demonstrate that the piping system is a candidate for LBB approval it is necessary to show the following:
  - There is no active degradation mechanisms which would undermine LBB assumptions, i.e., long surface flaws could not occur, and
  - There are no atypical high loading conditions (e.g., water hammer).
- With the above conditions satisfied, one can determine the smallest through-wall flaw (with some margin), which can be detected by facility's leakage-detection system.
- One needs to demonstrate by fracture analyses that the critical flaw (at design-basis loads) is larger than leakage flaw by a specified margin.

For the future LBB Regulatory Guide, a three-tiered analysis concept was recommended in NUREG/CR-6765. The following analysis levels were suggested, where the details of the approach were given in NUREG report and are also given in Appendix A, B and C in this report:

- Level 1 analysis - Simpler than current methodology (with potentially larger margins) that would accept piping systems which easily meet current requirements (i.e., most main coolant loops with good quality materials).

- Level 2 analysis - More complex than the current SRP methodology, i.e., include effects from pressure-induced bending, weld residual stresses, and unique material considerations like dynamic strain aging[22]. The safety factors or margins could be reduced by conducting a more detailed evaluation relative to Level 1 analysis.

- Level 3 analysis is significantly more complex than current methodology, but permits for nonlinear seismic stress/fracture time-history analyses that can allow the user to reduce the large margins in the SSE fracture analysis [20].

However, due to the occurrence of PWSCC cracking in Alloy82/182 nozzle welds in piping already approved for LBB, the Regulatory Guide activities for LBB have been put on hold.

The primary author of this report, had the occasion to make presentation of the suggested LBB Reg Guide analysis procedures to several organizations outside of the US. In general they liked the 3-level approach, but thought some of the requirements in Level 2 were too difficult and should be accounted for by the safety factors rather than requiring detailed non-standard testing, i.e., cyclic effects on ductile tearing.

As noted later in this report, the Level 2 approach has recently been adopted in Sweden for LBB with some modifications.

3.1.5 Current NRC LBB Assessments

There are several different on-going efforts at the USNRC relative to LBB. One is the Transition Break Size (TBS) effort, another is evaluation of mitigation efforts for piping susceptible to PWSCC but approved for LBB, and the final one is the development of a new NRC/EPRI probabilistic code. Summaries of these efforts are given below.

3.1.5.1 <u>Transition Break Size (TBS) Technical Basis and Plans for a Reg Guide</u>

The TBS efforts are aimed at the application of allowing the ECCS system to be downsized due to the large-diameter pipe-break opening area being an incredibly low probability event. The efforts involved a grueling elicitation process with about 12 international experts (author of this report was one of them) for normal operating conditions[34], and then a separate report on Seismic Considerations[35].

The elicitation efforts involved assessing the probability of failure for both PWR and BWR plants. Initial base cases were developed using service history data and probabilistic fracture mechanics analyses. (The probabilistic analyses did not include PWSCC as a failure mechanism.) The individual could then use either of these (or a combination or none of them) for creating their initial basis. There were about 12 pipe systems considered for BWRs and another 12 pipe systems for PWRs. Probability of failures of CRDM nozzles, steam generator tubes, RPV etc. were included in the evaluation as well as possible failure from indirect causes. The mean, 5%, and 95% failure probabilities were estimated by each person for all piping sizes. The NRC then conducted a statistical analysis of results providing the median, mean, and 95th percentiles from the group's evaluations, see Figure 15.

A separate follow-on effort assessed failure due to seismic considerations relative to the TBS. This involved:

- Assess failure probabilities using plant-specific seismic-hazard curves, which are using beyond-design-basis seismic events;

- Adjusting the seismic stresses from the original SSE designs for current seismic methodologies (generally making the amplitude of the seismic accelerations lower);
- Determining the failure probability of unflawed pipe with the seismic hazard curve (going up to seismic events with the probability of occurrence of 10^{-6});
- Determining the failure probability of flawed pipe with the seismic hazard curve (going up to seismic events with the probability of occurrence of 10^{-6} for two conditions);
    - o Circumferential through-wall cracks in the piping and if the LBB analyses conducted with standard methods with the SRP 3.6.3 safety factor would be sufficient for a 10^{-6} seismic loading with reduced safety factors and best estimate analyses.
    - o Circumferential surface cracks in the piping and if the ASME Section XI inspection criterion for a design SSE event with all associated safety factors, would be sufficient for a 10^{-6} seismic loading with reduced safety factors and best-estimate analyses.
- Failure by indirect means from a beyond-design-basis seismic event.

The above results showed the unflawed pipe had no problems with 10^{-6} seismic loading, and indirect failures from two cases studied had failure probabilities at or higher than 10^{-6}. For the flawed piping analyses, most of the standard LBB analyses with all imposed safety factors were sufficient to cover the 10^{-6} seismic event, but some had to show a more sensitive leak-detection capability of 0.5 gpm rather than 1.0 gpm. For circumferential surface flaws, in most cases the ASME Section XI criteria for Service Level D design loads (SSE) with all the associated safety factors, generally covered the 10^{-6} seismic loading. Generally, the worst-case circumferential flaw size for 10^{-5} seismic loading was a flaw 40% of the thickness and more than 60% of the circumference. For 10^{-6} seismic loading, the flaw depth limit decreased to 30% of the thickness. These flaw sizes were considered quite large and detectable relative to current NDE methods. A couple caveats were that these results did not apply to cast stainless steels that are highly sensitive to thermal aging degradation of the toughness, and a simple nonlinear correction was needed for some 10^{-6} seismic analyses to transform the simple elastic scaled seismic stresses for nonlinear fracture mechanics analyses. Nonlinear dynamic analyses of the type discussed in Section 2.3.5 would better quantify the simple nonlinear correction factor used.

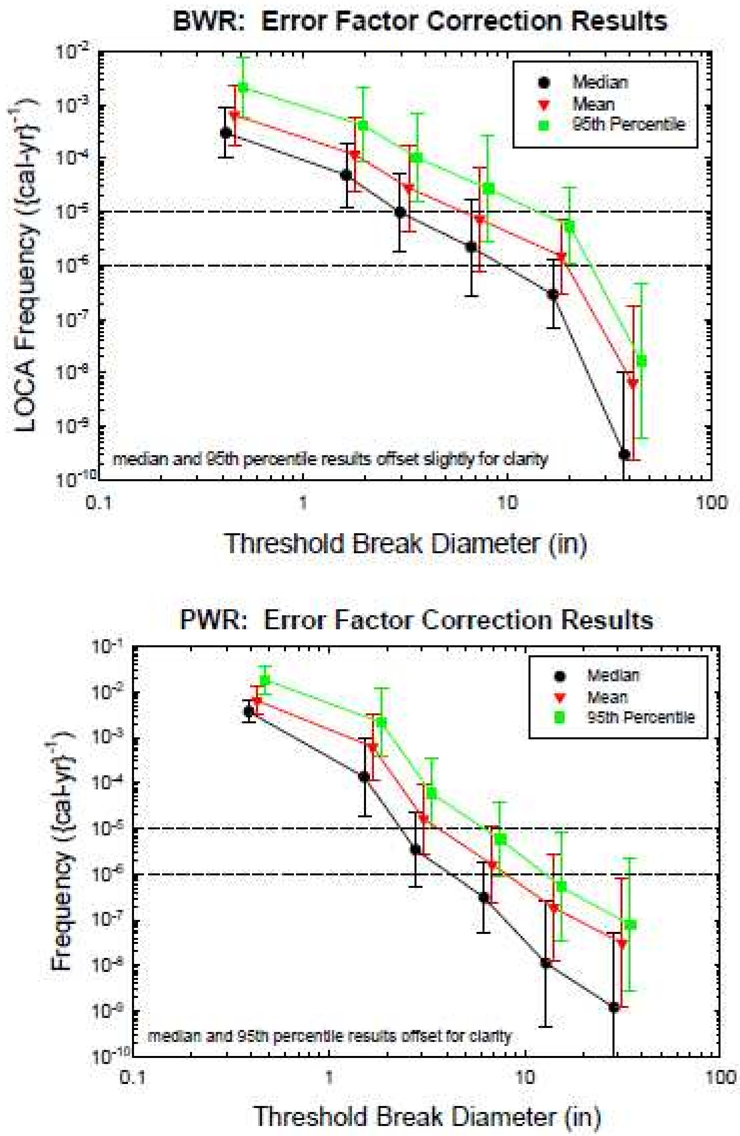


Figure 15 Failure probability estimates from NUREG-1829 for determining the TBS

3.1.5.2 xLPR Code Development to Assess Probability of Pipe Rupture for Piping Susceptible to PWSCC and Approved for LBB

Many software codes for modeling piping probabilistic failures currently exist. Perhaps the first was the PRAISE code. PRAISE was modified by Westinghouse for Risk-Informed In-service Inspection or RISI evaluations – with the code renamed as SRRA. Several European probabilistic codes also exist, i.e., NUBIT from Sweden, PRODIGAL from Rolls Royce in the UK, PROST from GRS, etc. It was recognized in the US that the PRAISE code was rather old and had many ad-hoc modifications over 20 years. Therefore, it was decided that it would be simpler to write a new well-documented probabilistic code with all the new developments from the many NRC piping integrity programs with features that readily allowed new degradation mechanisms to be incorporated (i.e., PWSCC). The USNRC first started this effort as part of their Large-Break Loss-of-Coolant-Accident (LB-LOCA) program at Battelle-Columbus and Emc^2. That code development was enhanced in an international group program effort called the MERIT program run by Battelle-Columbus with Emc^2. CNSC and COG were members of the MERIT program.

Towards the end of the MERIT program, it was recognized that PWSCC was becoming a major source of cracking in PWRs in the U.S. The USNRC initiated an extremely time-critical program to determine if nine US nuclear plants should be shut down early for repairs, or if they can continue to the next outage for inspection. This was called the "Wolf Creek Indications" effort, where Emc^2 conducted the work for the NRC and EPRI using advanced fracture mechanics analyses that allowed the modeling of PWSCC cracks to develop to a natural shape in the residual stress field with normal operating loads, rather than using the assumption of semi-elliptical flaw shapes. The time from leakage to potential failure (with a SSE load) was calculated. As a result of this effort and the follow-up evaluations, eventually all the plants were allowed to continue operation. Nevertheless, as a result of this "*regulatory scare*", the NRC decided to embark on development of its own open-source probabilistic code that will include not only piping, but also RPV and steam generators.

The initial kick-off meeting of the xLPR effort was held June 10-11, 2009 (during the course of this project for CNSC). There were about 30 people from NRC, EPRI, NRC contractors, and EPRI contractors. Many of the probabilistic programming aspects may be handled by Sandia National Labs for the NRC, with debate still going on if a probabilistic framework commercial code should be used, or to completely write the code and have it open source for longevity purposes. Copies of viewgraphs from three presentations on the first day are provided in Appendix D.

3.2 Summary of International LBB Procedures

3.2.1 Summary Prior to 2000

A detailed review of LBB procedures as of 2000 was provided in NUREG/CR-6765. This is included in Appendix E. The LBB procedures used in the following countries up to that time were described:

- France,
- Germany,
- Japan,
- Korea,
- Russia,
- United Kingdom,
- Canada, and
- Sweden.

3.3 Update of International Experience other than US NRC

In 2009, to obtain a quick assessment of LBB status in many countries, an informal questionnaire was sent to many key individuals in different countries that are involved with this technology. The countries contacted and the individuals and their organizations were:

- US – Tregoning (NRC-RES), Sullivan (NRC-NRR), Eric Reichelt (NRC-NRO), K. Wichman (Emc^2 and retired from NRC-NRR) and G. Wilkowski (Emc^2)
- Canada – Scarth (Kinectrics), Kozluk (AECL), Andrei Blahoianu (CNSC) and Ahmed Shalabi (CNSC)
- United Kingdom – Paul Harrop (NII)
- France – Dr. Claude Faidy (EDF), Dr. Eric Mathet (Atmea-sas), Dr. Philippe Gilles, Stephane Chapuliot and Ms. Catherine Migne (AREVA)
- Germany – Dr.-Ing. Karl-Heinz Herter (MPA-Stuttgart)
- Czech – Dr. Jiri Zdarek (NRI)
- Belgium – Robert Gerard (Tractebel)
- Sweden – Bjorn Brickstad and Karen Gott (SA)
- Finland – Rauli Keskinen (STUK)
- Spain – Carlos Garcia Cueto-Felgueros (Tecnatom)
- S. Korea – Youn-Hwan Choi (KINS)
- Japan – Hasegawa (JNES)
- Taiwan – Long-Chyuan Kang (INER)
- India – H.S. Kushwaha (BARC)
- Brazil – Jose Eduardo de Almeida Maneschy (Eletronuclear SA)
- Argentina – Nicholas Riga (ARN)
- South Africa – Kobus Smit (PBR)
- China – (some information from G. Wilkowski)

The reader should bear in mind that this was an informal survey that involved some people from regulatory agencies, while other were from the industry.

3.3.1 Summary of Questionnaire Replies

The questions asked and our synopsis of the response is given below. The actual replies provided by each country (or individual) follow the synopsis.

<u>Question 1 -- Is LBB applied to nuclear plants in your country?</u>
LBB has been applied in most countries, with the following details:

- LBB has been applied to only PWRs in the US, while several other countries allow LBB to be applied to BWRs with certain conditions to restrict IGSCC.
- The UK and Taiwan have not applied LBB at all for their LWR plants.
- France and Finland have not applied LBB to older plants, but they are doing so for the newer EPR plants.
- LBB is used for LMFBR plants and considered for the Pebble Bed Reactor plant in South Africa.

<u>Question 2 - Is there an English version of the LBB procedures in your country?</u>
Most frequently, countries followed the USNRC's SRP 3.6.3. Below is a summary.

- There are no English version of LBB <u>regulatory</u> documents in Sweden, Korea, Finland, India, or Czech.
- No formal regulatory documents on LBB (industry standards only) in Japan, Canada, France, Germany, and the UK.

- Belgium, Brazil, and Spain used the US NRC rules (SRP 3.6.3), although Brazil uses German rules for its German made plant.

Question 3 - Is the procedure based on US NRC SRP 3.6.3 (flaw tolerance approach) in your country?

- All analyses seemed to be deterministic flaw tolerance approaches similar to the USNRC's SRP 3.6.3.
- Germany, Japan and France have independent LBB type analyses procedures.
- The UK has a much less prescriptive regulatory system, so there is no regulatory document on LBB, but there is a very nice detailed summary of their procedures in Appendix F. There are LBB procedures in the R6 and BS-7910 industry standards.

Question 4 - If yes to the above question, are there any special considerations above SRP3.6.3? (For example; loadings, material requirements, allowing time to detect between leakage and reaching critical flaw size, SCC considerations, etc.)

- Frequently countries add some additional considerations to the USNRC's SRP 3.6.3 for material testing requirements, J-R curve extrapolation methods or other fracture toughness criteria, leak-rate analyses, residual stresses, fatigue crack growth analyses from a reference flaw size, limiting postulated crack growth within an inspection, defining degradation mechanisms, displacement monitoring in new plants, etc.

Question 5 - Do you accept LBB for other than the primary loop piping in your country?

- LBB accepted for primary PWR piping in all countries except UK and Taiwan.
- LBB accepted for surge lines and safety injection system/residual heat removal lines in many cases, and steam lines and feedwater lines (inside containment) on occasion.
- Some unique applications for in-service evaluations for pressure tubes, steam generator tubing, and feeder tubes in Canada.

Question 6 - Is the LBB application for only elimination of pipe whip restraints and jet impingement shields, or also for environmental qualification, containment sizing, ECCS, etc. in your country?

- Those countries that allowed LBB did so mainly for the elimination of pipe whip restraints and jet impingement shields.
    - Germany still uses a 10% cross-sectional flow area to calculate reaction and jet forces acting on pipes, components, component internals, and parts of buildings. This requirement was eliminated in the USNRC SRP 3.6.3 procedure.
- Some additional LBB applications cited were:
    - Environmental qualification can also be an issue if the consequences of a pipe break can imply damaging sensitive electrical equipment.
    - no more LBLOCA and steam line break in the design transient list;
    - allows for static simplified analysis of component support and stability;
    - no LBLOCA consideration for internal, core support, and fuel support;
    - refined the leak-detection system inside containment, no leakage detection system for steam line outside containment;
    - No consequences on either containment sizing or ECCS design rules and environmental qualification program; and

- Still open for discussion in France is the consequences of eliminating the LB-LOCA design requirements on the steam generator manholes.

Question 7 - Would your country ever be interested in adopting a probabilistic approach, or combined deterministic-probabilistic approach for LBB?

In general, there seemed to be interest in probabilistic approaches, but with some of the caveats listed below.

- Probabilistic approaches may be more important for application to existing plants, but not new plants.
- Some felt that a combined deterministic and probabilistic approach is preferred over a purely probabilistic approach.
- A few felt that probabilistic approaches would not be approved by regulators or are not needed.

3.3.2 Detailed Questionnaire Replies

The following section gives the detailed replies to the questionnaire and differentiates if the reply was from regulatory staff, industry staff, or in some cases government organizations that are not involved with regulatory decisions.

Question 1 -- Is LBB applied to nuclear plants in your country?

- **Canadian regulatory and industry reply:** Yes, we also think it is important to differentiate between the use of LBB concepts for design (of new plants) versus use of LBB concepts in fitness-for-service assessments for operating plants.
- **Belgium industry reply:** Yes, on primary loop piping (surge line not included).
- **Czech industry reply:** Yes, for the NPP Temelin.
- **Finnish regulatory reply:** For existing plants, there are no real applications. LBB is sometimes used to support defect assessments. For the new EPR plant of Olkiluoto 3 (start-up now scheduled for 2012), LBB is applied to the main coolant lines, main steam lines and main feed water lines.
- **French industry reply:** Yes for EPR, but only on main coolant loop and steam line (inside and outside containment).
- **German industry reply:** Yes, it is applied to piping system of PWR and BWR plants. In Germany, we called it "break exclusion" or "proof of integrity" and is part of the "Ageing management" for safety relevant systems (a new KTA safety standard is under preparation).
- **Spanish industry reply:** Yes, LBB is applied in Spanish NPPs, typically in the primary circuit and surge lines of PWRs.
- **Sweden regulatory reply:** We do not put any special requirements on BWRs compared to PWRs regarding LBB requirements. If all the requirements are fulfilled, we will approve LBB also for a BWR. Of course, it can be more difficult in a BWR-plant to prove that e.g., IGSCC will not be present. So far, we have only approved LBB for a PWR (Ringhals 2). Note that we have not approved LBB for the surge line nozzle weld made of Alloy 182. We also have an application from a BWR-plant (Ringhals 1) to apply LBB for the main circulation loops. Our review is ongoing.

- **UK regulatory reply:** The UK does not have a prescriptive regulatory regime, in particular, we do not have many specific technical regulations. Therefore, we do not have a "LBB regulatory document." Basically, the UK situation does not fit very well into the conceptual framework of the questions. For the Sizewell B plant (only PWR in the UK), LBB is not applied. (More information is given in Appendix F with a detailed summary of the UK regulatory system relative to LBB.)
- **China (by G. Wilkowski):** They have not done so yet, but plan to in future.
- **Japan regulatory reply:** We apply LBB to both BWR and PWR, in addition to FBR.
- **Korean regulatory reply:** Yes. LBB was approved for Korean Standard Nuclear Plants (KSNP, 1000MW) and APR 1400 (1400MW) in Korea. CE 80+ is the prototype of KSNP and APR 1400.
- **Taiwan reply:** No. We had planned to apply LBB in our PWR plant, but it did not work out.
- **India government reply:** Yes.
- **South Africa industry reply:** For the PBMR we intend to apply LBB to the following piping of the Helium Pressure Boundary:
    - Pipes connected to the top of the Reactor Pressure Vessel (RPV) with a diameter greater than 50 mm, and
    - Pipes connected to the bottom of the Reactor Pressure Vessel (RPV) with a diameter greater than 160 mm.
- **Argentina regulatory reply:** Yes, it is.
- **Brazilian industry reply:** Yes. For Angra 1 LBB was applied to the reactor coolant loop main piping. For Angra 2 LBB was also applied for the RCL main piping and, in addition, is applied for the residual heat removal and main steam piping.

<u>Question 2 - Is there an English version of the LBB procedures in your country?</u>

- **Canadian reply:** <u>Industry</u> -- There are no formal LBB documents, though I understand that there is one under preparation. In 1985, Mr. Brian Jarman of CNSC presented a paper "The Canadian Approach to Protection Against Postulated Primary Heat Transport Piping Failures," see Appendix I. The regulator made a presentation "LBB Applications to CANDU Piping" in 2008, see Appendix J. The regulator is also moving towards wider use of risk-informed methods.
  <u>Regulatory</u> **-** The only regulatory document on LBB is entitled "The Canadian Approach to Protection Against Postulated Primary Heat Transport Piping Failures", AECB INFO-0170, October, 1985. Since 1985, the AECB/CNSC has not adopted a formal position/requirement on but a conference paper entitled " CANADIAN REGULATORY PERSPECTIVE ON LBB APPLICATION FOR CANDU PIPING" presented at the SMiRT in 2007 outlines our non-mandatory position. CNSC is also developing an internal non-mandatory regulatory review guides and NRC's SRP 3.6.3 will be used by specialists for dispositioning pressure boundary issues.
- **Belgium industry reply:** No, we used the US rules (SRP 3.6.3).
- **Czech industry reply:** No, there is not, but the Czech version is practically identical with the SRP 3.6.3.
- **Finnish regulatory reply:** LBB is addressed in the guide YVL 3.5. An English version is still missing but preliminary translations of the relevant sections you will find in Appendix K.

- **French industry reply:** There is no regulatory requirements, but it is a part of Safety Analysis Report developed by the utility with the vendor; a specific document developed the requirements and the justification of these requirements (this document is in French, but a summary exists as a PVP paper - see Reference 36 not attached to this report since copyright protected.)
- **German industry reply:** There is no regulatory document on how to apply LBB, but in the RSK Guidelines for Pressurized Water Reactors, the leaks and breaks to be postulated are included in Chapter 21. Further requirements to material, design and manufacturing are included in Appendix 2 of Chapter 4 (Basis Safety, see Tables 1 to 4 in the document). The requirements are detailed in the nuclear safety standards KTA 3201 for "Components of the Reactor Coolant Pressure Boundary of Light Water Reactors" (KTA 3201.1 "Materials and Product Forms," KTA 3201.2 "Design and Analysis," KTA 3201.3 "Manufacture" and KTA 3201.4 "In-service Inspections and Operational Monitoring," (see Fig. 1 in the document). RSK-Guidelines (http://www.rskonline.de/downloads/8110dwr.pdf http://www.rskonline.de/downloads/7909dwr.pdf) Nuclear Safety Standards (http://www.kta-gs.de/common/regel_prog.htm). Furthermore, there are thoughts to call up a working group to prepare a new KTA safety standard dealing with implementation of break exclusion.
- **Spanish industry reply:** No. The reference document for LBB applications in Spain is the Standard Review Plan 3.6.3.
- **Sweden regulatory reply:** Our new regulation has the name SSMFS 2008:17 which deals with LBB in §12 and §13 together with some general advice. This has not been translated into English. However, our old regulation SKIFS 2004:2 has an English version and §12 and 13 are not changed (see Appendix H).
- **UK regulatory reply:** No explicit document, see detailed discussion in Appendix F. There are LBB procedures in both R6 as well as PD7910.
- **China (by G. Wilkowski):** Do not believe so at this time.
- **Japanese regulatory reply:** There is no English version of Japanese LBB procedures. After publication of JEAG 4613-1998 (in Japanese), we have JSME code (see Appendix G), published in Dec. 2002. The Japanese government does not endorse the JSME Code. The JSME Code is under revision. The chairperson is Dr. Yukio Takahashi of CRIEPI.
- **Korean regulatory reply:** Unfortunately, there is no English version. KINS has a Korean LBB regulatory guide for LBB based on SRP 3.6.3.
- **Taiwan industry reply:** We do not have such document.
- **India government reply:** No, however, we use procedure developed by USA.
- **South Africa industry reply:** There is no SA Nuclear Regulator document that governs or guides the application of LBB.
- **Argentina regulatory reply:** Yes, and it is based on both the USNRC SRP 3.6.3 and the Transition Break Size approach in NUREG/1829.
- **Brazilian industry reply:** There is no specific Brazilian regulatory document on this subject. For Angra 1 CNEN accepts the US NRC methodology defined in SRP 3.6.3 (draft) and NUREG 1061. For Angra 2, a German approach was adopted.

<u>Question 3 - Is the procedure based on US NRC SRP 3.6.3 (flaw tolerance approach) in your country?</u>

- **Canadian regulatory and industry reply:** The only application of LBB for large-diameter primary heat transport piping was for Darlington. The approach used was similar to NUREG-1061 Volume 3 (see also Mr. Brian Jarman's paper, see Appendix I. Also, see the reply to Question 2.)
- **Belgium industry reply:** It is based on USNRC's SRP 3.6.3. We also evaluated crack growth of a reference defect in 40 years of operation.
- **Czech industry reply:** Yes, it is.
- **Finnish regulatory reply:** Yes. SRP 3.6.3 margins for through-wall cracks are required. For the EPR, the vendor's supplementary life-time growth analyses for surface cracks have been approved. The applicable computational procedures are evaluated in Reference 4 in YVL 3.5 (see Appendix K). These range from estimation schemes to rigorous FEM (see Reference 37).
- **French industry reply:** No, refer to the PVP paper (Reference 36).
- **German industry reply:** The flaw tolerance approach is not based on USNRC's SRP 3.6.3, but it is a similar approach, see Figure 16, which shows how it is applied to the systems with break exclusion at the NPPs GKN and KKP. Postulated crack growth within an inspection interval shall be limited.
- **Spanish industry reply:** Yes. Current applications are based on deterministic approaches.
- **Sweden regulatory reply:** Yes, but we have also issued a report (only in Swedish) where we have put some guidelines for how to fulfill the LBB requirements. In this report, we are using the May 2002 version of NUREG/CR-6765, Level 2.
- **UK regulatory reply:** No, see detailed summary of UK regulatory process in Appendix F.
- **China (by G. Wilkowski):** Yes, but possibly with some adjustments as can be justified and accepted by the regulator.
- **India government reply:** Yes.
- **Japanese regulatory reply:** No, procedure is different. The procedure is shown in Figure D-1 in Appendix G.
- **Korean regulatory reply:** Yes.
- **Taiwan industry reply:** If we want to apply LBB, I believe we will follow USNRC's SRP 3.6.3.
- **South Africa industry reply:** We are designing to be able to satisfy the evaluation criteria in US NRC SRP 3.6.3, and are utilizing the guidance provided in NUREG-1061, Volume 3, and NUREG CR-6765 on calculation methodology and acceptable margins.
- **Argentina regulatory reply:** Yes, but we also used the Transition Break Size analysis procedures in NUREG-1829.
- **Brazilian industry reply:** As informed in the previous answer, the approach approved for Angra 1 was based on SRP 3.6.3 and NUREG-1061.

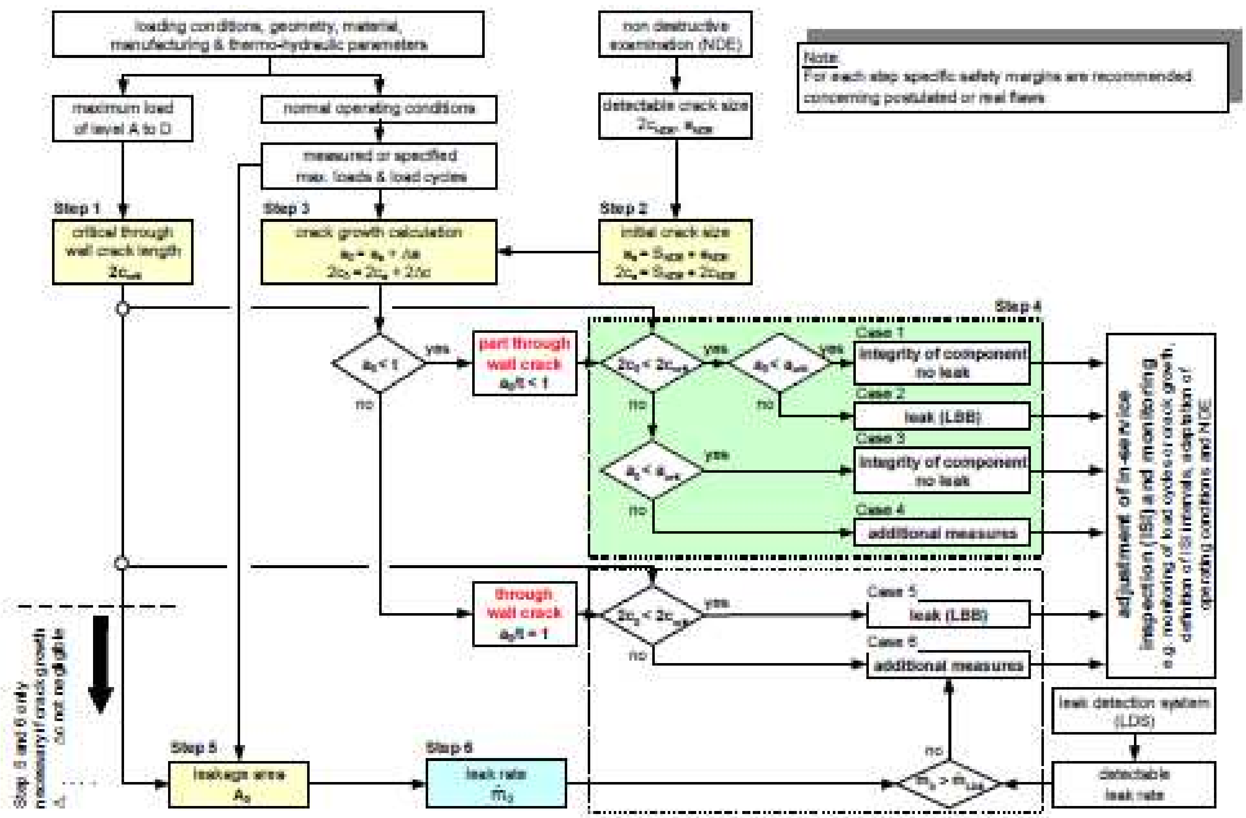


Figure 16 German LBB flaw tolerance flow chart

<u>Question 4 - If yes to the above question, are there any special considerations above SRP3.6.3? (For example; loadings, material requirements, allowing for time to detect between leakage and reaching critical flaw size, SCC considerations, etc.)</u>

- **Canadian reply:** <u>Industry</u> -- For CANDU plants the leak-detection sensitivity is better (ten times) than the 1 US gpm in technical specifications for PWRs. All pipes were SA-106 B (SA-105) and PWHT was mandated for all lines for which LBB was applied. LBB could not be used if the postulated rupture jeopardized fast shutdown systems or containment design (see Brian Jarman's paper in Appendix I.)
  <u>Government</u> -- No. However, for the LBB application for demonstrating fitness for service of degraded components, we expect that operating procedures be established to the sufficient level to ensure appropriate actions to be taken before the leaking crack reaches to the critical size with sufficient margin.
- **Czech regulatory reply:** Practically not.
- **Belgium industry reply:** One additional requirement from our safety authorities, which was to consider a steam generator manhole cover ejection in replacement of the LBLOCA eliminated by LBB (see paper by G. Roussel at the Lyon Specialist Meeting on LBB in 1995, in Appendix L).
- **Finnish regulatory reply:** See our replies to Questions 3 and 6. Technically, the J-R curve extrapolation approach has presented experimental difficulties in case of considerable weld strength mismatch (excessive base material deformation before

attaining desirable crack propagation in the pre-cracked weld). This relates also to the CT-specimen specifications in the applicable ASTM standard. Some progress would be needed to yield useful results with welded specimens matching the wall thickness.

- **French industry reply:** The following are additional requirements:
    - Higher quality requirement and justification than class 1 requirements,
    - Steam line design with class 1+ requirements,
    - More dimensioning justification and control: counterbore, elbows, etc,
    - Material toughness based on tests on real welding process,
    - Specific justification for dissimilar metal welds: toughness measurements at the metal interface,
    - Displacement surveillance system on lead plant,
    - Larger ISI program than class 1 requirement'
    - Thermal ageing justification: 60 years at 330°C, and
    - Cyclic effect of seismic loads for steam line analysis.
- **German industry reply:** Not applicable since German procedures are different from SRP 3.6.3.
- **Spanish industry reply:** No additional requirements.
- **Sweden regulatory reply:** SSM was not satisfied with the approach taken by Westinghouse (which was the main analyst regarding LBB for Ringhals 2) with regard to degradation mechanisms. We required the licensee to document all possible degradation mechanisms based on the specific material/environment combinations in their plant and to explain on what grounds they could eliminate those not included in the LBB analysis. We also required LBB to be investigated in sections along the pipe with both high and low nominal loadings. We recommend using COD-dependent crack morphology parameters. About weld residual stresses, it shall be quantified for determining the leakage flow rate and if the effect is significant, it should be taken into account.
- **UK regulatory reply:** Not applicable since UK does not use USNRC's LBB procedures.
- **China (by G. Wilkowski):** Not at this time, but possibly in the future.
- **India government reply:** We use material fracture toughness based on stress-zone-width measurement.
- **Japanese regulatory reply:** No reply. Editor note, the Z-factors used in the JSME code are different from those used in the ASME code.
- **Korean regulatory reply:** (1) Dynamic fracture test requirement for carbon steel piping (concern over dynamic strain aging); (2) Specific guideline for the number of fracture test for base metal, weld metal, and safe end; and (3) Database for fracture/tensile properties cannot be used for the newly constructed plants.
- **Taiwan industry reply:** Question is not applicable since LBB not used in Taiwan yet
- **South Africa industry reply:** No special considerations, apart from introducing the specifics of the helium coolant and the capabilities of helium leak-detection methods (as opposed to water-leak based regulation).
- **Argentina regulatory reply:** No, they are not.
- **Brazilian industry reply:** No. The main concern of the CNEN was the leakage detection system used in Angra 1. They required sufficient documentation to show that Angra 1 is able to detect 1 gpm.

Question 5 - Do you accept LBB for other than the primary loop piping in your country?

- **Canadian reply:** Industry -- For Zr-Nb pressure tubes, it is a requirement of the CSA Standard N285.8 (“Technical Requirements for In-service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors,” CSA-N285.8-05, Canadian Standards Association, 2005) that when the bulk hydrogen equivalent concentration in the pressure tube is at, or exceeds, the threshold level at which the material is susceptible to delayed hydride cracking at any Service Level A sustained hot condition, an LBB evaluation must be performed for the reactor core. The terminal solid solubility for hydrogen dissolution (TSSD) at the sustained hot condition temperature is used for this threshold level of bulk hydrogen equivalent concentration. Either deterministic criteria in Clause 7.4.2 or probabilistic criteria in Clause 7.4.3 of CSA N285.8 must be satisfied.

  We also employ LBB concepts in the industry’s Steam Generator Tube and Feeder Fitness-for-Service Guidelines.

  Government -- Yes. We have considered the LBB argument for pressure tubes and steam generator tubes and currently under review for the application for feeder cracking. The CNSC has also considered LBB submissions for main steam line on a case-by-case basis with the conditions of; improved material properties to Class 1 components, enhanced periodic inspection programs, or enhanced leak-detection systems.
- **Belgium industry reply:** In principle yes, for the surge line or large auxiliary lines, but it was not applied (we just did not have the need).
- **Czech industry reply:** LBB is usually applied to the whole primary circuit, which contains the main circulating line, the surge line, the part of purification line, the part of ECCS. We are going to apply the LBB also to the secondary side (feed water and steam lines).
- **Finnish regulatory reply:** LBB should be demonstrated for any high-energy piping whose dynamic break effects would jeopardize vital components not adequately protected via hardware, structural departmenting or distance. The particular pipe's qualification for LBB shall be assessed, though, considering the construction and degradation aspects, as well as the effectiveness/availability of leak detection and ISI. For instance, LBB has been applied to the main steam lines and main feed water lines of our EPR, which were constructed conforming to the RCC-M Quality Class 1 rules even though they belong to Safety Class 2. Most sensitive leak-detection technology will be installed, and the NDE systems will be qualified for the postulated surface crack sizes of LBB analysis.
- **French industry reply:** Yes, for steam line inside/outside containment, and the steam discharge line; not for the feedwater line.
- **German industry reply:** Yes, but it differs from plant to plant. For PWR’s LBB is applied to the main coolant lines (hot and cold leg), connecting lines to MCL (e.g., surge line, ECCS up to the first valve, volume control system), main steam and feed water lines up to the first isolation valve outside containment. For BWRs, LBB is applied to the main steam and feed water lines.
- **Spanish industry reply:** No LBB application was submitted in Spain for piping other than the primary circuit.
- **Sweden regulatory reply:** We have said that LBB can be applicable to pipe segments in Class 1 and 2 piping. Ringhals 1 is currently considering applying for LBB for the main steam line.
- **UK regulatory reply:** See detailed reply in Appendix F.

- **China (by G. Wilkowski):** They plan to start with just the primary loop for CN1000 plants, but perhaps more systems may be designed with LBB for the AP1000 plants.
- **India government reply:** Yes, on a case-by-case basis.
- **Japanese regulatory reply:** Yes, except wall-thinning area.
- **Korean regulatory reply:** Yes, LBB was approved for surge line, safety injection line (ECCS), and shutdown cooling line (RHR) for KSNP and APR 1400.
- **Taiwan industry reply:** Yes.
- **South Africa industry reply:** If the consequence of failure of these piping (other than the non-isolatable Helium Pressure Boundary) were found to be unacceptable, we would consider applying the LBB-principles to these. To date, there has not been the need for this.
- **Argentina regulatory reply:** Yes, we do.
- **Brazilian industry reply:** In Angra 1, the LBB acceptance was valid only for the primary loop. For Angra 2 other systems have had LBB accepted (main steam and residual heat removal, for instance).

<u>Question 6 - Is the LBB application only for elimination of pipe whip restraints and jet impingement shields, or also for environmental qualification, containment sizing, ECCS, etc. in your country?</u>

- **Canadian reply:** <u>Industry</u> -- For Zr-Nb pressure tubes, LBB is used to demonstrate that in the event of delayed hydride cracking initiation from a flaw in a pressure tube with limiting properties, followed by crack penetration through the wall, the leaking axial crack will be detected and the reactor unit will be shut down prior to the through-wall crack reaching the critical crack length. The LBB evaluation includes reactor operator response to moisture in the annulus gas system (AGS) that would indicate a leaking pressure tube. When the measures of moisture reach a threshold, the operator invokes reactor shutdown procedures. In the LBB evaluation, the LBB scenario is simulated over a number of hours from time of crack penetration through the wall and first leakage until the time when the reactor is in a cold shutdown state. The LBB evaluation is used to ensure that reactor-operating procedures are adequate for the postulated event of pressure tube leakage.

  LBB as applied for design of Class 1 piping systems was only for dynamic effects associated with the postulated rupture.

  Again, LBB concepts are employed in the industry's Steam Generator Tube and Feeder Fitness-for-Service Guidelines.

  <u>Government</u> -- Only for the local dynamic affects such as pipe whip restraints and jet impingement shields and for the purpose of supporting continued operation of degraded components. We do not allow for global effects such as harsh environmental qualification, containment sizing, and ECCS.
- **Czech industry reply:** LBB application is only for pipe-whip-restraint elimination.
- **Belgium industry reply:** LBB has been applied only for dynamic aspects (elimination of pipe whip restraints and jet impingement shields).
- **Finnish regulatory reply:** LBB is applied just for elimination of the hardware, protecting from local dynamic effects, as Para. 2.2.2 of YVL 3.5 explains. Global blowdown effects to RPV and its internals remain in the design basis. For their mitigation, the main coolant loops of our EPR will be supplied by the licensee's proposal with N4-plant-type whip restraints, though final analyses suggest adequate margins even

in case of postulated non-restrained DEGB. This is presently characterized as a 3-step defense-in-depth approach: (1) Break Preclusion/LBB technology enhances reliability, (2) the hardware provides design basis margins in case of postulated restrained DEGB, and (3) best-estimate design extension analyses demonstrate retaining the needed safety functions even in case of non-restrained DEGB. The final regulatory policy, possibly more favorable to LBB, will be established in the on-going revision of our YVL Guides.

- **French industry reply:** The consequences considered in LBB application are:
  - no more LBLOCA and steam line break in the design transient list;
  - no more pipe whip restraints and jet impingement shields needed;
  - allows for static simplified analysis of component support and stability;
  - no LBLOCA consideration for internal, core support, and fuel support;
  - refined the leak-detection system inside containment, no leakage detection system for steam line outside containment;
  - No consequences on either containment sizing or ECCS design rules and environmental qualification program; and
  - Still open for discussion in France is the consequences of eliminating the LBLOCA design requirements on the steam generator manholes.
- **German industry reply:** It is only for the elimination of pipe whip restraints and jet impingement shields. However, even applying LBB we still have to consider a leakage area of 10% of the pipe cross section area (0,1A) to calculate reaction and jet forces acting on pipes, components, component internals and parts of buildings, see Figure 17.
- **Spanish industry reply:** The aim of LBB applications in Spain had been only the elimination of pipe whip restraints and jet impingement shields.
- **Sweden regulatory reply:** LBB is for accounting for local dynamic effects which means mainly not having to install (or possibly remove) pipe whip restraints or jet impingement shields. Environmental qualification can also be an issue if the consequences of a pipe break can imply damaging sensitive electrical equipment.
- **UK regulatory reply:** See detailed reply in Appendix F.
- **China (by G. Wilkowski):** Probably only for pipe whip restraints and jet impingement shields.
- **India government reply:** Yes.
- **Japanese regulatory reply:** I think LBB is only applicable for pipe whip restraints, energy absorbers and jet impingement shields. Design for ECCS, etc. are safety and defense-in-depth issues.
- **Korean regulatory reply:** LBB application was only for elimination of dynamic effect. Other applications such as ECCS and environmental qualification were not considered.
- **Taiwan industry reply:** N/A
- **South Africa industry reply:** We are utilizing the LBB-application to exclude the sudden DEGB from the piping system in question. The result of a sudden DEGB is twofold:
  1. It generates internal pressure differentials across the core structures ceramics and core support structures.
  2. Inducing lift-off forces that would increase the potential for plated-out radionuclides and settled dust to contribute to the release source term.
- **Argentina regulatory Reply:** It is also for environmental qualification, containment sizing, ECCS, etc.

- **Brazilian industry reply:** In Angra 1, the LBB was approved to eliminate some whip restraints. In addition, because we introduced some modification in the primary system during the steam generators replacement (power up rate, SG snubbers elimination), LBB concept was used to do the structural qualification of the reactor vessel internals and primary equipment supports and nozzles.

Primary System (RSK Guidelines, Chapter 21.1, Version 3/1984)		
Component	*Leaks and Breaks to be postulated*	*Effects*
Reactor Coolant Lines	• 0,1A, 15 ms linear	• Pressure waves (RPV internals)
	• 0,1A, steady-state blowdown	• Jet forces (piping, components, building) • Reaction forces (piping, components, building)
	• 2A	• LOCA analysis • Containment (increase of pressure) • Pressure differences (building) • Qualification of I&C
Circumferential Nozzle Weld	• pAS, S=2	• Stability fo the components (e.g. RPV, SG, RCP, PRZ)
RPV Leak	• 20 cm² Leak	• RPV supporting • RPV internals • LOCA analysis
Austenitic connection lines with DN>200 mm (surgeline, ECCS up to the 1st isolation)	• 0,1A	• Jet forces (piping, components, building) • Reaction forces (piping, components, building)

RPV=Reactor Pressure Vessel; SG=Steam Generator; RCP=Reactor Coolant Pump; PRZ=Pressurizer; ECCS=Emergency Core Cooling System

Figure 17 Postulated leaks and breaks for the primary pressure boundary in Germany

Question 7 - Would your country ever be interested in adopting a probabilistic approach, or combined deterministic-probabilistic approach for LBB?

In general there seemed to be interest in probabilistic approaches, but with some of the below caveats.

- **Canadian reply:** Industry -- For Zr-Nb pressure tubes, Clause 7.4.3 of CSA N285.8 permits a probabilistic evaluation of LBB. The integrated probability over the evaluation period of delayed hydride cracking initiation from a flaw, followed by the subsequent increase in axial length of the growing crack exceeding the critical crack length, must be less than the maximum acceptable probability for the reactor core damage tolerance. The maximum acceptable probability is provided in Annex C of CSA N285.8. Probabilistic LBB is a part of the probabilistic core assessments of crack initiation that are performed on a regular basis for a number of Canadian CANDU plants. In general, the industry's Steam Generator Tube and Feeder Fitness-for-Service Guidelines permit the use of probabilistic approaches.
  Government -- We are currently reviewing the acceptance of probabilistic LBB approaches in terms of applicable fracture mechanic methodologies for assessing failure. These methodologies will need to be validated against regulatory QA requirements.
- **Belgium industry reply:** No specific need for the moment.
- **Czech industry reply:** Yes, we would be interested in any progress related to the LBB.
- **Finnish reply:** Probabilistic approaches are well implemented in Finland. For instance, YVL 3.8 prescribes RI-ISI application to ISI program planning to supplement deterministic approaches. According to YVL 3.5, PTS analyses shall be done using both deterministic and probabilistic methods. A similar trend could be anticipated around LBB in the long term.

- **French industry reply:** Not for design rules of future plants where large deterministic margins are required, but yes, it is under consideration for existing plants and a LBLOCA redefinition project start recently (TBS) in France.
- **German industry reply:** The German authorities and regulatory bodies do not accept probabilistic approaches for LBB or break exclusion and therefore it is not applied. Of course, industry is interested in probabilistic approaches and MPA is involved in some developments for future application.
- **Spanish industry reply:** Currently this is an open point in Spain to satisfactorily demonstrate the fulfillment of the LBB criteria for the Alloy 82/182 dissimilar metal weld between the pressurizer and the surge line after the mitigation by weld overlay. For this purpose, probabilistic approaches could be considered. However, industry and regulator are awaiting the developments in the US.
- **Sweden reply:** So far, probabilistic evaluations have been used to support the deterministic evaluations in the LBB-applications to SSM. I doubt that we will approve LBB based only on a probabilistic approach. A combined deterministic-probabilistic approach would possibly be a better alternative.
- **UK reply:** See detailed reply in Appendix F.
- **China (by G. Wilkowski):** Unknown, but probably not for new plants in the immediate future.
- **India government reply:** Yes.
- **Japanese reply:** JAEA (Japan Atomic Energy Agency, Old JAERI) and TEPCO are interested in Probabilistic LBB, see References 38 and 39.
- **Korean reply:** KINS is interested in adopting a probabilistic approach. In addition, KINS is interested in LBB application for the small diameter piping of ~6-inch to ~10-inch diameter pipes.
- **Taiwan industry reply:** I believe that both the probabilistic approach and combined deterministic-probabilistic approach will be accepted.
- **South Africa industry reply:** The regulator has indicated that due to the first-of-a-kind (FOAK) nature of the PBMR plant, and the lack of operating experience in high temperature helium-cooled reactors, he would not accept a probabilistic approach.

  I would say the best we can motivate at this stage of the project is a combined deterministic-probabilistic approach, and in future move to a probabilistic approach.
- **Argentina Reply:** Argentina would be interested in combined deterministic-probabilistic approach.
- **Brazilian industry reply:** We can consider this possibility for Angra 3 (under construction). However, we still need more information to decide.

4 POSSIBLE DIRECTIONS FOR FUTURE LBB PROCEDURES

This section discusses the main objective of this report. All the prior information was provided for background and for understanding the points discussed below. Possible future directions for LBB analyses can be classified into the following groups;

- Deterministic,
- Probabilistic, or
- Hybrid deterministic/probabilistic approaches.

It may also be necessary to give special consideration for LBB application to new plants versus LBB applications for older plants subjected to some new degradation mechanism; however, in the following work we are considering LBB application only to new plants.

4.1 Deterministic Procedures

Virtually all existing LBB procedures are deterministic and generally based on some flaw tolerance capability of the material. The US NRC SRP 3.6.3 LBB procedure effectively states that the pipe system of interest should not be susceptible to any degradation mechanisms, and that the stresses are low, but it is necessary to make the specified flaw tolerance analysis. Although there is a screening criterion for the degradation mechanisms and high-unknown loads (i.e., water hammer), there is still the possibility that some unforeseen degradation mechanism might develop. Accounting for potential future degradation mechanisms is probably the weakest part of existing LBB procedures and where improvements are necessary. A suggestion is given below.

4.1.1 Advantages

Deterministic procedures are generally quite straightforward. The most simplistic approaches should have large safety factors applied to account for unknown factors. For instance, the Option 1 approach in NUREG/CR-6765 was a very simplistic LBB approach that was designed to have significantly large enough margins that past piping systems that easily passed prior NRC 3.6.3 LBB analyses, would also pass the simple Option 1 analysis, but perhaps with not as much margin.

If the screening criterion from SRP 3.6.3 was really sure to be met throughout the life of the plant, then LBB would be an easy assessment. This is an easy regulatory approval, since the SRP 3.6.3 screening criterion effectively said that LBB can only be applied if no cracks will ever occur and that there will be no unknown high applied stresses, but the applicant must still conduct a simplistic flaw tolerance analysis even though there will never be a crack of that size.

If there were never any flaws occurring in service, then LBB would be applicable forever for that case. However, there are cases where unexpected degradation mechanisms have occurred.

4.1.2 Disadvantages and Significant Difficulties

The biggest disadvantages and difficulties for LBB analyses are the need to make it effective for any degradation mechanism that ever occurs. Additionally, plants are no longer being designed for just 40 years, but 60 year or longer lives. Some of these aspects are addressed below and how they might impact deterministic LBB analyses.

4.1.2.1 Accounting for Future Degradation Mechanisms

In some international LBB procedures, efforts are being made to determine how big a flaw might grow in the lifetime of the plants, but those are relatively simple degradation mechanisms like mechanical fatigue. Since most piping design codes are created with rules to prevent fatigue crack growth (and they do a good job for that), conducting an additional fatigue analysis for LBB seems redundant and unnecessary.

Perhaps one of the major improvements to LBB analyses is to ensure that they will be effective even if a new degradation mechanism occurs that is not accounted for in the piping design rules. As an example, the occurrence of PWSCC in primary loop of PWR plants started 20 years after initial operation and is currently a problem for LBB. Had LBB been developed earlier than the mid-1980s in the US, BWR plants in the US might have been approved for LBB prior to the development of IGSCC cracking.

The worst-case degradation mechanism that might occur is one that causes very long surface cracks that might not leak before breaking. Currently the mechanisms that nuclear piping has experienced that produce very long surface flaws are erosion-corrosion, corrosion-fatigue, and SCC. Creep and creep-fatigue degradation mechanisms in high temperature piping also have the potential to cause long surface cracks where there may be no protection from leakage detection before a failure. Fortunately, erosion-corrosion does not occur in primary nuclear piping made from austenitic material or clad with stainless steels. Additionally, LWR plants do not have problems with creep and creep/fatigue (except a rare possible occurrence in a CANDU feeder tube), so that mechanism can generally be eliminated. That leaves corrosion-fatigue and SCC.

Both of these mechanisms occur due to a combination of the susceptibility of the material, environment, and sufficiently high tensile stresses. Corrosion-fatigue has occurred in feedwater nozzles due to high thermal fatigue loading where cracks initiated from small pits or stress risers at nozzles[40]. Those nozzles had no stainless cladding to prevent the corrosion of the ferritic material. There was also large thermal stratification stresses in those nozzle-cracking cases, where the cyclic thermal stresses were not included in the original piping design analysis. Designers are much more astute about thermal stresses now compared to 30-years ago. Therefore, these types of failure occurrences from unaccounted thermal cyclic stresses are much less likely to occur. Vigilance in the review of the stress analysis of any piping system subjected to thermal cyclic loading is needed. Surge lines are susceptible to high thermal fatigue loads, but have not had failures do date. Significantly improved thermal and *insitu* monitoring has been conducted and continues today. As a result, operational procedures have been changed to reduce the thermal gradients in many plants to stay within the fatigue-life usage-factor limits from the piping design codes.

After 30 to 40 years of experience, the nuclear power industry knows what materials should have better SCC resistance, and in some cases, the industry has modified the water chemistry for better SCC resistance. However, there is no assurance that the SCC may just slow down. Furthermore, piping design codes are silent about SCC, which is the most prevalent failure mode for nuclear piping. What has not been done to date is to take advantage of current knowledge on how to fabricate welds so that there would be compressive or very low tensile residual stresses on the ID (wetted surface) of the piping. If the stresses were compressive, then material sensitivity and water chemistry are not a concern.

Consequently, a consideration for a possible improved LBB deterministic procedure is to require a more rigorous LBB analysis for avoiding breaks by SCC, unless weld fabrication procedures are used that induce compressive stresses on the ID surface at normal operating temperatures. This can be accomplished by careful control of weld sequencing and not allowing hard grinding on the ID surface that also produced biaxial tension stresses. Some ways of making "Fabrication Enhanced SCC Resistance Welds" are:

1. Weld sequencing in the past has been done to make a good-looking weld for passing the preservice NDE inspection. In many cases, the roots were ground out and final ID weld passes made to eliminate the root defects after the entire weld has been completed, see Figure 18a and b. The weld residual stresses are tensile and highest where the last weld pass was completed, so this procedure results in the most severe conditions. Instead, the procedure should involve completing part of the main weld, then grinding out the root defects and making the required ID weld passes, and then finally finishing the rest of the weld to the OD, see Figure 19.

2. If thermal shields need to be welded close to a girth weld, the shield should be welded first before finishing the entire weld. If ID fill-in welds are needed, then these should be completed before finishing the main weld, see Figure 18c.

3. If repairs are needed on the ID surface, they should be done before finishing the main weld.

4. If a safe end is used, then design the length of the safe end so that when making the second weld (field weld) the initial safe-end weld would have compressive stresses induced from the field weld.

5. Welds are frequently ground smooth on the ID to eliminate UT reflectors. Hard grinding that causes the surface to heat up significantly will produce tensile biaxial stresses that might be above the ultimate strength of the virgin base metal. This can accelerate crack initiation as was experienced in many BWR plants. The final ID grinding should be at a slower rate that just plastically deforms the surface without over-heating. This will produce compressive biaxial stresses on the ID surface.

Many of these welding procedures can be used to obtain "Fabrication Enhanced SCC Resistance Welds" and can easily be implemented with very little cost impact, see example in Figure 19.

If "Fabrication Enhanced SCC Resistance Welds" are not used, then the LBB procedure should become much more rigorous and involve assuming that a SCC crack can develop. Such analysis would consist of modeling the SCC crack growth as in Reference [41], using SCC crack morphology parameters for the leakage detection, and if appropriate account for constraint reductions in toughness due to a possible complex crack shape (through-wall crack with a surface crack in the rest of the circumferential plane) being developed[42,43].

If “Fabrication Enhanced SCC Resistance Welds” are used, then the LBB procedure can be reduced to a relatively simple flaw tolerance analysis akin to the SRP3.6.3, but perhaps with some of the improvements suggested in the Level 2 approach in NUREG/CR-6765. For the leak-rate analyses in such an approach, the use of corrosion-fatigue crack morphology parameter should be used as a worst-case assumption. Past practices of using an air fatigue crack for leak-rate analysis is not acceptable since cracks generally initiate on the ID surface where there will be additional roughness on the crack surface from water/inclusion impurity interactions.

4.1.2.2 Additional Deterministic Considerations

One additional area of consideration is having a better definition of material properties, and encouraging the use of better materials. Some considerations are given below.

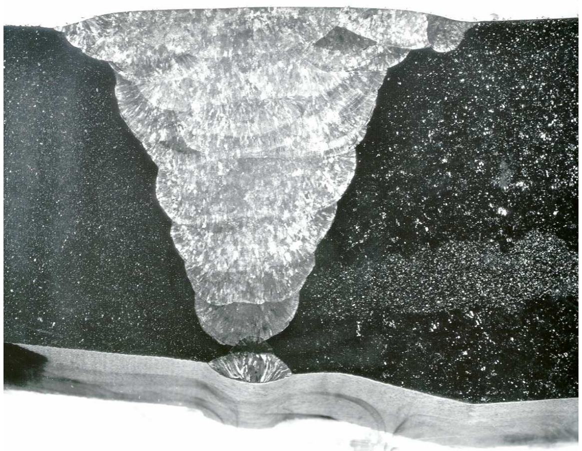
- Since future nuclear power plants will be realistically built for more than a 40-year life (at least 60 and maybe more), aging effects on the material properties should be included. It is well known that thermal aging can have a significant effect on reducing the toughness of some cast stainless steels[44]. However, all piping materials will expect thermal aging. Thermal aging will increase the strength and reduce the ductility. A little thermal aging might even be helpful in increasing the cracked piping load-carrying capacity due to the higher strength, and perhaps insignificant effect on the toughness loss. However, stainless steel welds that are already in the EPFM range have been noted to lose up to 40 percent of their toughness with thermal aging, which can be a significant effect for LBB(b).

- Ferritic steels are sensitive to toughness loss by dynamic strain aging[45,46]. By choosing the proper cooling rate in the steel processing, the detrimental effects can be eliminated. This screening of weld procedures to avoid detrimental effects of dynamic strain aging needs to be done for the weld metals as well as the base metals.

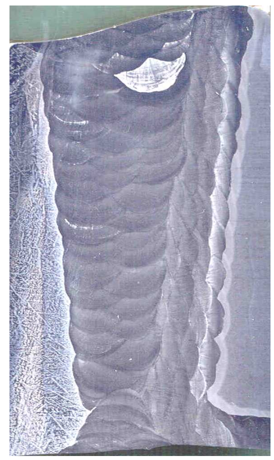
- Virtually all nuclear plant designs involve dissimilar metal welds (DMWs) at some locations in the piping system. The very early US plants and French plants used a stainless steel weld filler metal for DMWs. The concern with such stainless steel welds was the lower toughness of the ferritic/stainless weld HAZ and potential fatigue crack initiation/growth from the thermal strain. Later piping welds in the US used Inconel buttering with stainless filler, but most often in the US alloy 82/182 buttering and filler metal were used. SRP 3.6.3 Rev 1 prohibits the application of LBB to lines using alloy 82/182 weld metals and allows LBB for In52/152 welds. However, there is uncertainty if the In52/152 welds will really have the long-term SCC avoidance performance. An interesting hybrid approach we have seen was to use stainless weld metal for the root and two layers of hot passes over the root and then use Inconel weld metal for the filler passes. This hybrid approach is worth considering with additional thermal stress analysis.

(b) Briefly mentioned in a proprietary LBB submittal and examined in NUREG/CR-6428 “Effects of Thermal Aging on Fracture Toughness and Charpy Impact Strength of Stainless Steel Pipe Welds,” May 1996.

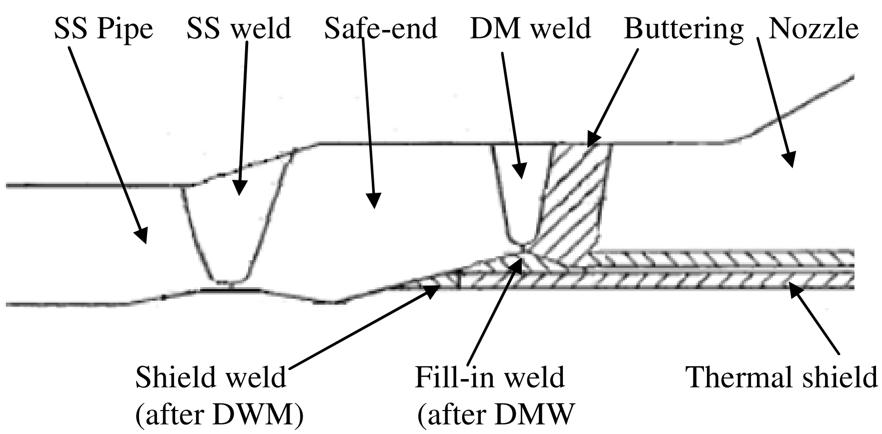
Nevertheless, weld metal selection alone might not provide the total answer here, and "Fabrication Enhanced SCC Resistant Welding Procedures" may be the best route short of putting a preemptive weld overlay on the DMW during construction.



(a) Weld from BWR plant with lots of IGSCC (ID weld made last and heavy ID grinding)



(b) DM Weld from cold-leg pipe (Back-gouged ID weld made last with grinding)



(c) Surge nozzle with thermal shield, fill-in weld, and shield weld

Figure 18 Examples of weld procedures in the past that were not good for SCC resistance

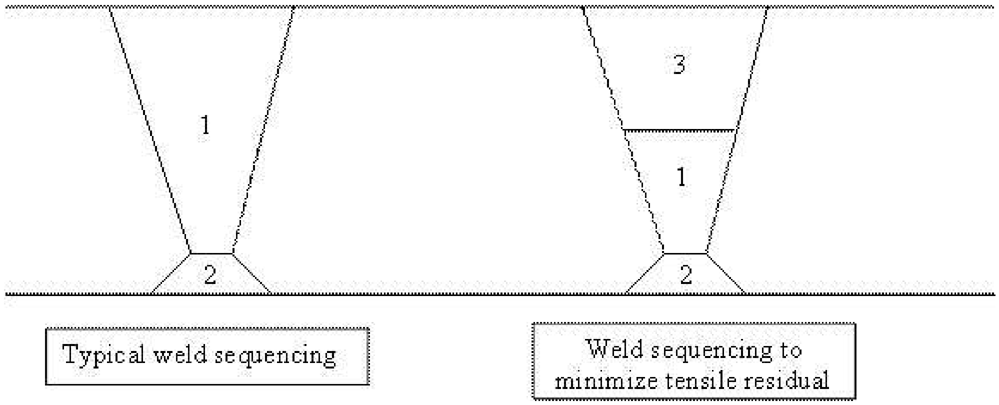


Figure 19 Example of changing weld sequencing to improve residual stresses

4.2 Probabilistic LBB Approaches

There are many probabilistic piping fracture mechanics codes as was noted earlier in this report. PRAISE was the first one developed and was done for the US NRC relative to the original Generic Issue A-2 on asymmetric blow-down loads[47]. It was updated several times and PC-PRAISE may be the latest version[48]. As noted previously, there are also a number of international piping fracture mechanics codes such as PRODIGAL, NURBIM, etc.

During an NRC project conducted jointly between Battelle-Columbus and Emc^2, a new probabilistic computer code called PRO-LOCA was developed[49,50]. The application of this code was for support of the Transition Break Size changes being proposed to the US 10FCR50.46 rule for ECCS line sizing. The TBS ruling is actually an extension of LBB to determine if the double-ended guillotine break (DEGB) failure probability of a large-diameter pipe so small as to be unlikely to ever occur (10E^{-6} probability of occurrence per year is a typical lowest probability for an unlikely event). The initial PRO-LOCA code was started in that effort, but much of the NRC funds needed to be focused on the elicitation efforts that resulted in NUREG-1829[34] and the technical basis for the draft Reg Guide on TBS to be developed. Nevertheless, the USNRC Commissioners required the USNRC staff to continue the development of a probabilistic code and make assessments every 10 years since they did not believe the elicitation results would capture new degradation mechanisms more than 10 years from the present time.

To continue with the PRO-LOCA probabilistic code development, the USNRC and other internal organizations (including CNSC and the COG) funded the MERIT group program at Battelle-Columbus. That program was completed in 2008, and involved development of a proprietary version of the PRO-LOCA code to members of that group program. The draft final report is being reviewed by the members of that project.

However, for those unfamiliar with these codes, the most important aspect to remember is that they are simply a series of deterministic runs. Consequently, it is essential that the deterministic model needs to be appropriately ideal with clearly defined uncertainties. The user also needs to know much more about the statistical variation of the inputs (applied loads, residual stresses, material properties (strength and toughness), subcritical crack growth behavior, and inspection capabilities). Furthermore, if the probabilistic models are trying to determine the absolute probability of failure, the degradation mechanisms that might occur over the lifetime have to be known and quantified statistically.

4.2.1 Advantages

The key advantage of having a probabilistic code is to quantify the risk and determine if the probability of failure is acceptable or not. The USNRC frequently uses risk-informed decision making, but may not depend solely on the probabilistic evaluation.

If one was aware of all of the degradation mechanisms, it would also be possible to separate the most uncertain aspects, and determine if the failure probability was acceptable even with the worst-case unknown assumption. For example, what would happen if no time was needed for crack initiation?

Probabilistic analyses are best when the statistical variation of the input variables are known. Hence, it is probably not good for first-of-a-kind reactor application as was noted by the South African Pebble Bed Reactor staff in their response to our questionnaire. For design purposes, simpler deterministic approaches are more desirable. However, for operating plants that have some unique problems, a probabilistic approach may be useful to assess the minimum time between detectable leakage and break at a postulated transient event.

4.2.2 Disadvantages and Significant Difficulties

One of the very difficult aspects in probabilistic coding is quantifying the time to crack initiation and possible initiation sites for corrosion fatigue or SCC. The crack initiation time is best validated by service history, so this is a difficult analysis step and conservative assumptions are frequently needed. Some probabilistic codes assume cracks only grow from weld defects, but that is seldom true for SCC. Other probabilistic codes may look at past SCC service history and put in a flaw distribution based on that result, but may only put in one flaw per pipe weld at most. In reality, if there is a susceptible weld to a degradation mechanism, then it is more likely that there may be a 2nd, 3rd, or 4th crack initiation site, see Reference 51. Those sites could be randomly located or biased based on service history experience with that mechanism.

Another aspect on the initiation behavior of cracks involves determining the number of initiation sites required to breakdown each pipe girth weld and the associated number of subunits. PC-PRAISE has used 2-inch long subunits for corrosion fatigue based on laboratory fatigue tests[52]. The issue of the subunit size was examined further during Emc^2 analyses(c) by considering one crack per pipe girth weld to 44 cracks per pipe girth weld. *The result (shown in Figure 21) was that the failure probabilities changed by five orders of magnitude with this single assumption in the PRO-LOCA code*. The most severe case used spacing comparable to IGSCC cracking in BWRs, but current service history PWSCC cracks do not have as many crack initiation sites as IGSCC cracks. Obviously, this is one of the parameters that needs careful review and a strong technical basis. It cannot be treated as a benign variable that can be adjusted by the user.

One of the disadvantages of a probabilistic code is that the embedded deterministic analyses need to be relatively simple since these procedures may be performed more than a million times. That

(c) Not previously published.

means some aspects, like variations in residual stresses, arbitrary growth of the shape of a SCC, nonlinear dynamic analyses of a cracked pipe system under seismic loading, etc., need to be very simplified or overly conservative.

Consequently, there has to be a great deal of validation and knowledge of the probabilistic code before one can have full confidence in their use. At a minimum, deterministic runs are needed to assess the reasonableness of the output, especially if the probabilistic analyses are oversimplified.

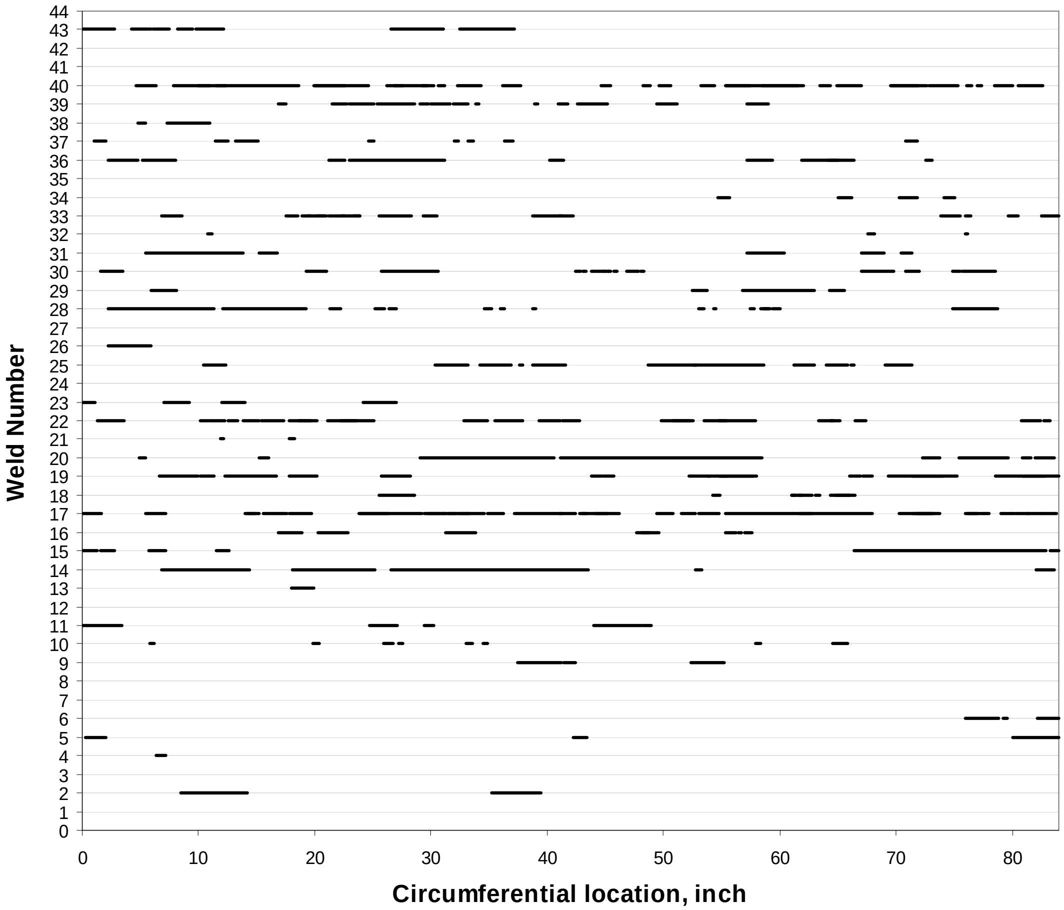


Figure 20 Dye penetrant results of IGSCC cracks from 28-inch diameter main recirculation-line pipe welds removed from service for inspection and evaluation

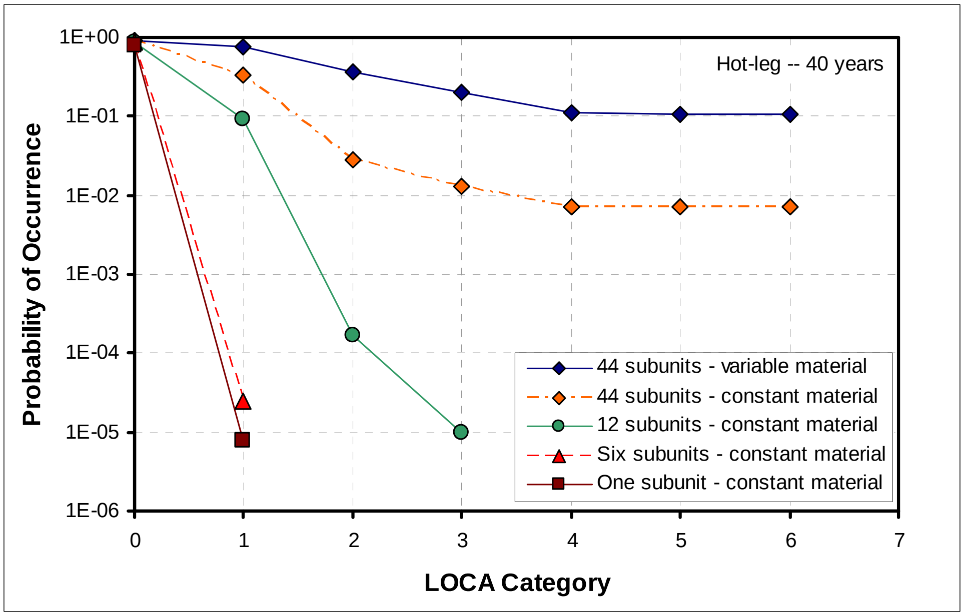


Figure 21 Illustration how the selection of number of subunits around the pipe circumference can affect the calculated failure probability

4.3 Hybrid Deterministic/Probabilistic Approaches

A hybrid deterministic/probabilistic approach is typically one where the analysis might be more computationally involved than could be achieved in a full probabilistic code, and hence incorporates only some probabilistic aspects.

The disadvantage of a hybrid deterministic/probabilistic approach is that it will not produce an absolute probability of failure and it would require more effort than just a simple deterministic analysis.

The advantage of a hybrid deterministic/probabilistic analysis is that it can be selectively designed to conduct more detailed analyses than are possible in probabilistic analyses alone for critical aspects (i.e., FE nonlinear analyses), and still include some probabilistic aspects. As an example, the fracture analyses could incorporate normal SSE design loads, as well as a seismic load that corresponds to a 10^{-6} event. Hence, if the analysis assumes the leakage flaw size occurred with a conditional probability of 1, and the flaw could tolerate the 10^{-6} loading, then the failure probability would be greater than 10^{-6} and could still be considered an unlikely event. Of course, this also assumes that other assumptions in the deterministic flaw size analysis are realistic or reasonably bounding.

One place where a hybrid deterministic/probabilistic approach was used for piping flaw assessment was in the “Seismic Considerations for the Transition Break Size” in NUREG-1903[35,53].

4.3.1 Example of a Hybrid Deterministic/Probabilistic Analysis

The following is an example of a hybrid deterministic/probabilistic approach called a "Robust LBB Approach"[54]. This "Robust LBB Approach" involved the steps outlined below, where perhaps only the first four steps are needed in most cases, but the later three might result in extra relief for additional considerations.

1. Use a conservative degradation mechanism assumption.
    a. Assume the crack growth mechanism is SCC, unless "Fabrication Enhanced SCC Resistant Welding Procedures" are used in construction. Otherwise, use a corrosion-fatigue (CF) mechanism.
    b. There may be large margins in such an assumption since stress corrosion cracking (SCC) has a more severe growth rate and flaw shape than simple fatigue mechanisms.

2. Determine time for the flaw to grow from workmanship flaw size (or UT detectable flaw size) to leakage detectable crack at normal operating conditions.
    a. This time period may actually be decades to develop the SCC cracks. Generally, such analysis does not include time to initiate the crack, which is common practice but a very conservative simplification.
    b. Multiple flaw initiation sites should be included in the analysis, which may change the leaking crack shape to a complex crack (long surface crack with much shorter through-wall crack in the same plane), see Figure 22b as an example.
    c. Conduct this analysis with the traditional safety factor of 10 on leakage, as well as a safety factor of three, which is the range of most leak-rate predictions compared to experimental results[42,43] (see Figure 23), and is also consistent with probabilistic analyses on varying the crack morphology parameters[55] as illustrated in Figure 24.
    d. This analysis may be helpful in determining the margins for in-service inspection intervals later in life.

3. Determine the time from leakage detection to the critical flaw size using static fracture mechanics analysis with <u>*design SSE*</u> loads and safety factor on load. This is a more traditional LBB type analysis as per NRC SRP 3.6.3 except the SCC circumferential flaw shape (see Figure 22) may be different from an idealized circumferential through-wall crack.
    a. The time from leakage detection to reaching the critical flaw size might be fractions of a year or longer.
    b. Note that since there could be a long surface crack from Step 2, there may be a constraint effect from a complex crack that can reduce the fracture resistance in the pipe[42,43].
    c. Use the design SSE stresses with a safety factor of 2 on the leakage crack size for the fracture analysis. The leakage crack size in this analysis should be the one with the SF of 10 applied from Step 3.
    d. This analysis result will generally be conservative because typically, plant piping stress analysis is elastically calculated, but the fracture analyses are all nonlinear.

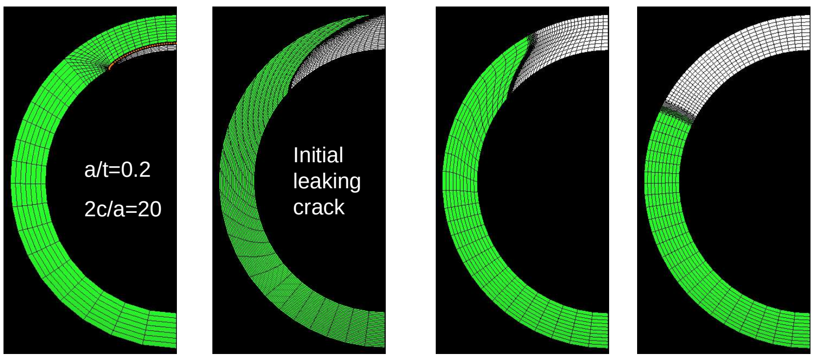
e. This analysis can determine the margins on time for the leak-detection system to shut the plant down. In discussions with US industry representatives, the actual shut down time might be on the order of just less than a week, so some margin in multiples of weeks might be in order.

4. Determine the time from leakage detection to the critical flaw size using static fracture mechanics analysis with $\underline{10^{-6}\text{ seismic loads}}$ and minimal safety factors.
    a. This is a check analysis to see if there is inherent protection from the more traditional LBB analyses with all safety factors applied, versus having a low-probability but high-amplitude seismic loading
    b. The time might be fractions of a year or longer.
    c. Note that there will be a constraint effect from a complex crack that can reduce the fracture resistance in the pipe[42,43].
    d. Since this is a low-probability event, the margins can be reduced. Use the $\underline{10^{-6}}$ seismic event amplitude with a safety factor of 2 on the leakage crack size.
        i. The plant-specific seismic-hazard curve can be used to elastically scale the seismic stresses to the peak ground acceleration (PGA), which would be very conservative.
        ii. A relatively simplistic nonlinear stress correction was developed in NUREG-1903 and may be needed for the 10^{-6} seismic stress. Figure 25 shows this nonlinear-stress correction, which is described in detail in NUREG-1903.
        iii. The leakage crack size in this analysis should be the one with the SF of 3 applied.
    e. This analysis result will generally be conservative because typically, plant piping stress analysis is elastically calculated, but the fracture analyses are all nonlinear. As determined by nonlinear dynamic cracked-pipe FE analyses in Reference [22], the margins in a full analysis are much greater than the simple nonlinear stress correction factors in Figure 25.
    f. This analysis can determine the margins on the leak-detection system to shut the plant down even under extreme seismic loading.

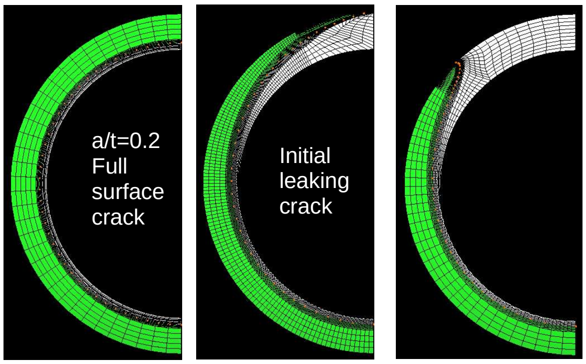
5. If needed, determine the extra margins on critical flaw size using a nonlinear dynamic analysis.
    a. Simple static analysis in the prior step assumes all the stresses are elastic and can be added and used in a nonlinear fracture analysis. In reality, plasticity will occur and the displacement-controlled stresses are not linearly additive to the load-controlled stresses. Using nonlinear dynamic analyses gives a longer critical flaw length and hence a longer time needed to grow a crack to the more realistic critical crack size.
    b. The margins gained here can be significant since there will also be higher damping in the whole pipe loop under the higher-amplitude loading[56].
    c. This step gives a larger margin on the leak-detection time for the $\underline{10^{-6}}$ loading, but also for the SSE loading.

6. If needed, determine the time from start of a seismic event to complete break of the pipe.
    a. For some applications, it is desirable to further know the time from the initial leakage in the seismic loading to a full break. This time can significantly reduce the thermal-hydraulic loads that are created from the typical safety analysis that assumes a 1-millisecond break time. Past IPIRG pipe-system tests showed that it could take 3 seconds of seismic loading from the start of major leakage to when the pipe is in two pieces.
    b. This assessment comes from the detailed FE analyses in Step 5.
    c. This analysis is easiest to conduct if it only goes up to the point when the crack has just barely gone completely around the pipe, but the pipe is not separated by the jet forces yet. The maximum opening area will be less than the DEGB at this instant in time, see schematic in Figure 26.
    d. This analysis gives the margins on time for reduced thermal hydraulic loads on the core internals, or time needed for the boron injection system, or control rods to start in the RPV depending on the reactor type.

7. If needed, determine the additional time from complete pipe break, to having the pipe move axially and radially (offset separation) from thrust forces to get to the DEGB opening area, see schematic in Figure 26.
    a. This time is probably in fractions of a second.
    b. There would be a slight additional margin for time for slowing down the dynamic decompression, or boron injection system to come on-line, or control rods to work.



(a) Assuming one initial crack with initial flaw of 20% of thickness and 2c/a = 20



(b) Assuming initial 360-degree flaw for bounding multiple initiation sites

Figure 22 Example of development of a SCC

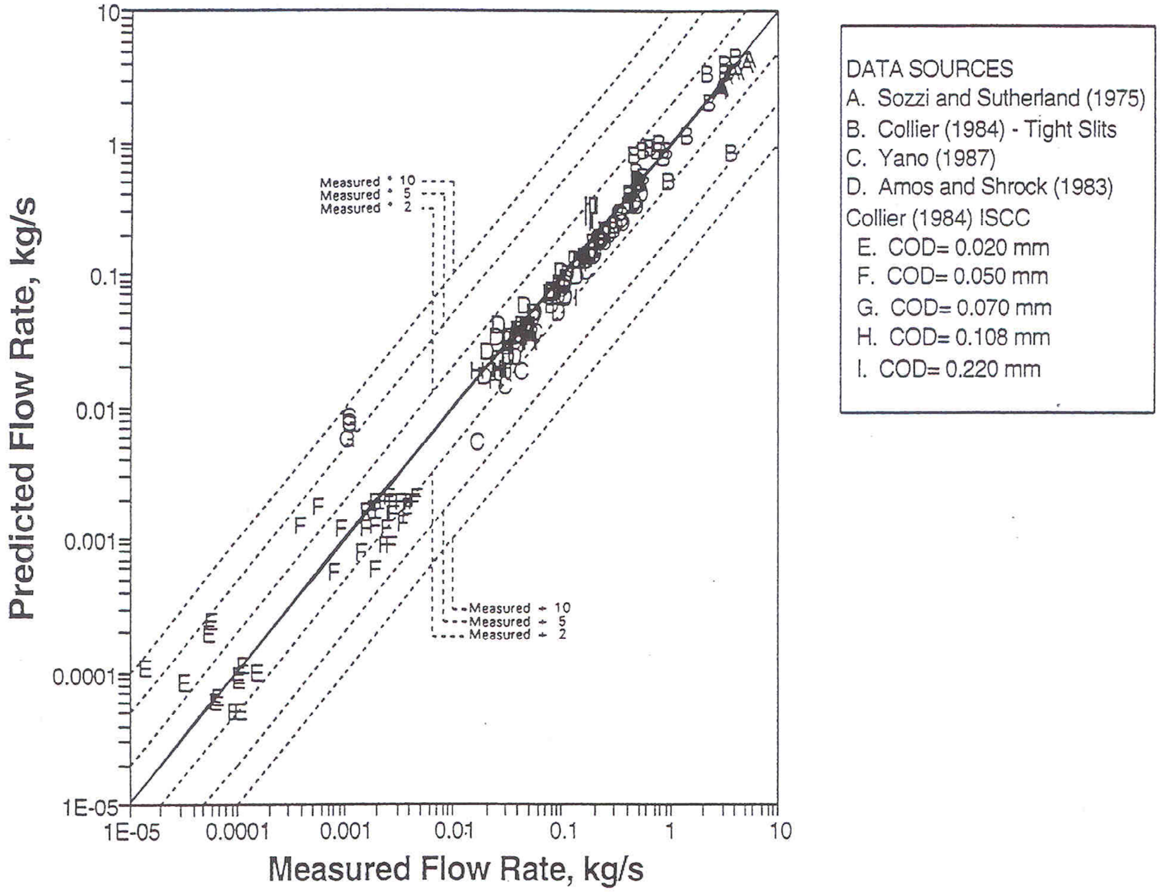
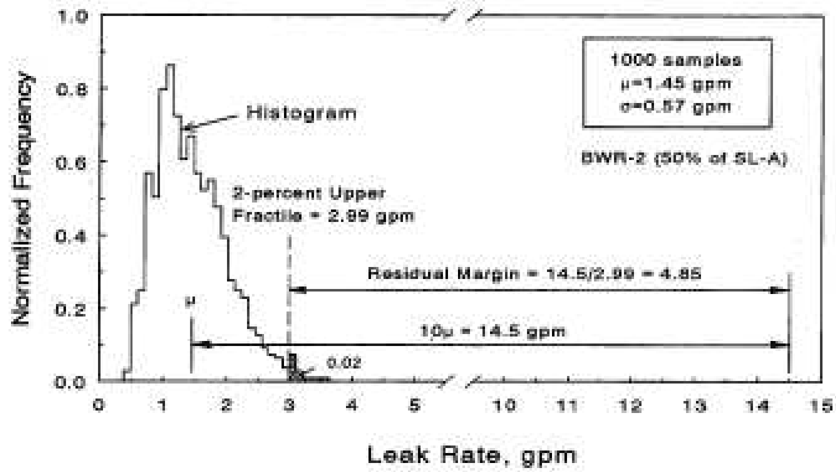
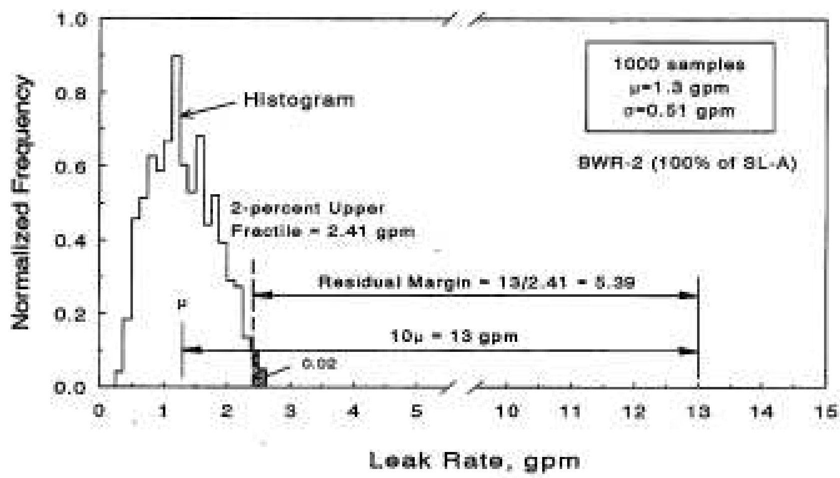


Figure 23 Comparison of various two-phase flow leak-rate tests used to validate the initial SQUIRT model



(a) At normal operating stress of 50% SL A



(b) At normal operating stress of 100% SL A

Figure 24 Effect of statistically varying the crack morphology parameters on the leak rate from Reference 55

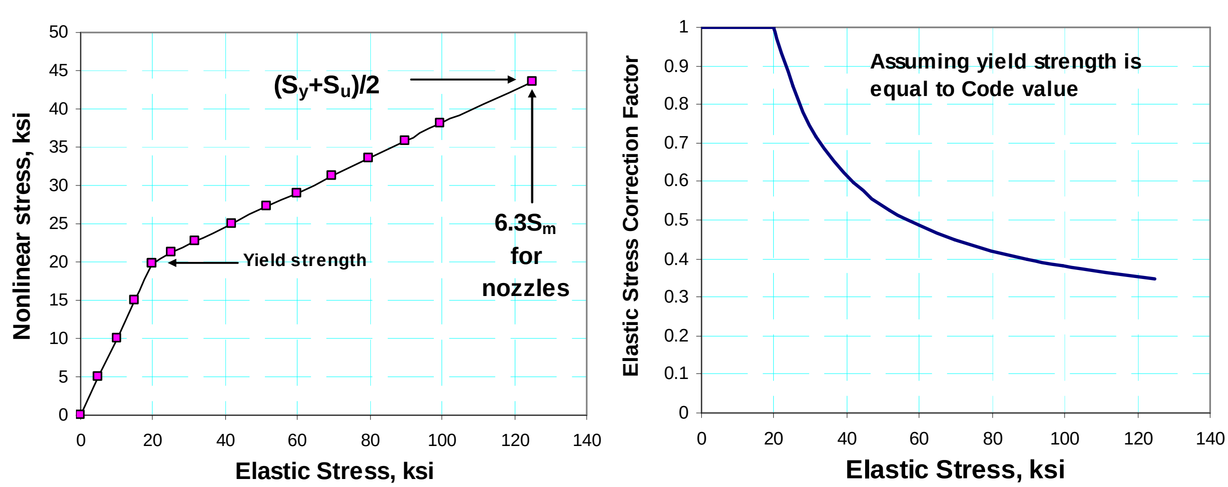
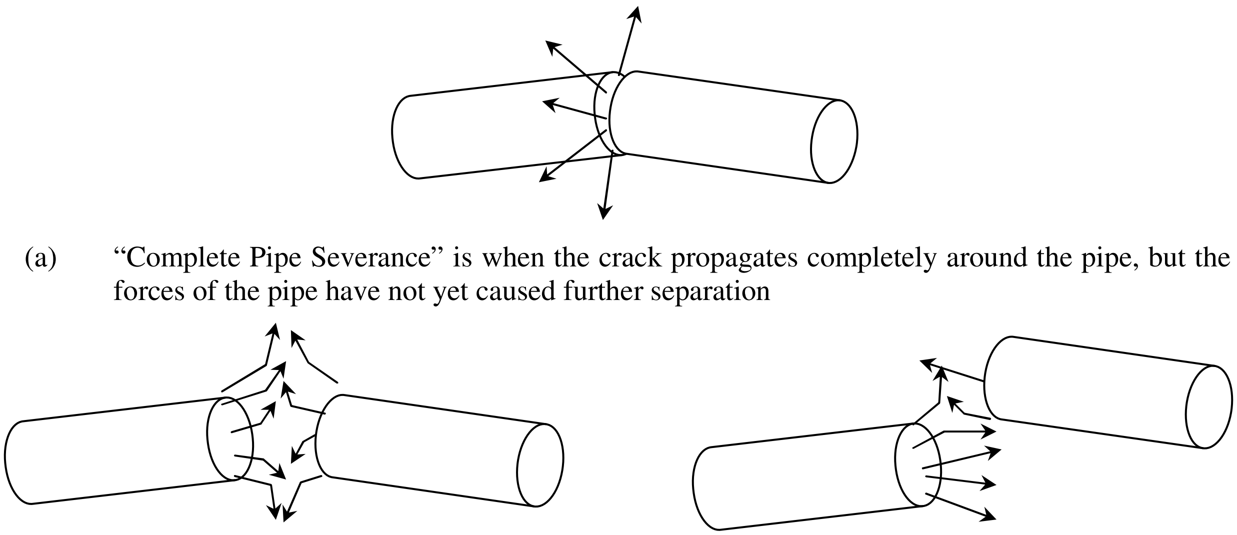


Figure 25 Example of nonlinear correction factor procedure set up in NUREG-1903 for high seismic loading



(a) "Complete Pipe Severance" is when the crack propagates completely around the pipe, but the jet forces of the pipe have not yet caused further separation

(b) Axial separation from jet forces

(c) Axial and radial separation from combined pipe-system restraint and jet forces

Figure 26 Schematic of limits of pipe severance versus full DEGB analysis of pipe motion after pipe severance

(Arrows indicate water jet direction from cracked pipe.)

5 RECOMENDATIONS

The General Design Criteria in the US initiated the concerns for designing to prevent damage from dynamic effects in nuclear power plants; however, that was back when it was believed that nuclear piping might fracture in a brittle manner in milliseconds. Significant efforts have been expended over the last 30 years on development of methodologies and application of LBB to eliminate hardware originally intended to prevent damage from dynamic pipe fracture.

There is now considerable operating experience since the first nuclear plants were designed. For primary loop piping, the main degradation mechanisms to date have caused circumferential flaws by girth welds not axial flaws. Experimental testing has shown that the extremely rapid fracture behavior concern in the early design days does not occur for circumferentially cracked pipes at normal operating temperatures.

The next generation of LBB applications will expand to what might be thought of as more critical to the plant design than just the removal of pipe whip restraints or jet impingement shields. For example, the transition break size (TBS) efforts are aimed at reducing the requirements for the ECCS flow requirements. The draft Reg Guide for the TBS is under preparation, and will be made for application to existing plants. Obviously, this is also an advantage in the cost savings in the design of new plants. However, the TBS also requires the LBB first be satisfied in the traditional NRC SRP 3.6.3 procedure, and then additional analyses are needed for the ECCS evaluations. Other future extensions to LBB might be for equipment qualification, containment sizing, etc. However, for these more significant extensions of LBB methodology, considerations that are more careful are needed since they must be valid for the life of the new plants. Already, there is discussion of plant life extension from 60 years to 80 years for existing plants.

As pointed out in this report, one of the greatest threats to LBB is the potential to develop long surface cracks from stress-corrosion cracking mechanisms. The occurrence of PWSCC in PWR plants was not thought of 20 years ago when LBB was approved for those plants, but it is a major issue at this time. Piping design codes are based on fatigue design resistance and they have performed well in avoiding fatigue failures in primary loop piping. Unfortunately, these design codes have not evolved to require the designer to use methods to avoid the potential for SCC cracking that is the dominate pipe degradation mechanism for primary loop piping. SCC issues have been handled in flaw evaluation procedures in Section XI of the ASME code (and comparable international codes), but that knowledge has not propagated forward to design improvements in the design codes and standards.

SCC improvements could be made by modifying; the water chemistry, materials, stresses on the wetted surface. Piping design codes do not traditionally deal with how to maintain water chemistry or use additives. Materials are generally selected and used from experience, but the Codes do not specify which materials should be used for better SCC resistance. The Codes are very concerned with design stresses, but weld residual stresses and fabrication procedures could readily be included as recommended practices.

Hence, one of the major recommendations is that if the design codes are unwilling to change to give guidance for SCC resistance, then LBB applications should assume that SCC might occur unless some fabrication enhanced SCC resistance procedures are used. This aspect is fundamental for the longevity of the plant and affects all LBB approaches, whether they are probabilistic, deterministic or some hybrid approach. Hence, it is recommended that weld joint designs be studied from a fabrication viewpoint to determine how that could be made with compressive stresses (or significantly reduced tensile stresses) on the ID surface, and still be economically fabricated. This includes not only welding procedures, but also surface grinding and repair welding strategy.

With regards to using deterministic, probabilistic or hybrid approaches, the unofficial survey taken of 17 different countries in this project showed that most of those contacted prefer that conservative deterministic approaches be used. At best, some would be willing to accept probabilistic analyses, but only if that case also passed a conservative deterministic approach first. Probabilistic analyses by themselves were the least favored approach for LBB (especially for new designs).

Probabilistic approaches provide a risk evaluation that provides direct regulatory guidance, but probabilistic analyses are always conditional on the type of analyses conducted and the expertise of the analyst. As an example, one of the worst flaws that ever occurred in nuclear plant piping were IGSCC cracks in the Duane Arnold safe ends by a thermal shield. Those particular welds had very unusual preparation, that is, there was supposed to be groove machined into the ID to fit the thermal sleeves in place, but the fabricator accidently made the groove on the OD, then weld-repaired them. He then cut the groove on the ID and then welded the thermal sleeve in place. This type of fabrication error probably would never be considered in a probabilistic analysis. This is one example to note caution about under estimating the risks from probabilistic analyses. Additionally, the probabilistic analyses generally require relatively simple underlying deterministic models. That is because millions of simulations are needed for Monte Carlo based approaches.

Deterministic analyses have evolved considerably, so that with better understanding the materials and the loads, perhaps the traditional safety factors could be reduced. However, SCC crack shape has never been considered in LBB evaluation procedures. Although conservative deterministic approaches are more readily acceptable to the international community, not having the risk quantified is a drawback.

A hybrid deterministic-probabilistic approach was also suggested, where perhaps more important aspects that require greater analyses aspect than could be done in a probabilistic code could be dealt with by some deterministic sensitivity studies. A couple examples that come to mind are; (1) the crack shape that can develop under SCC is important for flaw stability analyses, but might require weld residual stress analyses and SCC crack growth simulations using advance FEA methods. These can not be done probabilistically since one case may take 24 hours of CPU time on high-end multi-processor machines. Obviously one-million (or more) such analyses are not practical. (2) There are good results showing that piping is much more tolerant of flaws under seismic loading, but nonlinear dynamic analyses might take 10 to 15 hours of CPU time with high-end multiprocessor computers. Again doing such analyses for a million or more simulations is not practical. However, analyses could be conducted at several seismic levels representative of SSE (typically 10^{-3} to 10^{-4} probability of occurrence per year), 10^{-5} and 10^{-6} seismic event.

These are but a few aspects that might have high contributions to changing the risk. It is recommended that a procedure like the "Robust LBB methodology" presented in this report be refined and some sample case studies be conducted to determine the benefits over deterministic analyses, as well as how they could be used to improve the probabilistic analyses.

6 DISCUSSION AND CONCLUSIONS

The main application of the LBB approaches discussed was for primary pipe systems in new nuclear power plants, rather than dealing with existing piping with specific active degradation issues. Piping with active degradation mechanisms requires more than just leakage monitoring, and such procedures are really fitness-for-purpose analyses with methods to try to limit the degradation so LBB behavior would occur. LBB when applied to existing plants, particularly those with an active degradation mechanism that has significant cost impacts may be worthwhile to undertake probabilistically, but there must be much care in that development *for each degradation mechanism of interest.*

From the questionnaire on LBB sent out to 17 different countries, it was apparent that using probabilistic methods for LBB in the design of new plants was *not* a desired path. This is especially true for new plant types or first-of-a-kind designs. Probabilistic methods were felt to be useful for assessing risk if an actual degradation mechanism existed.

A few of the respondents from the different countries were interested in probabilistic analyses, but would still require deterministic analyses. A hybrid deterministic/probabilistic approach may be a more realistic compromise, where more elaborate analyses not possible in a probabilistic code could be conducted for key aspects of the assessment. One such hybrid approach for LBB was presented in this report, where the probabilistic nature of seismic loading was incorporated by conducting analyses at SSE loads (with comparable current safety factors) and then at 10^{-6} seismic event loads with reduced safety factors. Rather than assuming an idealized flaw type, the flaw size was determined from detailed crack growth analyses, such as the SCC analyses in used PWSCC cracking evaluations in the US, and was termed a "Robust LBB Approach". Of course, reasonable bounding material properties also need to be used, and some suggestions were given on improved selection of ferritic steels to eliminate detrimental effects of dynamic strain aging or accounting for thermal aging in all materials (not just cast stainless steels). This type of hybrid analysis is somewhat comparable to the approach used for "Seismic Considerations to the Transition Break Size" in NUREG-1903.

One of the main considerations for any new LBB procedure was to include additional considerations on protection against new degradation mechanisms that may develop. Mechanisms that allow long circumferential surface flaws to develop are the most threatening to leak-before-break behavior. Of these more threatening mechanisms, stress corrosion cracking is the most prevalent degradation mechanism in nuclear power plant piping, and unfortunately *SCC is not directly addressed by any nuclear pipe system design code*.

Since the life of nuclear plants is considered much greater than 40 years, analyses of possible degradation mechanisms in the LBB analyses seems prudent. Some international LBB analyses have involved fatigue analyses, however, the existing piping design rules are based on avoiding fatigue failures, and they are very effective (assuming all the loads are properly accounted for). Hence, conducting another fatigue analysis in the LBB evaluation seems like a uselessly redundant exercise.

It is difficult to know if the current SCC measures (i.e., substitute materials or water chemistry modifications) will be effective over the life of these plants that might cover the better part of a century. Consequently, one key suggestion was to include an incentive in the LBB procedure so that the plant fabricators prepare the welds in a manner that produces compressive longitudinal stresses on the internal surface (or ID) of girth welds through the used of "Fabrication Enhanced SCC Resistance Weld Procedures." Some weld sequencing procedures to produce "Fabrication Enhanced SCC Resistance Welds" were discussed, although more refinement is needed for actual application. These weld sequencing aspects in many cases could be adopted in existing weld procedures ***without*** much additional cost impact. If the plant uses "Fabrication Enhanced SCC Resistance Weld Procedures" during construction, then the deterministic and probabilistic approaches could be much simpler and easier to satisfy LBB considerations. If "Fabrication Enhanced SCC Resistance Weld Procedures" are not used, then the LBB application needs to consider all aspects of SCC in the deterministic or probabilistic LBB approach, which can be much more penalizing.

Finally, the two main recommendations from this project were;

3. Develop fabrication procedures that can be used to prevent high tensile stresses on the ID surfaces of primary loop piping, and
4. Conduct sensitivity studies on the hybrid deterministic-probabilistic "Robust LBB Procedure" for flaw shape development from SCC and seismic loading effects. Guidelines may evolve to better-improved deterministic as well as probabilistic analyses.

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APPENDIX A

Suggested Level 1 LBB Procedures from NUREG/CR-6765

The Level 1 LBB procedures will be the simplest of the three levels of Leak-Before-Break (LBB) procedures, requiring the least amount of information/data to apply. The safety factors associated with the Level 1 LBB procedure may be greater than the existing methodologies. The Level 1 approach was constructed such that piping systems that easily passed LBB using the draft SRP 3.6.3 procedure should be able to pass this Level 1 LBB procedure. If a piping system fails to pass the Level 1 LBB procedure, then the applicant can apply either a Level 2 or Level 3 LBB procedure in order to demonstrate LBB.

Whereas a Level 2 or Level 3 LBB procedure may require the use of a detailed leak-rate code for estimating the postulated leakage size crack and a detailed fracture mechanics code or finite element analyses for calculating the allowable moments or stresses, the Level 1 LBB procedure employs a series of simple algebraic equations to predict the:

- leakage area for a prescribed leak rate,
- crack-opening displacement,
- crack length, and
- allowable moment or stress.

The key elements of the Level 1 LBB procedure are described next.

A.1 Key Elements of Level 1 LBB Procedure

Upfront of all three LBB procedures will be a general screening criteria to eliminate those piping systems for which LBB is not applicable, i.e., piping systems susceptible to high undefined stresses (i.e., water hammer), or susceptible to cracking mechanisms causing long surface cracks (e.g., stress corrosion cracking). If a piping system passes this general screening criterion, then the user may elect to apply this Level 1 LBB procedure. The key elements of this Level 1 LBB procedure are:

- Data input requirements,
- Definition of critical locations for analysis,
- Prescribed safety factors,
- Simple algebraic equations for calculating the postulated leakage crack length,
- Level 1 screening criteria,
- Level 1 fracture analysis, and
- Level 1 LBB assessment.

Each of these elements is described in more detail in the subsequent sections.

A.1.1 Data Input Requirements

The data typically required to apply a Level 1 LBB procedure are shown in Table A.1. As a point of reference, Table A.1 also includes some of the typical data requirements for a Level 2 or Level 3 LBB procedure. Comparing the data requirements, the relative simplicity of the Level 1 approach is apparent when compared with either the Level 2 or Level 3 LBB procedure.

A.1.2 Definition of Critical Locations for Analysis

In applying a Level 1 LBB analysis to a subject piping system it will be necessary to make the necessary assessments at a number of critical locations along the piping system. At a minimum, each of the following locations should be considered in a Level 1 LBB analysis:

1. the location with the highest normal operating stresses (this is the location where a crack is more likely to occur),
2. the location with the highest safe shutdown earthquake (SSE), or transient, stresses,
3. the location with the highest ratio of the normal operating plus safe shutdown earthquake stresses (N+SSE) to the normal operating stresses (N), and
4. any other locations that have a material toughness with a J-R curve that is less than 75 percent of the J-R curve for the above material locations.

Normally, weld joint locations are selected as locations to be explicitly evaluated. Both the material properties of the weld material and the base material should be evaluated at these locations (particularly where cast stainless steel pieces are used). In addition, it is important to consider the case where the high stress occurs at a low toughness location.

A.1.3 Prescribed Safety Factors

With any of the three levels of LBB procedures there are certain values that must be prescribed by the NRC, most notably factors of safety on crack size and leak-rate detection capability. For the existing criterion in draft SRP 3.6.3, these prescribed factors of safety are typically 10 on leak rate and 2 on crack length (Ref. A.1). For this Level 1 LBB procedure, these safety factors may be increased by the NRC. For the Level 2 and Level 3 approaches, the safety factor on leak rate may be reduced due to more detailed analyses than conducted in the draft SRP 3.6.3. The safety factor on crack length for the fracture analysis may remain 2.

A.1.4 Postulated Leaking Crack Length Determination

The determination of the maximum postulated leaking crack length for the Level 1 LBB procedure is one of the major differences between the Level 1 LBB procedure and the Level 2 and Level 3 LBB procedures. (The other major differences are the fracture analysis used and potentially the factors of safety applied.) Instead of employing detailed computer codes for calculating crack-opening areas, crack-opening displacements, and postulated leakage crack lengths (as might be the case for a Level 2 or 3 analysis), the Level 1 LBB procedure employs a series of simple algebraic equations, that incorporate pre-established influence functions, to make these types of assessments. These influence functions have been established empirically through a series of sensitivity calculations in which each of the parameters that may have influenced the postulated leakage crack length were systematically varied while holding the other parameters constant.

Table A.1 Typical data requirements for a Level 1 analysis, with typical requirements for a Level 2 or Level 3 analysis shown for comparison

Level 1 requirements	Level 2 requirements	Level 3 requirements
Physical dimensions - Pipe diameter - Wall thickness	Same as Level 1	Same as Level 1
Thermohydraulic conditions - Temperature - Pressure	Same as Level 1	Same as Level 1
Material property data - Code or actual yield and ultimate strength values	Material property data - Code or actual yield and ultimate strength values - Stress-strain data - J-R curve data	Same as Level 2
Specialized computer codes required - None	Specialized computer codes required - Leak rate code, e.g. SQUIRT or PICEP - Fracture mechanics code, e.g., NRCPIPE or FEM analyses	Same as Level 2, except also need a finite element code for dynamic pipe system evaluations, e.g., ANSYS, ABAQUS, etc.
Stresses - Normal operating and transient stresses (i.e., SSE or transient thermal expansion stresses) from stress report	Same as Level 1	Stresses - Nonlinear finite element analysis
Elastic-plastic fracture analysis - Simplified procedures	Fracture analysis - J-estimation scheme - FEM analyses	Same as Level 2

In order to determine a postulated leakage crack length for the Level 1 analysis, one needs to calculate a leakage area (A) and a crack-opening displacement (COD). Then assuming an elliptical crack shape, one can calculate the total postulated leakage crack length (2c) using the expression:

$$2c = (4/B) \times (A/COD) \tag{A.1}$$

For the Level 1 analysis, the postulated leakage area (A) is calculated by dividing the piping system's leak-rate detection limit (LR), with an appropriate safety factor applied (LR w/SF), by the estimated flow rate per unit area (FR):

$$A = (LR\ w/SF)/FR \tag{A.2}$$

The flow rate per unit area (FR) is a function of the thermo-hydraulic conditions of the water, i.e., temperature (T) and pressure (P), the surface roughness of the crack (SR), and the wall thickness of the pipe (t). Mathematically it was found that for fatigue-type cracks, the flow rate per unit area could be expressed as a baseline value of FR ($FR_{baseline}$) times a series of influence functions that account for the effects of temperature, pressure, and wall thickness, see Equation A.3.

$$FR = (t_f)(T_f)(P_f)(FR_{baseline}) \tag{A.3}$$

The influence functions for wall thickness (t_f), temperature (T_f), and pressure (P_f) were empirically established through a series of sensitivity calculations using the SQUIRT leak-rate computer code (Version 2.4). The SQUIRT2 module was used to make these calculations. In Equation A.3, the baseline value of the flow rate per unit area ($FR_{baseline}$) is 950 lpm (250 gpm).

The pipe wall thickness influence function (t_f) was found to be:

$$\begin{gathered} t_f = 1.0 - (t-25.4) \times 0.0071 \quad \text{for } t > 25.4 \text{ mm} \\ t_f = 1.0 - (t-1.0) \times 0.18 \quad \text{for } t > 1.0 \text{ inch} \\ \text{or} \\ t_f = 1.0 - (t-25.4) \times 0.024 \quad \text{for } t < 25.4 \text{ mm} \\ t_f = 1.0 - (t-1.0) \times 0.6 \quad \text{for } t < 1.0 \text{ inch} \end{gathered} \tag{A.4}$$

The water temperature influence function (T_f) was found to be:

$$\begin{gathered} T_f = 1.0 - ((T - 288)/288) \times 2.37 \quad \text{for } T > 288 \text{ C} \\ T_f = 1.0 - ((T - 550)/550) \times 2.5 \quad \text{for } T > 550 \text{ F} \\ \text{or} \\ T_f = 1.0 - ((T - 288)/288) \times 0.95 \quad \text{for } T < 288 \text{ C} \\ T_f = 1.0 - (T - 550)/550 \quad \text{for } T < 550 \text{ F} \end{gathered} \tag{A.5}$$

The pipe system pressure influence function (P_f) was found to be:

$$\begin{gathered} P_f = 1.0 + ((P - 15.5)/15.5) \times 1.1 \quad \text{where pressure (P) is in terms of MPa} \\ P_f = 1.0 + ((P-2{,}250)/2{,}250) \times 1.1 \quad \text{where pressure (P) is in terms of psi.} \end{gathered} \tag{A.6}$$

Using the above influence functions, one can easily calculate the flow rate per unit area (FR). Knowing the flow rate per unit area (FR) and the leak-rate detection limit capability (with Safety Factor), i.e., LR w/SF, one can then calculate the leakage area (A) using Equation A.2. Then to calculate the postulated leakage crack length (2c) using Equation A.1, one only needs to be able to estimate the crack-opening displacement (COD).

For this Level 1 methodology, the crack-opening displacements are estimated using the Paris-Tada approach (Ref. A.2). The leak-rate code sensitivity study conducted as part of this program, found that the Paris-Tada method resulted in the most conservative predictions of COD, i.e., the Paris-Tada approach predicted relative smaller COD values for austenitic steels, which resulted in relatively large crack lengths for the same leak rate/crack opening area.

The crack-opening displacements (COD) based on the Paris-Tada approach can be estimated using Equation A.7:

$$COD = 2R_m^2 I_T(2_e)[\Phi_B(3 + \cos(2_e)/4 + \Phi_T]/cE$$

(A.7)

where,

R_m = mean pipe radius,
$I_T(2_e)$ = the tensile compliance function as defined in Reference A.2,
2_e = effective half crack angle accounting for the plastic-zone size,
c = half crack length,
E = elastic modulus,
Φ_B = nominal bending stress = $M/(BR_m^2t)$,
Φ_T = nominal tensile stress = $F_x/(2BR_mt)$, and
F_x = axial load on the pipe.

The effective half crack angle is:

$$2_e = c + [K_I/\Phi_y]^2/(\exists_1 BR_m) \qquad (A.8)$$

2 = half the total crack angle,
K_I = stress intensity factor,
Φ_y = yield strength, and
$\exists_1$ = plastic-zone size parameter.

The estimate of the plastic-zone size in Equation A.8 is only accurate for a small-plastic zone. In order to estimate J throughout the entire range between elastic and fully plastic conditions, Paris-Tada developed a method to interpolate between elastic and fully plastic conditions. This interpolation method amounted to modifying the $\exists_1$ term in Equation A.8. Therefore $\exists_1$ has to be determined in somewhat of a complicated fashion that depends on the current load as detailed in Reference A.2.

Comparisons of the Paris-Tada elastic-plastic COD values (using 2_e as defined in Equation A.8) were made with the linear elastic COD values (where $2_e = 2$) to see how much of an effect this plastic-zone size correction had on the COD values. In the range of load values typical of normal operating conditions for LBB, the difference was insignificant. Furthermore, even at the higher load levels (~75 percent of yield of the uncracked pipe), the differences were only on the order of 10 to 15 percent. In addition, the error was such that one would end up with a more conservative assessment of COD and crack length if the effect was ignored. As a result in order to simplify the Level 1 approach, the plastic-zone size correction was ignored, and

$$2_e = 2 \tag{A.9}$$

Consequently, using the empirically derived influence functions for flow rate per unit area (FR) and the Paris-Tada equations for crack-opening displacement, one can estimate the postulated leakage size crack (2c or 22) using Equations A.1 through A.9. This requires an iterative approach on crack length (2c) that is handled most efficiently using a spreadsheet.

Alternatively, one can make an estimate of the Level 1 leakage crack size using the expression

$$A = \alpha(\lambda)\left[\frac{\pi P_m (2c)^2}{2E}\right] \tag{A.10}$$

that is a shell-theory based equation used in the LBB procedures incorporated in the R6 document. It provides a conservative estimate of the crack opening area (A) as long as the through-wall bending stresses can be ignored.

In Equation A.10,

> 8 = a shell parameter = $[12(1 - \Lambda^2)]^{0.25}(c/(Rt)^{0.5})$,
> c = half crack length,
> R = shell radius,
> t = shell thickness,
> Λ = Poisson's ratio,
> P_m = membrane stress,
> E = elastic modulus, and
> ∀(8) = a correction factor to account for bulging which is a function of the shell parameter (8),

where,

> ∀(8) = $(1 + 0.1178^2)^{0.25}$ for circumferential cracks in cylinders.

One can rearrange Equation A.10 so that all of the terms which are a function of crack length (2c) are on one side of the equation and all of the known terms are on the other, such that

$$(2c)^2 \alpha(\lambda) = \frac{2EA}{\pi P_m} \tag{A.11}$$

The value of the crack opening area (A) is established using Equations A.2 through A.6. Then, Equation A.11 can be solved iteratively for the crack length (2c) using a simple spreadsheet.

The level of conservatism associated with this shell-based approach is about 30 percent greater than it is using the Level 1 influence expressions from Equations A.1 through A.9.

Consequently, the applicant has two options for calculating the leakage crack size to use in the LBB assessment. However, before proceeding to the fracture analysis/critical flaw size analysis, it is necessary to invoke the Level 1 LBB screening criteria to establish the appropriateness of employing a Level 1 LBB procedure.

A.1.5 Level 1 LBB Screening Criteria

Before proceeding further with the Level 1 LBB procedure , it is now time to check the values calculated up to this point to check the appropriateness of the assumptions invoked in a Level 1 LBB procedure. The five elements of the Level 1 screening criteria are:

- Check the ratio of the COD to the surface roughness. If this ratio is less than approximately 2.5 (Ref. A.3), then the validity of the analysis is questionable when using the standard crack morphology model from Reference A.4. For this standard crack morphology model, the surface roughness is approximately 40.5 µm (0.00159 inches) for corrosion fatigue cracks. The empirically derived influence functions discussed above were developed using the standard crack morphology model in SQUIRT. If the ratio of COD to the surface roughness if less than 2.5 (i.e., COD less than 0.10 mm (0.004 inches) for corrosion fatigue cracks), then one needs to go on to the Level 2 or Level 3 LBB procedure, and possibly invoke the COD-dependent crack morphology model from Reference A.4.
- Check the thermo-hydraulic conditions of the water. The influence functions used to estimate the leakage area, which in turn are used to estimate the leakage size flaw, are based on SQUIRT calculations that are only valid for two-phase flow from subcooled water. If the temperature and pressure are such that subcooled water conditions do not exist, then a more rigorous leak-rate analysis, using a code such as PICEP, will be required. This will involve a Level 2 LBB analysis.
- Check the ratio of the postulated crack length to the pipe circumference. If this ratio is greater than one-eighth of the pipe circumference, then there is the possibility that there may be restraint of the COD from the pipe system boundary conditions that need to be considered. (The definition of this predetermined value will be established as part of the Battelle Integrity of Nuclear Piping (BINP) program.) If this ratio is greater than one-eighth of the pipe circumference, then one needs to go on to a Level 2 analysis, which will account for these effects.
- Ascertain whether or not the piping system welds have been stress relieved or not. If not, then one needs to make an assessment as to whether or not weld residual stresses will impact the crack-opening displacements. For “thick-wall” piping the effects of weld residual stresses on the crack-opening displacements are probably minor. For “thin-wall” piping, the effects of weld residual stresses could be significant, and one will need to go on to a Level 2 analysis. The determination as to what is a “thick-wall” piping system and what is a “thin-wall” piping system still needs to be addressed as part of the BINP program.

A.1.6 Level 1 Fracture Analysis

The Level 1 fracture analysis is a simple limit-load analysis for which the allowable bending stress (S) is a function of flow stress (Φ_f), and postulated crack length (22), see Equation A.12. For the Level 1 LBB analysis, a factor of safety of at least 2 is applied to the postulated crack length. For convenience, this postulated total crack length with safety factor will be referred herein to as 22_1.

$$S = 2\Phi_f \, [2\sin(\exists) - \sin(2_1)]/B \qquad \text{(A.12)}$$

where,

$$\exists = [(B - 2_1) - BP_m/\Phi_f]/2 \qquad \text{(A.13)}$$

where, the flow stress (Φ_f) can be defined either in terms of Code properties (S_y and S_u) or actual material data (Φ_y and Φ_u), if available. The flow stress can be defined as either:

$$\Phi_f = (S_y + S_u)/2$$

or

$$\Phi_f = (\Phi_y + \Phi_u)/2 \qquad \text{(A.14)}$$

depending on whether actual material data are available. Typically for the leak-rate analysis used to estimate the postulated crack length, average data are used. Conversely, for the stability analysis, minimum values are typically used.

The allowable stress index ($SI_{allowable}$) can be found by adding the combined membrane stress (P_m) due to internal pipe pressure, deadweight, and seismic to the allowable bending stress (S).

$$SI_{allowable} = S + M\,P_m \qquad \text{(A.15)}$$

where,

M = margin associated with the load combination method selected for analysis (i.e., for absolute [M = 1.0] or for algebraic [M = 1.4]).

This allowable stress index is then compared with the applied stress index ($SI_{applied}$) for the normal operating plus safe shutdown earthquake stresses (N+SSE) from the stress report to determine whether the piping system passes the Level 1 type analysis. If the applied stress index at the faulted conditions is greater than the allowable stress index, then the piping system fails to satisfy the Level 1 criteria and one would need to move on to a Level 2 or Level 3 analysis.

$$SI_{applied} = M(P_m + P_b + P_e)Z \qquad \text{(A.16)}$$

where,

P_b = the combined primary bending stresses, including deadweight and seismic components,

P_e = the combined expansion stresses at normal operating conditions and seismic anchor motion.

For lower toughness materials, e.g., ferritic steels and lower toughness austenitic flux welds, one will need to apply a knockdown factor to the calculated allowable stress value. It is envisioned that this knockdown factor may resemble the Z-factors incorporated in the ASME Section XI pipe flaw evaluation criteria.

A.1.7 Level 1 LBB Acceptability Assessment

A piping system would pass the Level 1 LBB criteria if the applied stress index ($SI_{applied}$) at the faulted conditions is less than the allowable stress index ($SI_{allowable}$) for a flaw twice as long as the postulated leakage crack size at normal operating conditions. If the applied stress index is greater than the allowable, then one needs to go on to a Level 2 or Level 3 analysis in order to demonstrate LBB. This acceptance criterion maintains the factor of safety of 2 on crack size currently stipulated in the draft SRP 3.6.3 procedures. If the NRC chooses to invoke some other safety factor on crack size, then one need only to multiply the postulated leakage crack size by that factor (instead of 2) when calculating the allowable stress index.

A.2 References

A.1 Solicitations for public comment on "Standard Review Plan 3.6.3 LEAK-BEFORE-BREAK EVALUATION PROCEDURES," *Federal Register*, Vol. 52, No. 167, Friday, August 28, 1987, Notices, pp 32626 to 32633.

A.2 Paris, P. and Tada, H., "The Application of Fracture Proof Design Methods Using Tearing Instability Theory to Nuclear Piping Postulating Circumferential Through Wall Cracks," NUREG/CR-3464, September 1983.

A.3 Ghadiali, N., and others, "Deterministic and Probabilistic Evaluations for Uncertainty in Pipe Fracture Parameters in Leak-Before-Break and In-Service Flaw Evaluations," NUREG/CR-6443, pp 3-8 to 3-13, June 1996.

A.4 Rahman, S., and others, "Probabilistic Pipe Fracture Evaluations for Leak-Rate-Detection Applications," NUREG/CR-6004, pp 3-11 to 3-12, April 1995.

APPENDIX B

Suggested Level 2 LBB Procedures from NUREG/CR-6765

The Level 2 LBB procedures involve a more detailed analysis than the Level 1 LBB procedures in this document or the draft SRP 3.6.3 Leak-Before-Break (LBB) procedures. The factors of safety associated with the Level 2 LBB procedures may be less than the Level 1 LBB procedures. The Level 2 LBB procedures were developed to incorporate improvements to the draft SRP 3.6.3 LBB procedures (Ref. B.1), using the technologies from the various NRC and international programs developed since the introduction of draft SRP 3.6.3 (Ref. B.2). If a piping system fails to pass the Level 2 LBB procedures, then the applicant can apply the Level 3 LBB procedures in order to demonstrate LBB or choose to evoke certain options within the Level 2 LBB procedures.

The key elements of the Level 2 procedures are described next.

B.1 Key Elements of Level 2 Procedure

An initial requirement for all three LBB procedures is a general screening criterion to determine if LBB can be applied to the piping system. Piping systems not eligible for LBB are those where there may be large unknown stresses (e.g., water hammer) or where long surface flaws could occur (e.g., stress-corrosion cracking). The fatigue usage factor shall be below 1.0 for the life of the plant using design stresses and any service encountered stress cycles that have become known prior to the LBB application. Since the forces from a steam-hammer event can be calculated, lines susceptible to steam hammer can be considered for LBB. If a piping system passes this general screening criterion, then the user may elect to apply these Level 2 LBB procedures. The key elements of the Level 2 LBB procedures are:

- Data input requirements,
- Determination of critical locations for assessment,
- Applied safety factors,
- Procedures to calculate the postulated crack length for the acceptable leak rate, (some screening criteria are provided to circumvent unnecessary steps),
- Level 2 fracture analysis, and
- Level 2 LBB assessment.

Each of these elements is described in more detail in the subsequent sections.

B.1.1 Data Input Requirements

The data typically required for a Level 2 LBB assessment are shown in Table B.1. All of the input data are also needed for a Level 3 LBB assessment.

Table B.1 Typical data requirements for a Level 2 LBB assessment, with typical requirements for Level 1 and Level 3 LBB procedures shown for comparison

Level 1 requirements	**Level 2 requirements**	**Level 3 requirements**
Physical dimensions - Pipe diameter - Wall thickness	Same as Level 1	Same as Level 2
Thermohydraulic conditions - Temperature - Pressure	Same as Level 1	Same as Level 2
Material property data - Code or actual yield and ultimate strength values	Material property data - Code or actual yield and ultimate strength values - Stress-strain data - J-R curve data	Same as Level 2
Specialized computer codes required - None	Specialized computer codes required - Leak-rate code, e.g., SQUIRT, PICEP, - Fracture mechanics code, e.g., NRCPIPE or FEM analyses	Same as Level 2, except also need a finite element code for dynamic pipe system evaluations, e.g., ANSYS, ABAQUS, etc.
Stresses - Normal operating and transient stresses (i.e., SSE or transient thermal expansion) from the stress report	Same as Level 1	Stresses - Nonlinear finite element analysis
Elastic-plastic fracture analysis - Simplified procedures	Elastic-plastic fracture analysis - J-estimation schemes, or - FEM analyses	- Same as Level 2

B.1.2 Critical Location Determination

In applying a Level 2 LBB procedure to a subject piping system it is necessary to make assessments at a number of critical locations along the piping system. At a minimum, each of the following locations shall be considered in a Level 2 LBB procedure:

1. The location with the highest normal operating stresses (this is the location where a crack is more likely to start),
2. The location with the highest transient stresses, i.e., safe shutdown earthquake (SSE) or transient thermal expansion stresses at start-up or shut-down,
3. The next three highest stress locations with ratios of the normal operating plus transient stresses to the normal operating stresses (N) being greater than 80 percent of the location with the highest ratio, and
4. Any other location that has a material toughness with a J-R curve that is less than 75 percent of the J-R curve for the above material locations.

Postulated cracks can be in the straight pipes, girth welds, and fittings. For fittings, the most common type of fitting to develop cracks is an elbow. Elbow cracks can be either circumferential cracks on the extrados (closing moment applied), or axial cracks on the flank of the elbow.

B.1.3 Physical Dimensions

The entire pipe system, from anchor to anchor, needs to be included in the LBB evaluation. A detailed sketch with the pipe-system geometry (including pipe hanger locations, snubber locations, etc.) shall be included in the submittal. The pipe nominal diameters, thicknesses, and materials throughout the pipe system shall be identified. Actual thickness values can be used. If a weld location is considered as a critical location for an LBB application, then the thickness used in the evaluation shall be the minimum design or minimum actual thickness, i.e., the actual thickness at a counterbore without the weld crown.

B.1.4 Prescribed Safety Factors

With any of the three levels of LBB analyses, safety factors need to be prescribed by the NRC, typically on crack size and leak-rate detection capability. For the existing draft SRP 3.6.3 criteria, these prescribed safety factors are typically 10 on leak rate and 2 on crack length (Ref. B.1). For the Level 2 LBB procedures, the safety factor on leak-rate could be decreased since the leak-rate analyses are more detailed than in draft SRP 3.6.3. The safety factor for the fracture analyses will be on crack length only, not on stress level.

B.1.5 Postulated Leaking Crack Length Determination

The determination of the postulated maximum leaking crack length for the Level 2 LBB procedure is one of the major differences between the Level 2 LBB procedure and the existing draft SRP 3.6.3. In order to determine a postulated crack length for the Level 2 LBB procedure, one needs to first know the leak-rate detection capability with some safety factor. For instance, a leak detection capability of a PWR system is typically 1 gpm, and the safety factor of 10 has been typically applied. This would give a target 10-gpm leak rate for crack-size determination. The crack-opening displacement is then calculated for an initial crack length at the normal operating stresses, and then the leak-rate is determined. An iterative procedure is used until the crack length corresponding to the target leak rate is determined.

This is the basic step in this part of the Level 2 LBB procedures that is consistent with the draft SRP 3.6.3 approach. The additional requirements are:

1. The acceptable COD-analyses procedures are specified,
2. The effects of restraint on the COD from the pipe-system boundary conditions need to be included if simplified COD methods from Step 1 are used,
3. Crack-face pressure effects on COD can be included if desired by the applicant,
4. The COD-dependent crack-morphology parameters (surface roughness and number of turns) to be used in the leak-rate analyses are specified,
5. The effects of residual stresses need to be considered for certain cases, and
6. The acceptable leak-rate analyses and computer codes are given.

Because of these additional detailed COD and leak-rate evaluation steps, it is suggested that the safety factor on leak rate for the Level 2 LBB procedure be reduced from the value of 10 to perhaps a value of 5. This same analysis procedure would be used in the Level 3 LBB procedure; hence the new leak-rate safety factor should also apply for that LBB procedure.

B.1.5.1 Acceptable COD Analyses: The acceptable COD analyses are either the Tada-Paris analysis (Ref. B.3), the original GE/EPRI method (Ref. B.4), or by finite element analyses (Ref. B.5 gives results from numerous FEM COD analyses). The original GE/EPRI solutions for combined bending-and-pressure loading in both the PICEP (Ref. B.6), SQUIRT (Ref. B.7) and NRCPIPE (Ref. B.8) codes have been found to give comparable results to finite element analyses (Ref. B.5). The Tada-Paris method in the NRCPIPE code has also been benchmarked against finite element results in Reference B.5. Other COD estimation schemes can be used if appropriately benchmarked and documented in the submittal. These analyses consider that the pipe is a simple endcapped vessel, and hence do not account for pipe-system boundary conditions on restraining the induced bending from the axial stresses. For these analyses, a correction factor from Section B.1.5.2 of this appendix is needed.

Finite element solutions can involve relatively simple straight-pipe models that use end capped pipe boundary conditions as in Reference B.5, or could attempt to model the whole pipe system with the crack and the boundary conditions that might restrain the induced bending from axial tension loads. If a simple straight-pipe FE model is used, then the correction factor for pipe-system boundary conditions needs to be used.

B.1.5.2 Reduction of Axial Tension COD Due to Pipe-System Restraint: The COD estimation scheme analyses for combined loading typically consider that the pipe is free to rotate from the axial stresses applied. FEM analyses may also model only a straight section of pipe rather than the whole pipe system with the actual system boundary conditions for COD analyses. In a real pipe system, pipe anchors (such as vessel nozzles or nozzles to much larger pipes) will restrain the rotation that comes from the eccentricity of the crack section under axial tension loading. The following procedure shall be used to determine the reduction of the axial tension COD component. The axial tension stresses could be from pressure or other loads. This correction is only for the COD due to axial tension stress. This analysis step can be skipped if the following normal operating conditions can be met:

- If the axial tension stress is less than some percent[d] of the total stress, then there are negligible effects from the pipe-system boundary conditions,
- If the crack length is less than 1/8 of the pipe circumference, then this effect is negligible, or
- If the crack plane is more than 20 pipe diameters from an anchor or elbow in either direction, then these effects can be ignored.

If these conditions cannot be met and the entire pipe system with the crack was not considered in the FEM model for COD analyses, then the following steps shall be used.

1. Start with an estimated initial crack length and calculate the COD for combined bending and axial tension (pressure) forces using an estimation scheme like the GE/EPRI estimation scheme in PICEP, SQUIRT, or NRCPIPE[e],

[d] To be determined from future BINP program efforts.

2. Calculate the bending only COD for the same crack length using the same COD estimation scheme,
3. Subtract the bending only COD (in Step 2) from the total COD (in Step 1). This gives the unrestrained axial tension COD component,
4. Using the following equation[f], calculate the restrained axial tension component of the COD ($COD_{restrained}$);

$$COD_{restrained} = COD_{unrestrained} * fcn(R_m/t,\ \theta/\pi,\ L_1/D,\ L_2/D) \quad \text{(B.1)}$$

where,

R_m = pipe mean radius,
t = pipe thickness,
θ = half crack angle, radians,
L_1 = distance from crack plane to closest nozzle, pipe elbow, or pipe hanger on one side of the crack plane,
L_2 = distance from crack plane to farthest nozzle, pipe elbow, or pipe hanger on the other side of the crack plane, and
D = mean pipe diameter.

5. Add the $COD_{restrained}$ axial tension component to the bending only COD component from Step 2,
6. Calculate the leak rate using PICEP or SQUIRT with the crack morphology parameters given in Section B.1.5.4, and
7. Iterate on the crack length until the target leak rate is determined.

B.1.5.3 Effects of Crack-Face Pressure on COD: The effect of the pressure on the crack faces is to open up the crack further than if it was ignored. This effect will make it easier to meet LBB conditions, hence the applicant can ignore it and still be acceptable from a regulatory sense. This effect is probably only significant if the crack length is longer than a prescribed percent of the pipe circumference[g]. This effect may compensate for some of the restraint of pressure-induced bending effects required in Section B.1.5.2.

The following steps are acceptable for this analysis:

1. From the leak-rate calculations in Section B.1.5.2, determine the exit plane fluid pressure (Pressure at throat in PICEP or exit plane pressure in SQUIRT).
2. Assume the pressure distribution is linear through the thickness from the inside pressure to the exit plane (outside diameter).
3. Calculate the applied bending moment and axial tension forces on the pipe by integrating the pressure along the crack faces.
4. Add those moments and axial tension forces to the applied normal operating loads in Step 1 of Section B.1.5.2. Calculate the new leak rate. Check the pressure distribution through the thickness from the leak-rate code and iterate until there is convergence for that crack length.

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[e] Use the original GE/EPRI estimation scheme without plastic-zone correction in the elastic term in the NRCPIPE code. Do not use the Battelle-modified GE/EPRI estimation scheme method in the NRCPIPE code.

[f] Exact form of this equation is to be determined from future BINP work.

[g] Prescribed value to be determined from additional proposed work in this program.

5. Change the crack length and iterate through Step 1 of this section until the target leak rate is determined.

B.1.5.4 Crack Morphology Parameters: To maintain consistency with different LBB applications, specified crack morphology parameters shall be used. These parameters are the surface roughness and number of turns. As a crack opens up, then the number of turns decreases, and the surface roughness decreases. Hence, these parameters depend on the COD value. By having a COD-dependent roughness and number of turns, problems with the friction factor relationships in these leak-rate codes for tight cracks can be circumvented. The roughness and number of turns was chosen from the statistical evaluation of corrosion-fatigue cracks and thermal fatigue cracks found in service. The mean values are to be used, see Table B.2.

Table B.2 Mean and standard deviation of crack morphology parameters

Crack morphology variable	**Corrosion fatigue or thermal fatigue cracks**	
	mean	**standard deviation**
μ_L, μm (:inch)	8.814 (347)	2.972 (117)
μ_G, μm (:inch)	40.513 (1,595)	17.653 (695)
n_{tL}, mm^{-1} ($inch^{-1}$)	6.73 (171)	8.07 (205)

In Reference B.9, the following equations were established using engineering judgment. For the surface roughness (μ), the following equation should be used as a function of the center crack-opening displacement (δ).

$$\mu = \begin{cases} \mu_L, & 0.0 < \dfrac{\delta}{\mu_G} < 0.1 \\ \mu_L + \dfrac{\mu_G - \mu_L}{9.9}\left[\dfrac{\delta}{\mu_G} - 0.1\right], & 0.1 < \dfrac{\delta}{\mu_G} < 10 \\ \mu_G, & \dfrac{\delta}{\mu_G} > 10 \end{cases} \quad \text{(B.2)}$$

For the number of turns (n_t), the following equation should be used as a function of the center crack-opening displacement (δ).

$$n_t = \begin{cases} n_{tL}, & 0.0 < \dfrac{\delta}{\mu_G} < 0.1 \\ n_{tL} - \dfrac{n_{tL}}{11}\left[\dfrac{\delta}{\mu_G} - 0.1\right], & 0.1 < \dfrac{\delta}{\mu_G} < 10 \\ 0.1 n_{tL}, & \dfrac{\delta}{\mu_G} > 10 \end{cases} \quad \text{(B.3)}$$

B.1.5.5 Effect of Residual Stresses on Leak Rate: Weld residual stresses have been investigated and determined that they could possibly affect the leak rate under certain conditions. These conditions will be explored further in a future BINP program effort. What is now known about weld residual effects on crack opening and leak rates is summarized below.

1. Weld residual stresses can be either tension-compression through the thickness for a "thin-walled" weld or tension-compression-tension for a "thick-walled" weld, respectively.
2. The effect of weld residual stresses on the COD is to rotate the crack faces. Hence "thin-walled" welds (with tension-compression stresses through the thickness) will rotate the crack faces more than "thick-walled" welds.
3. The effect of crack-face rotation about a mean COD value has negligible effect on the leak rate. Past experimental results have documented this at Central Electric Generating Board (now Nuclear Electric), and could also be shown from calculations using the SQUIRT leak-rate code.
4. Weld residual stresses will only be significant for leakage detection purposes if the crack faces rotate enough to pinch off the flow.
5. If the applied normal operating loads give a COD that is much larger than the change in the COD due to the rotation of the crack faces from the residual stresses, then the weld residual stress effect can be ignored. The effect of the elliptical crack-opening shape should be considered in this evaluation.
6. Because of low crack-face rotations, the effect of residual stresses can be considered negligible for a "thick-walled" weld. The definition of a "thin-walled" versus "thick-walled" weld needs to be established.
7. Stress relieved welds can be considered exempt from weld residual stress effects on the COD.

Additional efforts need to be conducted to give more explicit guidance on how to handle cases when weld residual stresses should be considered. The developed relationship should take the form of

$COD_{residual}/COD_{base}$ = fcn[weld layer/thickness, weld bevel geometry, θ/π, $(P_m+P_b)/\sigma_y$, and weakly with R_m/t]

B.1.5.6 Acceptable Leak-Rate Codes: Computer codes that are acceptable for leak-rate analyses are PICEP and SQUIRT. Other codes that have been benchmarked against similar leak-rate data sets can be used if documentation is provided.

In these codes, an elliptical crack-opening shape shall be used.

The SQUIRT code should only be used for two-phase flow conditions. Only the original GE/EPRI COD analyses should be used in SQUIRT for COD analyses. Alternatively, the Tada-Paris COD analysis procedure or FEM COD values could be determined, and then used with the thermohydraulic options in SQUIRT (SQUIRT2 module) or PICEP (pick leakage only option). In the PICEP code, the GE/EPRI solution is the only option to use.

For single-phase flow through the cracks (either all-water or 100-percent quality steam lines), benchmarking of leak rates in this flow regime is desired for whatever computer code is used.

The surface roughness and number of turns used shall be those in Section B.1.5.4 in this report.

B.1.6 Level 2 Fracture Analysis

The Level 2 fracture analysis involves an elastic-plastic fracture analysis. Cracks could be either in straight pipes or in fittings. Based on service history, circumferential cracks are more likely to occur in straight pipes and in particular at girth welds near terminal ends or near fittings.

Circumferential through-wall cracks in straight pipes and at girth welds to fittings can be analyzed using the same analyses. Based on comparisons with full-scale pipe test data in Reference B.10 and B.11, the acceptable analyses for combined pressure and bending of a circumferential through-wall crack in a straight pipe are:

- ASME Section XI Z-factor equations (Refs. B.12 and B.13),
- Original GE/EPRI analysis (Ref. B.4 and in the PICEP, SQUIRT and NRCPIPE Codes),
- LBB.ENG2 analysis (Ref. B.10 and in the NRCPIPE Code),
- LBB.NRC analysis (Ref. B.14 and in the NRCPIPE Code), and
- Dimensionless Plastic-Zone Parameter (DPZP) analysis (Refs. B.2 and B.15).

For axial cracks in straight pipes, the analysis in Reference B.15 could be used.

The most common type of fitting where cracks have occurred is in elbows. Work is currently ongoing in the BINP program to assess methods to evaluate axial and circumferential through-wall flaws in elbows. Alternatively, one could use a finite element analysis for cracks in elbows or other fittings.

There are many common input parameters for these analyses. The following input parameters can be used.

B.1.6.1 Yield, Ultimate, Flow Stress, and Stress-Strain Curves: These properties should be determined for the operating condition of interest (temperatures may be different for normal operating versus transient loading conditions), and can be for quasi-static loading rates.

The yield and ultimate strength can be either the ASME Section II Code values (S_y and S_u) at the service temperature of interest, actual values at that service temperature (Φ_y and Φ_u), or reasonable bounding values[h] from a database at the service temperature of interest.

The flow stress (Φ_f) shall be defined by

$$\Phi_f = (S_y + S_u)/2$$

or (B.4)

$$\Phi_f = (\Phi_y + \Phi_u)/2$$

For weld metals, only the weld metal or HAZ toughness is needed. The weld metal strength is not needed. Some analyses[i] allow the weld metal strength to be incorporated in them, but these analyses are not required.

Typically, for the crack-opening analysis used to estimate the postulated crack length at normal operating conditions, the average strength data are used. Conversely, for the stability analysis, minimum values or reasonable lower-bound values[5] are typically used.

The stress-strain curve in these fracture analyses are typically represented by a Ramberg-Osgood curve, see Equation B.5.

$$\varepsilon/\varepsilon_o = (\sigma/\sigma_o) + \alpha(\sigma/\sigma_o)^n \tag{B.5}$$

In this equation, it is required that

$$\sigma_o/\varepsilon_o = E \tag{B.6}$$

where,

- E = elastic modulus from Section II of ASME Code
- ε_o = reference strain
- σ = any stress value
- σ_o = reference stress
- α = parameter from curve fitting of data
- n = strain-hardening exponent

σ_o is typically the yield strength, but could be any other value as long as Equation B.6 is satisfied and α and n are determined with this value. If a plastic-zone correction is used in the GE/EPRI analysis, then σ_o should be taken as the yield strength.

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[h] Mean minus one standard deviation value is considered a reasonable lower bound value.

[i] FEM analyses including the weld geometry, or the LBB.ENG3 J-estimation scheme (Reference B.11) using base and weld metal stress-strain curves.

The Ramberg-Osgood curve fit shall be obtained using the engineering stress-strain curve and fitting the data from 0.1-percent strain to the strain corresponding to 80-percent of the ultimate strength, Ref. B.2.

B.1.6.2 Fracture Toughness: Specimen orientation - The fracture toughness can be from actual test data, or representative lower-bound data[5]. For a circumferential through-wall flaw, the data should be from specimens machined in the C-L orientation[j]. For axial flaw evaluations, the data should be for specimens machined in the L-C orientation.

For crack locations at welds, the postulated crack location is in the center of the weld metal as well as in the HAZ and fusion lines. The HAZ/fusion line crack should be put in fracture specimens (i.e., bend-bar or C(T) specimens) fabricated so that the crack and HAZ/fusion line is normal to the specimen surface. Typically more specimens are needed for HAZ/fusion line testing than for base metal or weld centerline testing (Ref. B.17). It is suggested that five specimens be tested for HAZ/fusion line testing, and the lowest J-R curve from those five specimens should be used.

Loading rate - Data for austenitic base metals and weld metals can be at quasi-static loading rates.

If seismic loading or other dynamic loading is part of the transient loading condition for the fracture evaluation, then due to dynamic strain aging effects, the fracture toughness data for ferritic steels at temperatures greater then 149 C (300 F) should be tested at a dynamic loading rate comparable to the transient loading rate (Refs. B.2 and B.18). Steels with ultimate strengths at temperature that are greater than the ultimate strength at room temperature are susceptible to dynamic strain aging and should be tested at higher loading rates.

For a dynamic event, the loading rate should correspond to the time to get to crack initiation in one-quarter of the period of the first natural frequency of the piping system (Ref. B.2). The experimental time to crack initiation can be a factor of ±25% of the time corresponding to one-quarter of the period of the first natural frequency of the piping system.

Cyclic loading effects on toughness – Cyclic loading effects can be detrimental to the toughness of the material. Some results are still under development in the BINP program to ascertain if they should be considered as significant enough to be included.

Bimetallic welds - For bimetallic welds involving a stainless steel weld to a carbon steel pipe, the J-R curve of the HAZ/fusion line of the stainless steel weldment to the carbon steel (or low alloy steel) material should be considered in determining where the lowest toughness region is. For Inconel welds or welds using Inconel buttering on the low alloy or carbon steel materials, then the toughness of the Inconel weld or the carbon steel/low alloy steel can be used for the toughness of the bimetallic weld (Ref. B.19).

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[j] See Reference B.16 for specimen orientation definition.

Thermal aging - Thermal aging needs to be accounted for in cast stainless steel base metals. Trend curves with ferritic number or chemistry can be used to project the end-of-life toughness properties.

Thermal aging can also affect stainless steel welds. In cast stainless piping, the aged base metal properties may govern the toughness considerations over the weld metal. However, the thermal aging effects on the weld metal should also be considered for wrought stainless steel piping systems.

Extrapolation of J-R curves - Data for crack growth of up to 30-percent of the initial ligament of the fracture specimen can be used to establish the J-R curve. A significant amount of research results have shown that it is conservative to make a power-law extrapolation of the deformation theory J-R curve (Ref. B.20).

It has also been shown that the Modified J-R curve (J_M) gives good predictions for large crack growth in estimation schemes such as those mentioned at the beginning of Section B.1.6 (Ref. B.2). The J_M-R curve can only be used in cases where the slope of the J-R curve is linear, i.e., J_M-R curve should not be used if they exhibit an upward hooking behavior (power-law coefficient greater than 1.0).

B.1.6.3 Stress Definitions: For fracture analyses, the applied stresses from the plant stress report can be used to calculate a crack size that corresponds to that load-controlled instability. That is, the crack length can be increased so that maximum load is achieved at the transient loads (typically the N+SSE load). The stress components to be used in this evaluation are as follows:

1. All global secondary stresses and primary stresses shall be combined as an algebraic sum. A global secondary stress includes thermal expansion stresses and seismic anchor motion stresses. Primary stresses are dead-weight, pressure, and inertial stresses.
2. Weld residual stresses and through-thickness thermal stresses can be ignored if ductile fracture behavior is demonstrated in the J-R curve tests for the material at the temperatures of interest.
3. An equivalent bending moment (M_{eq}) shall be determined from a combination of the moments and torsion in the different directions using a Von Mises combination of these loads (Ref. B.21), i.e.,

$$M_{eq} = \{M_b^2 + [(3^{0.5}/2)*T^2]\}^{0.5} \tag{B.7}$$

Where

$M_b = (M_x^2 + M_y^2)^{0.5}$
M_x = Bending moment in one plane
M_y = Bending moment in the other plane
T = Torsion in x-y plane

B.1.6.4 Fracture Calculations: The critical crack lengths shall be calculated for the different postulated LBB locations. The critical crack length is the crack length at the maximum load (a load-controlled instability analysis). It is possible that some systems may not result in a double-ended guillotine break for applied displacements (from secondary stresses) that could go beyond the maximum load, but post maximum-load stability will be kept as an additional reserve margin.

The critical crack lengths shall be calculated for the service transient load (i.e., N+SSE) using the guidance in Section B.1.6 of this appendix.

B.1.7 Level 2 LBB Acceptance Criterion

A piping system would pass the Level 2 LBB acceptance criterion if the calculated critical crack length from Section B.1.6.4 is equal to or greater than twice the leakage crack length from Section B.1.5, i.e., there is a minimum safety factor of 2 on the leakage crack size. If it does not pass, then several of the options in Level 2 LBB procedure can be invoked, or a Level 3 LBB procedure can be employed.

B.2 References

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APPENDIX C

Suggested Level 3 LBB Procedures from NUREG/CR-6765

The Level 3 LBB procedure is the last option available to an applicant for demonstrating LBB in a piping system, and should only be considered as a last resort. Building upon the foundation of the Level 2 analysis, the Level 3 analysis looks for margin in the nonlinearity of the crack, the piping system, or both. Because such nonlinearities consume energy, this energy is not available for driving the crack. Thus, there may not be a large enough crack driving force to reach the critical crack load and hence, LBB is satisfied.

The key elements of the Level 3 procedure are described next.

C.1 Key Elements of Level 3 Procedures

Level 3 builds directly upon the Level 2. Thus, Level 3 has all of the same requirements for data inputs, applied safety factors and procedures to calculate the postulated crack length as listed in Appendix B. All of the screening criteria and exclusions of Level 2 apply. Where Level 3 differs from Level 2 is that a nonlinear stress analysis is performed in place of a pseudo-static, response spectrum, or dynamic linear analysis.

C.2 Nonlinear Stress Analysis Data Input Requirements

The data typically required for a Level 3 LBB assessment are as follows:

1. A piping system that qualifies for a Level 2 analysis but that does not meet the Level 2 LBB fracture margin requirement,
2. A finite element model of the piping run from anchor to anchor containing the hypothesized flaw,
3. A complete characterization of the loading in the time domain,
4. A load-displacement description of the crack behavior,
5. An assumed flaw orientation,
6. The stress-stain behavior of the pipe at the operating temperature, and
7. A nonlinear finite element analysis program.

C.2.1 Qualified Piping System

In applying a Level 3 LBB procedure to a piping system, all of the basic requirements for a Level 2 analysis must be met, except for demonstration of an adequate fracture margin. If any piping system is disqualified from consideration for LBB in Level 2 due to a violation of one of the Level 2 screening criteria stipulations, it is automatically disqualified from consideration for LBB in Level 3.

C.2.2 Finite Element Model

The entire pipe system, from anchor to anchor, needs to be included in the Level 3 model. A detailed sketch with the pipe-system geometry (including pipe hanger locations, snubber locations, etc.) shall be included in the submittal. The pipe nominal diameters, thicknesses, and materials throughout the pipe system shall be identified. Actual thickness values can be used. The characteristics of all supports (stiffness and damping properties) must be known.

C.2.3 Loading

All loads on the pipe system during the SSE event (pressure, dead weight, thermal expansion, cold springing, seismic anchor motion, inertial loading, etc.) must be known as a function of time. It is anticipated that the dead weight, pressure, and thermal expansion loads will be constant with time. The SSE loading, both the seismic anchor motion and inertial loading, will be time varying and must be known in three orthogonal directions. As appropriate, loads such as thermal stratification must be considered in combination with the SSE loading.

The three orthogonal directions of SSE time history loading (seismic anchor motion and inertial loading) must be known at a sufficiently small time increment that the nonlinear analysis will converge. In the event that the analysis fails to converge because the time step is too coarse, a finer time step must be used.

C.2.4 Postulated Crack Description

The hypothesized crack must be characterized in terms of a load-displacement behavior as part of the nonlinear analysis. For a circumferential crack, the crack behavior is generally given in moment-rotation coordinates. For axial cracks, a COD versus hoop load would be appropriate. The crack characterization must include the effects of all applicable loading (bending, pressure, tension) and unloading behavior and crack closure must be included.

In general, the required load-displacement characterization of the crack will come from the Level 2 leakage size crack calculations. A factor of safety of 2.0 must be applied to this Level 2 leakage crack size. J-estimation scheme or finite element analyses of some sort will then be used to define the crack behavior. The effect of yielding of the crack on unloading can be modeled. Crack closure must be included if the possibility of the crack faces touching exists. Because the LBB assessment is only concerned with whether or not the applied load is sufficient to reach the maximum moment of the crack, the crack load-displacement characterization is only needed up to the predicted maximum moment.

C.2.5 Crack Orientation

An orientation for the crack must be chosen for the Level 3 analysis. Unlike a Level 1 or Level 2 analysis, where there is a known applied load from the stress report that is given independent of direction, the Level 3 crack is fixed in a given orientation in the finite element model and will respond only to loads that will open/close the crack. Thus, if a Level 3 crack is oriented vertically and all of the loads are applied horizontally, LBB will be satisfied because the crack will not experience any crack-opening load.

It is important to correctly orient the crack so that a true LBB assessment is made. A conservative Level 3 LBB analysis would consider the largest possible leakage size flaw based on the normal operating loads, but oriented in the direction of the largest possible SSE loading in the nonlinear analysis. A less conservative, but technically defensible option would be to orient the crack for the time history finite element analysis solely based on the direction of the largest normal operating loads, since it would be the normal operating loads that would cause the crack in the first place. In this case, if the SSE loads were in a different direction from the crack orientation, LBB would be satisfied.

C.2.6 Remote Piping Material Properties

One of the possible sources of nonlinearity in a piping system that could contribute to LBB being satisfied is plasticity remote from the crack. In order to consider this possibility, the stress-strain characteristics of the pipe materials at all locations in the piping system at the appropriate temperature must be known. In general, true stress-true strain data are required. In the event that plasticity remote from the crack is not to be considered, modulus and Poisson's ratio at the operating temperature is all that is needed.

C.2.7 Nonlinear Finite Element Analysis Program

In order to successfully complete a Level 3 LBB analysis, a nonlinear finite element analysis program is required. In addition to having the standard features of a piping stress analysis program, the program must have:

- Time-history loading
- Option 1: A means to implement a nonlinear model of the crack
- Option 2: Means to conduct an analysis considering plasticity in all of the piping system.

The time history capability is needed because the crack/piping nonlinearities are load-path dependent. The nonlinear crack model is the finite element implementation of the postulated crack, see Figure C.1. In the event that the contribution of plasticity remote from the crack is to be considered in order to demonstrate that LBB is satisfied, the finite element program must have piping elements that permit yielding.

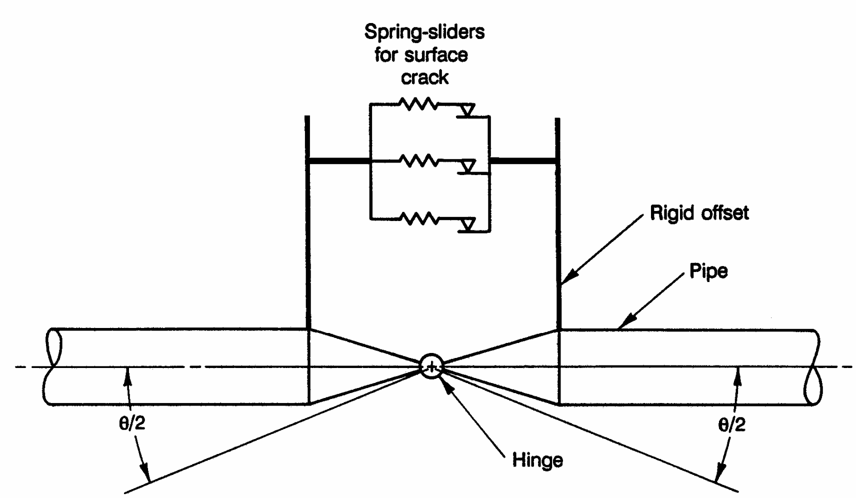

Figure C.1 Spring-slider model of a surface crack (or through-wall crack)

For analysis of circumferential cracks, standard pipe (beam) elements can be used for the bulk of the model. Shell elements can also be used for the circumferential crack analyses. For axial cracks, shell elements or beam elements with extra shell hoop behavior modes will be required since most beam-based pipe elements only consider beam bending behavior.

C.3 Level 3 LBB Acceptance Criterion

A piping system would pass the Level 3 LBB acceptance criterion if the load applied to the postulated leaking crack (with the safety factor of 2.0 applied to the crack size), as calculated in the finite element analysis, is less than the maximum load carrying capacity of the crack as calculated in Section C.2.4.

C.4 Level 3 Analysis Procedures

The procedures for conducting a Level 3 analysis are as follows:

General Set-Up

1. Make sure that the piping system meets all of the qualifications of the Level 2 analysis except for the fracture margin.
2. Build the basic piping finite element model including all boundary conditions (supports and anchors, snubbers, etc.). If the time history of loading is seismic anchor motion and inertial loads, the model only needs to consider the piping system from anchor to anchor. If the time history of loading is ground acceleration, the model must include a representation of the building foundation, the building, and the relevant members inside the building that affect the motion of the anchors of the pipe system. The piping model can be built from beam-type elements or shell elements. The building/foundation model, if needed, can be built from any number of different elements, so long as the correct interface to the pipe model is made. Structural damping, as appropriate to the type of system and construction should be included in the model.
3. Define the static loading – pressure, dead weight, thermal loading, etc. As appropriate, positional varying loads, such as thermal gradients (thermal stratification), must be considered.
4. Define the SSE loading as a time history at a suitably fine time step. Defining the loading at a fine enough time step may require interpolation. The interpolation should be done in the frequency domain (Ref. C.1) to preserve the spectral content of the interpolated signal. Failure to perform the interpolation in the frequency domain can introduce discontinuities in response, particularly if displacements (seismic anchor motions) are interpolated.

Analysis Considering Crack Nonlinear Behavior

5. Define the crack load-displacement behavior. In general, the crack behavior will come directly from the Level 2 analysis and will be given in moment-rotation or hoop load-COD coordinates.
6. Convert the crack load-displacement behavior into a finite element representation. For circumferential cracks, the load-displacement behavior can be converted to finite elements using a hinge with nonlinear springs across the hinge (Ref. C.2). Special considerations must be given to cracks when they unload. For axial cracks, the crack can

be modeled as a shell with nonlinear properties over part of the circumference. Line-spring elements in a shell model can be used to model either circumferential or axial cracks. The effect of crack closure can be modeled as very stiff springs with a gap that comes into play when the crack displacements go negative.

7. Put the finite elements representing the crack into the piping system model. The crack must be oriented in a direction that can be technically justified.

Analysis Considering Plasticity Remote from the Crack

8. Define the true stress-true strain behavior of the piping system materials.
9. Invoke the necessary plastic analysis procedures in the finite element analysis.

Finite Element Runs

10. Run the finite element time history analysis, ensuring that convergence has been met. Depending on the severity of the plasticity that the loading invokes, the time step increment may need to be reduced to a very small value (some small fraction of a millisecond) in order to have a successful run.
11. Extract the relevant applied load response data (load or moment) from the finite element time history at the crack location.

LBB Assessment

12. If the applied load in the nonlinear analysis is less than the maximum load capacity of the postulated crack (with the safety factor of 2.0 on crack size applied), then LBB is satisfied.

The Level 3 analysis considers all of the loads applied to the crack and correctly phased. Thus, there is no need to be concerned about how the various components of load are combined (algebraic, absolute sum, etc.) because they are always automatically summed algebraically.

It may be necessary to consider multiple nonlinear analyses to assure LBB because of the non-deterministic nature of the SSE loading. Experience has shown that multiple seismic time histories derived from the same response spectrum can have very different time history effects on a crack (Ref. C.3). A single time history can be used, provided that it meets certain duration, spectrum enveloping, frequency density, and PSD specifications (Ref. C.4).

C.5 References

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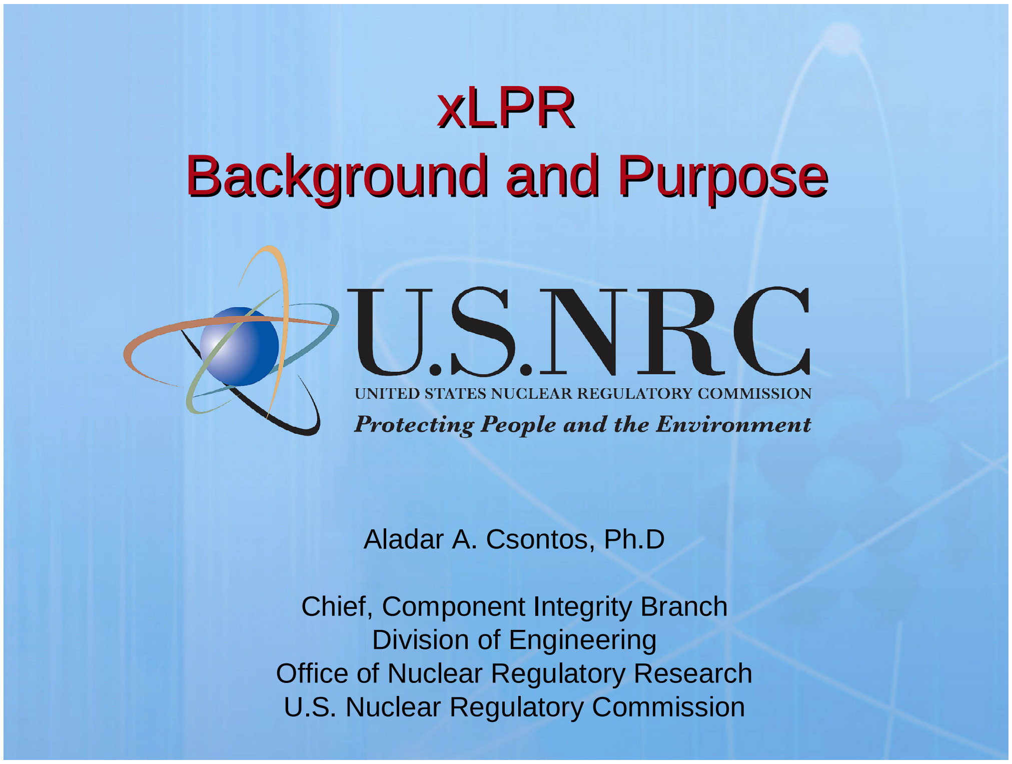
APPENDIX D

Selected Viewgraphs from xLPR Meeting



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Background

- Double-ended guillotine breaks postulated in high energy piping
- Pipe whip restraints and jet impingement shields installed
- NRC/Industry developed technical bases for demonstrating LBB
- Acceptance criteria established in NUREG-1061, Vol 3
- LBB review procedures formalized in SRP 3.6.3 (1987/2007)
- General Design Criteria 4 modified in 1987
  - allows dynamic effects of postulated pipe ruptures to be excluded from design basis when analyses approved by NRC demonstrate extremely low probability of rupture under design basis conditions
- LBB approved by NRC for several PWR piping systems
- Pipe whip restraints and jet impingement shields removed
- PWSCC is a challenge for LBB

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2

xLPR POP

- Purpose:
  - Develop probabilistic approach to ensure GDC-4 is satisfied, i.e. termed Extremely Low Probability of Rupture (xLPR)
- Expected Outcome:
  - Short Term – Deterministic Evals & Probabilistic Pilot Study
    - Assess industry's PWSCC mitigation activities in LBB lines
    - Complete pilot study - pressurizer surge line (May 2010)
  - Long Term - Probabilistic
    - Develop probabilistic tool (xLPR) for evaluating LBB in the presence of active degradation mechanisms to ensure that the probability of pipe rupture remains extremely low
- Process:
  - Joint NRC/Industry research program

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3

PWSCC: Piping Operating Experiences

- Plants with leaks
  - V.C. Summer – axial; reactor vessel nozzle (2000)
  - Tsuruga 2 – axial; pressurizer valve nozzle (2003)
  - Palisades – circ; pressurizer valve safe end HAZ (1993)
- Plants with cracks/indications
  - Ringhals 3 & 4 – axial; reactor vessel nozzle (2000)
  - V.C. Summer – circ and axial; reactor vessel nozzles (2000)
  - Tsuruga – axial; safety and relief nozzles (2003)
  - TMI-1 – axial; hot leg nozzle to surge line (2003)
  - Tihange 2 – axial; pressurizer nozzle to surge line (2003)
  - Calvert Cliffs 2 – axial; hot leg nozzle to drain line (2005)
  - D.C. Cook 1 – axial; pressurizer valve nozzle (2005)
  - Calvert Cliffs 1 – circs; hot leg nozzle to surge & drain lines (2006)
  - Calvert Cliffs 1 – axial; pressurizer relief nozzle (2006)
  - Wolf Creek – circs relief/safety nozzles, 3 circs surge nozzle (2006)
  - Farley 2 – axial and circ in surge nozzle (2007)

U.S. Nuclear Regulatory Commission

4

xLPR Uncertainty Workshop
Meeting Purpose and Goals

David Rudland
USNRC RES
June 10, 2009
Legacy Hotel, Rockville, MD

Purpose

- Developing probability of rupture in piping systems requires understanding uncertainties

- Simple sentence – Overwhelming concept

- How do we move forward??

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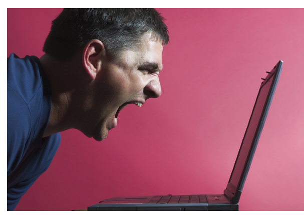
Goal

- Goal: Through this workshop become familiar with methodologies for classifying and quantifying uncertainties related to xLPR

- Develop initial consensus on how treat uncertainties

- Hopefully not…..



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Agenda

- 8:00 am - Welcome – Introduction by D. Rudland.
- 8:05 am - 8:15am - Background and Purpose of xLPR by A. Csontos
- 8:15-8:45 am – Meeting purpose and goals by D. Rudland
- 8:45 – 9:30 am –"*Overview of Uncertainty Characterization in Probabilistic Modeling - Constructing a Defensible Basis*" – Dr. S. David Sevougian
- 9:30 – 9:45am – break

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Agenda

- 9:35 – 10:35 am – "*Uncertainty and Sensitivity Analyses in Performance Assessment of Complex Systems*" – Drs. Jon Helton & Cedric Sallaberry

- 10:35 – 12:00 am – "*Sampling Based Methods for Uncertainty and Sensitivity Analysis*" – Drs. Jon Helton and Cedric Sallaberry

- 12:00 – 1:15 – LUNCH

- 1:15 – 2:15 - "*Examples of Uncertainty Analysis in Risk-Informed Applications Involving Physical Processes, Material Degradation and Fracture*" - Presentation by Dr. Mohammad Modarres, University of Maryland.

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Agenda

- 2:15 – 2:30 Summary: "*Some thoughts on probabilistic implementation in a complex system*." Helton & Sallaberry

- 2:30 – 3:30 OPEN DISCUSSION – Discussion of uncertainty methods, issues/questions with current modeling approaches. Development of consensus approach for quantifying uncertainties. Define list of initial requirements for computational group (e.g. define uncertain parameters, models, classification of uncertainty).

??

- 3:30 – 4:30 – xLPR code flow and initial structure – D. Rudland

- 4:30 – 4:45 - Plans for next day – D. Rudland

xLPR Code Flow – Working Document

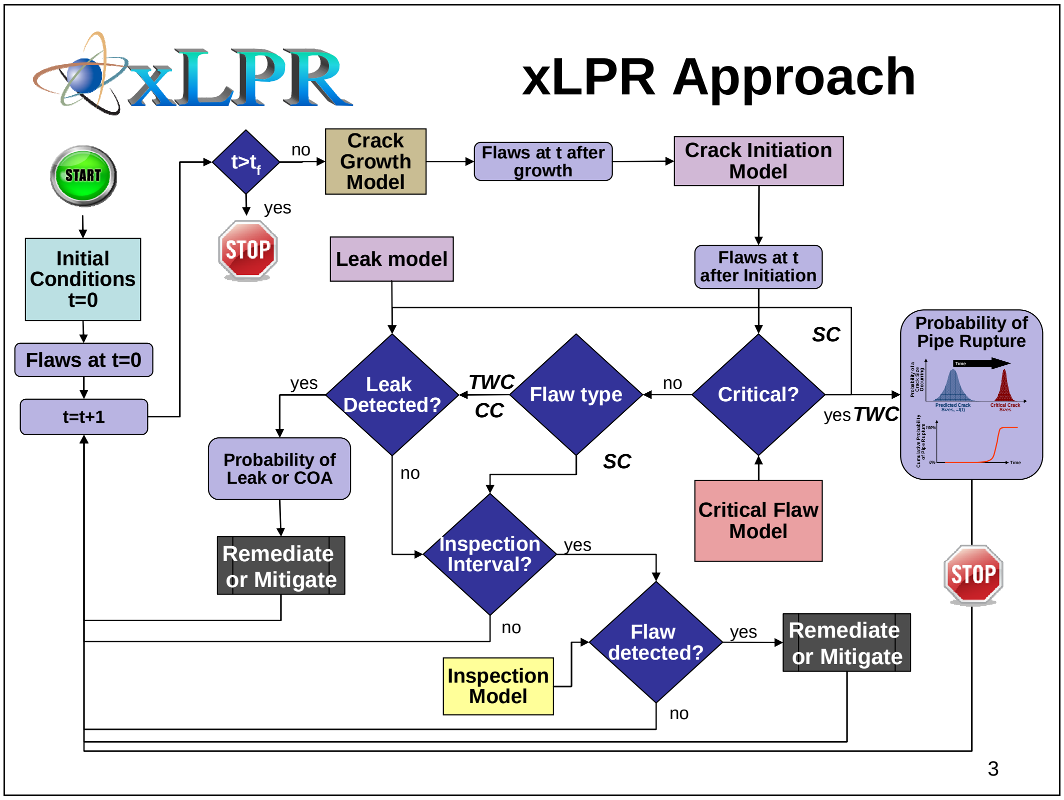
xLPR Team
xLPR Uncertainty Workshop
June 10, 2009
Legacy Hotel
Rockville, MD

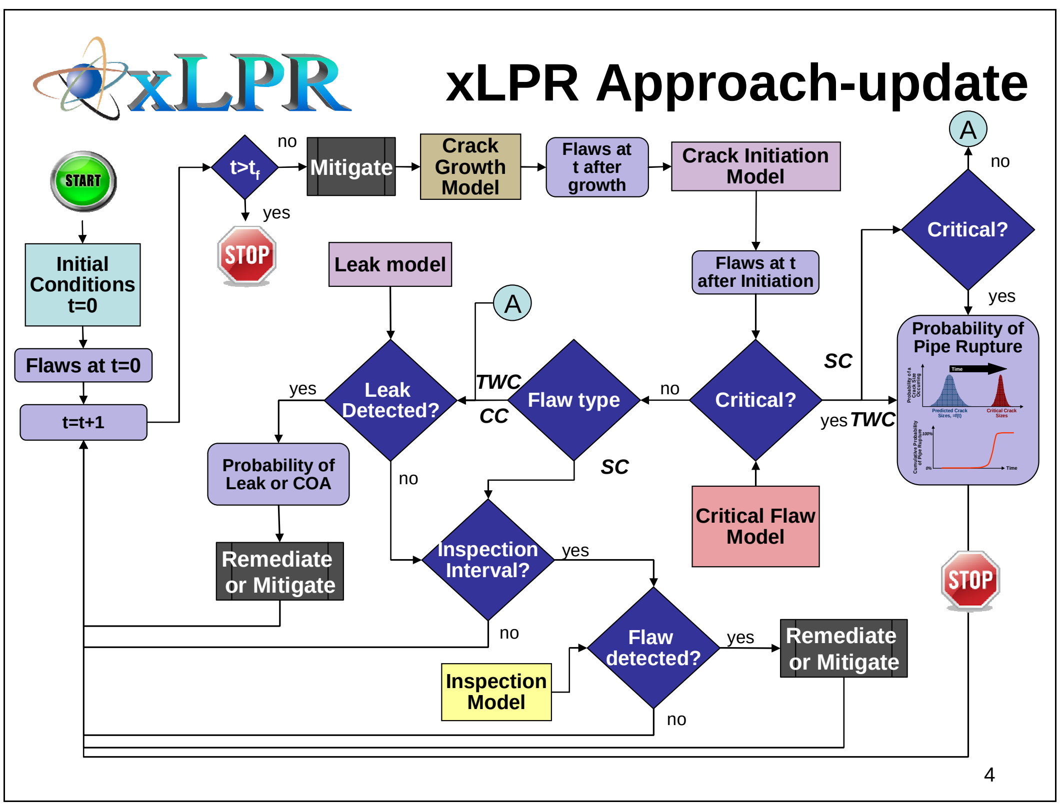
1

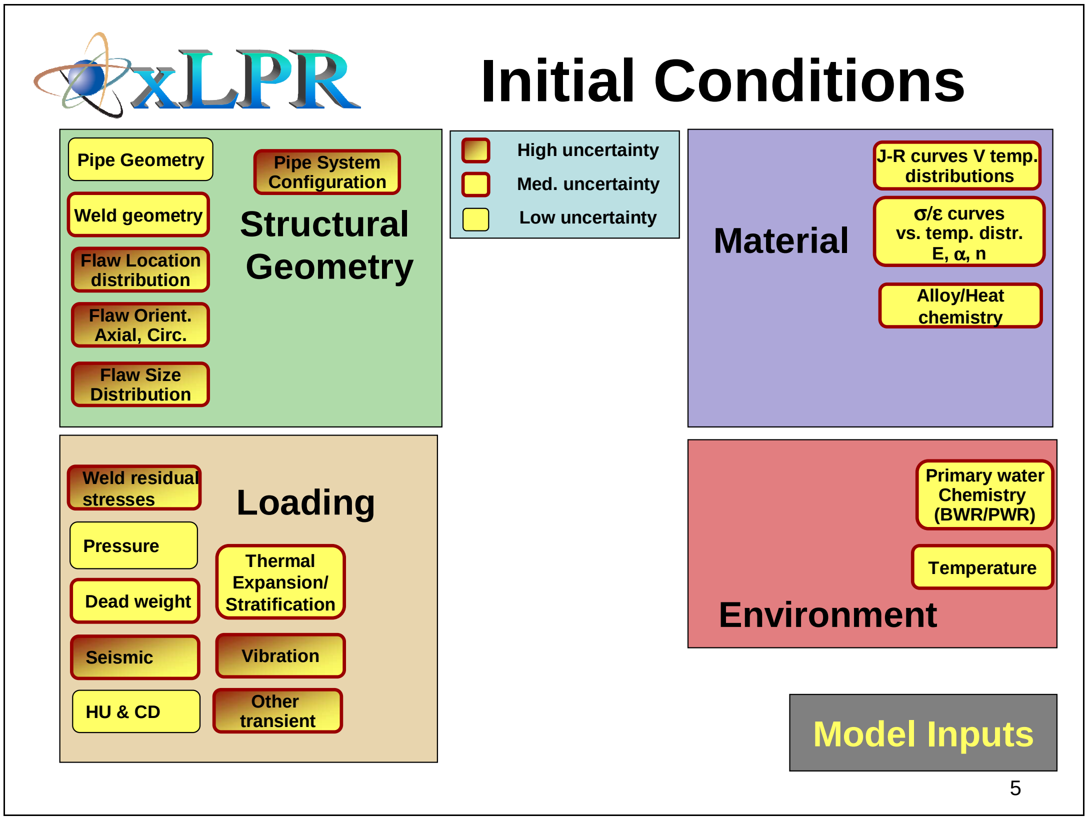
xLPR Flow

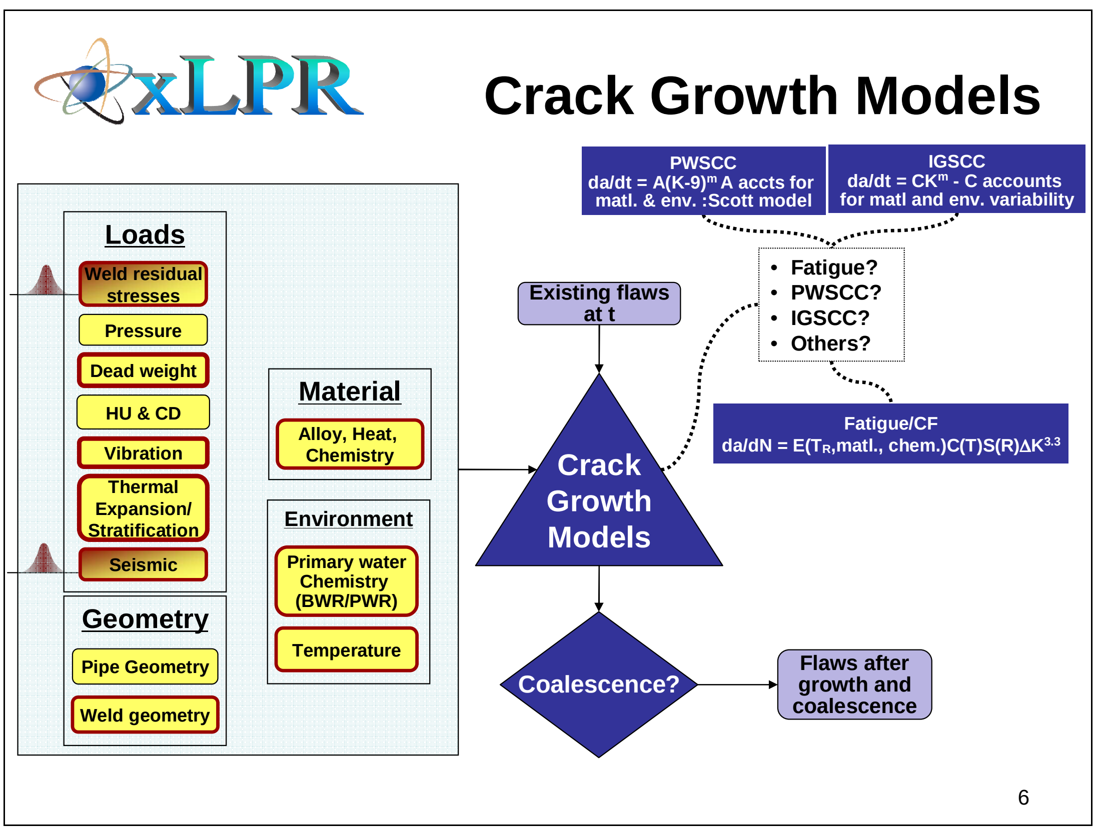
- Before beginning to characterize uncertainty, flow of code must be understood
- Basic xLPR flow is presented here, but it is still evolving
- Will be refined as working groups meet and discuss details

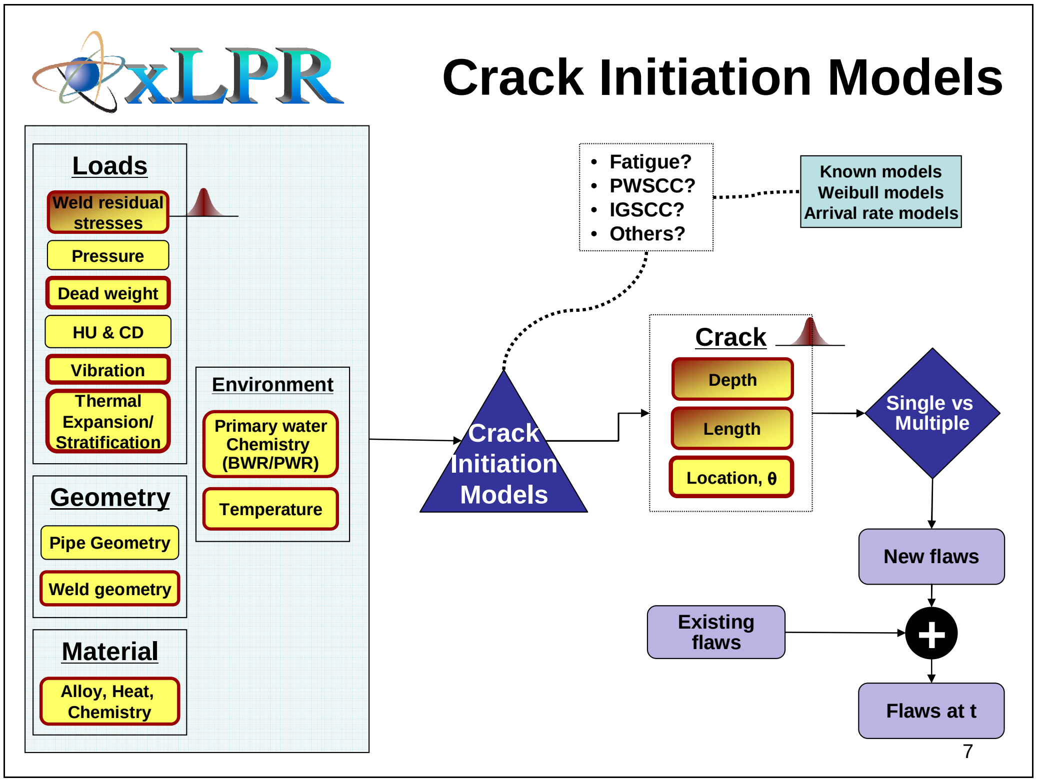
2

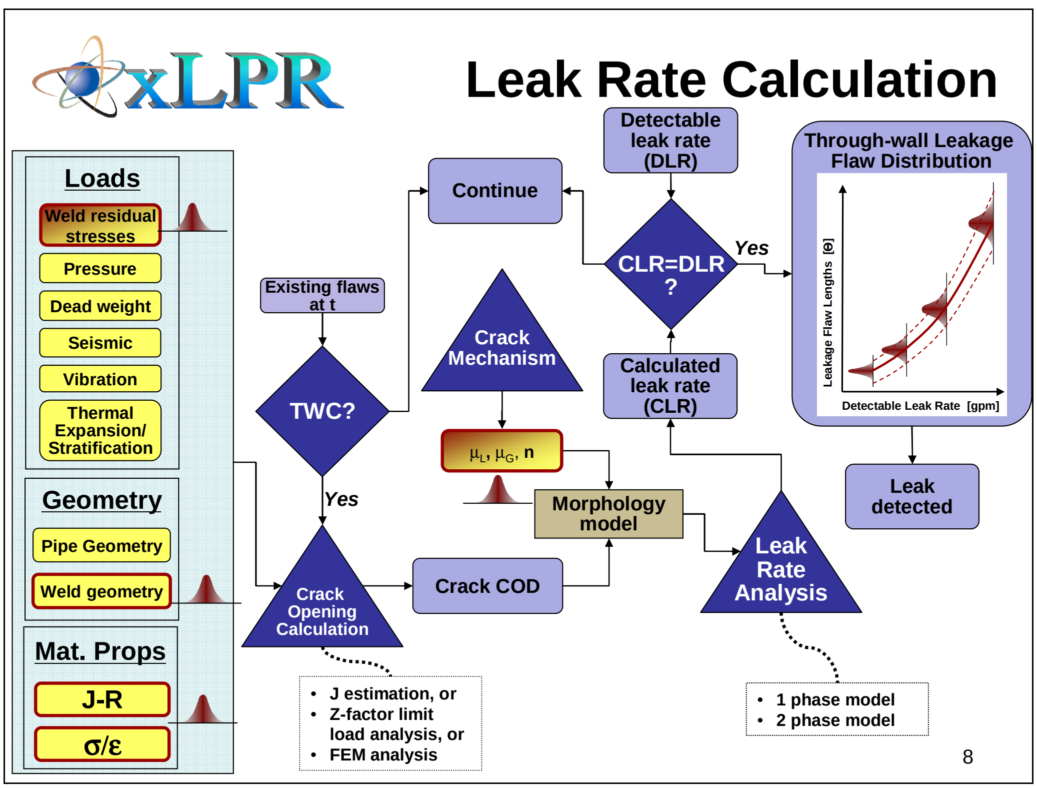

xLPR
xLPR Approach
START
Initial Conditions t=0
Flaws at t=0
t=t+1
$t>t_f$
no
yes
STOP
Crack Growth Model
Flaws at t after growth
Crack Initiation Model
Flaws at t after Initiation
Leak model
Leak Detected?
yes
no
Probability of Leak or COA
Remediate or Mitigate
TWC
CC
Flaw type
SC
no
Critical?
SC
yes TWC
Probability of Pipe Rupture
Critical Flaw Model
Inspection Interval?
yes
no
Flaw detected?
yes
no
Remediate or Mitigate
Inspection Model
STOP
3

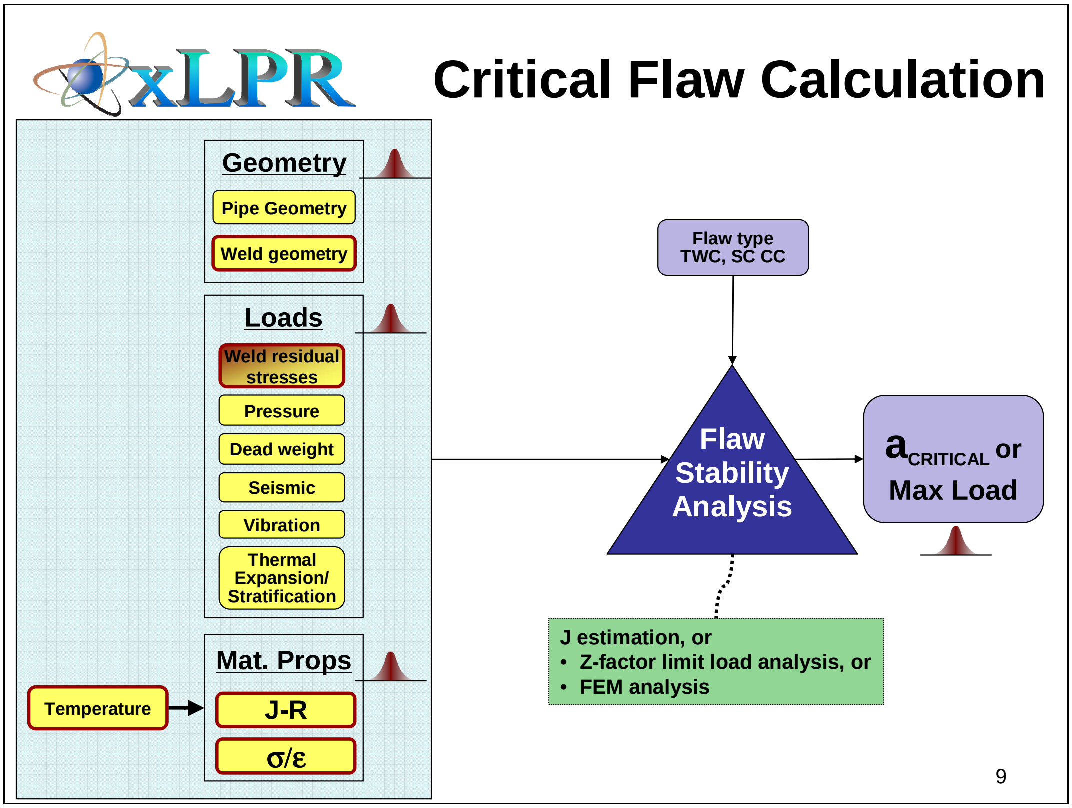

xLPR
xLPR Approach-update
START
Initial Conditions t=0
Flaws at t=0
t=t+1
$t>t_f$
no
yes
STOP
Mitigate
Crack Growth Model
Flaws at t after growth
Crack Initiation Model
A
no
Critical?
yes
Flaws at t after Initiation
Leak model
A
Leak Detected?
yes
no
Probability of Leak or COA
Remediate or Mitigate
TWC
CC
Flaw type
SC
no
Critical?
SC
yes TWC
Probability of Pipe Rupture
Critical Flaw Model
Inspection Interval?
yes
no
Flaw detected?
yes
no
Remediate or Mitigate
Inspection Model
STOP
4

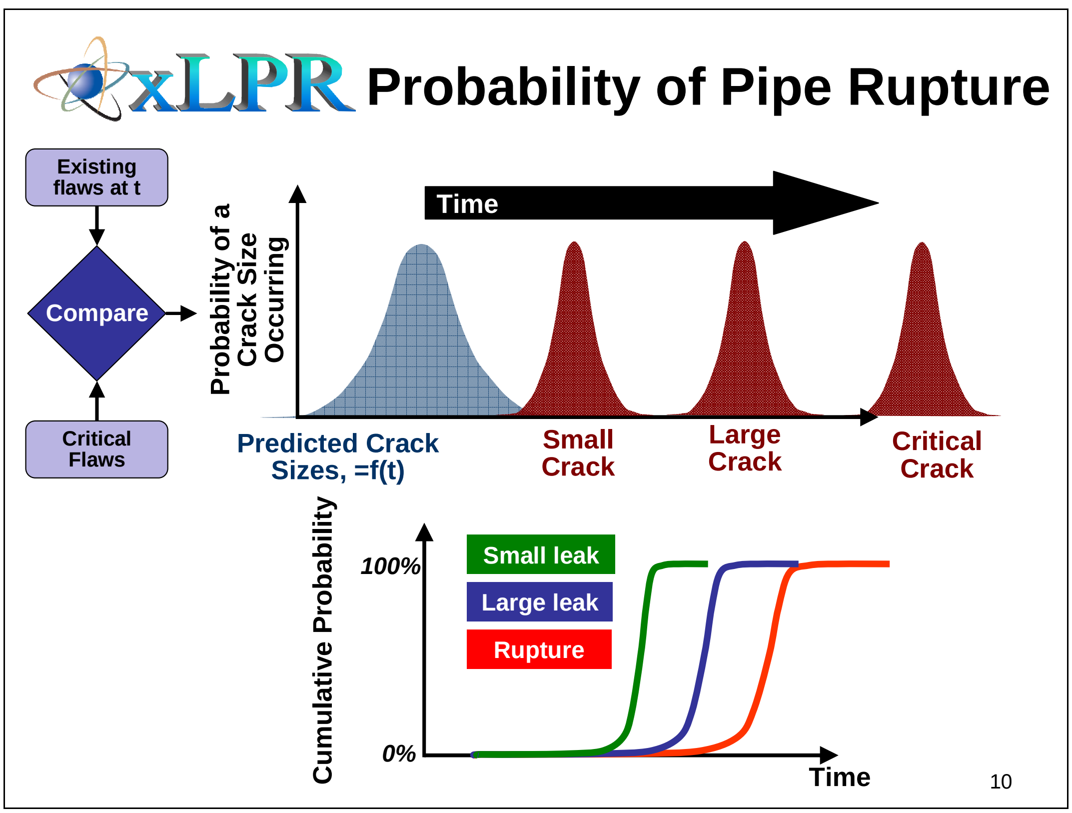

xLPR
Initial Conditions
Pipe Geometry
Pipe System Configuration
Weld geometry
Structural Geometry
Flaw Location distribution
Flaw Orient. Axial, Circ.
Flaw Size Distribution
High uncertainty
Med. uncertainty
Low uncertainty
Material
J-R curves V temp. distributions
σ/ε curves vs. temp. distr. E, α, n
Alloy/Heat chemistry
Weld residual stresses
Loading
Pressure
Thermal Expansion/ Stratification
Dead weight
Seismic
Vibration
HU & CD
Other transient
Primary water Chemistry (BWR/PWR)
Temperature
Environment
Model Inputs
5


xLPR
Crack Growth Models
PWSCC
da/dt = A(K-9)^m A accts for matl. & env. :Scott model
IGSCC
da/dt = CK^m - C accounts for matl and env. variability
Loads
Weld residual stresses
Pressure
Dead weight
HU & CD
Vibration
Thermal Expansion/ Stratification
Seismic
Geometry
Pipe Geometry
Weld geometry
Material
Alloy, Heat, Chemistry
Environment
Primary water Chemistry (BWR/PWR)
Temperature
Existing flaws at t
Fatigue?
PWSCC?
IGSCC?
Others?
Fatigue/CF
da/dN = E(T_R,matl., chem.)C(T)S(R)ΔK$^{3.3}$
Crack Growth Models
Coalescence?
Flaws after growth and coalescence
6


xLPR
Crack Initiation Models
Loads
Weld residual stresses
Pressure
Dead weight
HU & CD
Vibration
Thermal Expansion/ Stratification
Geometry
Pipe Geometry
Weld geometry
Material
Alloy, Heat, Chemistry
Environment
Primary water Chemistry (BWR/PWR)
Temperature
• Fatigue?
• PWSCC?
• IGSCC?
• Others?
Known models
Weibull models
Arrival rate models
Crack Initiation Models
Crack
Depth
Length
Location, θ
Single vs Multiple
New flaws
Existing flaws
Flaws at t
7


xLPR
Leak Rate Calculation
Loads
Weld residual stresses
Pressure
Dead weight
Seismic
Vibration
Thermal Expansion/ Stratification
Geometry
Pipe Geometry
Weld geometry
Mat. Props
J-R
σ/ε
Existing flaws at t
TWC?
Yes
Crack Opening Calculation
• J estimation, or
• Z-factor limit load analysis, or
• FEM analysis
Continue
Crack Mechanism
μL, μG, n
Morphology model
Crack COD
Detectable leak rate (DLR)
CLR=DLR ?
Yes
Calculated leak rate (CLR)
Leak Rate Analysis
• 1 phase model
• 2 phase model
Through-wall Leakage Flaw Distribution
Leakage Flaw Lengths [Θ]
Detectable Leak Rate [gpm]
Leak detected
8


xLPR
Critical Flaw Calculation
Geometry
Pipe Geometry
Weld geometry
Loads
Weld residual stresses
Pressure
Dead weight
Seismic
Vibration
Thermal Expansion/ Stratification
Mat. Props
Temperature
J-R
σ/ε
Flaw type TWC, SC CC
Flaw Stability Analysis
a CRITICAL or Max Load
J estimation, or
• Z-factor limit load analysis, or
• FEM analysis
9


xLPR
Probability of Pipe Rupture
Existing flaws at t
Compare
Critical Flaws
Time
Probability of a Crack Size Occurring
Predicted Crack Sizes, =f(t)
Small Crack
Large Crack
Critical Crack
Cumulative Probability
100%
0%
Small leak
Large leak
Rupture
Time
10

Open Discussion

- Can we develop an approach for quantifying uncertainties?
  - Models/Input group take initial stab at quantifying uncertainties
  - Computational group to concur
  - Iterations may occur
- How is the best (accurate and efficient) way to code uncertainty handling?

11

Open Discussion

- Input needed from models/inputs
  - Model/Input name
  - Model/input type
    - Database
    - DLL
    - Other?
  - Input/output
    - List of variables, arrays needed
    - Iteration with computational group on these
  - Uncertain parameters
  - Classification of uncertain parameters

12

Some thoughts on probabilistic implementation in a complex system

xLPR Uncertainty Workshop
June 10-11, 2009
Rockville, MD

Jon C. Helton
Cédric J. Sallaberry

Sandia is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the United States Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

Summary of Uncertainty Workshop

- Uncertainty is an important component of the analysis of any complex system and needs to be addressed accordingly

- Every group (Data Group, Model Group, Computational Group) has responsibilities in insuring a consistent and reasonable treatment of uncertainty. **Communication is essential between the groups**

- The process is iterative. Nothing will be perfect the first time. However uncertainty and sensitivity analyses will help determine the areas that need study/ improvement

2

Data Group

Responsibilities

Characterizing the uncertainty in the input parameters

Key-points to remember

•Consider the **context** in which a parameter will be used in a model
•Be aware of the nature of the available data *(e.g. spatial variability)* w.r.t. the use of a parameter in a model *(e.g. uncertainty in a spatially averaged value)*
•Maintain distinction between aleatory uncertainty and epistemic uncertainty
•Goal is to be neither pessimistic nor optimistic in an assessment, but honest w.r.t. the uncertainty
•Consider the implications of your choices *(e.g. normal distribution has infinite tails)*
•Document information, procedure and rationale used to characterize uncertainty
•This uncertainty characterization is iterative. **It is OK to have an initial assessment and later modify it on the basis of additional information**

3

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Model Group

Responsibilities

Helping characterizing the uncertainty in the input models
Characterizing model uncertainty

Key-points to remember

•Provide the data group descriptions of how parameters will be used in individual models/modeling contexts
•Maintain distinction between aleatory uncertainty and epistemic uncertainty
•Provide feedback on implication of parameter uncertainty
•You **do not have to necessarily choose** between two or multiple models: alternative models can be treated as an epistemic uncertainty
•Identify combinations of uncertain parameters that can lead to non-physical situations *(use correlation to reduce or suppress non-physical combinations)*
•Document the meaning of your inputs and outputs, as well as the uncertainty associated with each *(your output may be the input data for another model.)*
• Perform/document appropriate **verification and validation studies**

4

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Computational Group

Responsibilities

Insuring a consistent treatment of uncertainty in the system

Propagate uncertainty

Analyze results and provide feedback

Key-points to remember

•Maintain distinction between aleatory and epistemic uncertainty

•Develop/document clear conceptual model for an analysis: characterization of aleatory uncertainty; characterization of epistemic uncertainty; model/models for prediction of consequences

•Choose the appropriate propagation of uncertainty technique *(LHS, stratified sampling, discretization followed by interpolation …)*

•Check on the consistency on parameters and model output

•Perform Uncertainty and Sensitivity Analysis to check the validity of the model and provide feedback to the other groups on the influence of their choices

•Perform/document appropriate **verification and validation studies**

5

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Concluding Thoughts

You do not necessarily need to become an expert in statistics and uncertainty treatment (although it's nice if you do)

However, you need to document your approach and be comfortable with the choices you have made

You may not know the consequence of choosing a distribution or a range until you see the results of an uncertainty and sensitivity analysis. Iteration and feedback are essential steps of a successful probabilistic approach

6

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APPENDIX E

Summary of International Experience as of 2000

From NUREG/CR-6765

Appendix E

International LBB Experience as of 2000

A number of other countries have developed, or are in the process of developing, their own LBB procedures. Some of the countries that have or are developing LBB procedures include:

- France,
- Germany,
- Japan,
- Korea,
- Russia,
- United Kingdom,
- Canada, and
- Sweden.

Like the NRC's LBB procedures, many of these foreign procedures are still in draft form. For the most part, the procedures in these other countries are very similar to those in the United States. However, one striking difference is that they oftentimes start by postulating the existence of a part-through surface flaw, instead of a postulated through-wall crack, and then conduct a fatigue crack growth analysis of that postulated surface flaw up to the instant of surface crack penetration. Some of the other differences will be discussed in the sections that follow.

E.1 France

Chapter 4 of the draft French A16 Report (Ref. D.1), prepared by the Commissariat A L'Energie Atomique (CEA), the NRC's Office of Research counterpart in France, provides a set of draft procedures for conducting LBB analyses. The purpose of such an LBB analysis is to determine if it is possible to detect, under in-service conditions, a leak in a fluid-filled structure prior to the associated flaw causing a rupture of the structure. Procedures are provided in Reference E.1 for both the case where creep damage would not be expected and for the case where the potential for creep damage is deemed significant.

The key steps in the procedures are:

- The highest stressed regions need to be selected.
- The initial surface flaw, including the position, orientation, shape, and dimensions, needs to be defined. Typically a semi-elliptical initial flaw of size a_i and $2c_i$ is assumed.
- The fatigue crack growth of the initial semi-elliptical flaw (a_i, $2c_i$) under normal operating conditions and the analysis of the avoidance of a fast rupture or instability of the final semi-elliptical flaw (a_f, $2c_f$) need to be analyzed, for both the normal operating and normal operating plus faulted load conditions.
- The evolution of the semi-elliptical flaw size (a_f, $2c_f$) under cyclic loading up to a detectable through thickness flaw ($2c_{det}$) corresponding to a detectable leak rate (Q_{det}) needs to be calculated. The evolution of the flaw can be determined in two stages: up to the instant the surface flaw penetrates the pipe wall thickness, and up to the situation

where the length of the through-wall flaw on the external surface reaches a value equal to the detectable flaw length ($2c_{det}$).

- Analysis needs to be conducted to demonstrate the avoidance of a fast rupture or instability of the detectable flaw ($2c_{det}$) under the normal plus faulted conditions.

In calculating the evolutionary crack size (a, c) as a result of the cyclic loading, and the length of the associated through-wall crack at the instant of surface-crack penetration, an approach is presented in Reference E.1 to estimate the relationship between the length of the surface crack (c_s) and the wall thickness (t). For this approach, the ratio of c_s/t is a function of the ratio of the cyclic bending stress to the cyclic membrane stress, i.e., $)\Phi_b/)\Phi_m$. From this approach, it can be seen that pure tension loadings result in relatively short cracks while pure bending loadings result in relatively long cracks.

Chapter 4 provides a series of closed-form equations to calculate the detectable flaw length ($2c_{det}$) from the detectable leak rate (Q_{det}). First, the crack-opening area (A_L) for the detectable through-wall crack is calculated from the detectable leak rate (Q_{det}) and the fluid velocity (V) through the crack.

$$A_L = \frac{Q_{\det}}{V} \qquad \text{(E.1)}$$

where, the detectable leak rate (Q_{det}) is equal to the minimum detectable leak rate (Q_{min}) with a safety of factor of 10 applied, i.e.,

$$Q_{\det} = 10 Q_{\min} \qquad \text{(E.2)}$$

The crack-opening area of an elliptically-shaped through-wall crack is:

$$A_{TWC} = \frac{\pi \delta c}{2} \qquad \text{(E.3)}$$

where the crack-opening displacement (*) is a function of the applied stress, crack length (2c), and the dimensions of the component under consideration, i.e., mean radius (R_m) and wall thickness (t). In the third draft version of this document, it was indicated that a simplified expression for * was forthcoming.

Two equations are provided for the fluid velocity (V) depending on whether the fluid flow is in the laminar (Reynolds Number, R_e, < 2300) or turbulent (R_e > 2300) flow regime.

For laminar flow,

$$V = \frac{\Delta P \left(D_H^2 \right)}{48 \mu t} \qquad \text{(E.4)}$$

where,

ΔP = pressure difference across the crack, i.e., typically internal pipe pressure,
D_H = hydraulic diameter, approximated in Chapter 4 as B∗/2 for an elliptical crack,
μ = dynamic viscosity of the fluid at the temperature under consideration, and
t = pipe wall thickness.

For turbulent flow,

$$V = \left(\frac{2\Delta P}{\rho \left(1.5 + \frac{\lambda t}{D_H} \right)} \right)^{1/2} \tag{E.5}$$

where,

Δ = fluid density at the temperature and pressure under consideration, and
8 = a function of the rugosity and hydraulic diameter.

Rearranging Equations E.1 and E.3,

$$c_L \delta V = \frac{2Q_{\det}}{\pi} \tag{E.6}$$

The three terms in the left hand side of Equation E.6 (c_L, ∗, and V) are all functions of the crack length (c), thus Equation E.6 has to be solved iteratively.

In order to demonstrate the avoidance of a fast rupture or a crack instability, both limit-load and elastic-plastic J-based analysis routines are provided in Reference E.1 for both surface cracks and through-wall cracks.

E.2 Germany

In Germany, LBB is applied for many of the same reasons as it is applied in other countries, i.e., to justify the elimination of the design requirements that account for the dynamic effects during a pipe rupture. The elimination of these design requirements allows for the elimination of hardware, such as pipe whip restraints and jet impingement shields. This hardware can impede accessibility to pipes for inspections and increases radiation exposure during maintenance operations. As with other countries, to demonstrate LBB in Germany, it has to be shown that any crack will lead to a leak, and that this leak will be detected long before it could possibly grow to a critical size that it would grow unstably at the faulted load conditions.

In Germany, the LBB procedures are part of the break-preclusion (BP) or basis safety (BS) concept. There are two main prerequisites of the BP (or BS) concept: basic safety and independent redundancies (Refs. E.2 and E.3). The independent redundancies required for break preclusion are: (1) in-service inspection, (2) load monitoring, and (3) leak-detection systems. The process of demonstrating that a break will not occur is based on the following points:

1. Stress corrosion cracking, thermal fatigue, and water hammer need to be shown that they are not relevant failure mechanisms for the piping system under consideration. Thus, the only failure mechanism that needs to be considered is potential ductile failure resulting from a large load (emergency and faulted conditions, e.g., earthquake).
2. The fracture resistant material properties used in the fabrication of the piping system make a rupture of the piping system highly unlikely.
3. The pre-service and in-service inspections will detect any flaws. If a flaw goes undetected, its growth over the life of the plant will be insignificant, i.e., no mechanisms exist to develop a through-wall crack.
4. If an unlimited number of plant lives are assumed, a theoretical through-wall crack may develop, but that through-wall crack will not become unstable under the worst case loading conditions.
5. This stable through-wall crack will leak at a rate such that the leak can be detected by the plant's leakage detection equipment, and the plant subsequently shutdown, so that the appropriate repairs completed.

Fracture mechanics principles and criteria are used to demonstrate LBB behavior, according to the last three steps above (Steps 3 through 5 above). The initial flaw (or reference flaw) used in the fatigue crack growth analysis (Step 3) and in the LBB fatigue crack growth demonstration (Step 4) is a semi-elliptical surface flaw with a depth (a) and total length (2c). This flaw is postulated to exist in a highly stressed weld. The size of this reference flaw is based on an envelope of allowable flaws for pre-service examination and in-service inspection. Performance of inspection technologies and accumulated experience are taken into account when defining the size of this reference flaw.

The fatigue crack growth analysis for the reference surface flaw is performed using the normal and upset transient loadings, using the Paris-law fatigue crack growth model (Ref. E.4) with a conservative fatigue crack growth curve (da/dN versus)K) accounting for environmental effects. The criterion for acceptance is to demonstrate negligible fatigue crack growth of the reference flaw during the course of the projected life of the plant (Step 3 above). Assuming the piping system passes this first level of acceptance, a similar analysis is performed, except an unlimited number of plant lives are assumed. For this case, the acceptance criterion is that if the crack grows through the pipe wall by fatigue, or the ligament tears through the pipe wall, without an instability in the circumferential direction, then the LBB fatigue crack growth condition is demonstrated (Step 4 above). If on the other hand, the crack reaches a critical length before it tears through the wall, then LBB is not demonstrated. For this condition, additional safety measures (e.g., additional in-service inspections) may be incorporated in order to ensure the proof of integrity.

Next, the stability of the end-of-life surface defect, and the stability of the through-wall crack that exists once the reference surface flaw penetrates the pipe wall thickness ($2c_{Leak}$), must be demonstrated for the normal operating plus maximum accident load condition (e.g., SSE loads), i.e., the resultant leakage size crack ($2c_{Leak}$) must be less than the critical through-wall crack size ($2c_{crit}$) at the normal plus SSE load condition. Frequently, fully plastic limit-load analyses are used for these stability assessments. Given that this end-of-life surface defect and the resultant leaking through-wall crack after surface crack penetration are found to be stable, crack opening area and leak-rate analyses are performed to establish a detectable crack length. The length of this detectable crack ($2c_{LDS}$) is a function of the sensitivity of the leak detection system, as well as the applied loads on the piping system. To demonstrate LBB, this detectable through-wall crack (at normal operating loads) must be smaller than the critical through-wall crack (at normal plus SSE loads), and there must be enough time to detect the leak by the leak detection system before the crack could possibly grow to a critical length, i.e., the growth rate of the through-wall crack is not excessive at the normal operating loads.

In summary, LBB is satisfied if: (1) the leakage crack size ($2c_{Leak}$) after the fatigue crack growth of the reference defect (after unlimited plant lives) is less than the critical crack size ($2c_{crit}$); and, (2) the detectable crack size ($2c_{LDS}$) is less than the critical crack size ($2c_{crit}$); and, (3) the growth of the resultant through-wall crack is slow enough that there is enough time to detect the leak by the leak detection system, see Figure E.1.

At the time of the publication of Reference E.2, there were no prescribed safety factors (or margins) on the leakage detection capability or on the leakage or detectable crack to the critical crack size relationship. Discussions had been initiated with the German KTA with the goal of achieving a common understanding on the subject of LBB. One of the main items of these discussions will be establishing prescribed safety factors.

At the time of publication of Reference E.2, the LBB concept had been applied to a number of Siemens/KWU plants in Germany, as well as in the Netherlands, Brazil, and Argentina. For all of these applications, including the German applications, the safety factors had been set by Siemens.

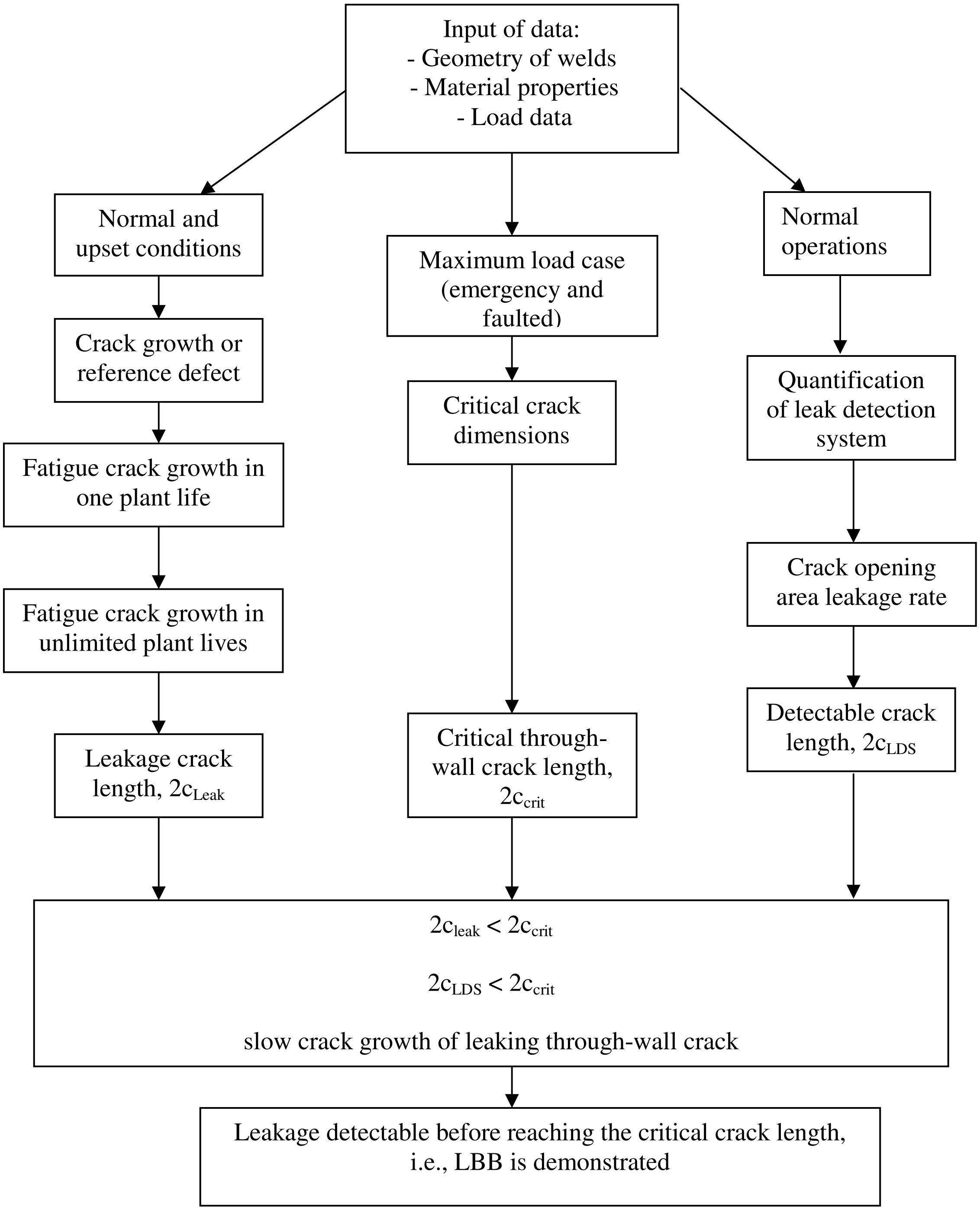

Figure B.1 Flow diagram for German LBB analysis

E.3 Japan

The Japanese LBB procedures are published in the Appendix to Reference E.5. Reference E.5 is applicable to reactor coolant pressure boundary (RCPB) piping systems. The basis concept is as follows:

1. A single initial flaw is assumed to exist on the inner surface of the pipe. The size of this initial flaw is based on the ultrasonic testing (UT) detectable limits for pre-service inspection (PSI), with an appropriate margin.
2. A fatigue crack growth analysis for this initial flaw is conducted up to the point when the growing surface flaw penetrates the pipe wall thickness.
3. The length of the resultant through-wall crack at the instant of surface crack penetration is compared with the length of a through-wall crack required to cause a 19 lpm (5 gpm) leak, and the larger of the two cracks is assumed in the crack stability analysis.
4. The stability of the assumed crack is evaluated for Operational Conditions I, II, and III and Operational Conditions I plus an S_1 earthquake.
5. If the resultant through-wall crack from Step 3 is deemed to be stable in Step 4, then LBB is satisfied.

Some of the key details associated with these basic steps outlined above are discussed in the following sections.

E.3.1 Assumed Initial Surface Flaw – It is assumed that the integrity of the base pipe materials is ensured by strict quality control and material inspection when taking delivery of the pipes from the mill. Consequently, if a flaw does exist in the piping system under consideration, it would most likely be located in the one of the circumferential girth welds. As such, only circumferentially-oriented flaws in girth welds are considered for evaluation. Further, the presence of a significant flaw in a weld prior to service need not be considered because of the inspections imposed prior to putting the plant into operation. The depth and length of this assumed initial flaw, based on limits of UT detectability, are 0.2t and 1.0t, respectively, for pipes with wall thicknesses (t) greater than 15 mm (0.59 inch). For pipes with wall thicknesses less than 15 mm (0.59 inch), the assumed flaw length is 3 times 15 mm (0.59 inch) or 45 mm (1.78 inches). In each case, the flaw shape is assumed to be semi-elliptical.

These postulated initial surface flaws are assumed to exist at locations where the applied stresses or cumulative usage factors (CUF) for fatigue are large. For this application, failure is assumed to be foreseeable if the applied stress is greater than 2.4 S_m or if the CUF is greater than 0.1. Moreover, terminal ends are assumed to be places where it is possible that relatively high-applied stresses will exist because of the existence of a structural discontinuity.

E.3.2 Applicable Damage Mechanisms Considered – Propagation and failure of a flawed pipe in service is attributed to fatigue. Water quality control has been sufficient in Japanese plants since some of the original plants were first put into operation such that incidences of stress corrosion cracking have not been observed. Corrosion and erosion/corrosion are not applicable damage mechanisms since these LBB procedures are for RCPB piping made of austenitic stainless steel,[k] and these materials have excellent resistance to general corrosion and erosion/corrosion. As such, incidences of these damage mechanisms have not been observed in the past. Creep is not a concern since the operational temperatures are less than the creep regime, and irradiation embrittlement is not a concern because of the sufficient shielding provided. Finally, water hammer can be excluded from the list of potential damage mechanisms due to precautions taken during design and optimized operational management control measures taken once the plants were placed in operation. As such, through the process of elimination, the only known applicable damage mechanism is fatigue.

E.3.3 Loads Used in Evaluation – As part of these Japanese LBB procedures, fracture mechanics calculations are made as part of the crack propagation analysis, the crack stability analysis, and the crack-opening-area analysis. The loads assumed for the crack propagation analysis are based on Operational Conditions I and II and 1/3 of the S_1 earthquake load. The loads assumed for the stability analysis are based on Operational Conditions I, II, and III and an Operational Condition I plus an S_1 earthquake. For the crack-opening area analysis, the normal operating loads are used.

Crack propagation analysis is conducted based on the design stress cycle. However, since it is an onerous task to consider differences in design conditions, pipe configurations, and earthquake resistance conditions for a variety of pipes, stress cycles that are simplified to represent a stress cycle pattern based on the design transient conditions are used. The document provides separate representative stress cycles for BWRs and PWRs, in terms of the design stress intensity (S_m). The Operational Conditions I and II and a 1/3 S_1 earthquake should be considered when setting the stress cycle for the crack propagation analysis. The number of load, or stress, cycles to be used in the crack propagation analysis is not to be specified, but instead, the crack propagation analysis is carried out until the surface crack penetrates the pipe wall thickness.

For the crack stability analyses, the stresses (or loads) considered for analysis are the primary stresses. However, for the sake of safety, the thermal expansion stresses, which are secondary stresses, are also considered. Torsional stresses should not be included, only the bending stresses. As far as a method of combining these stresses (or loads), the directional components, or signs, of each of the applicable loads can be considered as a means of superposition. Draft SRP 3.6.3 allows for a similar load combination approach, however, when doing so, the draft SRP plan procedures specify the application of safety factor of 1.4 on load. The draft SRP procedures allow this safety factor for the stability analysis to be decreased to 1.0 if the loads are combined on an individual absolute basis.

E.3.4 Material Issues – For the crack propagation analysis, corrosion fatigue crack growth rate data (da/dn versus ΔK data) for a light water reactor environment should be used. The Paris Law (Ref. E.4) expression, see Equation E.7, for the fatigue crack growth rate should be used, using the Newman and Raju (Ref. E.6) K-solution for a flat plate.

$$\frac{da}{dN} = C(\Delta K)^m \qquad \text{(E.7)}$$

(k) The Japanese LBB standards for carbon steel and low-alloy steel piping are under development.

where,

da/dN = fatigue crack growth rate,
ΔK = $K_{max} - K_{min}$, and
C and m = experimentally derived fatigue crack growth rate constants for a specific material and environment.

Values for C and m for austenitic stainless steels in an LWR environment are provided in the Japanese LBB document.

Limit-load analyses are used to predict the stability of the resultant through-wall crack. As such, only strength data are needed; there is no need for fracture toughness data. This is probably an adequate assumption when considering a stainless steel piping system fabricated with higher toughness TIG welds, but some sort of stress multiplier, such as the ASME Z-factors used in Section XI, are needed if the piping system is fabricated from lower toughness SAW or SMAW welds. The strength parameter used is the flow stress, taken to be the average of the Code specified yield and ultimate strengths at the temperature of interest:

$$\sigma_f = \frac{S_y + S_u}{2} \tag{E.8}$$

where,

Φ_f = flow stress, and
S_y and S_u = Code specified yield and ultimate strength values, respectively, at the temperature of interest.

E.3.5 Crack-Opening-Area and Leak-Rate Analyses – As part of the generalized LBB analysis procedures, the length of a through-wall crack that would cause a 19 lpm (5 gpm) leak rate must be calculated. This 19 lpm (5 gpm) leaking through-wall crack is compared with the resultant through-wall crack at the instant of surface-crack penetration, and the longer of the two crack lengths is used as the postulated crack for later use in the crack stability analysis. The basis of this 19 lpm (5 gpm) criterion is the application of a factor of safety of 5 to the plant's leak-rate detection limit capability of 3.8 lpm (1 gpm). This factor of safety of 5 is half of that specified in the USNRC draft SRP 3.6.3 on LBB.

In Reference E.5, a rather prescriptive method is provided for calculating this 19 lpm (5 gpm) leakage crack length. The method involves an iterative approach on crack length (c). As part of this methodology, a volumetric flow-rate analysis is conducted to calculate the flow rate per unit area of crack opening. Dividing the prescribed 19 lpm (5 gpm) leak rate by this volumetric flow rate per unit area, one can calculate the necessary crack opening area for a 19 lpm (5 gpm) leak. Two separate models are provided in Reference E.5 for this mass, or volumetric, flow-rate analysis. A model proposed by Henry is to be used for the case where subcooled water conditions exist, while a model developed by Moody is to be used for the case of saturated water or saturated vapor.

Having established the crack-opening area necessary to sustain a 19 lpm (5 gpm) leak, the crack-opening area (COA) of a through-wall-cracked pipe subjected to normal operating loads is calculated using the Paris-Tada method (Ref. E.7). (As shown in Section 5 of this report, the Paris-Tada method is the most conservative of the COA analyses, especially at the higher applied load levels.) The resultant COA, based on the Paris-Tada method, is a function of the pipe geometry (R and t), the applied load or stress (Φ_m and Φ_b), and the crack length (c). At this point, it is a rather simple matter of iterating on the crack length so that the Paris-Tada calculated COA equals the crack area required to sustain a 19 lpm (5 gpm) leak rate.

One final point with regards to the leak-rate analyses, the prescribed methodology specifies that the inlet losses, acceleration losses and friction losses along the crack flow path be taken into account. The surface roughness value specified is 30 μm (0.0012 inch), which is comparable to the global roughness value of 33.6 μm (0.0013 inch) reported in Table 3.3 of Reference E.8 for an air fatigue crack in a stainless steel pipe. No data for corrosion fatigue cracks in stainless steel pipes were reported in Reference E.8.

E.4 Republic of Korea

Leak-Before-Break has been approved in Korea for high energy piping systems inside containment[l] of the recently constructed pressurized water reactors (PWRs). The stated purpose behind the application of LBB for these piping systems is the removal of the dynamic effects associated with the postulated double-ended-guillotine-break from the design basis, as well as the elimination of the need for pipe whip restraints and jet impingement shields so as to increase access for inspections. Reference E.9 describes the procedures followed in these applications. These procedures are fundamentally based on the US Nuclear Regulatory Commission (USNRC) requirements as detailed in NUREG-1061 Vol. 3 (Ref. E.10) and the USNRC draft SRP 3.6.3 (Ref. E.11). However, in applying LBB for these piping systems, the Koreans imposed a number of additional special requirements and addressed a number of issues of concern not specifically addressed in NUREG-1061 Vol. 3 or the draft SRP 3.6.3. These requirements and concerns are discussed below.

E.4.1 Dynamic Fracture Toughness Tests – For carbon steel piping applications, the Korean regulators required that both static and dynamic fracture toughness tests be performed. This stipulation was added to address the concern that the fracture properties of carbon steel piping materials are known to decrease as the loading rate increases at PWR operating temperatures. This phenomenon has been attributed to dynamic strain aging effects, as discussed previously in Section 5.3.3.3.

E.4.2 Thermal Stratification Considerations – The pressurizer surge line at the Yong Gwang Nuclear Units 3 and 4 (YGN 3&4) barely satisfied the required margin of 2 on crack size when the thermal stratification loads were added to the normal and faulted loads. As a result, the following additional requirements were stipulated prior to the approval of LBB for the surge lines in these plants:

- The thermal stress due to thermal stratification had to be considered in the piping design stress analysis and had to be considered as a special load in the LBB evaluation.
- The effects of thermal stratification in the surge lines had to be measured during the hot function test of these units to verify the conservatism of the assumptions used in the calculation of the thermal stresses. Intensive measurements of the temperature distribution and piping deflections were made during the start up of YGN Unit 3. The results from these measurements showed that the assumptions used in the thermal stress calculations were indeed conservative.

E.4.3 Thermal Stripping in the Pressurizer Surge Line – Because thermal stripping in the surge line has the potential to cause fatigue damage, and it was felt that such a crack might go undetected during in-service inspections (ISI), the applicant was required to evaluate the fatigue behavior of a small crack due to thermal stripping. The behavior of a crack located in the thermal stripping zone in a thermally stratified pipe was numerically investigated. The results of that analysis showed that the behavior of such a crack would depend strongly on the oscillation frequency and the heat transfer coefficient. However, the crack was not expected to grow

(l) Primary coolant lines, pressurizer surge lines, safety injection system lines, and shutdown cooling lines.

because the magnitude of the thermal stripping stresses is highest on the inside surface and attenuates rapidly through the wall thickness.

E.4.4 Water/Steam Hammer in the Main Steam Line – The applicant of the YGN Units 3 and 4 submitted an application for LBB for the main steam lines. However, that application was not accepted for two main reasons. For one, the required margins could not be satisfied when the water/steam hammer loads were considered. Secondly, for the carbon steel pipe material used for these steam lines, there were a number of uncertainties in the material fracture properties that had to be considered, e.g., dynamic load effects, cyclic load effects, weld/HAZ effects, etc. The necessary data to address each of these concerns did not exist at the time of the application.

E.4.5 Nozzle/Pipe Interface Considerations – In some of the LBB analyses considered, the highest stress locations were at the nozzle-to-pipe interface location at the terminal end. At these locations there are asymmetries due to both geometry and material considerations. The concern was that these asymmetries may affect the crack-opening behavior. The effect of asymmetry on the crack-opening behavior, and resultant leak rate, was numerically investigated. The results showed that the traditional simplified finite element model, in which the asymmetry due to geometry and material properties was not considered, still resulted in a conservative assessment when compared with the 3D model in which this asymmetry was considered.

E.4.6 Leak-Rate Detection Limit Capability - An additional stipulation on LBB imposed in these applications was that it was not acceptable to use a 1.9 lpm (0.5 gpm) leak-rate detection limit capability with a margin of 10 in order to reduce the size of the postulated leakage crack even though the leak-rate detection system has the detection capability of 1.9 lpm (0.5 gpm). This is more restrictive that the criteria imposed in the draft SRP 3.6.3. Draft SRP 3.6.3 merely stipulates a margin of 10 on leak-rate detection limit capability, regardless of the detection limit capability. Numerous applications have been approved in the US in which the leak-rate detection limit capability was reported to be 1.9 lpm (0.5 gpm).

E.5 Russia

Some of the early generation WWER-440/230 nuclear power plants (NPPs), built in Russia and some of the Eastern Bloc nations, were designed and built with emergency core cooling systems (ECCS) which were able to cope with only a limited scope of breaks, and were also designed and built without an appropriate containment system. As a result, a large pipe break in some of these plants would result in the loss of two main safety functions: cooling of the fuel and containment of the radioactive material. Therefore, the applicability of LBB was identified as an issue of major safety significance for their continued operation. Successful application of LBB was a must to justify their continued operation. LBB was considered as the only feasible approach for providing for the reduction of the probability of the primary breaks that these Russian plant designs are not currently able to cope.

In 1994, the International Atomic Energy Agency (IAEA) published some guidelines for the application of LBB to these types of plants (Ref. E.12). The LBB guidance/guidelines provided by IAEA are similar in nature to those used in the US. Basically, LBB can be applied to

WWER-440 Model 230 type reactors if it can be demonstrated that sufficient margins exist between a through-wall flaw of a size that can be reliably detected by the plants leakage detection systems at normal operating conditions and a through-wall flaw of a critical size at the faulted loading conditions.

As is the case with the US procedures, the IAEA guidelines postulate the existence of leaking through-wall cracks at discrete locations for analysis along the piping system. At these locations, it must be demonstrated that this leaking crack can be detected by the plant's leakage detection systems. Furthermore, if undetected, this leaking through-wall crack would be of such a size that it would not grow in an unstable manner under the faulted loading conditions (SSE loadings) specified for the plant. The IAEA guidelines specify the same margins (i.e., 10 on leak-rate detection limit capability, 2 on crack size, and 1 or 1.4 on loads [depending on the method of load combination] for the crack stability analysis) as incorporated in the draft SRP 3.6.3. In addition, as is the case with the draft SRP, the IAEA guidelines require that it be demonstrated that fatigue, corrosion, and stress corrosion cracking are not active degradation mechanisms for the piping system under consideration.

At about the same time the IAEA was publishing their guidelines for LBB for WWER-440/230 plants, engineers in Russia were attempting to apply LBB to the main coolant loop piping systems for WWER-1000 plants (Ref. E.13). Besides the obvious desire for a higher safety level, these engineers were attempting to build a case for the abandonment of a number of the costly protective measures needed to mitigate the consequences of a hypothetical DEGB in a high-energy piping system. The procedures they followed were similar to those advocated by the IAEA (as well as the USNRC), except that they also stipulated the evaluation of a postulated part-through surface crack (0.1t deep and 0.5t long, where "t" is the pipe wall thickness) for fatigue crack growth and surface crack instability analyses. For this particular application, they concluded that the surface crack growth due to fatigue could be neglected. The surface crack would not grow unstably (for all crack lengths) as long as the crack depth was less than 50 percent of the pipe wall thickness, and would not grow unstably for cracks less than 90 degrees, as long as the crack depth was less than 75 percent of the pipe wall thickness. Overall, they concluded that LBB could be applied to the main coolant loop piping of WWER-1000 designs.

E.6 United Kingdom

In the United Kingdom, Chapter III.II of the R6 document (Ref. E.14) is one of the documents that deals with the subject of LBB. British Standards document BS7910 and its predecessor PD6493 are two others. The technical details of the LBB procedures in each of these documents are essentially the same. In many instances, the wording is identical. Unlike some of their counterparts in other parts of the world (e.g., the draft SRP 3.6.3 procedures in the United States), the BS7910 and R6 procedures are generic procedures applicable to a variety of industries, not just nuclear. Both BS7910 and R6 set out two alternative methodologies for making an LBB assessment and recommend methods for carrying out each. The first method common to both is a simplified detectable leakage approach based on a postulated through-wall crack, much in the motif of the USNRC draft SRP 3.6.3 procedures. The second method is a full LBB procedure that sets out a more rigorous approach, which considers the development of a part-penetrating defect.

E.6.1 Detectable Leakage Approach – The simplified type of LBB argument in both BS7910 and R6 aims to demonstrate that a leaking through-wall crack is detectable long before it grows to a critical length. This type of detectable leakage argument is the type of assessment made in a USNRC NUREG-1061 Vol. 3 or draft SRP 3.6.3 type of LBB analysis. The starting point for this type of assessment is to postulate the existence of a full-penetrating crack, and demonstrating that, should that crack arise, the leakage would be detectable well before the crack grew to a critical length.

While the detectable leakage approach in BS7910 and R6 is fundamentally similar to the USNRC LBB procedures in NUREG-1061 and draft SRP 3.6.3, there are some fundamental differences of note. Because NUREG-1061 is specifically intended for light water reactor piping, some of its recommendations and safety margins are rather specific. On the other hand, in keeping with the basic philosophy of BS7910 and R6, margins are left to the judgment of the user with due regard to the methodology used, the assumptions made, the sensitivity studies conducted, and the specific application.

Implicit in this type of analysis is the assumption that once a through-wall crack develops that results in a leak of size equal to the minimum detectable leakage by the plant's leakage detection systems, that such a leaking crack will be detected almost immediately. However, the authors of BS7910 and R6 recognized the fact that for certain applications, the piping system under consideration is only monitored at set intervals, perhaps by personnel on scheduled inspection tours. As such, these documents stipulate that allowances must be made for any fatigue or creep crack growth that might occur between the instant the crack first penetrates the pressure boundary with a detectable leak rate and the time of the next scheduled inspection.

E.6.2 Full Leak-Before-Break Approach – Whereas the starting point for the detectable leakage approach is a postulated through-wall crack, the starting point for the full LBB approach is usually a surface defect that has yet to break through the pipe or vessel wall. In order to make such an assessment, it is necessary to show that:

- the defect will penetrate the pressure boundary before it can lead to a catastrophic failure; and
- the resulting through-wall crack leaks at a sufficient rate to ensure its detection before it grows to a critical length at which time a catastrophic failure occurs.

In order to carry out such an assessment, several steps are involved. First, the defect must be characterized as a surface crack or through-wall crack, and the mechanisms by which it can grow identified. The next step is to assess the crack shape development as the surface crack grows through the pipe wall in order to calculate the length of the through-wall crack formed as the initial defect penetrates the pressure boundary. Where crack growth occurs by fatigue, methods are provided in the documents to predict the increase in both the depth and length of the defect. Procedures are also provided for the treatment of creep crack growth. The crack length at breakthrough is then in turn compared with the critical crack length of a fully-penetrating crack. Finally, it is necessary to estimate the crack-opening area and the associated leak rate of fluid from the crack, and whether or not the leak will be detected by the plant's leakage detection system before the crack grows to a critical length.

E.6.3 Crack Opening Area Analyses – R6 provides a relatively simple set of closed-form equations for estimating the crack opening area (A) of a through-wall crack in a pipe if through-wall bending stresses are absent or can be ignored, see Equation E.9.

$$A = \alpha(\lambda)\frac{\pi P_m (2c)^2}{2E} \qquad \text{(E.9)}$$

where,

P_m = membrane stress,
c = half crack length,
E = elastic modulus, and
Φ_f = flow stress.

where,

∀ is a correction factor to account for shell bulging, i.e.,

$$\alpha(\lambda) = 1 + 0.1\lambda + 0.16\lambda^2 \qquad \text{(E.10)}$$

for axial cracks in cylinders, and

$$\alpha(\lambda) = \left[1 + 0.117\lambda^2\right]^{1/2} \qquad \text{(E.11)}$$

for circumferential cracks in cylinders,

where,

8 = shell parameter = $\left[12\left(1-\nu^2\right)\right]^{1/4} c/(Rt)^{1/2}$

These expressions were derived using thin-walled, shallow-shell theory, and are strictly valid only for pipes with $R/t \geq 10$, and the crack length does not exceed the least radius of curvature of the shell.

These closed-form expressions could be used in a Level 1 type LBB analysis in the prediction of the postulated leakage crack size. On the surface they appear to be somewhat easier to use than the empirically-derived influence functions specified for Level 1 type analyses. In addition, they may be more theoretically sound due to the fact that they are based on readily recognized shell theory.

These expressions are generally conservative as long as the through-wall bending stresses are negligible. It is recognized in the British documents that through-wall bending stresses can induce crack face rotations that reduce the effective crack opening area. If complete crack closure occurs, a case for LBB cannot be made. In such a case, it may be necessary to invoke a more complicated Level 2 type analysis. Significant through-wall bending stresses may be associated with thick-walled shells under internal pressure loading, or be associated with weld residual stresses, geometric discontinuities, or thermal gradients. A series of references that may be useful in estimating the elastic crack-face rotations in simple geometries are provided.

It is also recognized that if the crack is close to a significant geometric constraint (e.g., a pipe nozzle intersection), then local effects can influence the amount of crack-opening area. This is the same effect recognized during the IPIRG program referred to as the restraint-of-pressure-induced bending effect on crack-opening displacements. The impact of this effect on LBB analyses is currently being investigated as part of the BINP program. Again, if such a restraint exists, then the user would most likely need to invoke a Level 2 type analysis in lieu of a Level 1 type analysis.

For cracks in complex geometries (such as elbows), reference is made for the need to resort to finite element analyses to obtain an accurate crack-opening-area assessment. Until recently, this was one of the few possible means of estimating the crack opening area of a through-wall crack in an elbow. However, recently, Battelle as part of the USNRC LBB Reg. Guide and BINP programs developed a finite-element based J-estimation scheme that can be used for such assessments. In addition, lots of work in this area has been conducted in India (Refs. E.15 and E.16).

Finally, it is recognized in the British documents that off-center loads and crack-face pressure can influence the crack-opening-area predictions. With regards to the crack-face pressure effect, it is recommended that 50 percent of the internal pressure should be added to the membrane stress on the crack face. This value should then be reassessed when undertaking the leakage calculations, and the results iterated, if necessary.

E.6.4 Leak-Rate Calculations – The calculation of the leak rate through a crack is a complex problem involving the crack geometry, flow path length, friction effects, and the thermodynamic conditions of the fluid through the crack. For two-phase flow, references are made in the British documents to both the PICEP (Ref. E.17) and SQUIRT (Ref. E.18) leak-rate codes as being state-of-the-art codes for predicting the leak rate through a crack. These British documents also recognize friction effects, as described by local crack morphology parameters, as being an important consideration in any leak-rate analyses. These parameters vary with the type of cracking mechanism. In addition, at least one of the British documents comments that consideration should be given to the potential for flow reduction mechanisms due to particulate blocking or plugging, but offers no firm advice as how to assess such effects.

E.7 Canada

Ontario Hydro has developed an LBB approach for application to the large diameter heat transport piping for the Darlington nuclear generating stations, as an alternative to the provision of pipe whip restraints. This approach, which is described in detail in Reference E.19, has been applied to pipe sizes that are equal to or greater than 21 inches in diameter. A comprehensive and systematic review of pipe failure mechanisms is considered the first important step in establishing the role and applicability of the LBB concept. The intent, at this first step, is to provide assurance that adequate protection from failures attributable to each relevant potential failure mechanism is provided for, or that sufficient provisions are incorporated into the program to preclude the occurrence of failures from any mechanism evaluated as being credible. The failure mechanisms assessed included: stress corrosion cracking, corrosion, erosion and erosion-corrosion, cavitation and cavitation accelerated corrosion, conventional and corrosion-assisted fatigue, material aging, external effects (such as fretting, impact, pipe whip, and snubber malfunctions) and excessive loading. Based on this assessment, it was concluded that fatigue was the only mechanism that could be active in these piping systems.

The Ontario Hydro LBB approach incorporates assessments at several levels to provide assurance against catastrophic rupture. As part of the normal design process for Class 1 nuclear piping, stress analyses are performed to show that the piping system can accommodate the defined service loads with large margins of safety. At a second level, it is further demonstrated that the largest part-through surface flaw that can be detected, will not grow through the pipe wall during its design life, and that such flaws are stable for the maximum credible piping loads. At a third level of assurance, application of elastic-plastic-fracture-mechanics (EPFM) methods are used to show that a postulated leaking through-wall crack will not extend in an unstable manner, and that the leakage rate from that postulated crack is well within the capabilities of the leakage detection systems.

For the evaluation of crack stability, the J-integral/tearing modulus (J/T) approach was used. The finite element program ABAQUS was used to perform the EPFM analyses. The analyses were performed not only for circumferentially-oriented cracks at girth welds in straight pipe runs, but also for longitudinally-oriented cracks in fittings, namely, elbows, tees, and branch connections. Extensive material property data were developed from actual large diameter piping, forgings, welds, and heat-affected-zones for the Darlington nuclear generating station.

With respect to leakage, operating policies in place at similar Ontario Hydro facilities require immediate shutdown actions to be initiated upon detection of a 0.5 kg/s (1.1 lbm/s) leak rate from the heat transport system[m]. Based on operating experience, leak rates from the heat transport system significantly less than 0.05 kg/s (0.11 lbm/s) are within the capability of the leakage detection systems in the current design. Thus, there is at least a margin of 10 between detection capability and required action, similar to that in the USNRC draft SRP procedures. A special purpose leak rate code (LEAK RATE) was used to make the leak rate calculations. The crack opening displacements (COD) used in this code are calculated by assuming that only the normal operating pressure in the pipe acts to open the crack, i.e., crack opening due to the bending moments is not accounted for. This approach assures margin on leak rate, and thus provides additional confidence that the overall assessment is conservative. Other crack-opening-displacement aspects that might affect the leak rate calculations that were considered by Ontario Hydro included: crack lipping, surface roughness, and crack face pressure. Crack lipping is a bulging related effect in which the presence of a through-wall crack in a shell structure results in a redistribution of the stresses, which results in a relative rotation (lipping) of the two crack, faces. The results from studies conducted as part of Reference E.19, showed that the crack opening area at the outside surface is 50 percent larger than that at the inside surface. Furthermore, it was shown that the leakage rate corresponding to the actual crack geometry was 25 percent larger than when lipping was not accounted for, i.e., when the middle surface crack opening area was used in the analysis. Thus, not accounting for this lipping behavior results in a conservative prediction of the COD from an LBB perspective.

(m) For the operating pressure assumed in Ontario Hydro's analysis [9.6 MPa (1,400 psi)], this mass leak rate of sub cooled water of 0.5 kg/s equates to a volumetric leak rate of 43 lpm (11 gpm).

With regards to surface roughness, it was shown as part of Reference E.19 that the assumed surface roughness can significantly influence the calculated leakage rate. It was shown that an order of magnitude change in surface roughness results in a 50 percent change in the calculated leakage rate.

Finally, Reference E.19 provides some very useful insights as to the effects of crack face pressure on the crack-opening-displacements, and thus the calculated leakage rates. As stated earlier, the pressure acting on the faces of the through-wall crack will tend to open the crack, which will increase the crack opening area and associated leak rate. Ignoring this effect will result in a conservative assessment of LBB. However, for cases that barely fail to satisfy LBB, accounting for this effect may be all that is needed to successfully demonstrate LBB. Unfortunately, no concrete means of accounting for this effect have been proposed, until now. However, Reference E.19 proposes a simple equation to correct for this effect, see Equation E.12:

$$\frac{COD_{cf}}{COD_{wo}} = \left(1 + \frac{P_{cf}}{\sigma}\right) \tag{E.12}$$

where,

COD_{cf} = crack opening displacement corrected for the crack face pressure,
COD_{wo} = crack opening displacement not accounting for crack face pressure,
P_{cf} = pressure acting over the crack faces, and
Φ = far field component of the membrane stress perpendicular to the crack plane.

Comparisons were made between this simple correction factor (Equation E.12) and finite element results, and it was found that Equation E.12 slightly underpredicted (1 to 7 percent) the finite element calculated corrected COD term. It was also shown that this effect (crack face pressure) could result in an additional 25 to 40 percent in margin on COD, depending on the component geometry (straight pipe versus elbow), crack orientation, and crack size. Consequently, this may be an effect worth considering if LBB cannot be demonstrated using the more conventional LBB methods.

E.8 Sweden

In corresponding with Dr. Bjorn Brickstad, the former IPIRG TAG representative from Sweden, SKI (the Swedish Inspectorate) has recently issued a report on the subject of LBB (Report Number SKI-PM 98:39, 2000-03-27, in Swedish). SKI now allows LBB in accordance with the draft SRP 3.6.3 procedures with the following amendments:

- Weld residual stresses should be accounted for when determining the shape of the crack and when evaluating the leak rate.
- There should be strict requirements for leak rate detection and limiting values of detected leak rates above which the plant has to shut down.
- In the fracture mechanics evaluation, the SSE load should be replaced with “the worst emergency faulted load” if such a load exists that is worse than the SSE load.

- The pipe system under consideration for LBB should have been previously subjected to a full volumetric inspection with a qualified procedure, either after construction or later as part of an in-service inspection (ISI).

According to Dr. Brickstad, there are other amendments to consider, but they are of less importance.

E.9 References

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E.3 Wellein, R., and Preu∃er, G., "State of Engineering of the Leak-Before-Break Procedures in Germany," 7th German-Japanese Joint Seminar, MPA-Stuttgart, September 1997.

E.4 Paris, P. C., and Erdogan, F., "A Critical Analysis of Crack Propagation Laws," *Journal of Basic Engineering*, Vol. 85, pp 528-534, 1960.

E.5 Kyokai, N., and Bukai, G., "Technical Guidelines for Protection Design Against Postulated Piping Failures in Nuclear Power Plants," JAEG 4613-1998, December 1998.

E.6 Newman, J. C., and Raju, I. S., "An Empirical Stress-Intensity Factor Equation for the Surface Crack," *Engineering Fracture Mechanics,* Vol. 15, No. 1-2, pp 185-192, 1981.*

E.7 Paris, P. C., and Tada, H., "The Application of Fracture Proof Design Methods Using Tearing Instability Theory to Nuclear Piping Postulating Circumferential Through-Wall Cracks," NUREG/CR-3464, September 1983.

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E.9 Lee, J. B., and Choi, Y. H., "Application of LBB to High Energy Pipings of a Pressurized Water Reactor in Korea," *Nuclear Engineering and Design,* Vol. 190, pp 191-195, 1999.

E.10 Report of the U. S. Nuclear Regulatory Commission Piping Review Committee – Evaluation of Potential for Pipe Breaks," NUREG-1061, Vol. 3, November 1984.

E.11 "Leak-Before-Break Evaluation Procedures," draft Standard Review Plan 3.6.3, August 1987.

E.12 "Guidance for the Application of the Leak Before Break Concept: Report of the IAEA Extrabudgetary Programme on the Safety of WWER-440 Model 230 Nuclear Power Plants," IAEA-TECDOC-774, November 1994.

E.13 Kiselev, V., A., and others, "Application of Leak-Before-Break Concept to Integrity and Safety of PWR Primary Piping with WWER 1000," *Nuclear Engineering and Design*, Vol. 151, pp 409-424, 1994.'

E.14 Milne, I., and others, "Assessment of the Integrity of Structures Containing Defects," Nuclear Electric Report R/H/R6, Revision 3 with updates to June 2000.

E.15 Rastogi, R., Bhasin, V., and Kushwaha, H. S., "Fracture Assessment of Straight Pipes and Elbows with Through-wall Cracks: Using R-6," Paper G05/5, Division G, *Transactions of the 14th International Conference on Structural Mechanics in Reactor Technology,* August 1997.

E.16 Chattopadhyay, J., and others, "Limit-Load Analysis of Straight Pipes and Elbows with Through-Wall Cracks," Paper G05/6, Division G, *Transactions of the 14th International Conference on Structural Mechanics in Reactor Technology,* August 1997.

E.17 Norris, D., and others, "PICEP: Pipe Crack Evaluation Program," EPRI Report NP-3596-SR, 1984.

E.18 Paul, D. D., and others, "Evaluation and Refinement of Leak-Rate Estimation Models," NUREG/CR-5128, Rev. 1, June 1994.

E.19 Nathwani, J. S., and others, "Ontario Hydro's Leak Before Break Approach: Application to the Darlington (CANDU) Nuclear Generating Station A," *Nuclear Engineering and Design*, Vol. 111, pp. 85-107, 1989.

APPENDIX F

Detailed Explanation of UK Regulatory System Relative to LBB Questionnaire – by P. Harrop (NII)

Below is a summary of the publicly available documents that are relevant to the general topic of "leak before break." The two relevant documents published by the NII (on the web site of its parent body, the Health and Safety Executive (HSE)) are the Safety Assessment Principles and a Technical Assessment Guide (TAG). The web links to the documents are below:

> Safety Assessment Principles 2006 edition, revision 1 (SAPs):
> http://www.hse.gov.uk/nuclear/saps/saps2006.pdf (about 2MB)
>
> Technical Assessment Guide (TAG) "Integrity of Metal Components and Structures" (html version): http://www.hse.gov.uk/foi/internalops/nsd/tech_asst_guides/tast016.htm (one of a number of TAGs listed at:
> http://www.hse.gov.uk/foi/internalops/nsd/tech_asst_guides/index.htm).

Note both the SAP document and the TAG are intended to cover all types of nuclear installations in the UK, not just nuclear power plants. So for instance, these documents are written to include nuclear chemical plant (e.g., fuel reprocessing plant).

The TAG on the web site is in html format and is not particularly easy to read. The Safety Assessment Principles are the 'top level' document; the TAGs are lower level documents. It is important to note that both the SAPs and the TAGs are advice to NII Inspectors in carrying out their regulatory decision-making regarding licensees' safety cases; they are not regulations with which licensees have to comply.

'Leak Before Break' does not make an appearance in the SAPs. Paragraph 252 on pages 42/43 of the SAPs gives a list of evidence topics that could form the basis of a structural integrity safety case. "Leak Before Break" does not appear in that list. By implication leakage might be part of in-service monitoring - item (j) in the list. Principle EMC.25 on page 47 of the SAPs deals with monitoring for leakage. Principle EMC.26 on page 47 deals with forewarning of failure.

The TAG has section 4.12 – "Leak detection and leak-before-break," on pages 24 and 25. The first paragraph of section 4.12 of the TAG reads:

> "Where high reliability in structural integrity needs to be claimed and justified, a "leak-before-break" argument may not be appropriate as the main thrust of the safety case argument. However, it depends on what is in the argument, rather than simply the label attached to it. For very high integrity (for instance where there is no 'line of protection' for the consequences of failure), a "No Break" argument or a "No Leaks or Break" argument might best summarise or label the sort of structural integrity safety case required. If some consequences are still protected, for example loss of fluid by providing emergency injection, but other consequences are not, for example pipe whip and jet forces, the inspector should expect the Licensee to present a clear justification for the apparent inconsistency."

The only PWR NII has licensed in the UK to date is Sizewell B. This entered commercial operation in 1995. Sizewell B was built over the period (roughly) 1987 to 1994. My understanding is the safety case for Sizewell B pipework in general is based on assuming full guillotine breaks (there are limited locations where an "Incredibility of Failure" argument is used and some locations where "No Break Zone" arguments are made, neither uses "leak-before-break"). I know of no application of a leak-before-break analysis being a principal element of a safety case considered for any plant licensed by NII.

APPENDIX G

Title page and Summary of JSME Code relative to LBB

JSME

日本機械学会

発電用原子力設備規格
配管破損防護設計規格

Codes for Nuclear Power Generation Facilities

－Rules on Protection Design against Postulated Pipe Rupture for Nuclear Power Plants－

JSME S ND1 – 2002

2002年12月

社団法人 日本機械学会

Summary of JSME LBB Criteria provided by K. Hasegawa

The JSME (The Japan Society of Mechanical Engineers) LBB Code (JSME S ND1-2002) was published in December 2002 as *Rules on Protection Design against Postulated pipe Rupture for Nuclear Power Plants*. The Rules are consistent with three articles; "General", "Design", and "Attachments".

Article "General" prescribes objective, application, review and nomenclatures. The designs are applied for austenitic stainless steel pipes, ferritic pipes and low alloy steel pipes, which constitute reactor coolant pressure boundaries, to protect pipe whipping, jet impingement.

Article "Design" consists of procedures of protection design and assessment of LBB. Pipe failure modes, failure locations, crack opening areas, jet impingement, applied loads, etc. are described in the procedures. Assessment of LBB provides applicable conditions, hypothetical cracks, leak rates, fatigue crack growth calculations, failure analyses, applied loads-service levels, and crack opening area for jet impingements. One of the applicable conditions shall have effective countermeasures performed for stress corrosion cracking for austenitic pipes and erosion/corrosion wall thinning for ferritic pipes.

Article "Attachment" provides more concrete calculations for assessment of jet impingement, leak rates, crack opening area, fatigue crack growth and failure analyses.

One of the examples of failure modes and crack opening areas for BWR austenitic stainless steel pipes in the JSME Code is tabulated in Table G-1. Flow chart of LBB procedures is shown in Figure G-1.

Table G-1 Failure mode and crack opening area for austenitic stainless steel pipes for BWR

Nominal pipe diameter, inch	1.5	2	4	6	8	10	12	14	16	20
Pipe diameter, mm	48.6	60	114.3	165.2	216.3	267.4	318.5	355.6	406.4	508.0
Pipe wall thickness, mm	5.1	5.5	8.6	11.0	12.7	15.1	17.4	19.0	21.4	26.2
Critical crack angle, $2\theta^o$.	-	-	109.2	94.4	81.2	71.6	63.6	59.2	54.0	46.0
Critical stress, P_f/S_m	-	-	1.34	1.60	1.83	2.01	2.16	2.24	2.34	2.49
$\boldsymbol{P_m = 0.5S_m, P_b = 0}$	B	B	L(27)	L(34)	L(37)	L(39)	L(40)	L(42)	L(43)	L(37)
$\boldsymbol{P_m = 0.5S_m, P_b = 0.5S_m}$	B	B	L(100)	L(110)	L(108)	L(109)	L(108)	L(109)	L(110)	L(112)
$\boldsymbol{P_m = 0.5S_m, P_b = 1.0S_m}$	B	B	B	L(345)	L(294)	L(273)	L(255)	L(251)	L(245)	L(239)
$\boldsymbol{P_m = 0.5S_m, P_b = 1.5S_m}$	B	B	B	B	B	L(694)	L(599)	L(569)	L(535)	L(494)
$\boldsymbol{P_m = 0.5S_m, P_b = 2.0S_m}$	B	B	B	B	B	B	B	B	B	B
Opening at critical stress	-	-	L(250)	L(439)	L(582)	L(707)	L(791)	L(845)	L(910)	L(999)

Note: "L" is leak and LBB design is applicable; "B" is break; Number of L( ) is crack opening area in units of mm^2.

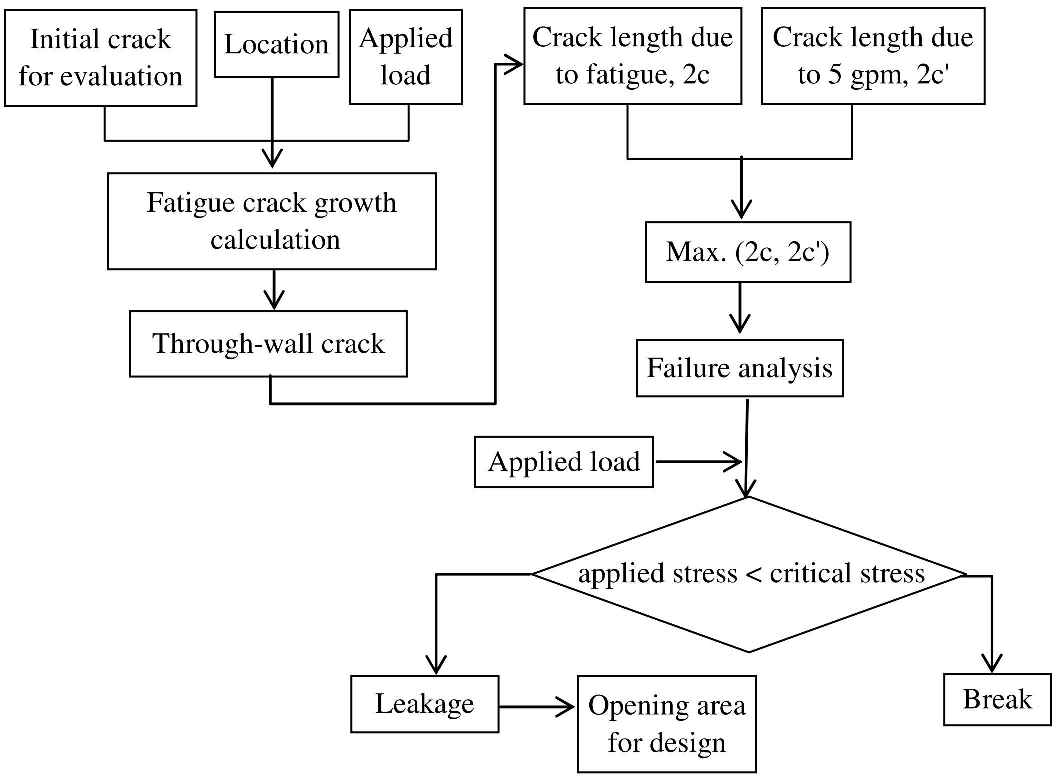

Figure G-1   Flow chart for LBB assessment provided by JSME Code

APPENDIX H

Older Swedish regulation on LBB - SKIFS 2004:2 (newer version essentially unchanged for LBB)

The Swedish Nuclear Power Inspectorate Regulatory Code

ISSN 1400-1187
Publisher: Ingvar Persson

SKIFS 2004:2

Date of printing
November 18, 2004

The Swedish Nuclear Power Inspectorate's Regulations concerning the Design and Construction of Nuclear Power Reactors

Decided on October 7, 2004.

On the basis of 20 and 21 §§ of the Ordinance (1984:14) on Nuclear Activities, the Swedish Nuclear Power Inspectorate has issued the following regulations and general recommendations.

Applicability and definitions

1 § These regulations apply to measures required to maintain and develop safety in the design and construction of nuclear power reactors with the aim of, as far as reasonably achievable, taking into account the best available technology, preventing nuclear accidents. The regulations comprise provisions on technical and administrative measures.

These regulations supplement, for application to nuclear power reactors, what has been said about design and construction as well as safety analysis in Chapters 2, 3 and 4 of the SKI's Regulations (SKIFS 2004:1) concerning Safety in Nuclear Facilities.

2 § In these regulations, a nuclear power reactor is the same as the definition provided in 2 § of the Act (1984:3) on Nuclear Activities.

In these regulations, barrier, defence in depth, nuclear accident and safety function are the same as the definitions provided in SKI's Regulations (2004:1) concerning Safety in Nuclear Facilities.

The following terms and definitions are also used in these regulations

Diversification: two or more alternative systems or components that independently of each other perform the same safety task, but in essentially different ways or through having different characteristics.

Single failure: a failure which means that a component cannot fulfil its intended safety task, as well as any consequential failure that arise.

Common cause failure: a failure which simultaneously occurs in two or more systems or components due to one specific event or cause.

Functional separation: systems or components that do not affect each other's function unintentionally.

Physical separation: systems or components that are physically separated, through distance or barriers or a combination of these.

Event class: classification of events conducted in connection with safety analysis and which reflects an expected probability of an event occurring and affecting reactor performance. The following event classes are used in these regulations:

Normal operation (H1)
Includes disturbances that are successfully managed by regular operations and control systems, without interrupted operation.

Anticipated events (H2)
Events that can be expected to occur during the lifetime of a nuclear power reactor.

Unanticipated events (H3)
Events that are not expected to occur during the lifetime of a nuclear power reactor, but which can be expected to occur if several reactors are taken into account.

Improbable events (H4)
Events that are not expected to occur. This also includes a number of overall events that are analyzed to verify reactor robustness, independently of the event frequency. These events are often called design basis events.

Highly improbable events (H5)
Events that cannot be expected to occur. If the event should nevertheless occur, it can result in major core damage. These events are the basis of the nuclear power reactor's mitigating systems for severe accidents.

Extremely improbable events (residual risks)
Events that are so improbable that they do not need to be taken into account as initiating events in connection with safety analysis.

Nuclear fuel bundle: nuclear fuel pins with accessories for load-bearing structures, as well as with such boxes that in boiling water reactors surround the fuel pins and load-bearing structure components[1].

Reactor core: part of the reactor where nuclear fission occurs and which includes the nuclear fuel bundles, control rods and neutron detectors.

Reactor primary system: comprises the reactor pressure vessel and other pressure-bearing devices which are a part of the reactor coolant system or which are connected to the coolant system including

- the external isolation valve in a pipe penetrating the containment wall,
- the reactor pressure relief and blowdown valves,
- the second of two, during operation, normally closed valves in pipes which do not penetrate the containment wall,
- the second of two automatically closing valves which do not penetrate the containment wall.

Redundancy: two or more alternative – identical or different – systems or components that independently of each other perform the same safety task.

Safety system: systems that have the function of ensuring reactor shutdown and residual heat removal, as well as systems that are needed to mitigate con-

1 The term "fuel assembly" is used synonymously with "nuclear fuel bundle" in connection with both boiling water reactors and pressurized water reactors. However, one difference is that pressurized water reactors do not use fuel boxes.

sequences of events, to and including the event category improbable events.

Design principles for the defence-in-depth

3 § The nuclear power reactor shall be designed so that the safety functions reactivity control, protection of the primary system integrity, emergency core cooling, residual heat removal and the containment function[2] can be maintained, to the extent needed depending on the operational state, in all events to and including the event class improbable events.

The design shall take into account events in the event class, highly improbable events in accordance with 4-9 as well as 18-20 §§.

4 § The following design principles shall be taken into account in the design of the reactor's defence-in-depth to the extent reasonably practicable

(a) Simplicity and durability in the design of the safety systems
(b) Redundancy, including diversification as well as physical and functional separation in the design of the safety functions
(c) Automatic control or passive function in necessary activation and operational changeovers of the safety functions
(d) Failure in safety classified equipment leads to an acceptable safety level
(e) Failure in operations classified equipment may not affect the performance of equipment with safety function
(f) When sharing of safety systems between reactors, a failure in one of the reactors shall not affect the possibility to perform shutdown and residual heat removal in the other reactors

Manual measures in connection with necessary activation and operational changeover of the safety functions may only be applied if the personnel is given sufficient time – time for consideration – in order to conduct the measures in a safe manner.

2 The containment function is for boiling water reactors the containment leaktightness function and pressure suppression function, for pressurized water reactors, it refers to the leaktightness function.

5 § The reactor containment shall be designed taking into account phenomena and loads that can occur in connection with events in the event class highly improbable events, to the extent needed in order to limit the release of radioactive substances to the environment.

6 § Instrumentation shall exist which makes it possible to monitor the parameters that are essential for handling of all events to and including the event class highly improbable events.

7 § It shall be possible to cool the reactor core through spraying or sufficent water cover, in all types and sizes of loss of coolant that can result from breaks in connections to the reactor pressure vessel.

8 § It shall be possible in all events, to and including the event class highly improbable events, to achieve a stable end state with a water-covered core/core melt and established residual heat removal. It shall be possible to cool a molten core in a long-term sequence.

Withstanding of failures and other internal and external events

9 § The safety functions in accordance with 3 § shall be able to withstand single failures in all events to and including the event class improbable events. In connection with events in the event class highly improbable events, the active components that belong to the mitigating systems shall be able to withstand a single failure.

10 § Reasonable technical and administrative measures shall be taken in order to counteract common cause failures, in connection with design, manufacturing, installation, startup, operation and maintenance of safety systems.

11 § In order to counteract simultaneous failure of redundant parts of safety systems, the nuclear power reactor shall be designed so that the redundant parts and their support functions have sufficient physical and functional separation.

The degree of separation shall be determined based on the consequences in the facility of the initiating events, which result in the need to take the safety system into operation.

12 § The nuclear power reactor shall be able to withstand global and local loads and other effects, which can occur in connection with a pipe break.

The consequences of a pipe break as initiating event shall be analyzed and assessed with respect to how such effects affect barriers and those safety functions that are credited in connection with the pipe break.

13 § Local dynamic effects do not need to be taken into account in those parts of the facility where the pipe systems have been given such a design, such operating conditions and environmental conditions that the conditions for damage to the piping, as a result of known and identifiable degradation mechanisms, have been reduced as far as possible and where measures have been taken so that damage which, in spite of this, can arise leads to detectable leakage before pipe break occurs.

Further regulations concerning the design, manufacturing and control of pipe systems are stipulated in SKI's Regulations (SKIFS 2000:2) concerning Mechanical Components in Certain Nuclear Facilities.

14 § The nuclear reactor shall be dimensioned to withstand natural phenomena and other events that arise outside or inside the facility and which can lead to a nuclear accident. In the case of such natural phenomena and events, dimensioning values shall be established. Natural phenomena and events with such rapid sequences that there is no time to implement protective measures when they occur, shall also be assigned to an event class. For each type of natural phenomenon that can lead to a nuclear accident, an established action plan shall exist for the situations where the dimensioning values run the risk of being exceeded.

15 § Equipment with readiness for operation requirements may be taken off line for planned maintenance during operation, if the nuclear power reactor is designed so that the safety systems concerned can withstand a single failure in connection with the measures, and the applied diversification and separation of the safety function concerned can be maintained.

16 § Equipment with readiness for operation requirements may be taken off line for repair and testing during operation, if the nuclear power reactor is designed so that the safety functions, in accordance with 3 § can withstand single failure in connection with the measures. Such repair and testing may be applied, even if a safety function does not withstand a single failure in connection with the measures, on condition that a safety analysis shows that the risk contribution that arises in such a way is very small.

Environmental durability and environmental impact[3]

17 § The barriers and equipment which belong to the safety systems of the nuclear power reactor, shall be designed so that they withstand the environmental conditions that the barriers and equipment can be subjected to, in the situations where their function is credited in the safety analysis of the reactor.

Equipment in the nuclear power reactor shall not make such an environmental impact that the performance of the safety functions of the reactor is reduced.

Provisions concerning control rooms

18 § It shall normally be possible to control and monitor the nuclear power reactor from the main control room during all operational states, and it shall be possible to take measures from the main control room to bring the reactor to a safe state, and to keep the reactor in this state, during all events to and including the event class improbable events.

19 § Events that can be a threat to continued activity in the main control room shall be identified and an established action plan shall exist for how these threats shall be handled with maintained reactor safety.

20 § In the case of events where the main control room is not available, an emergency control post shall exist with adequate instrumentation and maneuvering possibilities so that the reactor can be brought to hot shutdown, the residual heat removed and necessary safety parameters can be monitored. The emergency control post shall be physically and functionally separated from the main control room. Monitoring from the emergency control post shall be possible also in the event of a single failure in one of the systems that are necessary for the safe shutdown and cooling of the reactor.

3 Section 17 with general recommendations has been notified in accordance with the European Parliament's and European Council's Directive 98/34/EG.

When bringing the reactor to cold shutdown, other local maneuver posts besides the emergency control post may be used. However, it shall be possible to perform the supervision and monitoring of cold shutdown from the emergency control post.

Safety classification

21 § Structures, systems, components and devices of the nuclear power reactor shall be divided into safety classes. The detailed quality and functional reqluirements, resulting from this safety classification, shall be defined and controlled by specifying sub-classes, including mechanical quality class, electrical function class as well as classification with respect to seismics and environmental durability.

Further provisions concerning quality classification are stipulated in SKI's Regulations (SKIFS 2000:2) concerning Mechanical Components in Certain Nuclear Facilities.

Event classification

22 § In order to analyse safety, the initiating events included in the deterministic safety analysis, in accordance with Chapter 4. 1 § of SKI's Regulations (SKIFS 2004:1) concerning Safety in Nuclear Facilities, shall be divided into a limited number of event classes, with specified analysis assumptions and acceptance criteria.

These event classes shall cover normal operation, anticipated events, unanticipated events, improbable events and highly improbable events. When analysing events that have not been taken into account in the reactor design, realistic analysis assumptions and acceptance criteria may be applied.

Provisions concerning the reactor core

23 § The reactor core and connecting systems shall be designed so that
- design limits for the core can be met with adequate margins in all events to and including the event class anticipated events,
- power transients are not possible, or can reliably be detected and mitigated

without exceeding the design limits of the nuclear fuel bundles.

24 § The reactor core and connecting cooling systems shall be designed so that the net impact of the core's immediate reactivity feedback counteracts a reactivity increase during power operation.

25 § The reactor core and reactivity control systems shall be designed in such a way that the reactivity addition is limited in all events to and including the event class improbable events, in order to prevent

- the design limits for the nuclear fuel bundle coolability from being exceeded,
- the reactor pressure vessel internals from being damaged so that core coolability is degraded,
- the acceptance limits in the design specifications for the pressure-bearing parts of the reactor's primary system from being exceeded.

26 § An established limit shall exist for the highest power output from the fuel bundles during normal operation.

In connection with the highest power output in accordance with the first paragraph, it shall be possible to cool the core in the event of a loss of coolant accident. The limit for the highest power output shall be determined so that

- overheating and embrittlement of the fuel cladding and hydrogen production from the bundles are limited in the event of a loss of coolant accident,
- the core geometry is not changed in such a way in the event of a loss of coolant accident that cooling is prevented,
- the residual heat from the nuclear fuel bundle can be removed.

27 § For each fuel design and configuration of the core, established operating limits and parameters shall exist which shall be monitored and followed up during the operation of the core, to the extent needed for the provisions in 23-26 §§ to be met.

The analyses of the design and operating limits for the reactor core shall be reported in the safety report of the nuclear power reactor, in accordance with Chapter 4. 2 § of SKI's Regulations (SKIFS 2004:1) concerning Safety in Nuclear Facilities.

Exceptions

28 § The Swedish Nuclear Power Inspectorate may grant exceptions from these regulations if particular grounds exist and if this can be done without neglecting the purpose of the regulations.

Entry into force and transitional regulations

These regulations enter into force on January 1, 2005.

Without any impediment from the first paragraph, measures for complying with the provisions in accordance with 3-17 and 20 §§ shall be taken no later than on the deadlines established by the Swedish Nuclear Power Inspectorate for each nuclear reactor. The same applies to 18 § with respect to the introduction of additional monitoring equipment, as well as 23 § with respect to the introduction of equipment for detection and automatic protective measures against power transients.

On behalf of the Swedish Nuclear Power Inspectorate

JUDITH MELIN

Erik Jende

The Swedish Nuclear Power
Inspectorate's Regulatory Code

GENERAL RECOMMENDATIONS

The Swedish Nuclear Power Inspectorate's General Recommendations concerning the Application of the Regulations (SKIFS 2004:2) concerning the Design and Construction of Nuclear Power Reactors

General Recommendations:

Such general recommendations on the application of regulations which specify how someone can or should act in a certain respect.

[1 § Regulatory Code Ordinance (1976:725)]

The Swedish Nuclear Power Inspectorate's General Recommendations concerning the Application of the Regulations (SKIFS 2004:2) concerning the Design and Construction of Nuclear Power Reactors

Comments on Certain Sections

3 §
This requirement means that the reactor pressure vessel internals, which are also important for maintaining the core geometry, are designed to withstand the loads that can arise during events to and including the event class improbable events.

4 §
The equipment included in safety systems should be designed and positioned in such a way that the probability of deficiencies and malfunctions is low, and that safety is adequate even if deficiencies and malfunctions should arise in the equipment. In connection with failures such as loss of power or with external environmental impact, the equipment should assume a fail-safe position.

The provision [b] on reasonably practicable separation in the design of the safety functions means for instance that safety functions should be independent at an initial stage, in connection with all events to and including the event class anticipated events, namely the execution of the function should not be dependent on the execution of other functions. In this analysis, realistic analysis assumptions and acceptance criteria can be applied. One example of initiating independence in boiling water reactors is that it should be possible for the reactor to be made sub-critical without reliance on pressure relief, and it should be possible for pressure relief to occur without reliance on scram.

The provision [b] also means that equipment, with the main task of functioning in order to limit radioactive releases in connection with severe accidents, shall not be affected by a malfunction in other equipment in the facility.

The provision [c] on automatic control or passive function means, as a rule, that necessary activation and changeover of the safety functions shall be automatic. If this is not possible or reasonable, prepared manual measures can be accepted. No initiating events that require activation of the reactor protection system should, however, result in demands on rapid operator action. Information and time should always be granted to the operator so that he/she can understand the event sequence, the facility status and have time for thought, before the design requires manual action to be taken. Measures required within the first thirty minutes after the initiating event, in order to bring the reactor to a safe state, should be automated for all events, to and including the event class improbable events.

Reasonable time for consideration should exist for operator action also in connection with anticipated and postulated events resulting from the initiating events.

The following time for consideration should apply in the event of severe accidents[4]:

- Manual measures should not be needed for the first 8 hours.
- The manual measures that may be needed after 8 hours should be well prepared and controlled by procedures.

Other measures, which are not prepared, should not be needed until after 24 hours.

If an automatic safety function should not be activated when needed, it should be possible to manually activate the function in the main control room. If an automatic function were to jeopardize safety, possibilities outside the control room should exist to interrupt or block the automatic function. Such an extraordinary measure should be thoroughly analyzed and controlled by procedures.

5 §

The design basis for the reactor containment is events, to and including the event class improbable events, as shown in 3 §. To meet the requirement in 5 §, a safety evaluation should be performed of events and phenomena which may be of importance for containment integrity in highly improbable events. Examples of such events and phenomena, which can result in need to take measures, include high pressure melt-through of the reactor pressure vessel, steam explosion, re-criticality, hydrogen fire and containment underpressure.

4 Included in the event class highly improbable events.

8 §
The coolability of a molten core should be included in the safety evaluation mentioned in the general recommendation to 5 §.

9 §
A single failure should be postulated to occur in any component, at the most unfavourable time, in connection with the initiating event or thereafter. A single failure in passive components does not need to be assumed until 12 hours after the initiating event.

Certain components, such as check valves, as well as software and circuit card components have properties which should be subjected to safety assessment before they are considered to be active or passive components in individual cases. A check valve, which has to change position in order to fulfil its safety task, should primarily be considered to be an active component.

The requirement on the ability of consequence-mitigating systems to withstand a single failure can be considered to be fulfilled, if the ability to withstand a single failure exists for active components whose function may be needed within 8 hours after the initiating event, and for components which may be difficult to access for corrective measures when their function is demanded.

10 §
Technical measures are measures for diversification. A suitable and reasonable diversification should be applied to the design of the safety functions in accordance with 3 §, with realistic analysis assumptions and acceptance criteria for events to and including the event class unanticipated events, pipe breaks excluded. When designing such a diversification, all existing power supply to all plant systems can be credited.

The reactor protection system should, as far as reasonably practicable, be designed so that the need for protection is identified and so that protective measures are initiated through at least two different parameters, for example pressure and neutron flux, in connection with all events, to and including the event class unanticipated events. The various ways of detecting an event should be functionally separated.

12 §
Examples of global effects in connection with pipe break include pressure and temperature loads in the area where the pipe break occurs, as well as in

the adjacent areas to which pressure relief occurs, global vibrations due to condensation loads, loads due to flooding and steam release, including other environmental impact.

Examples of local dynamic effects are pipe whips, reaction forces and jets. The ability to withstand such events, especially in the case where a pipe break can result in the failure of an entire safety function, should be achieved through pipe whip restraints, missile shields or changes in pipe configurations.

When analyzing the measures that must be implemented, a pipe break should be assumed to occur where it is important for safety, as well as

- where there are basic conditions for such damage that can lead to pipe break, and
- in accordance with the criteria in SRP 3.6.1 and 3.6.2[5].

14 §

Examples of natural phenomena that should be taken into account are:

- extreme winds,
- extreme precipitation,
- extreme icing,
- extreme temperature,
- extreme sea waves,
- extreme seaweed growth or other biological conditions that can affect the cooling water intake,
- extreme water level,
- earthquake.

Examples of other events that should be taken into account are:

- fire,
- explosion,
- flooding,
- aeroplane crash,
- disturbances to or loss of the offsite grid.

5 US Nuclear Regulatory Commission Standard Review Plan: (SRP) 3.6.1 – Plant Design for Protection Aginst Postulated Piping Failures in Fluid Systems Outside Containment, NUREG 0800. SRP 3.6.2 – Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, NUREG 0800.

In these regulations, the events mentioned in the second paragraph are considered to be of the accident type and not intentionally initiated. Work is underway to issue regulations, on dimensioning and procedures to withstand terrorist attacks and sabotage, in special regulations concerning physical protection of nuclear facilities.

In the analysis of a fire in the facility, a fire that causes all equipment in a fire cell[6] to fail should be assumed to occur. If a fire hazards analysis can show that the probability of failure of an entire fire cell is low, through the fact that protective measures have been taken to prevent fire spreading, the burn out of the entire cell does not have to be assumed. Such a fire hazards analysis should encompass all measures necessary until the fire is extinguished. In the first instance, passive protective measures should be applied such as room dividers, encapsulation or shielding of equipment, minimized fire loads and distance separation between equipment.

If distance separation alone is counted as a protective measure between redundant equipment, this should apply to sufficiently large areas and on condition that the fire hazards analysis confirms that the separation is sufficient to prevent fire spreading.

Furthermore, fire should be taken into account in the following way when analyzing initiating events

- When analyzing fire as an initiating event, an additional fire does not have to be assumed in the facility.

- When analyzing other initiating events besides fire, which in turn can result in a fire, a fire should be assumed to occur as a possible consequential failure of the initiating event.

- When analyzing other events besides fire, which in turn cannot result in a fire, a fire should be assumed to occur no earlier than 12 hours after the initiating event. This event sequence does not have to be combined with a single failure. This applies to initiating events, to and including the event category unanticipated events, apart from pipe breaks.

6 Corresponds to "Fire Compartment" in accordance with IAEA Safety Guide NS-G-1.7: Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants. International Atomic Energy Agency. Vienna, 2004.

17 §

This requirement means that structures, systems, components and devices included in safety systems shall be environmentally qualified. Environments that can affect safety systems should be followed up as long as the systems are utilized for their purpose.

In environmental qualification of electrical equipment in safety systems, the principles for handling of ageing should be applied as specified in IEC 60780[7], Reg. Guide 1.89[8] or IEEE 323[9]. In connection with this, acceleration factors for thermal ageing exceeding 250 times, ionizing radiation lasting less than 10 days or a dose speed greater than 5 Gy/h should be avoided, or the applicability of the results should be justified.

In the case of fuel bundles and control rods, the requirement means that these should be able to withstand the irradiation and the environmental conditions in general, which can occur during all events, to and including the event class anticipated events.

Analyses of how equipment, from the environmental standpoint, can affect the reactor safety functions, should cover all events that are taken into account in the safety analysis of the reactor.

18 §

It should also be possible, from the main control room, to monitor the readiness of the safety functions to operate, namely that the equipment has assumed the correct position for operation. At events in the event category highly improbable events, it should be possible to perform an overall assessment of the facility's safety status.

The interface between the operator and the technical process of the facility should be designed so that the operator is given adequate, reliable and integrated information, which is sufficient to effectively monitor the reactor safety functions, make decisions within the available time, as well as receive feedback on

7 International Electrical Commission. Qualification of electrical equipment of the safety system for nuclear power plants.

8 US Nuclear Regulatory Commission Regulatory Guide. Environmental Qualification of certain Electric Equipment important to safety in Nuclear Power Plants.

9 The Institute of Electrical and Electronics Engineers Inc. Standard for qualifying class 1 E equipment for nuclear power generating stations.

automatic and manual measures. A suitable way of designing the annunciator presentation is pattern recognition.

The adequacy of the main control room and emergency control post should be evaluated and documented within the framework of the periodic safety review of the facility, in accordance with Chapter 4. 4 § of SKI's Regulations (SKIFS 2004:1) concerning Safety in Nuclear Facilities, as well as when operating experience shows that an evaluation is warranted.

An evaluation should comprise experience from the operation of the facility and similar facilities and simulator training, evaluations of the interfaces in relation to ergonomical requirements, as well as evaluations of how well the control room design supports the work of the operators. Local control rooms in the facility should be evaluated in connection with modifications, as well as when experience shows that an evaluation is warranted.

Ergonomical requirements and other conditions, that need to be taken into account in the interaction man-technology-organisation, should be specified at an early stage and taken into account in connection with such modifications to the main control room that relate to these conditions. Recurrent verification and validation of the new solutions should be conducted during the design process so that needed corrections can be made successively. Furthermore, verification and validation should be performed of the entire control room function, before modifications are introduced which essentially affect ergonomical or other conditions in the interaction between the operators and the technical process of the facility[10].

19 §

The threats against continued activity in the main control room, to which the regulations refer, are events like fire, steam release and flooding. A nuclear accident in another reactor at the same site should also be taken into account here. Requirements concerning procedures in connection with threats, such as armed intrusion and sabotage, will be stipulated in the special regulations on the physical protection of nuclear facilities, mentioned in the general recommendations to 14 §.

10 Examples of methodology for the evaluation of control room modifications are to be found in US Nuclear Regulatory Commission: Human Factors Engineering Program Review Model, NUREG 0711.

20 §

When designing the emergency control post, the events and conditions that result in the unavailability of the main control room should be taken into account. The personnel should be able to reach the emergency control post in a protected way. The interface should be designed to facilitate the transfer to working at the emergency control post.

Examples of other local maneuvering posts, besides the emergency control post, include relay rooms, switchgear rooms and local control rooms that do not include the emergency control and monitoring function.

21 §

The classification provides the basis of fulfilling the provisions of Chapter 3. 4 § of SKI's Regulations (SKIFS 2004:1) concerning Safety in Nuclear Facilities, through the design, manufacturing, installation and testing of structures, systems, components and devices with requirements that are adapted to their safety importance. The division into safety classes should be conducted in accordance with the principles provided in ANSI/ANS-51.1 for pressurized water reactors and ANSI/ANS-52.1 for boiling water reactors[11].

22 §

The selection of the initiating events to be included in each event class should be based on an analyzed probability with which the event is expected to occur. However, certain initiating events should be included as postulates, in order to verify the robustness of the facility, independent of the probability of these events occurring. An example of such an event is loss of coolant at a break of the largest pipe or connection to the reactor pressure vessel.

23 §

In the design of the core, the impact of changes in coolant temperature, coolant flow, reactor power and reactor pressure should be taken into account. In the case of pressurized water reactors, changes in the boron concentration of the coolant should also be taken into account.

In addition to design measures, boiling water reactors should have procedu-

11 ANS-51.1: American National Standard: Nuclear Safety Criteria for the Design of Stationary Pressurised Water Reactor Plants. American Nuclear Society, 1983. ANS-52.1: American National Standard: Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants. American Nuclear Society, 1983.

res for measures which need to be taken in the event of core instability. The procedures should state what characterizes instability, how it is detected and how it is mitigated. Concerned personnel should be well acquainted with the procedures and should be trained in handling instability.

The stability margins should be calculated for new core loadings.

25 §
In order to ensure nuclear fuel bundle cooling, the design limits stipulate that the nuclear fuel is not fragmented in connection with a reactivity accident. The reactivity value of the control rods should be limited, so that the energy accumulation in the fuel bundles will not become too high.

26 §
When analyzing the limit for the highest power output, the acceptance limits specified in 10 CFR 50.46[12] should be used.

27 §
In addition to limits for the highest power output, limitations should exist which provide margins for fuel bundle overheating and limits for conditions that can lead to stress corrosion cracking of fuel bundles. For pressurized water reactors, there should also be limits for asymmetrical power generation in the core.

In analysis of the limitations that provide a margin for overheating of the nuclear fuel bundles, acceptance criteria in accordance with SRP 4.4[13] should be used.

Further guidance for handling of nuclear fuel bundles at different stages and situations during operation and core configuration modifications, as well as analysis, monitoring, followup and documentation, is provided in the IAEA safety standard: Core Management and Fuel Handling for Nuclear Power Plants[14].

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12 Section 50.46 – Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors. US Code of Federal Regulation. Energy Parts 0 to 50.

13 US Nuclear Regulatory Commission Standard Review Plan (SRP) 4.4 – Thermal and Hydraulic Design, NUREG 0800.

14 Safety Guide NS-G-2.5: Core Management and Fuel Handling for Nuclear Power Plants. International Atomic Energy Agency, 2002.

APPENDIX I

Mr. Brian Jarman of CNSC 1985 paper
"The Canadian Approach to Protection Against Postulated Primary Heat Transport Piping Failures"

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THE CANADIAN APPROACH TO PROTECTION AGAINST POSTULATED PRIMARY HEAT TRANSPORT PIPING FAILURES

by

B.L. Jarman

Atomic Energy Control Board
Ottawa, Canada

October 28-29, 1985

THE CANADIAN APPROACH TO PROTECTION AGAINST POSTULATED PRIMARY HEAT TRANSPORT PIPING FAILURES

A paper presented at a seminar on Leak-Before-Break, sponsored by the U.S. Nuclear Regulatory Commission and the Battelle Memorial Institute, Columbus, Ohio, October 28-30, 1985.

SUMMARY

In Canada, the Atomic Energy Control Act and Regulations stipulate in broad terms the requirements to be met by licensees. In addition, AECB staff have prepared licensing guides to amplify those requirements. For nuclear reactors, these include providing suitable protection against the consequences of failure of any pipe in the reactor cooling system. The suggested means of limiting the damage caused by whipping pipes or steam jets is by separation of equipment, installing barriers, or restraining piping. If, however, the designer can demonstrate that restraints are impractical or detrimental to safety, AECB staff may consider alternate arguments based on a demonstration that piping is likely to crack and then leak for a long time prior to rupture. This alternative approach would not be considered for ruptures having a high probability of defeating containment, damaging essential safety systems, or of disrupting flow to the core to the extent that fuel cooling could not be maintained.

RÉSUMÉ

Au Canada, la Loi sur le contrôle de l'énergie atomique et les règlements stipulent, dans les grandes lignes, les exigences demandées des titulaires de permis. Le personnel de la CCEA a préparé aussi des guides de réglementation pour développer ces exigences. Dans le cas des réacteurs, on demande une protection adéquate contre les conséquences d'un bris de n'importe quel tuyau du système caloporteur. Des dommages peuvent être causés par des jets de vapeur ou par le fouettement de tuyaux brisés, et les moyens de minimiser ces dommages sont de séparer les équipements, d'installer des barrières, ou de restreindre les tuyaux. Si un requérant peut démontrer que l'installation des contraintes n'est pas rentable, ou même à l'encontre de la sécurité, le personnel de la CCEA peut quand même accepter des arguments tendant à démontrer la probabilité, pour la tuyauterie en question, de fissurer et subir une fuite avant sa rupture. Cette méthode ne s'applique pas aux ruptures susceptibles d'endommager l'enceinte ou les systèmes essentiels à la sécurité, ou de mener à une diminution sensible du refroidissement du combustible du réacteur.

The Canadian Approach to Protection Against Postulated Primary Heat Transport Piping Failures

In Canada, the Atomic Energy Control Act and the Regulations made thereunder stipulate in very broad terms the powers of the Atomic Energy Control Board (AECB) and the requirements to be fulfilled by licensees. Since the Board has chosen to issue only skeletal regulations, the specific regulatory requirements are applied through the licensing process. For example, a licence condition requires the licensee to consider any type of failure at any location of the reactor main coolant system; draft regulatory guides C-7, C-8, C-9 (refs 1 to 3) describe the requirements for limiting the consequences of the postulated rupture to containment, shut-down systems, and emergency core cooling systems.

Application of the Act, Regulations, and guides in the licensing process is best illustrated by reference to the Darlington Nuclear Generating Station which is being designed and built by Ontario Hydro (the licensee).

The construction licence for Darlington includes the following condition:

Except as otherwise approved by the Board, all piping and headers which form part of the primary heat transport system shall be restrained to the extent necessary to ensure that their failure could not cause consequential damage which would render invalid the analyses in the Safety Report or other submissions to the Board.

In addition, Ontario Hydro was advised that the draft licensing guides, now in trial use, would be applicable to Darlington. The relevant sections from these guides are:

- from C-7; Requirements for Containment System

  The containment system shall be designed such that, ... dynamic effects or jet forces caused by the event cannot result in impairment of the containment system to an extent that the relevant requirements in sections 2.2 (dose limits), 2.3 (structural integrity), and 2.4 (leakage) would not be met.

- from C-8; Requirements for Shutdown Systems

  Each shutdown system shall be designed such that, ... dynamic effects or jet forces caused by the event cannot result in impairment of the shutdown system to an extent that relevant requirements in section 2.2 (performance) would not be met.

- and from C-9; Requirements for Emergency Core Cooling

  The ECCS shall be designed such that, ... dynamic effects or jet forces caused by the event cannot result in impairment of the ECCS to an extent that relevant requirements in section 2.2 (cooling) would not be met.

During the evolution of the Darlington Station design, AECB staff discussed with the licensee the specific items in the guides. For example, the requirement that breaks be postulated at any location was perceived by the licensee as generating an infinite number of break locations. The AECB staff, however, considered that the consequences of rupture at certain 'critical' points could be shown to bound several others. Staff adopted this position because they did not believe that any alternate position based on the calculated stress and fatigue in piping could be supported by relevant pipe failure data. Jet forces, rates of flashing, impact dynamics, and damage scenarios were not specified but left to the designers to establish. Board specialists would then review the licensee's submissions for adequacy.

In order of desirability as the means of coping with the consequences of pipe failures, the licensee proposed: separation of equipment, installing barriers, or restraining piping. On Darlington, designers added the following specific features:

- some areas of containment, which could be subject to impact or jet loads, have been strengthened;
- boiler and piping layout has been separated into four quadrants to limit consequential damage to a local sector of the primary heat transport system;
- various structures, such as the pressurizer supports, have been strengthened to withstand the impact loads from pipe whip;
- concrete and steel barriers were introduced to protect shut-down systems from missiles and jets;
- some restraints notably on the headers and balance lines have been incorporated, to restrict reaction movements.

Despite these modifications, there remained 1100 break locations, on 250 mm lines and larger, which would require approximately 250 restraints costing several million dollars. Consideration of smaller lines would increase the number of restraints considerably.

The licensee asked Board staff to reconsider the licensing requirements for Darlington. The basis of the request was:

- the probability of piping failure is very low;
- restraints may increase that probability;
- restraints are difficult and costly to install;
- some restraints would impede or prohibit inspection;
- the piping will always leak for a considerable time before rupturing.

A review of the licensee's case led AECB staff to conclude that the licensee had not demonstrated that piping failures, including rupture, are sufficiently improbable that their consequences need not be considered. However, the staff were prepared to consider proposals to omit restraints in certain circumstances. Some of the considerations which led to this position were:

- a continuing awareness that catastrophic failure of nuclear standard piping is an improbable event. The U.S. Nuclear Regulatory Commission claims that a best-estimate probability of a double-ended guillotine break in a Pressurized Water Reactor (PWR) is 10^{-12} events per plant year. The failure rate for CANDU will be of a similar order of magnitude.

- the good performance of nuclear-grade piping worldwide. This despite there being corrosion- and welding-related problems in a large proportion of the world's reactors. None of these problems is evident in CANDU primary heat transport piping, mainly because the primary heat transport water chemistry is dedicated to controlling corrosion, whereas PWRs and BWRs also use the water chemistry for reactivity control.

- elimination of pipe-whip restraints would improve access to pipe welds for in-service inspection and thereby reduce occupational radiation exposure.

- the moderator and primary heat transport fluid are heavy water. Because this is very expensive, there is close control on and monitoring of leakage. Typically, station procedures require that leak rates above 50 kg/h (0.2 USGPM) be repaired within 24 hours or the plant shut down. This leak rate is an order of magnitude smaller than that recommended as an action level for PWRs and BWRs in the U.S.A.

- all the primary heat transport piping, feeders, headers, and pressure vessels are ferritic, typically 516 Gr70 and SA106 Gr. B. The in-service experience with these steels has been good; there have been no signs of in-service degradation, no problems associated with welding and, because they are ductile, they are tolerant of spurious loading.

It was not the intention of AECB staff that acceptance of a case to omit constraints be construed as acceptance that pipe breaks cannot occur and hence that containment or other safety systems need not be designed to withstand any of the consequences of pipe rupture. Consequently, the staff proposed that the consequences of pipe rupture at all locations in the main coolant loop should still be analyzed by the licensee.

It was also proposed that the plant design should be modified if necessary so that the consequences of failure are acceptable. The suggested means of achieving this continues to be, in order of desirability, separation of equipment, providing barriers, or restraining piping. If the licensee can show that these are impractical, an alternative safety analysis, to demonstrate that there is a long time between leaking and rupture (Leak-Before-Break), may be considered by AECB staff for certain break locations.

The degree to which restraints are considered impractical will be judged against the relative importance and likelihood of occurrence of the consequential damage. AECB staff is unlikely to accept alternative safety analysis for ruptures which have a high probability of breaching containment, or of damaging essential safety systems (Shutdown Systems one and two, and the Emergency Core Cooling System), or of disrupting flow to the core to the extent that the licensee cannot demonstrate to the AECB staff's satisfaction that fuel cooling can be maintained.

Staff of the AECB also proposed that alternative safety analysis should include:

- a description of the consequences of the break(s) for which protection is not provided;

- reasons why protection is impractical;

- test data to show that material ductility and fracture toughness are adequate to ensure leak-before-break behaviour. This should also include consideration of weld material;

- an In-Service-Inspection program for the section of piping for which restraints are not provided. This would replace Canadian Standard 285.4, which defines the areas of inspection using stress and fatigue criteria, with the volumetric requirements of ASME XI for class 1 piping plus wall-thickness measurements of the bends.

- a fracture analysis demonstrating that leak-before-break will occur for design transients causing the greatest stress at the point, and in the direction, of interest. The design transient should be selected from ASME level A (design) to level D (emergency) events. The fracture analysis should include consideration of residual stress, material degradation, 90° circumferential through-wall cracks, and axial cracks as long as four times the wall-thickness.

- a leak detection system capable of responding to the leak rate through a crack, as long as twice the wall-thickness, under ASME level A (design) loading. The crack should be postulated at the least detectable position on the piping for which restraints are not to be provided.

These proposals are applicable only to ferritic steels used in class 1 piping, and to class 1 welds which have been stress relieved. Piping or welds which include defects dispositioned prior to service are also excluded.

In conclusion Board staff have reviewed the practice worldwide for the protection of reactor equipment against the consequences of breaks in large diameter pipes. This led to an approach, for the licensing of Darlington, which while recognizing that large diameter pipe failures are infrequent, and are likely to leak-before-break, would maintain the requirements for the protection of essential safety systems. In practice this permits the number of piping restraints to be reduced while maintaining the principle of defence-in-depth.

References

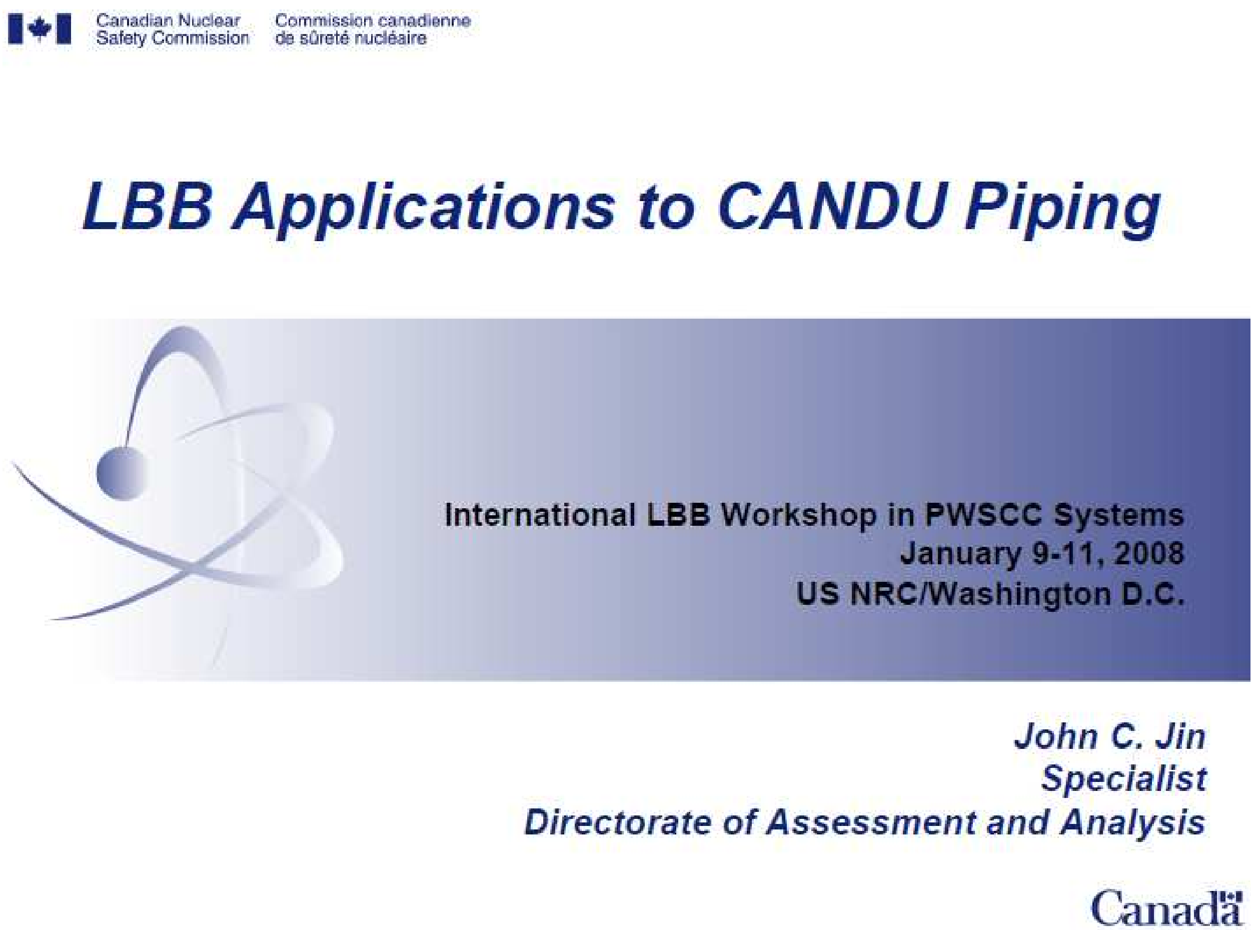
1. AECB Consultative Document C-7: Proposed Regulatory Guide - Requirements for Containment Systems for CANDU Nuclear Power Plants May 1982

2. AECB Consultative Document C-8/Rev. 1: Proposed Regulatory Guide - Requirements for Shutdown Systems for CANDU Nuclear Power Plants May 1982

3. AECB Consultative Document C-9/Rev. 1: Proposed Regulatory Guide - Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants May 1982

APPENDIX J

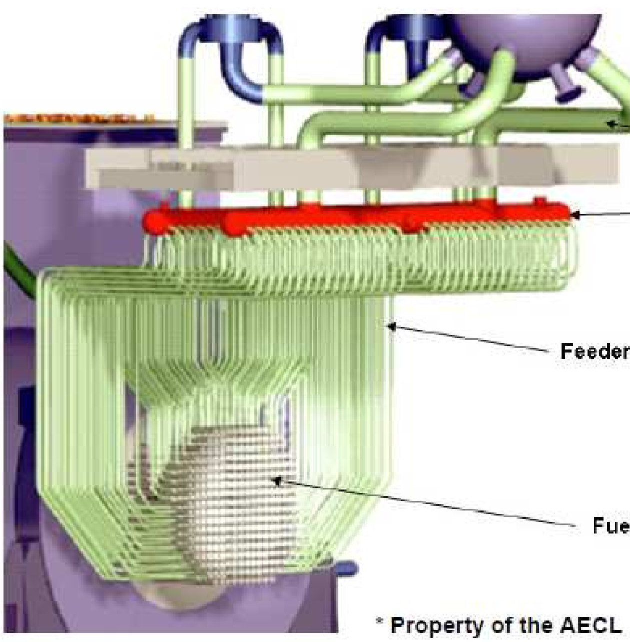
2008 Presentation by J. Jin on "LBB Applications to CANDU Piping"



LBB Applications

- To address dynamic effects of postulated pipe break (for new constructions or life extension)
- To support fitness for service assessment of degraded components (of operating plants)

CANDU Piping System (Schematic)



* Property of the AECL

I. LBB Application to address dynamic effects of postulated pipe break

- LBB application for new construction
  - In 1980's, the LBB was accepted for large diameter piping (larger than 20 inches) of PHT piping at Darlington NGS to eliminate the requirement for pipe whip restraints
  - SA-106 Gr.B Pipes and SA-105 fittings
  - Consistence with NUREG 1061, Vol.3
  - Enhanced in-service inspection program

➢ Possibility of active degradation mechanisms such as FAC in the LBB applied piping is a concern.

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- LBB Applications for Life Extension:
  - Requested to conform to modern safety standards and practices (Integrated Safety Review).
  - Proposed the application of LBB, in lieu of plant modifications, as a means of demonstrating the adequacy of piping systems of existing CANDU plants for postulated pipe rupture.
  - Condition assessment for assurance of no active degradation mechanisms, such as IGSCC and FAC, recently discovered in some CANDU piping.

- LBB Applications for New Plants
  - To smaller size piping up to 6 inches (ACR)
  - NUREG 1061, Vol.3, SRP 3.6.3

II. LBB Applications to support fitness for service assessment of degraded components

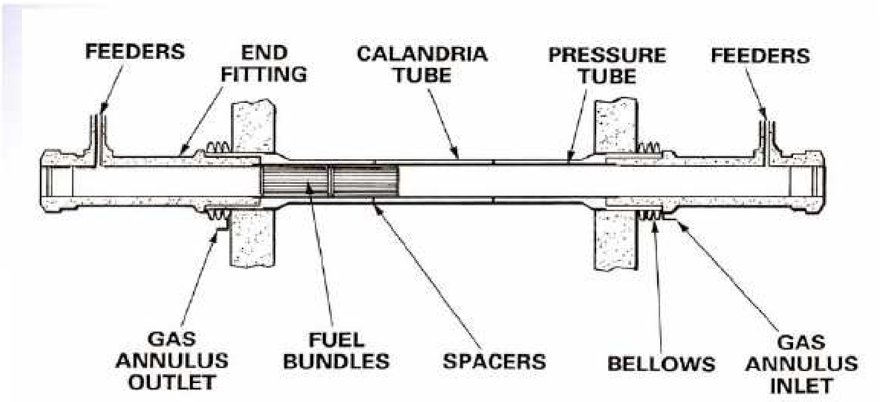
- The LBB assessment is applied for enhancing the defence-in-depth of continued service of components with potential flaws for a specified operating period by demonstrating pipe rupture is not a likely event.
  - Pressure tubes
  - Feeder pipes
  - S/G tubes

Same philosophy, different application

- LBB to address postulated pipe break
  - Design life
  - Postulated crack (screening criteria)
  - Margins on crack sizes & leak rate
- LBB to support fitness for continued service
  - Operating period
  - Actual crack characteristics
  - Margins on the time from leaking to instability
  - Reliable CGR, leak rate, leak detection system and operating procedure are crucial elements

LBB application to Pressure Tubes

- Zr-2.5Nb, 4mm thk, 103mm ID,
- Internal Pressure = 10 MPa
- Temp. = 250 °C to 300 °C



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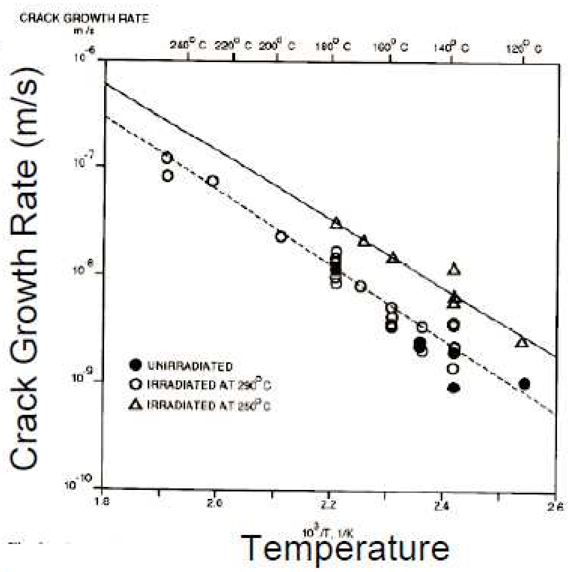
- **Susceptible to Delayed Hydride Cracking (DHC)**
  - **Tensile residual stress produced during rolling process**
  - **Hydrides**
- **LBB applied as a defence-in-depth**
  - **it is required to demonstrate the DHC is not likely**
  - **even if the flaw is propagated by DHC, unstable rupture should be avoided by LBB**

*Methodology**

- Demonstrating that the time available to detect a leak and take actions is much greater than the time required to detect the crack
- LBB Parameters
  - Max. crack length at penetration (Lp)
  - DHC Velocity (V)
  - K_{DHC}
  - Critical Crack Length (CCL)
  - Leak rate
  - Leak detection by AGS

* Ref. [1] G. D. Moan, et. al, 1990, Leak Before Break in the Pressure Tube of CANDU Reactors", Int. J. Pressure Vessels & Piping, 43, 1-21

Crack Growth Rate



* Fig. 5 of Ref.[1]

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Aspect Ratio

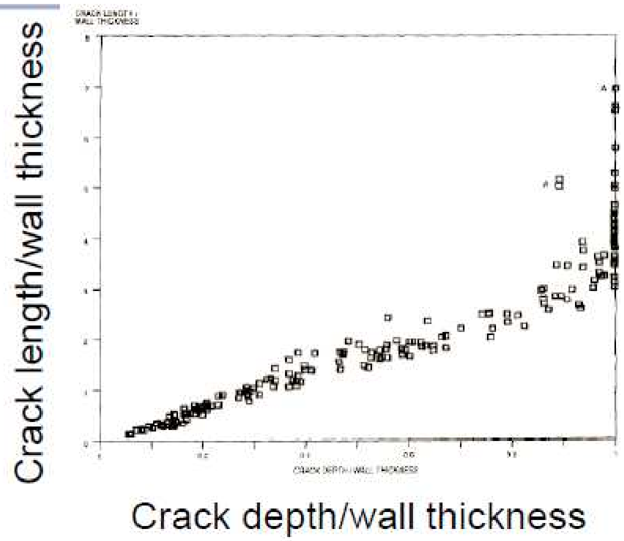

Fig. 7 of Ref.[1]

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The time available to detect the leak and take action, Ta

Ta = (CCL-Lp)/2V

= 18 hours for

CCL = 50mm

V = 2.7×10^{-7} m/s

Lp = 4W

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The time required to detect the leak

❑ Leak rate from actual cracks >> 1 kg/h

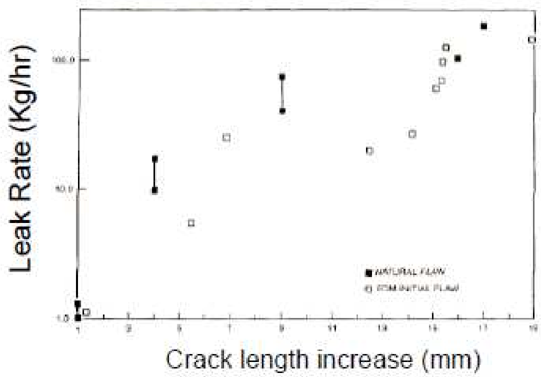

Fig. 12 of Ref.[1]

Effect of crack length increase on the water leak rate measured from cracks in pressure tubes in laboratory tests after tubes were removed from the reactor

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The time required to detect the leak (con't)

❑ Leak detection by the AGS

- **Rate of change of dewpoint alarm**
  - **less than 1 hour for the addition of D_2O at a rate of 10 g/hr**
- **Beetle alarm for the presence of liquid**
  - **less than 1 hour for the addition of D_2O at 2.3 Kg/hr**

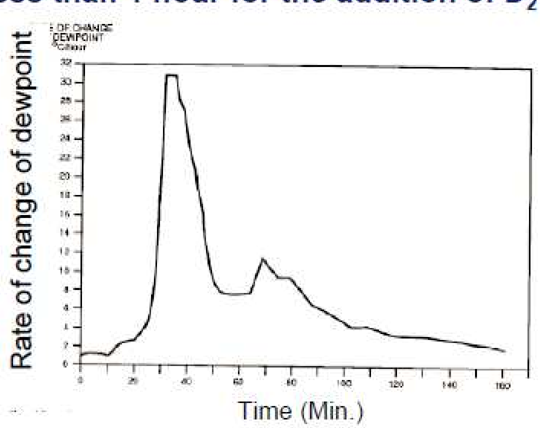

Fig. 15 of Ref.[1]

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CCL to Crack growth

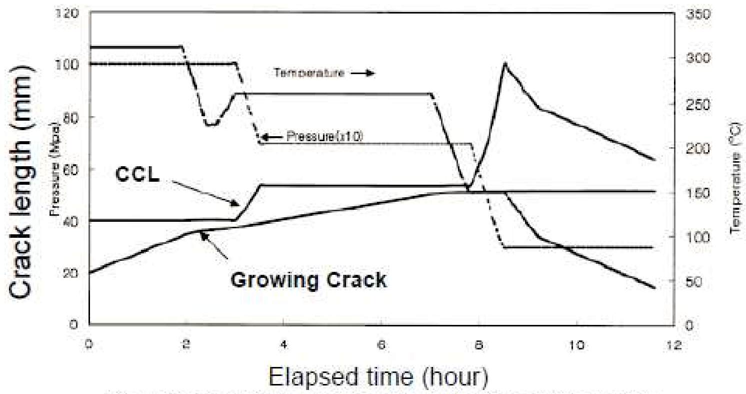


Fig. 9. Results of LBB assessment based on the modified operating procedure.

* Ref. [2] Y. W. Park and Y. K .Chung, 1999, Leak-Before-Break Assessment of CANDU Pressure Tube Considering Leak Detection Capability", Nuclear Engineering and Design, 191, 205-216

Issues

- ❑ Possibility of longer crack
  - due to through wall thermal gradients preventing crack growth in the radial direction while axial crack growth continues
- ❑ Degradation of margins
  - deterioration in fracture toughness due to hydrogen/deuterium pickup
- ❑ Stability of part-through wall crack

Feeder pipes

- 380~480 inlets/outlets
- 1.5 ~ 3.5" NPS 80
- Carbon Steel, SA-106, Gr.B
- Bend or Fitting
- Pressure: 1,475 - 1,550 Psig
  - Temperature: 515 - 585 °F

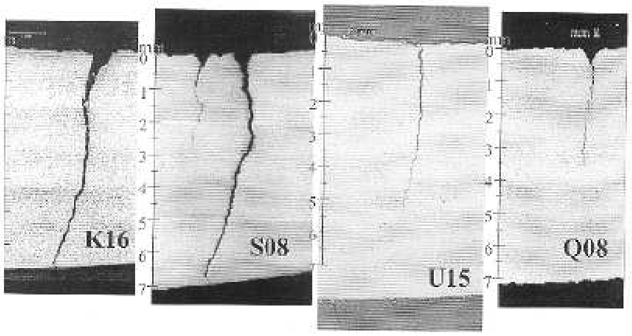
❑ Degradation Mechanisms
- IGSCC/LTCC
- FAC Wall Thinning

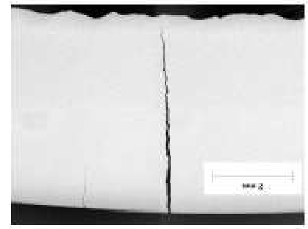
Cracking history

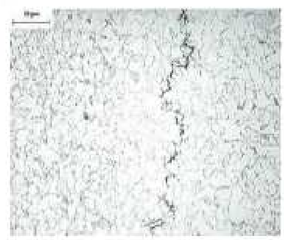
❑ Bend Cracking
- 13 confirmed cases of feeder cracking in bends at the PLGS
- Of these, 2 cracks went through-wall resulting in leaks in 1996 and 2001
  - Reactor was shutdown before the cracks reached critical sizes.

❑ Weld Cracking
- 1 leaking crack at a feeder repaired field weld at Gentilly-2 station

Metallographic Images of Feeder Cracks







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Cracking mechanism

- ❑ **IG stress corrosion cracking caused by exposure to mildly oxidizing hot coolant (Inside surface cracking)**
- ❑ **Low temperature creep cracking exacerbated by hydrogen (inside and outside surface cracking)**
- ❑ **One of Possible Causes: *Residual stress produced during bending process***

Leak rate history

- Bend cracking
  - 20~37 days from when the crack first penetrate the all until the leak rate increases to 12kg/hr
  - 5~20 days for the leak rate increase from 2kg/hr to 20kg/hr
- Repaired weld cracking
  - < 3kg/hr
- Shutdown leakage limit: 20kg/hr

Bases of continued operation

- Prevention of leakage no matter how small the leak may be
- Flaws to be detected by ISI and detected flaws to be repaired/replaced
- Full scope inspection upon discovery of cracks
- Assessment of risk increment (del CDF) associated with feeder degradation

➢ Issue: Capability of ISI

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LBB application

- Licensee claimed the applicability of the LBB, at least as a contributor to defense in depth, for the demonstration of the fitness for service of feeder pipes
- Issues:
  - Long surface crack
  - Uncertain crack growth rate
  - Axial crack in carbon steel pipe bends
  - Tight crack morphology
  - Material issues (cyclic load effect, dynamic strain ageing, toughness degradation)
  - Sensitivity and diversity of leak detection system
  - Plant operating procedures based on LBB assessment

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Regulatory Position *(under consideration)*

- Detected service induced cracks shall be repaired.
- Inspection scope shall be expanded upon discovery of the cracks.
- Limitations in ISI capability and mechanistic understanding of the cracking shall be addressed:
  - It shall be demonstrated that a pipe break is an unlikely event even if undetected crack could grow through wall during the operating period :
    - Actual crack characteristics
    - Margins on the time from leaking to instability
    - Reliable CGR, leak rate, leak detection system and operating procedure are crucial elements

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APPENDIX K

LBB in guide Finnish YVL 3.5

JOINT CSNI/CNRA WORKSHOP ON
"REDEFINING THE LARGE BREAK LOCA:
Technical Basis and Its Implications".
June 23-24, 2003 – Zurich, Switzerland

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.

<u>Provision against Pipe Breaks (para. 2.2.2)</u>

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 license 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 categorize 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:

1. *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.*

2. *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.*

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

APPENDIX L

Paper by G. Roussel at the Lyon Specialist Meeting on LBB in 1995 on Belgium LBB Efforts (NUREG/CP-0155 published 1997)

ADDITIONAL REQUIREMENTS FOR LEAK-BEFORE-BREAK APPLICATION TO PRIMARY COOLANT PIPING IN BELGIUM

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ABSTRACT

Leak-Before-Break (LBB) technology has not been applied in the first design of the seven Pressurized Water Reactors the Belgian utility is currently operating. The design basis of these plants required to consider the dynamic effects associated with the ruptures to be postulated in the high energy piping. The application of the LBB technology to the existing plants has been recently approved by the Belgian Safety Authorities but with a limitation to the primary coolant loop.

LBB analysis has been initiated for the Doel 3 and Tihange 2 plants to allow the withdrawal of some of the reactor coolant pump snubbers at both plants and not reinstall some of the restraints after steam generator replacement at Doel 3. LBB analysis was also found beneficial to demonstrate the acceptability of the primary components and piping to the new conditions resulting from power uprating and stretch-out operation. LBB analysis has been subsequently performed on the primary coolant loop of the Tihange 1 plant and is currently being performed for the Doel 4 plant.

Application of the LBB to the primary coolant loop is based in Belgium on the U.S. Nuclear Regulatory Commission requirements. However the Belgian Safety Authorities required some additional analyses and put some restrictions on the benefits of the LBB analysis to maintain the global safety of the plant at a sufficient level.

This paper develops the main steps of the safety evaluation performed by the Belgian Safety Authorities for accepting the application of the LBB technology to existing plants and summarises the requirements asked for in addition to the U.S. Nuclear Regulatory Commission rules.

INTRODUCTION

Under the amendment to GDC-4 (Ref.[1]), the U.S. Nuclear Regulatory Commission (NRC) allows the use of an LBB analysis to exclude from the design basis the "dynamic effects" associated with postulated pipe ruptures of primary coolant loop piping in Pressurized Water Reactors.

Before authorizing the Belgian utility to apply the LBB technology to existing plants, the Belgian Safety Authorities reviewed the benefits of the LBB analysis as set forth by the U.S. NRC rules. Their review was made with reference to the defence-in-depth principles and led to define the conditions and limitations under which the LBB technology was allowed to be used for the reactor coolant circuit of existing plants.

POSTULATION OF THE LOCA

LOCA as a Design Basis Accident

The third level in the defence-in-depth concept is achieved by providing the plant with additional systems (Engineered Safety Features -ESF - systems) - as well as with the part of the Reactor Protection System necessary to initiate these systems - in order to limit the consequences of extremely unlikely accidents to an acceptable level for the public. In addition to the ESF systems, the reactor core and internals in conjunction with the reactor coolant system will be designed to ensure sufficient core reactivity control and core cooling during these events.

The postulation of the Loss-Of-Coolant-Accident (LOCA) originates so from the technical safety objective which requires to consider in the design of the plant those accidents of low probability. A design basis accident is then defined for each range of relating possible initiating events which could challenge the safety of the plant. The design basis accidents include the Loss-Of-Coolant-Accident. A deterministic analysis is performed to predict the course of the event and all its reallistically conceivable consequences. The analysis shall define the design parameters of the ESF sytems which are necessary to halt the progress of the LOCA and, when necessary, to mitigate its consequences.

Safety Design Principle

Most aspects of safety design are connected with the three functions that protect against the release and dispersal of radioactive material :(i) controlling core power /core shutdown, (ii) core cooling, and (iii) confinement of released radioactive fission products.
For the purpose of designing a nuclear power plant to cope with the postulated design basis accidents, design requirements are set forth to ensure that these functions are not impaired by the LOCA.

Core shutdown
In order that the boron delivered by the Emergency Core Cooling System (ECCS) together with the control rods provide sufficient negative reactivity for safe shutdown after LOCA, the reactor core and internals shall be designed so that their geometry is maintained after LOCA to allow the control rods to fall in the reactor core and the borated water to be delivered to the core.

Core cooling
- The ECCS shall be designed to ensure adequate core cooling in the event of a LOCA.
- The reactor vessel internals shall be designed to ensure the capability of the core to be cooled after the occurrence of a LOCA.
- The primary loop supports shall be designed to prevent large distorsion of the piping during a LOCA in order to ensure that water from the ECCS enters the reactor vessel.
- The containment structures and containment systems shall be designed to absorbe the energy released in the containment after a LOCA.

Confinement of released radioactive fission products
- Containment shall be designed to contain radioactive material leakage or releases from equipment located within the containment after the occurrence of a LOCA.
- The steam generator tube bundle shall be designed to ensure its integrity after the LOCA and so to avoid containment bypass and escape of radioactive fission products directly to the environment.

Two specific principles are also included for safety reasons, (i) equipment qualification and (ii) non-increase of the severity of the accident.

Equipment qualification
Mechanical and electrical equipment that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal shall be qualified to the environmental conditions that would prevail if they were required to function after a LOCA.

Non-increase of the severity of the accident
Design provisions shall be made at the design stage to maintain the LOCA accident within the design basis. Design provisions shall therefore be taken in order that :
- a Reactor Coolant System (RCS) pipe break is limited to the leg in which the break started
- an RCS pipe break does not cause a steam or feedwater line break
- propagation of a "small" break to a "large" break is prevented
- an RCS pipe break does not cause a steam generator tube rupture.

MECHANICAL ANALYSIS TO LOCA (PRIOR TO RULE CHANGE)

According to the Appendix A to 10 CFR Part 50, "Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest of the pipe of the reactor coolant system".

The containment design, ECCS performances and qualification of mechanical and electrical equipment are based on a complete spectrum of breaks type and size.

From the 1970s, the analysis of the reactor coolant circuit (piping, components and supports) and the reactor vessel internals is based on the postulation of specific pipe breaks in the primary circuit. In a generic analysis performed by Westinghouse (Ref.[2]), the postulated locations and types of pipe breaks are derived from the results of a stress and fatigue analysis. Eleven pipe breaks are postulated in each loop, ten of which are double-ended guillotine breaks and one of which is a longitudinal break. The loads resulting from a LOCA depend on the size of the break area and on the opening time of the break. The full cross sectional flow area of a circumferential break is not considered in the analysis if the presence of restraints limits the displacement of piping and components and so allows to justify a lower value of the break area. In the conventional break assumptions based on conservative estimates of equipment/piping displacements, the circumferential breaks on the primary piping -with the exception of the break postulated at the reactor coolant pump (RCP) outlet nozzle - have an opening area of less than 144 square inches. At the RCP outlet nozzle, a guillotine break of a double-ended pipe cross sectional flow area (2 × 4.125 square feet) is assumed. The circumferential breaks postulated at the connections of the auxiliary lines (Pressurizer Surge line, Safety Injection line, Residual Heat Removal line) with the primary piping have double-ended pipe cross sectional flow area. The opening area of the longitudinal break postulated on the intrados in the elbow of the steam generator inlet is equal to one time the flow area (5.241 square feet). The conventional opening time of the pipe breaks is assumed to be 1 msec.

RULE CHANGE TO GDC-4

The final "limited" scope rule published on 11April 1986 (Ref.[1]) amends GDC-4 by permitting the use of LBB analyses to eliminate from the design basis the dynamic effects associated to postulated pipe ruptures of primary coolant piping of PWRs. On 27 October 1987, a final "broad scope" rule (Ref.[3]) amends GDC-4 to permit the use of LBB analyses in all high energy piping.
Limiting the LBB analysis to the primary coolant piping leads to postulate breaks only at the branch connection of the auxiliary lines (Pressurizer Surge line, Safety Injection and Residual Heat Removal) with the reactor coolant loops.
Analysis of the U.S. NRC documents (GDC-4, Statement of Consideration, SRP 3.6.3 in Ref.[4]) and examination of the available documents (i.e., Safety Analysis Reports of U.S. plants and U.S. NRC Safety Evaluation Reports) lead to the plausible interpretation that the application of LBB allows to not consider :
(i) the loading of the primary component supports due to the pipe break reactions
(ii) the subcooled blowdown loading of the reactor vessel internals
(iii) the subcooled blowdown loading of the steam generator internals (divider plate and tube bundle)

(iv) the asymmetric pressurization of the reactor cavity
(v) the effects resulting from pipe whipping, jet impingement and missiles.

The U.S. NRC rules clearly exclude the containment design, the ECCS performances and the qualification of the mechanical and electrical equipment from the benefits of the LBB analysis.
The consequences of the LBB analysis on the protection of the unbroken loops against the effects from the broken leg (by the physical separation with concrete structures and by the decoupling of the mechanical effects at the reactor vessel) are not clearly stated in the U.S. NRC documents.

CONSEQUENCES OF THE MODIFIED GDC-4

Inconsistency in the Mitigation Measures

Before the GDC-4 was amended, the design bases for the reactor coolant circuit (piping, heavy components and their internals, supports), the containment systems, and the ECCS and the requirements for qualification of the mechanical and electrical equipment were coherent.
The modified GDC-4 introduces an inconsistency in the mitigation measures to face a LOCA. Firstly, it does not seem logical not to consider a double-ended guillotine break for designing the reactor internals and core whereas this break is assumed in the design basis of the ECCS. Secondly, the question can be raised why the ECCS should be designed for a double-ended guillotine break if the mechanical effects impair the core assembly geometry to such an extent that control rods cannot be dropped and the core cannot be adequately cooled or cause such large distorsion of the primary piping that the ECCS water cannot enter the reactor vessel.
The U.S. NRC acknowledged this inconsistency and clarified its position by introducing the distinction between the local and the global effects (Ref.[5]). However this clarification does not address the consequences on core reactivity control and core cooling of the large distorsions of the reactor core and internals or primary piping.

Non Increase of the Accident Severity

The safety requirement for non increasing the severity of the accident does not seem to have been considered.

Protection Against Non-Identified Events

For each plant condition a limited number of events is defined. These were analyzed to ensure that they enveloppe other (non identified) related possible initiating events belonging to the same plant condition. By eliminating from the design basis the dynamic effects associated with the postulated LOCA, the protection against some effects of the related possible initiating events could have been lost.

The consequences of the elimination of the protection against the dynamic effects of the LOCA on the protection against the related possible initiating events do not seem therefore to have been taken into account.

BELGIAN SAFETY AUTHORITIES POSITION

Applying Modified GDC-4 vs Retaining Safety Margins

The concept of defence-in-depth relies first on preventing the event and then mitigating the consequences. There is so far no reason to change this concept.

The consequences of the amendment to the GDC-4 on the measures mitigating the design basis LOCA should be analyzed. The modified GDC-4 does not change the design bases for the containment systems and the ECCS nor the requirements for qualification of the mechanical and electrical equipment. The elimination of the dynamic effects from the design basis of the reactor coolant circuit have potential consequences which cannot be accepted as such. The elimination of the mechanical effects associated to the postulated primary pipe breaks could result in unacceptable consequences in terms of core shutdown, core cooling and non-increase of the accident severity. Indeed, the modified GDC-4 results in decreasing the structural capacity of the

- primary component supports
- reactor cavity
- reactor core and internals
- steam generator tube bundle

and it furthermore does not consider the pipe whip nor jet impingement effects.

In a situation where an LBB analysis is only performed on the main primary piping but not on the auxiliary lines, the design basis circuit includes pipe breaks up to about 100 square inches. If the application of the LBB is extended to all auxiliary lines, as permitted by the modified "broad scope" GDC-4, the consideration of the dynamic effects of any pipe break shall be excluded from the design basis of the reactor coolant circuit. This would lead to an unacceptable safety loss in terms of core shutdown, core cooling and non-increase of the accident severity.

(Limited) Reevaluation of the Present Situation

Nevertheless a reevaluation of the conventional situation, i.e. before the amendment to the GDC-4, is deemed necessary and this leads to some suggestions for adjusting the mitigating measures. The key points of this reevaluation are :

(i) The protection against the pipe whip and impingement effects is somewhat theoritical. Pipe breaks occurring at locations different from the postulated locations cannot be excluded. Moreover experiments have shown that severance schemes different from the schemes postulated (circumferential or longitudinal break) can also be expected. The actual restraints are not demonstrated to ensure protection against breaks different from the postulated breaks with respect to location or severance scheme.

(ii) The assumed conventional opening time of 1 msec is very penalizing for the calculation of the blowdown loads and is also believed to be unnecessarily conservative. The use of more realistic opening times should lead to lower loads.

Acceptable Modifications to the Design Bases

The suggestions for adjusting the mitigating measures are based on :
(i) the acknowledgment that by removing the restraints and some of the snubbers, some of the construction features installed to ensure the non-increase of the severity of the event and the core cooling are eliminated,
(ii) the requirement that all the remaining features to mitigate the consequences of the LOCA shall be maintained because LOCA sources other than the primary pipe breaks and the related possible initiating events envelopped by the design basis LOCA are still to be considered.

The suggestions for adjusting the mitigating measures are :
(i) The LBB analysis can be considered as an acceptable method for removing the restraints. However some precaution against pipe whip and jet impingement effects resulting from primary pipe breaks remains required.
(ii) The LBB analysis can be considered as an acceptable method for not designing the heavy component supports (steel and concrete structure) to the postulated LOCA reaction loads. This may result in elimination or decrease in load rating of existing snubbers. However the ability of the component supports to avoid excessive distorsion of the reactor coolant piping under the dynamic loadings of the LOCA related possible events shall be maintained.
For plants initially designed for conventional LOCA breaks, the reactor cavity concrete structures and the steel supports of the heavy components are believed to have sufficient margin to accomodate any dynamic loadings during LOCA related possible initiating events.
(iii) The design basis of the reactor core and internals and of the steam generator tube bundle shall include the rapid rupture (opening time of 1 msec) of the steam generator manway covers (hot leg and cold leg) and a slow break (opening time of 3 sec) of one time the flow area anywhere in the primary coolant piping.
These breaks are postulated because they are physically acceptable and coherent with the design bases of the other ESF systems. They are also believed to induce hydrodynamic loads which cover the loads resulting from the full spectrum of the conceivable and realistic sources of LOCA in the reactor coolant pressure boundary other than the double-ended pipe rupture and to enveloppe the dynamic loads resulting from the LOCA related possible initiating events. They shall therefore be considered as design basis breaks for the reactor core and internals and the steam generator tube bundle.
(iv) The existing physical separation shall be maintained.

CONCLUSION : KEEPING THE GLOBAL SAFETY OF THE PLANT

The global safety of the plant should not be decreased. It is not believed that the removal of the restraints and of some of the snubbers after the LBB analysis of the primary piping reduces significantly the global safety of the plant. Some safety decrease at the third level of the defence-in-depth could be expected in the protection against the non-increase of the severity of the event or in the core cooling capability. However, as mentioned hereabove, the protection against the pipe whip and jet impingement effects by means of the actual restraints is somewhat theoretical and the structural capacity of the concrete and steel supports of the heavy components is not affected as long as their original design basis is maintained. A potential safety increase of the defence-in-depth can be expected from the removal of some of the snubbers and also to a certain extent (and with caution) by the LBB analysis itself.

Potential safety increase of the safety at the first level might be achieved by reinforcing the in-service inspection of the primary piping and at the second level by installing an improved system to detect or locate leaks from the primary circuit. Such requirements were however not imposed to the Belgian utility.

REFERENCES

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[2] WCAP-8082-P-A Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop, January 1975.

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[5] Leak-Before-Break Technology : Solicitation of Public Comment on Additional Applications *Federal Register*, Vol.53, No66, published by the Office of the Federal Register, Washington, DC, 6 April 1988, pp.11311-2.