IAEA-TECDOC-710

Applicability of the leak before break concept

Report of the IAEA Extrabudgetary Programme on the Safety of WWER-440 Model 230 Nuclear Power Plants

Status report on a generic safety issue

INTERNATIONAL ATOMIC ENERGY AGENCY IAEA

The IAEA does not normally maintain stocks of reports in this series. However, microfiche copies of these reports can be obtained from

INIS Clearinghouse
International Atomic Energy Agency
Wagramerstrasse 5
P.O. Box 100
A-1400 Vienna, Austria

Orders should be accompanied by prepayment of Austrian Schillings 100,– in the form of a cheque or in the form of IAEA microfiche service coupons which may be ordered separately from the INIS Clearinghouse.

APPLICABILITY OF THE LEAK BEFORE BREAK CONCEPT
IAEA, VIENNA, 1993
IAEA-TECDOC-710
ISSN 1011-4289

Printed by the IAEA in Austria
June 1993

FOREWORD

Within the framework of the IAEA Extrabudgetary Programme on the Safety of WWER-440 Model 230 NPPs, a list of safety issues requiring broad studies of general interest have been agreed upon by an Advisory Group which met in Vienna in September 1990. The list was later revised in the light of the programme findings.

The information on the status of the issues, and on the amount of work already completed and under way in various countries, needs to be compiled. Moreover, an evaluation of what further work is required to resolve each of the issues is also necessary. In view of this, the IAEA has started the preparation of a series of status reports on the various issues.

The main objective of the reports is to provide a clear overview of each issue and of the work which is still needed before final conclusions can be reached. The reports should provide the basis for defining the scope of the studies still required to resolve each generic safety issue. They will also compile information on the work already performed or under way in various countries, to prevent duplication of efforts.

This report on the generic safety issue "Applicability of the Leak Before Break Concept" presents a comprehensive survey of technical information available in the field and identifies those aspects which require further investigation.

The preparation of the report was co-ordinated by the IAEA Secretariat and it was later circulated internationally for review. Valuable comments have been received from A. Alonzo (E.T.S.I.I., Spain), S.K. Bhandari (Framatome, France), J. Darlaston (Nuclear Electric, United Kingdom), A.F. Getman (VNIIAES, Russian Federation), S. Kawakami (NUPEC, Japan), M. Kozluk (Ontario Hydro, Canada), P. Milella (ENEA/DISP, Italy), H. Schulz (GRS, Germany), J. Strosnider (OECD/NEA), G. Wilkowski (Battelle Columbus, USA), K. Wichman (United States Nuclear Regulatory Commission, USA) and M. Erve (Siemens KWU, Germany). These were very helpful for the preparation of the final version of the document and are greatly appreciated. J.Zdarek of the Nuclear Research Institute, Řež, Czechoslovakia was responsible for drafting the report. J. Höhn of the Division of Nuclear Safety was the IAEA staff member responsible for co-ordinating the draft and review of the document.

PLEASE BE AWARE THAT
ALL OF THE MISSING PAGES IN THIS DOCUMENT
WERE ORIGINALLY BLANK

EDITORIAL NOTE

In preparing this material for the press, staff of the International Atomic Energy Agency have mounted and paginated the original manuscript and given some attention to presentation.

The views expressed do not necessarily reflect those of the governments of the Member States or organizations under whose auspices the manuscripts were produced.

The use in this book of particular designations of countries or territories does not imply any judgement by the publisher, the IAEA, as to the legal status of such countries or territories, of their authorities and institutions or of the delimitation of their boundaries.

The mention of specific companies or of their products or brand names does not imply any endorsement or recommendation on the part of the IAEA.

Throughout the text names of Member States are retained as they were when the text was compiled.

CONTENTS

SUMMARY AND CONCLUSIONS

Summary

The leak before break (LBB) concept is widely used in the nuclear industry to describe the idea that in the piping carrying the coolant of a power reactor a leak will develop before a catastrophic break will occurred.

The LBB concept could be accepted as a technically justifiable approach for eliminating postulated double ended guillotine breaks (DEGBs) in high energy piping systems with respect to resulting dynamic effects. This conclusion has resulted from extensive research, development, and rigorous evaluations by the United States Nuclear Regulatory Commission (USNRC) and the German Reaktorsicherheitskommission (RSK) and the commercial nuclear power industry and its organizations since the early 1970s. Efforts in other European countries (France and the United Kingdom) have not resulted to date in similar regulatory actions.

The LBB concept is based on analysis which demonstrates by deterministic fracture mechanics that a crack would grow through the wall, resulting in a leak, and that this postulated small 'through wall' flaw in plant specific piping would be detected by the plant's leakage monitoring systems long before the flaw could grow to unstable size. Leakage exceeding the limit specified requires operator action or plant shutdown. Necessary prerequisites are the high toughness and resistance to unstable crack growth of austenitic and ferritic steels used in nuclear high energy piping and the exclusion of possible degradation.

Historically, the hypothetical DEGB has been taken as the most severe reactor loss of coolant accident (LOCA) in nuclear power plant (NPP) design. The original purpose and intent of the postulated DEGB was to provide a limiting basis for the emergency core cooling and containment systems. However, the postulate was extended to the design of the high energy piping system, resulting in the construction of massive pipe whip restraints and jet impingement shields, because no alternative acceptable design measure was available. The DEGB postulate was further extended to the design for environmental qualification of safety equipment. For many years the commercial nuclear industry has recognized that a DEGB is highly unlikely even under severe accident conditions, and that a LOCA based on DEGB is too restrictive a design requirement.

In the USA a new document, Modification of General Design Criterion 4, released by the USNRC, which defines the LBB case and assigns legislative consequences to the LBB status, represented a break from this attitude. To provide review guidance for the implementation of the LBB concept, Standard Review Plan 3.6.3 was issued.

There are differences in the legislative status of the LBB concept in Member States. In the USA the LBB concept and Section XI of the American Society of Mechanical Engineers (ASME) Code are sufficient to relieve the facilities from having to design for the dynamic effects of a DEGB.

In Germany the break preclusion concept was defined, based on basic safety and independent redundancies related to LBB behaviour. This concept was introduced into the RSK Guidelines, resulting in the use of leaks instead of breaks in the design of plants (e.g. jet and reaction forces, pressure waves).

In other Member States the LBB concept is generally viewed as having possible applications. However, permission is not granted for the exclusion of DEGBs for dynamic effect calculation. Recently, changes in this position could be observed in Spain and Czechoslovakia.

Application of the LBB concept

In the USA, approximately two thirds of all pressurized water reactors (PWRs) have approval for the application of the LBB concept in the primary coolant loop. There are also four PWRs which have such approval for their auxiliary lines; specifically, pressurizer surge, accumulator, and residual heat removal lines. One of the four plants also has approval for the safety injection lines and the reactor coolant loop bypass lines. The approved auxiliary lines are all inside the containment, fabricated from austenitic stainless steel, and at least 150 mm in diameter. The application of LBB concept has not yet been approved for any boiling water reactor (BWR).

In Germany break preclusion was implemented for all PWRs for the main coolant line (MCL) and for nearly all PWRs and BWRs for other main lines (e.g. feedwater line, main steam line, surge line).

The LBB concept cannot be applied to piping where severe degradation in service could occur, such as by water hammer, creep, erosion, corrosion and excessive fatigue. The rationale is that these degradation mechanisms challenge the assumption in the LBB acceptance criteria and their effects are difficult to quantify. Water hammer may introduce excessive dynamic loads which are not accounted for in the LBB analyses. Corrosion and fatigue crack growth may introduce flaws whose geometry may not lead to a leak and whose fracture not be bounded by the postulated 'through wall' flaw in the LBB analyses.

The procedure has two steps:

Screening:

To demonstrate that the candidate piping is not susceptible to failure from these degradation mechanisms, the operating history and measures to prevent or mitigate this degradation must be reviewed.

Fracture mechanics:

After the LBB candidate piping has been reviewed for degradation mechanisms and found to be acceptable, it is subjected to a rigorous fracture mechanics evaluation. The purpose of this evaluation is to show that there is flaw stability and the resulting leakage will be detected in the unlikely event that a flaw should develop.

The current acceptance criteria established by the USNRC are expected to remain the same. These are as follows:

(1) The loading conditions should include the static forces and moments (pressure, deadweight and thermal expansion) due to normal operation, and the forces and moments associated with the safe shutdown earthquake (SSE). Critical sections are determined on the basis of coincidence of the highest resulting stresses, and the poorest material properties, for base materials, weldments and 'safe ends'.

(2) A 'through wall' flaw should be postulated at the locations determined from criterion (1). The flaw should be large enough that any leakage from this postulated flaw is sure to be detected. When the pipe is subjected to normal operational loads, it should be demonstrated that there is a factor of at least 10 between the leakage from the so-called leakage size flaw and the plant's installed leak detection capability. A leakage analysis which has been benchmarked against experimental or plant data is required. The margin on leakage is required to account for uncertainties such as the crack opening area, crack surface roughness, two-phase flow and leak detection capability.

(3) It should be demonstrated that the postulated flaw is stable under normal and SSE loads. It should be demonstrated that there is a factor of at least 1.4 between the loads that will cause flaw instability under the normal and SSE loads. The factor of 1.4 stems from the factor of the square root of two on stress intensity for flaw evaluation under Service Level C and D Loadings in IWB-3610 of Section XI of the ASME Code. Service Level Loadings are defined in NCA-2140 of Section III of the ASME Code. However, the broad scope rule permits a reduction of the factor of 1.4 to 1.0 if the individual normal and seismic (pressure, deadweight, thermal expansion, SSE, and seismic anchor motion) loads are summed absolutely. This is because the USNRC considers that an absolute load summation sufficiently conservative to warrant a margin reduction.

(4) The flaw size margin should be determined by comparing the leakage size flaw with the critical size flaw. Under normal and SSE loads, it should be demonstrated that there is a factor of at least two between the critical size flaw and the leakage size flaw. The factor of two stems from the factor of two on flaw size for flaw evaluation under Service Level C and D Loadings in IWB-3610 of Section XI of the ASME Code.

To summarize, the first step in the application of the LBB concept is to screen the candidate systems for the susceptibility to known degradation mechanisms. Next, for systems that potentially qualify for LBB, data on the fracture and material properties, piping loads, weld procedures and locations, and leak detection capabilities are gathered. At those locations in each system which have the least favourable combinations of material properties and stress, a crack is postulated. Calculations, including fluid mechanics, are performed to determine leak rate values and fracture mechanics analysis is used to demonstrate corresponding size flaw stability.

Clearly, the fact that leak detection methods will be the only hardware to be added to operating plant is important for the application of the LBB concept. No single leak detection method currently available combines optimal detection sensitivity, location ability and measurement accuracy. Although quantitative leakage determination is possible with condensate flow monitors, sump monitors and primary coolant inventory balance, these methods are not adequate for locating leaks if this would be of benefit (as for WWER-440 reactors) and are not necessarily sensitive enough to meet given requirements.

The technology is available to improve leak detection capability at specified sites by use of acoustic emission monitors or moisture sensitive tape. However, current acoustic emission techniques still have difficulties with source discrimination (e.g., distinction between leaks from pipe cracks and valves) and leak rate information (a small leak may saturate the system). Moisture sensitive tape provides neither quantitative information on leak rate nor specific information on the location other than the location of the tape.

In the USA, Regulatory Guide 1.45 (issued 1981), the only available code, recommends the use of at least three different methods to detect system leaks. Monitoring of both sump flow and radioactivity of airborne particulate is mandatory. A third method can involve either monitoring of condensate flow rate from air coolers or monitoring radioactivity of airborne gases. Although current methods used for leak detection are up to date, other techniques may be developed and used. Recommendations on threshold sensitivity and installation of indicators and alarms in the main control room are also given.

WWER-440 model 230 candidate system

WWER-440 model 230 primary circuit and the emergency core cooling system (ECCS) was not designed to take into account a DEGB and the confinement does not meet corresponding requirements. Plants were intended to be constructed exclusively on low seismicity sites in accordance with former Soviet regulations. The safety concept of these reactors was based on accident prevention through design, high quality fabrication of the primary circuit and control, with only a limited scope design basis accidents (DBAs).

The primary circuit with inner diameter of 500 mm consists of six loops. Each loop has two main isolation valves. Compared to PWR designs, the piping length is greater, with several bends and elbows. Piping was manufactured using austenitic stainless steel type 0Ch18N10T and 0Ch18N12T respectively (corresponding to ANSI A321). All piping components were forged except the cast main isolation valve and the body of the reactor coolant pump. The elbows were longitudinally welded from forged halves, their wall thickness often exceeding significantly the design requirement. Antiseismic measures, pipe whip protection or jet impingements shields are not included in the original design.

There are 78 circumferential welds which have been made on the site. These, together with longitudinal elbow welds, are a source of great concern. Dissimilar welds between the piping and reactor pressure vessel (RPV) and pressurizer nozzles are also of major concern.

There are three pressurizer surge lines made of austenitic stainless steel type 0Ch18N10T; two of these are connected to the bottom side of the pressurizer, the third to the top. All piping components were manufactured in the former USSR and there is very little information on wall thickness deviation, ovality, etc. In the calculations performed, the maximum code allowable values are used. Data on the assessment of these lines are not available from the former USSR. A comprehensive database has been developed within the Czechoslovak LBB programme. An attempt is being made to prepare a database of defects left in on-site made welds and defects found by in-service inspection (ISI).

Sections of the pressurizer surge lines could be affected by thermal stratification. It is desirable to use applicable data from other WWER plants (Loviisa, etc.) in any LBB assessment.

There is also a dissimilar weld between the pressurizer nozzle made of 22K ferritic steel and 'safe end' of the austenitic steel which should be included in the LBB analysis.

There are heavy components supports on the RPV, steam generator (SG), MCL and main isolation valve. The RPV support is also a major concern with regard to the seismic loading stability.

Results available

In order to determine the most critical areas within the primary circuit, it is necessary to consider the combination of material and stress state. The aforementioned welds and their positions within the circuit provide the material aspect. The stress state aspect could be identified when stress analysis is performed for normal operating conditions (NOCs) and for the safe shutdown earthquake (SSE) (corresponding to site specific seismic activity).

The fatigue damage assessment demonstrates that the most critical locations are the primary piping to surge line nozzles and elbow to pipe connections and RPV 'safe ends'.

From analysis performed for Greifswald and Bohunice NPPs, it is possible to conclude that data on NOC assessment, including fatigue damage, are now available. All the piping components have been assessed analytically and in detail using three dimensional finite element analysis. The stress state in all NOCs can be described for all components with a high degree of confidence. The assessment, including NOCs, seismic loading, of occurrence water hammer, etc., was not carried out until recently. Specific data from the former USSR are not available but there has been some work carried out.

For the assessment of defects including fatigue crack growth, it is necessary to have details of non-destructive examination (NDE) and defects classification. Owing to technical difficulties and the lack of advanced techniques for NDE, there are very few volumetric data available. Most of the data derive from inspections of the outside surfaces. Only recently, modified inspection techniques were introduced for inspecting internal surfaces of critical sections. However, no systematic database is available for the inside surface and 100% volumetric control. The database of all defects left in circumferential welds made on-site (all of them within the allowable limit) is being developed for the Bohunice plant and used for the LBB assessment as well.

In 1989 an extensive theoretical and experimental programme was prepared by the former All-Union Scientific Research Institute for Nuclear Power Plant Operation (VNIIAES) in co-operation with other institutions in the former USSR. The programme consisted of 19 tasks aimed at providing data to assess the probability of a DEGB in the primary piping of WWER-440 reactors. According to the schedule, most of the results should have been available by the end of 1992.

Data for the ageing assessment could be also derived from the '100 000 hours programme', as well as from the '150 000 hours programme'. Within these programmes, parts of piping components (elbow and straight pipe with circumferential welds) had been cut out from the main circulating pipe line at Kola NPP after 100 000 hours of service. A similar programme has been carried out at Greifswald NPP which includes not only pipe material but also selected parts of other piping components. Unfortunately, comprehensive results of these programmes are not available.

Siemens Kraftwerk Union has performed NOC analysis of the main primary loop and surge lines for the Bohunice and Greifswald NPPs. Results available suggest that LBB analysis is likely to be successful providing all necessary data will be available. The pressure vessel nozzle to 'safe end' dissimilar weld and the longitudinal to circumferential weld area of the elbows seem likely to be the most critical locations.

The application of the LBB concept for the WWER-440 model 230 reactors also has strong relevance to other reactor types. It will form the basis for the application of the LBB concept to WWER-440 model 213 plants. For the majority of these plants, reassessment of antiseismic measures and PSA studies will be needed and, in both cases, the plants concerned will benefit if the LBB concept is analysed.

Conclusions

On the basis of the detailed review of the present status of the application of the LBB concept to operating WWER-440 model 230 plants, the following could be concluded:

- The status of the application of the LBB concept provides input for safety assessment of nuclear plants. For older plants, where it is necessary to reassess safety margins, LBB can provide the basis for the safety philosophy as well as means for a backfit assessment with significant economic benefits and a reduction of collective exposure.

- Only the USNRC and the German RSK have developed a legislative procedure for application of the LBB concept taking into account dynamic effects. Other countries use the USNRC procedure as a reference approach. Information available suggests that the LBB concept could be applied to the WWER-440 model 230 plants. However, additional data and work are needed to demonstrate safety upgrading requirements.

- The fatigue damage, corrosion-erosion damage and occurrence of water hammer have to be shown to be not significant. The stability of all heavy components supports needs to be reanalysed in detail as their performance under accident loading conditions such as seismic loading is crucial. The need to perform detailed seismic analysis arises because the basic design applies for non-seismic sites/conditions and most of the plants have installed or are developing antiseismic measures. The proposed or installed measures have to be reanalysed. Additional safety margins could be demonstrated as well as significant economical benefits provided.

- Pressure and thermal stresses, static expansion stresses, dead weight, etc., are well established and, in principle, do meet the design and additional (e.g. LBB) analysis criteria.

- Without sophisticated NDE methods, it is difficult to perform reliable and comprehensive inspection of coarse grained austenitic stainless steel. Therefore, fatigue crack growth analysis is very important; apart from detected defects also hypothetical ones also have to be considered. The multiple life assessment as applied in Germany is used with great advantage and is the basis for the LBB analysis necessary for the break preclusion concept.

- A reliable defect parameter database have to be developed, at least for the most critical sections, on the basis of 100% volumetric control.

- The leakage crack size and crack stability could hopefully be determined with a high level of confidence. However, special attention has to be paid to dissimilar welds (reactor and pressurizer vessel nozzles), where validation of data on small size specimen fracture mechanics has to be performed in full scale experiments.

- Successful application of the LBB concept depends strongly on the capabilities of the leak detection diagnostic system. Three independent leak detection methods have to be used in conjunction with a reliable expert system to decide on the priority of detected leak signal. A leak detection system providing reliable quantitative data in terms of leak rate and crack location is still needed.

- Installation of such a leak detection system in accordance with results of LBB analysis (critical sections, leakage crack size, calculated leakage rate) as well as its careful calibration is required.

Appropriate operating procedures need to be developed and implemented in operator training programmes.

1. PRESENT STATUS

1.1. BACKGROUND

The major components in PWRs can be ranked on the basis of their importance to preventing the release of fission products and their relevance to overall plant safety. On this basis, the components can be ranked as follows: RPV, containment, reactor coolant piping and 'safe ends', steam generators, reactor coolant pump body, pressurizer, control rod drive mechanisms, RPV internals and RPV supports and biological shields [1].

Historically, the hypothetical DEGB has been taken as the most severe reactor LOCA in NPP design. The original purpose and intent of the postulated DEGB was to provide a clearly limiting basis for sizing the reactor containment system. However, the postulate was extended to the design of the high energy piping system, resulting in the construction of massive pipe whip restraints and jet impingement shields, simply because no alternate acceptable design basis was available. The DEGB postulate was further extended to the design for environmental qualification and even in the sizing of the ECCS. For many years the commercial nuclear industry has recognized that a DEGB is highly unlikely, even under severe accident loads, and that a design basis LOCA based on DEGB is unnecessary and undesirable design restriction.

The LBB methodology could be accepted as a technically justifiable approach for eliminating postulated DEGB in high energy piping systems. This conclusion has resulted from extensive research, development and rigorous evaluations by the USNRC and the German RSK and the commercial nuclear power industry and its organizations since the early 1970s. Efforts in other European countries (France and the United Kingdom) have not resulted in regulatory action, as is the case in the USA.

Applications of the LBB concept gathered momentum in the latter part of the 1980s. The benefits of LBB are multifold with maximum economic benefits derived if the concept is applied right from the early design stages. Owing to lack of new construction orders for NPPs in recent years, the LBB concept has been generally applied in a backfit mode rather than right from the preliminary design stage. In Germany, however, for all advanced Convoi NPPs the break preclusion concept, including LBB was introduced during the design phase of the plants and was successfully achieved during their construction.

Application of the LBB concept has saved hundreds of millions of dollars in backfit costs on many operating plants. Application of this concept to plants under construction has resulted in cost savings of tens of millions of dollars owing to the elimination of the need for pipe whip restraints and jet shields. Added cost savings result from reduced collective exposure during in-service inspection and maintenance [3].

The RPV and primary coolant pipework of PWRs are usually manufactured from tough fracture resistant steel. For this material the load corresponding to rupture of the component would be significantly influenced by cracks, which could occur in the component wall. This has been demonstrated in numerous studies, e.g. in an extensive experimental programme carried out in Germany [4].

The results suggest that there is more probably an extensive leakage for a component rather than a net section break (in the case of primary piping a so-called DEGB) and that there would be safe shutdown of the power plant if leakage is reliably monitored. However, for safety reasons, and for lack of a reliable treatment of the above argument, the DEGB has been considered as a possible accident in the guidance documents of the Regulatory Commissions of countries with nuclear power programme. Pertinent safety arrangements to deal with a DEGB accident were therefore obligatory for the utilities.

A breakthrough for this attitude resulted in a new document, Modification of General Design Criterion 4 (GDC-4): Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures [5], released by the USNRC, which uniquely defines the leak before break case and assigns legislative consequences to the LBB status. For the utilities in the USA, which can prove the LBB case according to Ref. [5], extensive measures to handle the dynamic consequences of the DEGB accident are no longer obligatory. This had sweeping practical implications for the layout of the primary piping (for example pipe whip restraints and jet shields are no longer needed), and economic savings were significant.

The issue of the document [5] in the USA was initiated by a meeting, at which nine countries with nuclear industries commented on and criticized the approach taken in the USA. To our knowledge, no country has yet issued a statement contrary to the document. The evaluation of the LBB case still contains some shortcomings and debatable issues in the view of the European and Japanese participants. There are principally two different attitudes adopted by countries other than the USA.

In some countries (e.g. in the United Kingdom), the application of the LBB concept is still to be submitted to development and validation before strong legislative conclusions can be drawn up. Before that, LBB can be seen and used as an independent safety case argument. In other cases (e.g. for Germany [6]), the basic safety concept is preferred, which also includes such phenomena as leak and loss of integrity, with, however, a different logic and different legislative consequences.

In Germany, RSK issued Guidelines in 1979 [44], dealing with requirements for protection against dynamic effects. These Guidelines were commented on in detail in [45], describing the LBB analysis.

In the WWER-440 model 230 reactors, the main primary piping is made from 500 mm diameter titanium stabilized austenitic steel. There are six loops and the length of one leg is over 15 metres. A horizontal steam generator on every loop allows relatively free thermal expansion. The supports of the main isolation valves and pumps also allow free thermal expansion. The system is therefore flexible rather than rigid (Figs 1(a) and 1(b)). There are three surge lines from the pressurizer to the main piping loop No. 1. The surge line is made also from austenitic steel.

The number of loops and their length means that a large number of welds must be on-site (78 for main circulating pipes), together with several elbows bends and T-joints. The elbows are fabricated with longitudinal welds.

The choice of high quality austenitic steel for this piping has advantages from the point of view of high ductility, crack growth resistance and crack stability, very good corrosion resistance and ageing characteristics. On the other hand this creates problems with dissimilar welds with the RPV nozzles as well as with the pressurizer and steam generator nozzles.

The design of first generation type WWER-440 model 230 did not consider the DEGB principle. The containment and the ECCS did not compensate for a DEGB and there were no

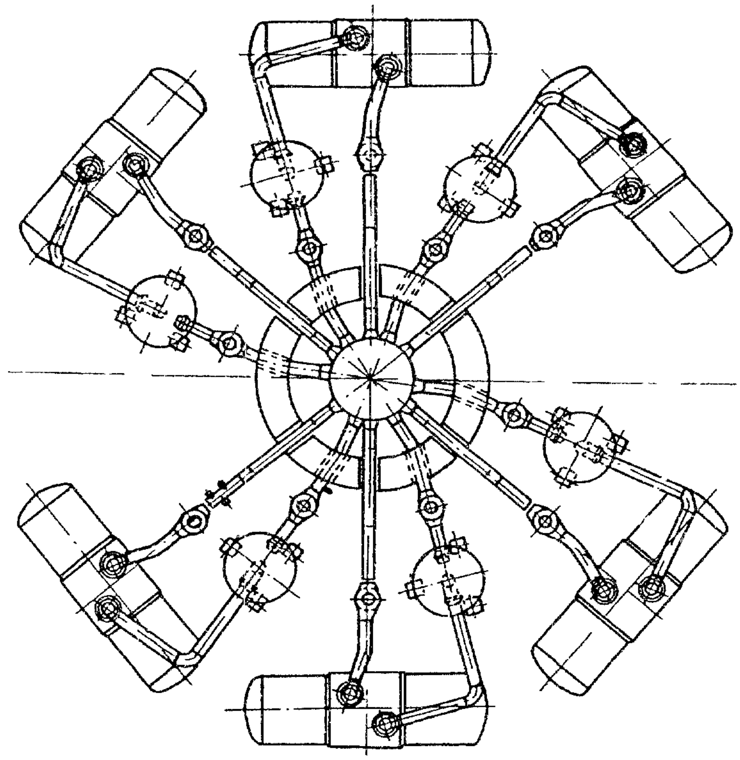


FIG. 1(a). Schematic view of the primary piping layout of the WWER-440 model 230 reactor (six loops).

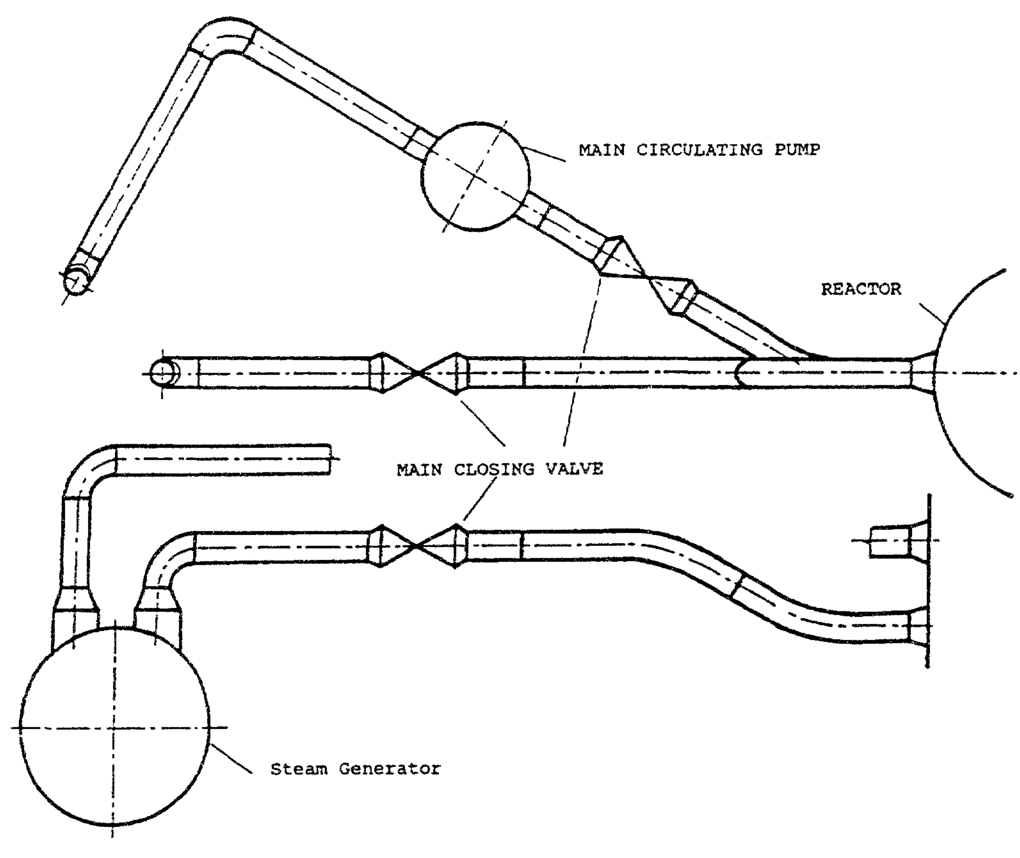


FIG. 1(b). Schematic view of the WWER-440 model 230 one loop arrangement.

pipe whip restraints or antiseismic measures to cope with higher intensity earthquakes (above 5° Medvedev–Sponheuer–Karnik (MSK-64)). In order to reach minimum safety requirements necessary for present NPPs, it is important to take corrective actions. For the safety related piping, it is necessary to provide evidence that probability of failure is extremely low.

In the piping system fully ductile properties of base material can be expected. In highly stressed areas there are dissimilar welds which could show less ductile behaviour or even ductile to brittle transition behaviour. At present, knowledge of defect sizes and their distributions within volumes of critical welds is limited. On the other hand the piping systems are flexible, with membrane and bending stresses low due to the thicker walls.

As the original design is without antiseismic measures, higher seismic loads could be expected in critical sections. Supports for all heavy components must be reassessed to assure their stability. Similarly, reassessment has to be done if antiseismic measures have already been introduced. There is a strong possibility that steam lines and feed water lines are influenced by this measures and have to be included in assessment.

Fatigue damage assessment has been studied extensively and in general low damage values have been indicated. Stratification problems, however, have not been studied extensively and there is a need for further assessment.

All the aforementioned topics have to be considered as requirements which must to be assessed in detail in order to avoid the influence of such damage mechanisms on LBB behaviour.

Provided that existing defects or conservative (hypothetical) defects will not grow to exceed critical crack length under single (or multiple) end of life loading, stability and growth of 'through wall' cracks has to be assessed. In order to do so we have to determine leakage crack length and critical crack length. It is important to realize, that leakage crack length depends on the sensitivity of the leak detection system. Three independent detection systems should be used with specified sensitivities.

The critical crack length must be stable under specified loading conditions and with use of conservative safety coefficients. This analysis has to be performed for the most critical situation in terms of material, loading and stress state combination.

1.2. LEAK BEFORE BREAK APPLICATION IN THE USA AND EUROPE

1.2.1. Approach to the LBB concept in the USA

In the USA the DEGB concept was originated by the US Atomic Energy Commission for the multiple purpose of sizing containments and establishing 'accident' doses and, later, for the sizing of ECCSs. This safety philosophy was supported by the following:

Regulations

- Appendix A to "Modification of General Design Criterion for NPP" to 10 CFR Part 50, Criteria 4, 30, 31 and 32.

Regulatory Guides

- RG 1.45: Reactor Coolant Pressure Boundary Leakage Detection Systems.
- RG 1.46: Protection Against Pipe Whip Inside Containment.
- RG 1.116: QA Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems.
- RG 1.124: Service Limits and Loading for Class-1 Linear Type Components Supports.
- RG 1.130: Service Limits and Loading for Class-1 Plate and Shell Type Components Supports.

Standard Review Plans

- SRP 3.6.1: Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment.
- SRP 3.6.2: Determination of Rupture Locations and Associated Dynamic Effects.
- SRP 5.2.1.1: Compliance with the Codes and Standards Rule 10 CFR Part 50.55a.
- SRP 5.2.1.2: Applicable Code Cases.
- SRP 5.2.3: Reactor Coolant Pressure Boundary Materials.
- SRP 5.2.4: Reactor Coolant Pressure Boundary Inspection and Testing.
- SRP 5.2.5: Reactor Coolant Pressure Boundary Leakage Detection.

Codes and Standards

- ASME Section XI: IWB 3640.
- ANSI Draft Standard: Leak Detection.

The direct result of such postulated piping ruptures was the establishment of unresolved safety issue (USI) A-2 "Asymmetric Blowdown Loads on PWR Primary Systems" (1975), described in detail in NUREG-0609. The resolution of this issue would require some licensees for operating PWRs to add both massive pipe whip restraints to prevent postulated large pipe ruptures from resulting in full DEGBs and jet impingement shields. The same problem concerns newly designed NPPs. The estimated industry costs to install plant modification to withstand this effect is over US $100 million.

Subsequent to the identification in 1975 of the generic safety concern that initiated USI A-2, the fracture mechanics methods regarding the potential rupture of tough piping such as is used in PWR primary coolant systems has advanced considerably. The conclusions reached from many theoretical and experimental studies and from many reactor-years of operating experience is that flawed piping is much more likely to leak before it breaks. These advanced fracture mechanics techniques deal with relative small flaws in piping components (either postulated or real) and examine their behaviour under various pipe loads. The objective is to demonstrate by deterministic analyses that the detection of small flaws either by in-service inspection or by a leakage monitoring system is assured long before the flaws can grow to critical or unstable sizes which could lead to large break areas such as in a DEGB.

Because the application of the LBB concept was at variance with USNRC regulations, during 1983 some actions were taken by the USNRC Committee to Review Generic Requirements (CRGR). As a result, the CRGR in its recommendations to the USNRC Executive Director for Operation endorsed the position that an acceptable technical and regulatory basis exists to grant exemptions to General Design Criterion 4 (GDC-4). The scope and bases for these exemptions were specified in Generic Letter 84-04 [14].

At present, the regulatory position to the applicability of the LBB concept is established in Standard Review Plan, Section 3.6.3. The following specific effects are excluded from GDC-4:

- pipe whip or other pipe break reaction forces;
- jet impingement forces;
- vessel cavity or subcomponent pressurization, including asymmetric transient effects;
- pipe break associated transient loadings in functional systems or portions thereof whose pressure retaining integrity remains intact.

The procedure in the USA for establishing an LBB case for the primary pipework, approved by the USNRC, can be summarized as follows:

(1) Identify critical sites in the pipework. These are defined as the sites where highest stresses occur in combination with the poorest material properties.

(2) Show that a postulated defect, located in the critical site of the pipework, would not grow significantly during service. The defect is to be postulated in accordance with the acceptance criteria of Section XI of the ASME Code.

(3) Postulate 'through wall' cracks at critical sites, the size of which is sufficient to ensure their detection by way of the resulting leakage; demonstrate that these crack will remain stable even when subjected to the loads imposed by an SSE occurring during normal operation.

Remarks:

(1) The fatigue crack growth analysis is included in order to exclude the part-through cracks from the logical structure of LBB. It is intended to demonstrate that acceptable defects cannot result in leaks nor growth to critical size cracks during the remaining lifetime of the plant.

(2) The critical site should be determined for base materials, weldments and safe ends. Load combination associated with normal operation and the simultaneous occurrence of an SSE should be taken into account when identifying the critical site.

(3) The procedure defines margin coefficients: 10 for leak and either 2 for crack size or 1.4 for stresses. The margin of 1.4 on stress can be reduced to 1.0 if all the dynamic stresses are algebraically added as tensile components.

1.2.2. Approach to the LBB concept in the United Kingdom

The most characteristic feature of the European attitudes to LBB relies on direct treatment of the part-through cracks within the logic of the LBB concept. Although there are differences between LBB concepts within Europe, we shall assess the concept of Nuclear Electric (United Kingdom) (see e.g. Ref. [9]). This can be summarized as:

(1) Characterize the defect at an suggested critical site. The ultimate LBB case then relates to defects in the framework of this characterization and to the preselected critical sites.

(2) Determine defect length of 'through wall' defect. Here the lower bound material conditions are to be used as well as the most severe loading.

(3) Estimate defect length at breakthrough.

(4) Calculate crack opening area.

(5) Calculate leak rate from defect.

(6) Calculate time to grow to limiting length.

(7) Assess results

Some characteristic features are:

(1) Lower bound J–resistance (J–R) curves and best estimate stress strain data are strictly prescribed to pertinent stages of evaluation.

(2) Margin coefficients are not prescribed but left to the judgement of the user.

(3) The procedure is logically closed, but limited only to the postulated defects. The defects out of postulate are to be covered by other methods (see e.g. Ref. [7]).

(4) The LBB analysis is seen as an potentially strong tool of the safety case assessment. Currently, no restriction on dynamic effects measures are drawn.

1.2.3. Procedure in Germany for break preclusion with LBB [44–47]

In Germany, the RSK Guidelines for Pressurized Water Reactors have come into operation since 1979 [44] on LBB for primary system, e.g. Fig. 2(a).

- Basic Safety Concept, see Fig. 2(b) (Basic Safety and Independent Redundancies);
- Realization in the German Code (see Fig. 2(c));
- Evaluation of LBB (see Fig. 2(d));
- Principle of LBB (see Fig. 3).

Procedure:

(1) Defects to be considered.
(2) Fatigue crack growth analysis.
(3) Demonstration of LBB fatigue crack growth beyond design ($2c_f'$), (see Fig. 4(a)).
  (a) If the crack grows through the wall by fatigue or the ligament breaks without instability in the circumferential direction: ($2c_f' < 2c_c$), then 'leak before break',
  (b) If the crack becomes critical before it grows through the wall ($2c_f' > 2c_c$), then 'break before leak'.
(4) Analysis of surface crack for end of life.
(5) 'Through wall' crack (TWC) stability analysis under maximum loads ($2c_c$).
(6) Completeness of loads used for fatigue crack growth and crack stability.
(7) Leak rate of stable TWC (leak detection).

Fracture mechanics procedure (see Figs 4(a) and 4(b))

Fracture mechanics methodology and criteria demonstrate LBB. In Germany, Siemens KWU performed numerous analyses to preclude breaks based on simplified elasto-plastic fracture mechanics (flow stress concept, plastic limit load). The acceptability and applicability of these concepts were discussed with and accepted by the authorities in Germany.

Primary System (RSK Guidelines Chapter 21.1, version 3/1984		
	Leak and break postulates	effects
Reactor coolant lines	•0.1 A, 15 ms, linear	•pressure waves (RPV internals)
	•0.1 A, steady–state blowdown	•jet force (pipings, components, building) •reaction force (pipings, components, building)
	•≤2A	•LOCA analysis •containment •pressure differences (building) •qualification of I&C
Circumf. nozzle weld	•p·A·S, S=2	•stability of the components (e.g. RPV, SG, RCP, PRZ)
RPV leak	•20 cm^2	•RPV supporting •RPV internals •LOCA analysis
Austenitic connection lines with DN > 200 (surgeline, ECCS up to the 1st isolation)	•0.1 A	•jet force (pipings, components, building) •reaction force (pipings, components, building)

FIG. 2(a). Postulated leaks and breaks for piping systems.

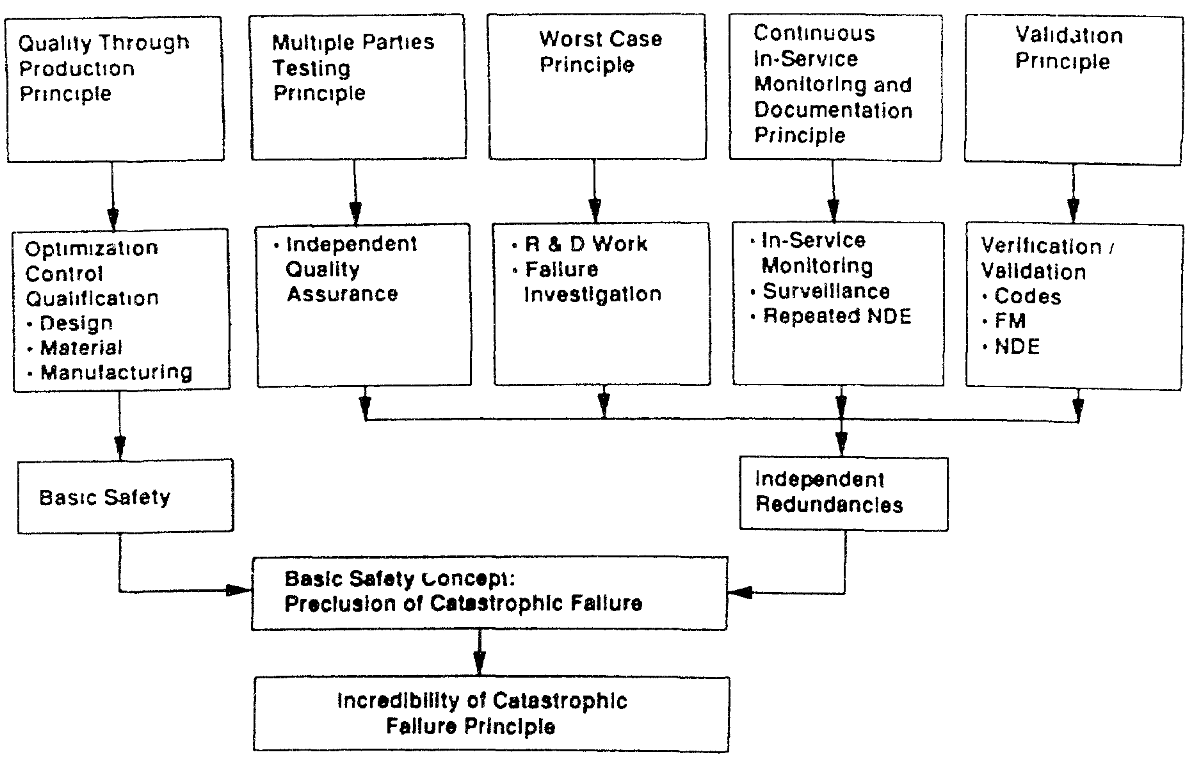


FIG. 2(b). General concept for break preclusion.

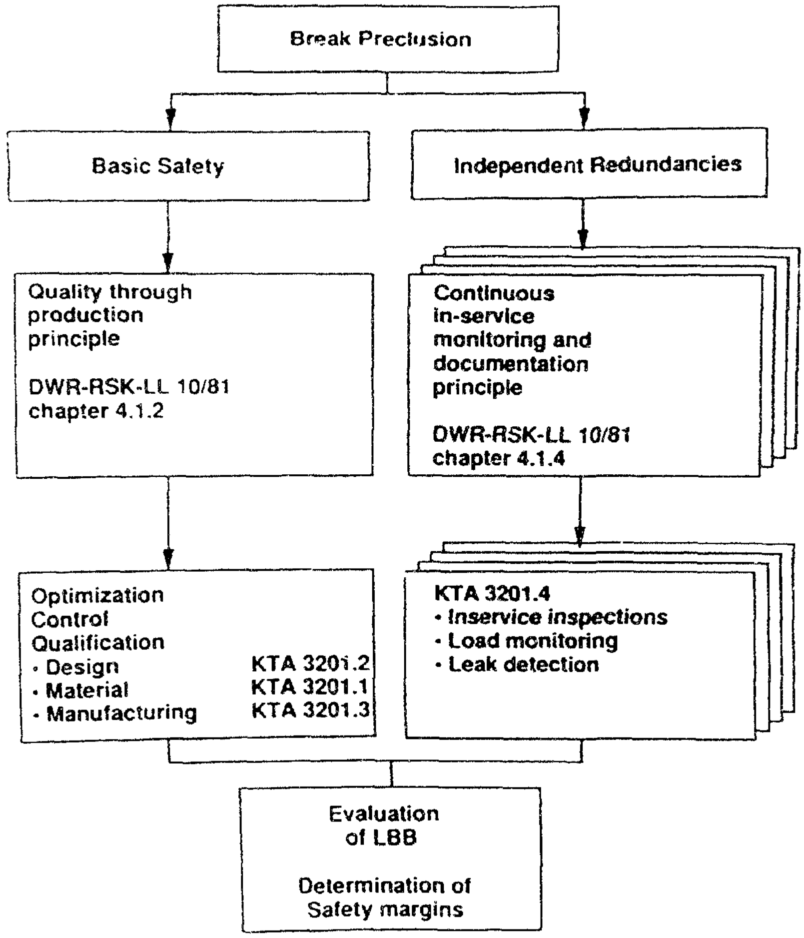


FIG. 2(c). Realization of break preclusion concept.

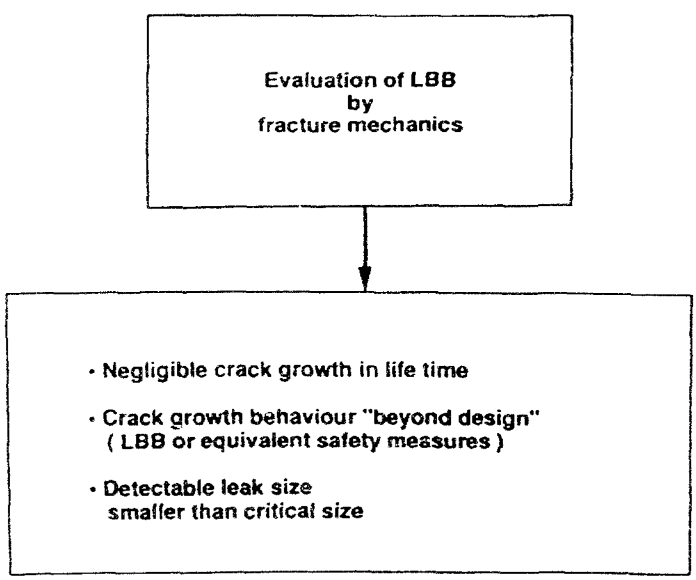


FIG. 2(d). Evaluation of LBB and determination of safety margins.

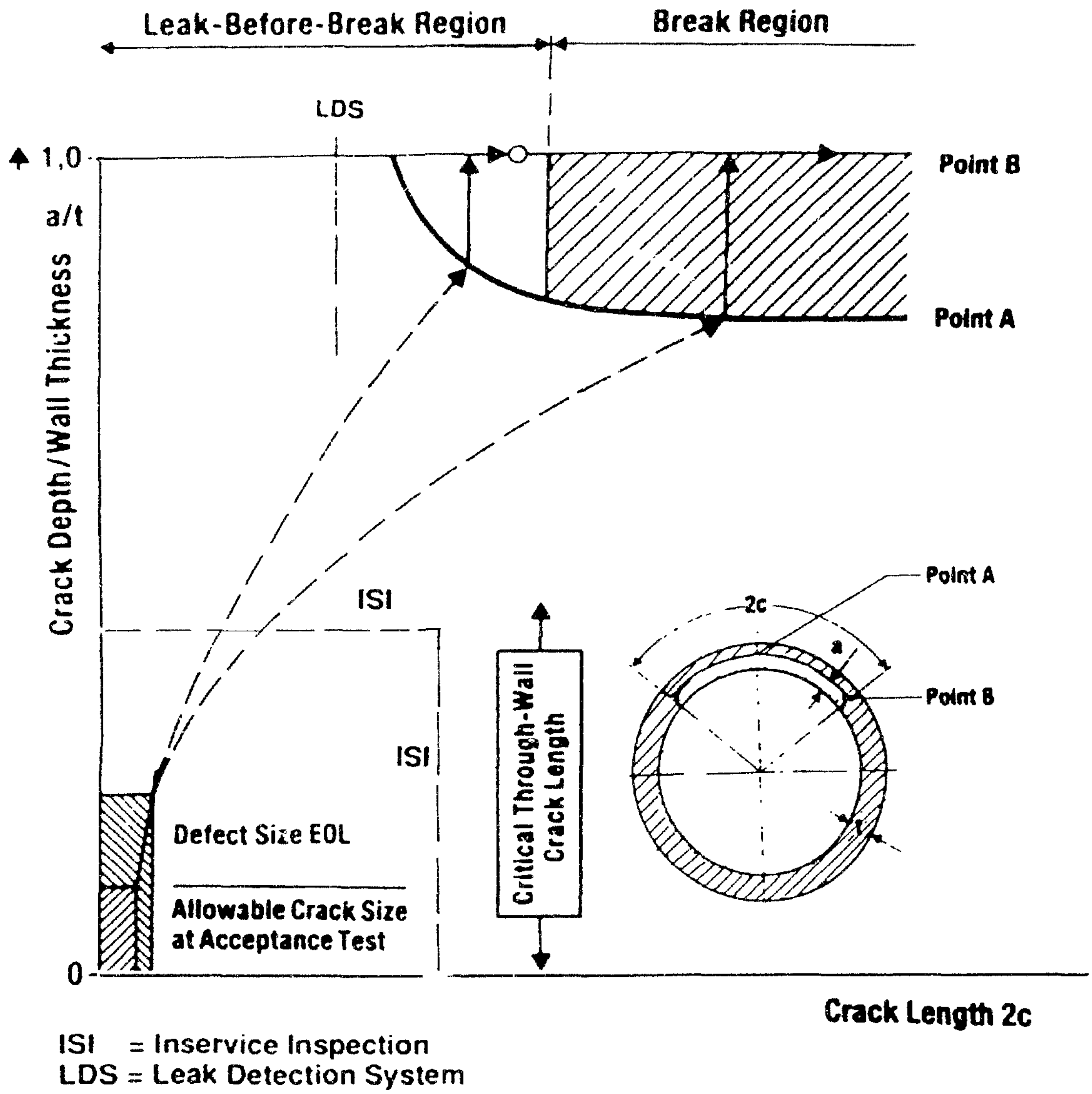


FIG. 3. Schematic diagram of the LBB concept.

1.2.4. The Czechoslovak LBB concept and the application to primary piping for WWER reactors

The Czechoslovak LBB programme was started in October 1991. The first objective of the programme is to prove the LBB status of the primary piping of the WWER model 230 in Jaslovske Bohunice (first generation of the WWER NPPs in Czechoslovakia). This objective is due to be completed by January 1993. The second objective is to prove the LBB case for primary piping of the WWER type NPPs which are under construction.

The Czechoslovak LBB concept is pursued as a part of the strategy to meet western style safety standards. The Czechoslovak Atomic Energy Commission (CSAEC) recognizes that Czechoslovak NPPs are lacking in some standard safety facilities [2] such as ECCS and western style containments. The status of the Czechoslovak LBB is as follows [8]:

(1) The LBB concept follows the LBB outline [5].

(2) Application of the LBB concept at Jaslovske Bohunice is aiming to show that the probability of major failure (due to brittle behaviour or the critical crack size is exceeded) is extremely low.

(3) Some parts of the LBB evaluation according to Ref. [5] are supplemented by additional, internationally recognized evaluation procedures.

(4) The safety margins requirements are taken over from Ref. [5].

(5) The LBB evaluation is accompanied by extensive stress state calculations including seismic analysis of all safety related piping systems.

(6) The fracture mechanics database and corrosion database is based on tests of archive material including base metal, weld metal and heat affected zone.

(7) The leak rate calculations will be verified by experiments together with calibration tests of leak diagnostics system.

(8) The LBB evaluation is accompanied by an extensive large scale experimental validation programme.

(9) Validation and verification of analytical and experimental procedure will be undertaken. Quality assurance procedures will be to IAEA and international standards.

Description	Methodology	Loads	Results	Criteria
Definition of reference defects	Based on allowable defects + performance of inspection technologies		a_0 , $2c_0$	allowable defects
Fatigue crack growth (F.C.G.)	Integration of crack growth law until end of life (EOL)	Normal and upset trans. for 1 specif. load collective	$a_f = a_0+\Delta a$ $c_f = c_0+\Delta c$	Small F. C. G. in 1 specif. load collective
a) LBB Fatigue crack growth	Same as previous $2c_c$ computation with qualified method	Normal & upset trans. Unlimited specif. load collectives	a_f' $2c_f'$	• LBB is demonstrated, if $2c_f' < 2c_c$ for unlimited specified plant load collectives
b) LBB Fatigue crack growth			a_f' $2c_f'$	if $2c_f > 2c_c$ equivalent safety measures

FIG. 4(a). Fracture mechanics method.

Description	Methodology	Loads	Results	Criteria	Margin
Stability of EOL surface crack	qualified methods: Flow stress concept and / or Plastic limit load	all trans. + SSE	$2c_c$ a_c	Stability of ligament	sufficient margin
T.W.C. stability analysis	qualified methods: Flow stress concept and / or Plastic limit load	all trans. + SSE	$2c_c$	Stability of ligament	sufficient margin
Leak rate of T.W.C.	qualified methods	Full power	A, $\dot{m}(2c_c)$	$\dot{m}(2c_c) >$ y x Z l/h	Y on the detectable leak rate Z

A : crack area of critical crack length $= 2c_c$ at full power

$\dot{m}(2c_c)$: Leak rate of critical crack $= 2c_c$ at full power

FIG. 4(b). Fracture mechanics method (cont.).

1.2.5. Differences between the LBB concepts in Europe and in the USA

The differences in LBB concepts in Europe and the USA are obvious from Sections 1.2.1 and 1.2.2 and can be summarized as follows:

(1) Legislative status

USA: The LBB concept [5] and Section XI of the ASME Code are sufficient to relieve the facilities from coping with the dynamic effects of a DEGB.

Europe: The LBB case can be seen as a powerful independent method of safety case establishment which does allow for the omission of the DEGB dynamic effect arrangement.

(2) Variability of the procedure

USA: Because of the legislative consequences the procedure is strictly prescribed, including the safety margins.

Europe: The logical structure is prescribed, but the particular evaluation is left to legislative judgement.

(1) Further development and validation is required for predicting the growth and stability of 'through wall' cracks in PWR primary circuit components (including branches and elbows) subjected to combined axial, bending and torsional loads.

(2) If the likelihood of failure of cracked primary coolant loop pipework due to seismic loads is considered a design basis event for PWRs there will be a need to consider methods for predicting crack growth and stability under combined static and dynamic loadings.

(3) The adequacy of existing models for predicting the leak rates from cracks in PWR piping materials, the performance and sensitivity of the various leak detection systems available and the evidence for plugging of leaks to occur in the PWR environment all need to be reviewed.

(4) The role of inspection must be clarified and the nature, location and minimum sizes of the defects that will have to be reliably detected by non-destructive evaluation methods in order to make an acceptable LBB case must be determined.

(5) The ability of pre-service and in-service inspection methods to reliably detect the defects defined in (4) must be demonstrated.

(6) A probabilistic model for primary circuit pipework failures should be developed which could be used to ensure that the work carried out under items (1) to (5), when combined with existing data and methodologies, was sufficient to meet safety requirements for primary coolant loop integrity.

(7) The available data on the nature, causes and consequences of major failures in PWR primary coolant loop and similar piping should be reviewed. The frequency, causes and likely consequences of such failures form an essential input to the probabilistic model described in (6).

1.3. LEAK DETECTION SYSTEMS

No single leak detection method currently available combines optimum leakage detection sensitivity, leak locating ability, and leakage measurement accuracy. For example, although quantitative leakage determination is possible with condensate flow monitors, sump monitors, and primary coolant inventory measurements, these methods are not adequate for locating leaks and are not necessarily sensitive enough to meet Regulatory Guide requirements.

For German plants localization can be made in respect to compartments and sensitivity is enough to meet German regulatory requirement.

It is possible to improve leak detection capability at specified sites by use of acoustic emission monitors or moisture sensitive tape. However, current acoustic emission monitoring techniques provide no source discrimination (e.g., distinction between leaks from pipe cracks and valves) and little or no leak rate information (a small leak may saturate the system). Moisture sensitive tape provides neither quantitative leak rate information nor specific location information other than at the location of the tape [10].

1.3.1. Current practice in the USA

Regulatory Guide 1.45 [11], which is the only established guide, recommends the use of at least three different methods to detect system leakage. Monitoring of both sump flow and radioactivity of airborne particulates is mandatory. A third method can involve either monitoring of condensate flow rate from air coolers or monitoring of radioactivity of airborne gas. Although current methods used for leak detection reflect the state of the art, other techniques may be developed and used. Regulatory Guide 1.45 also recommends (but not requires) that leak rates from identified and unidentified sources be monitored separately to an accuracy of 3.8 L/min (1 gal/min) and that indicators and alarms for leak detection be provided in the main control room.

Because the recommendations of Regulatory Guide 1.45 are not mandatory, Argonne National Laboratory reviewed the Technical Specifications for 74 operating plants, including many PWRs. The purpose of the review was to determine the types of leak detection methods actually used, the range of limiting conditions for operation and the surveillance requirements for the leak detection systems.

All plants used at least one of the two systems specified by Regulatory Guide 1.45. All but eight used sump monitoring and all but three use particulate monitoring. Many plants also monitored the condensate flow rate from drywell air coolers and radioactivity of atmospheric gases.

The plant Technical Specifications' limit on unidentified coolant leakage for all PWRs was 3.8 L/min, whereas the limit for BWRs was 19 L/min. The limits on total leakage (unidentified plus identified) were generally 38 L/min for PWRs and 95 L/min for BWRs. [Regulatory Guide 1.45 does not specify limits but does suggest that the leakage detection system be able to detect a 3.8 L/min leak in 1 hour.] In some cases, limits on rates of increase in leakage were also stated in the plant technical Specifications. Two BWRs had a limit of 0.38 L/min/h; four had a limit of 1.9 L/min/h.

Generally speaking, reactor operators rely on sump monitoring to establish the presence of leaks. Other methods appear to be less reliable or less convenient. In most reactors, the surveillance periods are too long to detect 3.8 L/min leak in 1 hour, as suggested by Regulatory Guide 1.45, but this sensitivity apparently could be achieved if monitoring procedures were modified. Simply tightening the current leakage limits to improve sensitivity is not adequate, however. It might produce an unacceptably high number of spurious shutdowns because current leak detection systems cannot identify leak sources. None of the systems provides any information on leak location, and leak must be located by visual inspection after shutdown. Because cracks may close when the reactor is shutdown (or flow rates are reduced considerably), the ability to locate cracks during plant operation would be desirable.

It has still not been shown that the leak rate can be measured with sufficient accuracy. The possibility exists that large cracks may initially produce only low leak rates. This situation could arise because of corrosion plugging or fouling of relatively slow growing cracks or the relatively uniform growth of a long crack before penetration. In such cases the time required for a small leak to become a significant leak or rupture could be short, depending on crack geometry, pipe loading, and transient loading (due to a seismic or water hammer event).

1.3.2. Current practice in Germany

Approach to leakage monitoring is based on the integral methods and localization methods. The integral methods allow leakage to be localized in compartments or plant sections. The physical nature is based on monitoring of temperature, pressure, moisture in compartments, condensate accumulation in the recirculation air coolers and sump water or gully water levels.

The typical monitoring system for the 1300 MW Kraftwerk Union PWR plant uses the dew point temperature changes. The sensors are installed in every loop at these positions: steam generator, main coolant pump, pressurizer surge lines and in the neighbourhood of the RPV.

The localization methods used an acoustic leakage monitoring system or a moisture detection system on the basis of permeable hose. The theoretical basis of the acoustic leakage monitoring system is fracture mechanics and thermohydraulic analysis. The following quantities are calculated: critical 'through wall' crack length, leakage crack length and leakage flow rate. Basic parameters of the acoustic leakage monitoring system are:

- the leakage indication in the time interval (5; 120 s).
- localization of the leak (1–3 m).
- sensitivity detection less than 0.025–0.065 kg/s.

The main features of the design concept are:

- Continuous monitoring of high frequency structure borne noise (>100 kHz) directly on the components by registering the effective levels of the measuring signals.
- In the event of an alarm issue, calculation of the filtered leakage noise level from momentary measured value and background level.
- Estimation of leakage rate with plus or minus 50% accuracy by comparison of the noise level extrapolated to the leakage location with empirical data. The procedure described is shown in Figs 5 and 6 [12, 13].

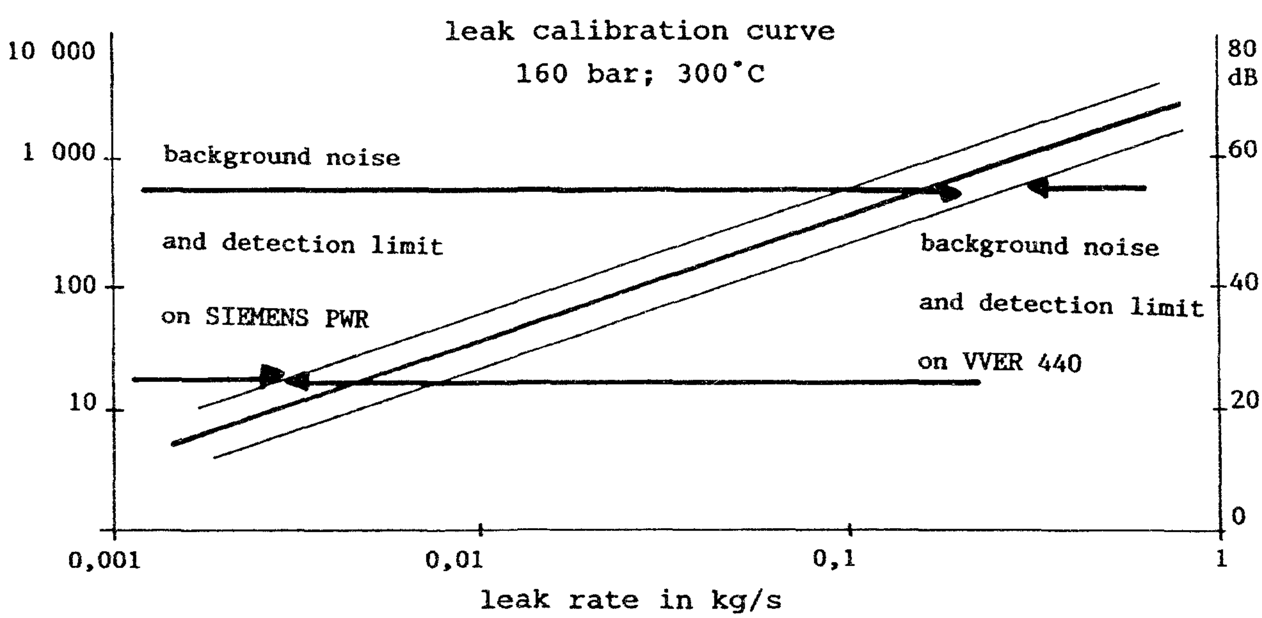


FIG. 5. Leakage noise versus leakage rate for ALMS.

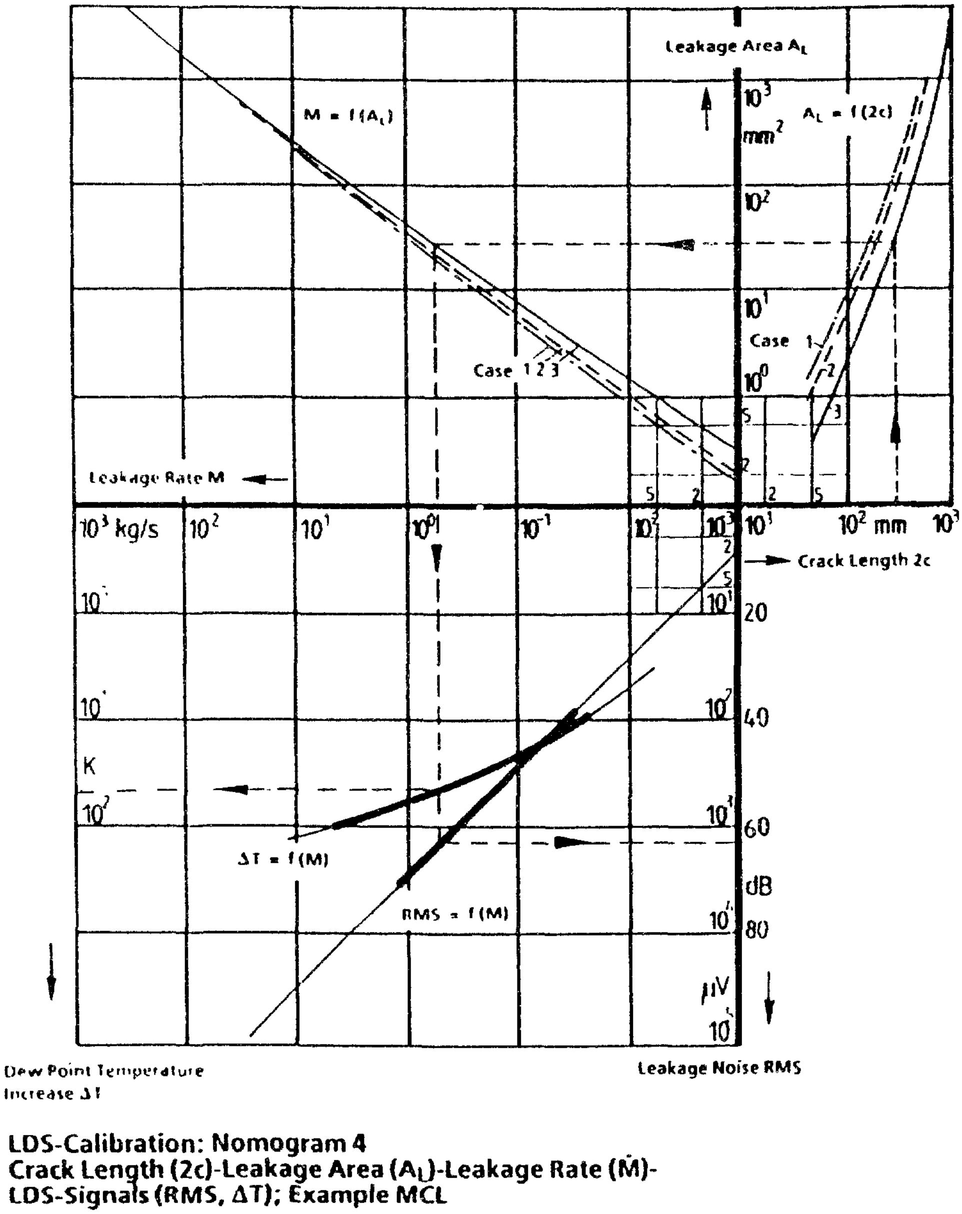


FIG. 6. Calibration nomogram No. 4 for the leak detection system (LDS).

1.4. RELEVANCE TO OTHER REACTOR TYPES

The application of the LBB concept application for the WWER-440 model 230 reactors will have strong relevance to other reactor types. First of all it will form the basis for the application on the WWER-440 model 213. There are ten 440 MW model 213 in Czechoslovakia and many more in the former USSR. Most of them will need the reassessment of antiseismic measures and PSA studies. The simplification of the antiseismic measures could save millions of dollars in equipment, labour and collective exposure. The results of the LBB assessment can and will be readily used in the PSA data input.

There is work in the LBB programme in Czechoslovakia which has direct application on western style PWRs. Results of large scale experiments, especially tests on 'safe ends', are of great importance, as is fracture mechanics data, on the dissimilar welds. Recently the IAEA Programme "Pilot Studies on Selected Components for the Management of Power Plant

Ageing and Life" was started. One of the four major subtasks is concerned with data collection on the RPV 'safe end' ageing. The corresponding work in the LBB programme should be used as part of this.

Siemens KWU has performed analysis of LBB for the main piping systems of Greifswald and Bohunice. The results are published in Refs [48, 49].

The application of the LBB concept to BWR type reactors has also a developing trend. However, the US regulatory approach excludes direct application, mainly because of possible corrosion damage mechanisms. In Germany the LBB concept is applied for BWRs and the fast breeder reactor.

Also necessary are the evaluation and calibration of diagnostic systems, particularly of the acoustic emission (AE) leak detection systems. There is a strong need for calibration data between the leak rate data and the corresponding AE signals in real piping systems. There is a comprehensive experimental programme using steam water loop with verified leak rate through a fatigue 'through wall' crack, and with instrumented AE system. This could give more confidence to assessment work as well as to the NPP operator. Experience from these experiments will have strong relevance to all reactor types and will be of significant value in applying LBB concept.

2. REGULATORY APPROACHES

2.1. THE APPROACH IN THE USA

It is permissible in the USA to eliminate the dynamic effects of a postulated high energy pipe rupture from the design basis of nuclear power plants utilizing the USNRC's LBB concept. This regulatory approach has been adopted by some countries, including Spain (for Westinghouse plants) and Czechoslovakia [8], and is used as the basis for their assessments.

The USNRC issued a Standard Review Plan to provide guidance for the implementation of the LBB concept. The development of the USNRC's Standard Review Plan 3.6.3 brought together the concept which forms the LBB Regulatory Approach. The development is best understood through the following brief description by K. Wichman and S. Lee [20]:

> "General Design Criterion 4 (GDC-4) [5] of Appendix A to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50) of the United States regulations requires postulation of pipe breaks and provision of appropriate protection against associated dynamic effects. However, the USNRC staff issued Generic Letter 84-04 [14] accepting that the DEGB of the PWR primary loop piping was unlikely to occur, provided it could be demonstrated by deterministic fracture mechanics analyses that postulated small 'through wall' flaws in plant specific piping would be detected by the plant's leakage monitoring systems long before the flaws could grow to unstable sizes. Leakage exceeding the limit specified in plant Technical Specifications requires operator action or plant shutdown. A detailed discussion of limitations and acceptance criteria for LBB used by the USNRC staff is provided in NUREG-1061, Vol. 3 [15]."

The USNRC initiated a rule change to GDC-4 [16, 17] because of the extremely low probability of pipe rupture and the potential for adverse safety implications resulting from the installation of protective devices. The placement of pipe whip restraints degrades plant safety when thermal growth is inadvertently restricted, reduces the accessibility for and effectiveness of in-service inspections, increases radiation dosages during in-service inspection, and adversely affects the economics of construction and maintenance. On 11 April 1986, a final limited scope rule [5] was published amending GDC-4 to permit the use of analyses to eliminate from the design basis the dynamic effects of postulated pipe ruptures of primary coolant piping in PWRs. On 27 October 1987, a final broad scope rule [18] was published amending GDC-4 to permit the use of LBB analyses in all qualified high energy piping, i.e. pressure exceeding 1.9 MPa or temperature exceeding 93°C, in nuclear power units.

Currently, approximately two thirds of the PWRs in the USA have approval for the application of the LBB concept in the primary coolant loop. There are also four PWRs which have approval for the application of the LBB concept for their auxiliary lines, specifically, pressurizer surge, accumulator, and residual heat removal (RHR) lines. One of the four plants also has approval for its application for the safety injection lines and the reactor coolant loop bypass lines. The approved auxiliary lines are all inside containment, fabricated from austenitic stainless steel, and at least 150 mm in diameter. The application of the LBB concept has not been approved as yet for any boiling water reactor.

2.1.1. Standard Review Plan 3.6.3

A new USNRC Standard Review Plan (SRP) section, numbered 3.6.3 [19] and entitled Leak Before Break Evaluation Procedures, providing review guidance for the implementation of the revised GDC-4, was published for public comment on 28 August 1987. The development of the final form of SRP 3.6.3 by the USNRC staff is continuing. The application of LBB is limited to piping that is not likely to be susceptible to failure from various degradation mechanisms in service [15]. From the USNRC experience, a significant portion of any LBB review involves the evaluation of the susceptibility of the candidate piping to various degradation mechanisms.

The LBB approach cannot be applied to piping that can fail in service from such effects as water hammer, creep, erosion, corrosion excessive fatigue. The rationale is that these degradation mechanisms challenge the assumption in the LBB acceptance criteria. For example (1) water hammer may introduce excessive dynamic loads which are not accounted for in the LBB analyses, and (2) corrosion and fatigue crack growth may introduce flaws whose geometry may not be bounded by the postulated 'through wall' flaw in the LBB analyses. In line with the 'defence in depth' principle, piping susceptible to failure from these potential degradation mechanisms has to be excluded from LBB application.

To demonstrate that the candidate piping is not susceptible to failure from these degradation mechanisms, the operating history and measures to prevent their operation or mitigate their effects must be reviewed.

2.1.2. LBB acceptance criteria (USA as example)

After the LBB candidate piping has been reviewed for degradation mechanisms, as discussed previously, and found to be acceptable, the piping is subjected to a rigorous fracture mechanics evaluation. The purpose of this evaluation is to show that there is flaw stability and the resulting leakage will be detected in the unlikely event that a flaw should develop. The current acceptance criteria established by the USNRC are expected to remain the same. These are as follows:

(1) The loading conditions should include the static forces and moments (pressure, deadweight, and thermal expansion) due to normal operation, and the forces and moments associated with the SSE. These forces and moments should be located where the highest stresses, coincident with the poorest material properties, are induced for base materials, weldments, and 'safe ends'.

(2) A 'through wall' flaw should be postulated at the locations determined from criterion (1). The flaw should be large enough that the leak from this postulated flaw is assured of detection. When the pipe is subjected to normal operational loads, it should be demonstrated that there is a factor of at least 10 between the leakage from the leakage-size flaw and the plant's installed leak detection capability. A leakage analysis which has been benchmarked against experimental or plant data is required. The margin on leakage is required to account for uncertainties such as the crack opening area, crack surface roughness, two phase flow, and leak detection capability.

(3) It should be demonstrated that the postulated leakage size flaw is stable under normal plus SSE loads. It should be demonstrated that there is a factor of at least 1.4 between the loads that will cause flaw instability and the normal plus SSE loads. The factor of 1.4 stems from the factor of the square root of two on stress intensity for flaw evaluation under Service Level C and D Loadings in IWB-3610 of Section XI of the ASME Code. Service Level Loadings are defined in NCA-2140 of Section III of the ASME Code. However, the broad scope rule permits a reduction of the factor of 1.4 to

1.0 if the individual normal and seismic loads (pressure, deadweight, thermal expansion, SSE, and seismic anchor motion) are summed absolutely. This is because the USNRC considers that an absolute load summation is sufficiently conservative to warrant a margin reduction.

(4) The flaw size margin should be determined by comparing the leakage size flaw with the critical size flaw. Under normal plus SSE loads, it should be demonstrated that there is a factor of at least two between the critical size flaw and the leakage size flaw. The factor of two stems from the factor of two on flaw size for flaw evaluation under Service Level C and D Loadings in IWB-3610 of Section XI of the ASME Code.

2.2. THE APPROACH IN GERMANY

The preclusion of breaks of piping in the German assessment route is based on the basic safety concept [41–43], (see Figs 2(a) and 2(b)). The concept consists of the basic safety (design, material, manufacture) and the independent redundancies (multiple testing, worst case principle, in-service inspection and verification by fracture mechanics). The basic philosophy of LBB concept as used in Germany is that a leak occurs long before the critical 'through wall' crack length is reached (see Fig. 3). Break preclusion based on the basic safety concept can be achieved with LBB behaviour. Break preclusion without LBB behaviour, however, requires equivalent safety measures.

2.3. THE APPROACH IN JAPAN

The standardization work on the application of the LBB concept to stainless steel pipes in Japan was started in December 1984 at the Sub-Committee on Structural Components within the Technical Advisory Committee of Nuclear Power Plant Operation of the Ministry of International Trade and Industry (MITI), and the Sub-Committee finished writing of a draft of LBB standards in March 1986. A review of the draft from the standpoint of the MITI text was then started, and it was expected that the draft would be opened as one of the MITI private regulations in 1992 [50].

The following items were decided to be introduced in new draft LBB standards:

(1) In order to make it possible to deal with an event having probability beyond the design assumption, the fatigue crack propagation analysis should be continued, without specifying the load cycles, until the crack grows to a full wall thickness.

(2) The assumed crack is to be determined as a larger one by comparing the crack length based on the crack propagation analysis with the crack length bringing about a 5 gal/min leak.

(3) The completion of the countermeasures against fluctuation in the thermal stratification phenomena is a premise for application of the LBB concept.

The basic concept is as follows:

(1) Position to be evaluated: The circumferential welded joint is to be calculated. Accordingly, the initial defect is assumed to be oriented in the circumferential direction.

(2) Initial defect: The initial defect is assumed to be a single defect since it is rare to have plural defects distributed intermittently or a long continuous defect covering a wide range of the whole circumference. By taking a margin of two times the detecting limitation by ultrasonic testing (UT) in pre-service inspection (PSI), the size of the initial defect in evaluation calculation is defined to have a semielliptical shape, that is 1.0 t long and 0.2 t deep on the inner surface of a pipe, where t denotes pipe wall thickness. When the wall thickness is not more than 15 mm, the size of the defect is to be defined as 3 mm deep and 15 mm long.

(3) Fatigue crack growth analysis: An analysis is to be made of the growth of a fatigue crack starting from the initial defect up to the full thickness penetration of the pipe wall. The Paris formula, where the stress intensity factor is the effective factor, and Raju-Neuman's solution are to be used.

(4) Loads for evaluation: The crack propagation analysis is to be made for loading conditions in the operating conditions I and II, and for 1/3 of S_1 earthquake load. In the crack stability analysis, evaluation is to be made on the loads of the operating conditions I, II and III and/or the combination load of the operating condition I and the S_1 earthquake load. In calculating the area of the crack opening, normal operating load is to be considered.

(5) Size of a leak detectable crack: Although the leak detection system in the PWRs has a capability of detecting a 1 gal/min leakage within an hour, the size of a leak detectable crack was defined to have a margin of 5 times. In the calculation of the leak amount, Henry model and Moody model are used depending on sub-cool water and saturated water/saturated steam, respectively. The calculation of the opening area is based on Paris–Tada's formula.

(6) Crack stability analysis: The stable limit stress for the assumed crack is based on the net stress concept, where the critical stress is the flows stress.

(7) Classification of events: The assumed event as providing criteria for the protection design is to be determined on the basis of the result of stable or unstable analysis. If the crack is stable, the event is determined as a leak, whereas if it is unstable the event is determined as a break. When a case is judged as a leak, the opening area under the design condition is to be determined.

3. WORK IN MEMBER STATES

3.1. PROGRAMMES OF THE OECD ON THE LBB CONCEPT

Leak before break is now widely applied in the nuclear industries of the Member States of the Organisation for Economic Co-operation and Development (OECD) as a means of assessing the susceptibility of pressurized components to failure by unstable crack propagation. While the detailed nature of the applications differ widely, depending on the country and the component, the basic concept does not. A crack is postulated to exist in a structure and assumed to grow stably by some mechanism to a size at which it penetrates the wall of the component resulting in leakage of the pressurized fluid. When the crack, through continued growth, reaches a size at which leakage is deemed to be detectable (with margin), an assessment is done to show the margin against failure. The margin can be expressed either as a factor of safety on the stress or crack length for instability or on the time available to effect a safe shutdown of the reactor before the crack grows to a critical size. Several meetings were held on the topic of LBB:

USNRC/CSNI	September 1983	Monterey, CA, USA	[21]
USNRC/IPIRG	November 1985	Columbus, OH, USA	[22]
USNRC/IPIRG	May 1987	Tokyo, Japan	
USNRC/IPIRG	May 1989	Taipei, Taiwan	[23]
OECD-NEA/CANDU	October 1989	Toronto, Canada	[24]

At the last meeting it was concluded that the LBB concept is a route to guaranteeing structural integrity of reactor components. The uses of the LBB have been discussed for various systems and to varying degrees as a part of a defence in depth procedure. These systems include zirconium alloy pressure tubes, piping systems and steam generator tubing. Application on the vessel in a sodium cooled fast reactor was also included.

Often the application, of LBB concept was underpinned by extensive research and development programmes with the following main themes:

- Methodologies: leakage calculations for stable and unstable crack growth;
- Qualification of LBB procedures;
- Materials data for input into assessment.

The following main issues have been identified for future effort [25]:

Issue 1: Sensitivity and reliability of detection devices.
Issue 2: Factors that effect leakage and make detection difficult.
Issue 3: The gradual development of a part-through crack, its shape and its effect on instability:
(a) the importance of the development of the part-through crack and its shape;
(b) instability criteria for the part-through crack, which is receiving much less attention than the 'through wall' crack.
Issue 4: Consideration of weld properties.
Issue 5: Application of LBB concept to non-ideal regions.
Issue 6: Probabilistic approach.
Issue 7: When the LBB concept should be used.
Issue 8: Incorporation within codes.

3.2. THE APPROACH ADOPTED IN THE FORMER USSR

According to Russian specialists at a recent IAEA meeting in Vienna, the Codes and Standards of the former USSR [26–30] deal with the prevention of pipeline in-service leaks and breaks.

The elimination of pipe break in the course of operation is ensured by the following measures:

(1) The choice of materials and technologies assuring the specified chemical and mechanical properties in operation.

(2) The selection and validation of the pipe design, eliminating the possibility of high levels of in-service thermal-mechanical loads and unacceptable corrosion effects.

(3) Demonstration of the impossibility of pipe deformations due to reaching the material yield strength (except at some local points).

(4) Demonstration of the impossibility of in-service crack generation.

(5) Demonstration of impossibility of pipe break due to any operational loads on pipe, including the unlikely transients caused by the maximum design basis earthquake.

The practice of demonstrating the impossibility of in-service pipe break initiation has been confirmed by extensive operational experience with piping at NPPs in the former USSR [32–35].

To investigate the possibility of an instantaneous in-service break of large diameter pipe, research has been conducted for the Novovoronezh and Kola NPPs. The aim of this research was to check the quality of the design, architect-engineering, manufacturing and installation effort, the adequacy of the pipe operation and maintenance and actual status and operating conditions of the pipe. It also sought to determine the possibility of initiation of gross pipe defects resulting in penetrating cracks causing leaks, to study the stability of 'through wall' defects and trends in their evolution to the critical defects which eventually result in pipe break. The research was performed using the methods of qualitative and quantitative analysis. Research was supported by extensive use of mathematical statistical techniques and probabilistic methods of fracture mechanics.

The proposed research to demonstrate the applicability of the LBB concept can be divided into three phases:

(1) Baseline reliability. Studies have been made of the compliance of pipes from the point of view of design, metal condition, manufacturing process and operation conditions with requirements of Codes and Standards [26–30] applied in the nuclear power industry.

(2) The quantitative deterministic analyses have been performed of hypothetical pipe leak or break events.

(3) Assessment have been made of the possibility of in-service leak or instantaneous break and if possible, estimations of the probability of such events. Recommendations have been developed for their prevention and elimination from operational practice.

The system approach was extensively used and the majority of the parameters affecting the pipe reliability were included in the analysis. Quantitative calculations using fracture mechanics techniques were performed assuming that all the discontinuities including the spatial ones are cracks.

Discontinuities with sizes not exceeding those specified in the weldment test procedures and documents for base metal quality control after manufacturing are neglected and considered acceptable.

Acceptable discontinuities are those with sizes (taking into account their growth to the end of operation or during specified period of time) not exceeding the allowable values determined according to calculation methodology.

These calculations are performed using fracture mechanics methods (linear and non-linear taking into account the limiting ductile behaviour).

Components made of corrosion resistant austenitic steels and not affected by irradiation are evaluated only on the basis of ductile behaviour. For the brittle domain the evaluation is performed by means of methods of linear elastic fracture mechanics using the stress intensity factors and fracture ductility temperature dependence coefficients. For the quasi-brittle and ductile domains the evaluation is performed by means of non-linear fracture mechanics methods using the elastic stress rate factor and other factors.

The probabilistic assessment includes the determination of the probability of the following events:

(1) Event A: existence of at least one defect with parameters (depth: a, length: c) exceeding critical levels e.g., a total break.

(2) Event B: existence of at least one penetrating defect in the absence of critical defects (leak). To determine the critical size defects two equations are used (for axial and circumferential type of defects).

The question of categorizing the defect as a critical one is considered with the interpretation of strength condition. For the case of a probabilistic treatment of the strength condition there is a need to determine the probability that the defect (a, c) is a critical one.

The present calculation approach assumes that the yield strength is a random value subject to normal distribution. Events A and B are due to existence of defects. It was concluded that accident loading conditions and earthquake loadings do not have significant direct effect on the probability of pipe fracture. However, they contribute significantly to growth rates of hypothetical cracks and with the increase of the effective loading cycles the probability of pipe fracture also increases.

The most effective technical measures to reduce the probability of pipe break at operating NPPs are hydrostatic pressure tests and better NDE systems, 100% volumetric NDE reduces probability of fracture by one order of magnitude and more.

3.2.1. The experimental programme of the former USSR

In 1989 an extensive theoretical and experimental programme [40] was prepared by the research institute VNIIAES in co-operation with other major institutions in the former USSR.

The programme consisted of 19 major tasks aimed to provide data to assess the probability of DEGB in the primary piping of the WWER-440 reactors. According to the proposed work schedule, most of the results should be available by the end of 1991. It is not possible to confirm if this programme was carried out completely or only partially as proposed. Nevertheless results are not yet available and at this moment only data already discussed [32–35] can be used.

Very important data for assessment of the ageing phenomena could be derived from the 100 000 hours programme and later from the 150 000 hours programme as well. Under this programme parts of piping components (elbow and straight pipe with circumferential welds) had been cut out from the MCP line at Kola NPP after 100 000 hours service. A similar programme has been carried out at Greifswald NPP which includes not only pipe material but also selected parts of other piping components.

Although comprehensive details are not available, the following conclusions can be summarized:

- mechanical properties: yield and ultimate strength due to long term exposure increased up to 10%;
- plasticity properties: changes are not significant, but fracture mechanics database is not available;
- cyclic crack growth resistance: is affected — comprehensive results are not yet available;
- no changes in submicrostructure;
- microstructural changes: are seen on grain boundaries including dispersion distribution of particles;
- assessment of changes in corrosion properties: not yet completed;
- changes in the properties of dissimilar welds: not included in the programme.

3.3. THE LBB PROGRAMME IN CZECHOSLOVAKIA

The Czechoslovak Atomic Energy Commission issued a legislative procedure [8] based on the USNRC Regulatory Procedures [5, 18, 19]. In order to be able to apply the LBB concept in full, several requirements have to be met. Firstly it has to be shown that fatigue damage is low. The original assessment of the plant based on Soviet as well as ASME codes was finished in 1979 [36]. A fatigue damage assessment was recently carried out with a more detailed analysis and also using the actual loading regimes and numbers of cycles. The condition for low fatigue damage was shown to have been met.

The second requirement deals with the demonstration of the absence of corrosion damage. There is no indication of corrosion damage in the plant piping. However, although this condition has been met, a corrosion damage database is being prepared as part of the current programme and results are now becoming available.

The third requirement concerns water hammer and associated damage assessments. There has been no experience of water hammer to date in the plant. The very unlikely case of sudden failure of the main shaft of the circulating pump has been analysed. The resulting reactions on the RPV nozzle and the steam generator (SG) elbow have been assessed and the stresses shown to be very low.

Assessment of the seismic margin

For the seismic margin assessment the high confidence of low probability of failure (HCLPF) concept has been used [37]. The safety factors and their standard logarithmic deviations have been estimated using the fragility approach. As the expected seismic event intensity is 8° of MSK-64 scale it has been proved that the calculated HCLPF is higher and that a seismic margin for this piping system exists. However the seismic upgrading of safety related piping systems will be completed by the end 1992 and early 1993.

Stress state analysis of higher stressed components

Information on expansion stresses primary, secondary and peak stresses including fatigue damage evaluation is available [36, 37]. The higher stress sections resulting from NOC and in combination with seismic loading are also available [37]. An in-depth three dimensional finite element analysis has been carried out. All the higher stressed piping components and sections, particularly the welds, are being analysed in terms of stress state and integrity analysis.

Full scale experiments with 'through wall' cracks

The full scale experiments include the reactor pressure vessel 'safe end', pressurizer nozzle 'safe end' and the SG elbow with circumferential and part of longitudinal welds. 'Through wall' cracks are located in the most critical cross-sections of dissimilar welds and T-type welds in the elbow. The preparation of all tests is well advanced [38].

Experiments are carried out in two stages. Stage 1 has three objectives. The first objective is to show that under seismic and operating loading there would be no initiation of crack growth. If the seismic loading is high and the initiation of crack growth and even some crack growth is expected, the tests will be stopped at the initiation level. Crack initiation is an important event with respect to the repeated loading under seismicity. The second objective is to compare the predicted values with the real behaviour of the component. The third objective is the measurement of crack opening displacement (COD) to improve the leak rate computations. Each test consists of the following steps using a model which is identical to the component from the power plant.

A 'through wall' crack of a length related to the leak rate criteria is machined into the critical section. This 'through wall' crack is sealed to prevent leakage when internally pressurized. The component is heated to the operating temperature and loaded to operating pressure. The bending moment is increased in small steps. The increase of the applied moment is stopped either if crack initiation is recognized by acoustic emission or if the total load is equivalent to the superimposed seismic load, but no more than 0.5 of the limit load.

The objective of the stage 2 tests is to prove that the margins predicted from the J–R curve are in agreement with the component behaviour. The initial crack length chosen is about 2 cm in order to check the validity of the J–R curve of large components.

Leak rate tests

These tests are to show that the predicted leak rate is representative of component behaviour. For this purpose the experimental set-up in steam water loop is used. The main features can be characterized as follows [39]:

- the thermodynamic parameters of fluid are 12.5 MPa, 280°C;
- the small diameter pipe (outside diameter 89 mm) and the full scale pressurizer surge line pipe (outside diameter 246 mm) with the same t/R ratio;
- 'through wall' circumferential cracks of various lengths are introduced and sharpened by fatigue loading in one type of dissimilar weld;
- the tested pipes are loaded by bending moments (three different values) to provide crack openings;
- the leak rate is measured by various methods and predictions will be compared with measured values.

The LBB concept requires the installation of three independent leak diagnostic systems; the baseline calibration of background noise induced by fluid flow is mandatory on this set–up for all acoustic emission systems which will be installed on the primary piping.

Fracture and corrosion database

The fracture and corrosion database will include specific J–R curves and sigma–epsilon curves for different specified temperatures. Where archive material is available three J–R and sigma–epsilon curves at room temperature and maximum temperature will be obtained. In cases where archive material is not available, material representative of the plant specification will be used. In this latter case three heats and two samples at each temperature will be tested.

In the case of the Jaslovske Bohunice plant, some archive material is available. For components and welds where such material is not available, three representative type materials will be tested. There will be additional support from tests on service exposed material and these will be evaluated to provide conservative lower bound curves.

The initial requirement on the level of corrosion damage has been mentioned already. It is necessary to prove that corrosion mechanisms, mainly stress corrosion cracking and combined corrosion fatigue damage are not dominant mechanisms.

Demonstration that there is no corrosion cracking can be carried out by the examination of plant components and by supporting experimental work utilizing the known plant history conditions of water chemistry, etc. The present work is aimed at extending the database in this context. Further work is also under way to extend the water chemistry conditions to establish the margin that exists between no corrosion cracking and the conditions which could lead to such cracking.

3.3.1. Overview of available WWER-440 model 230 data from the Czechoslovak programme

The application of the LBB concept generally covers the primary circuit piping and components. The assessment of these structures is crucial from the point of view of the safety of the plant. For WWER-440 model 230 plants, consideration also has to be given to the steam and feedwater lines. In a seismic margin review assessment the HCLPF coefficients for these lines show lower safety margins (lower HCLPF values) than for the primary piping. In addition to this, a main steam line break in a WWER-440 model 230 unit could have implications for a core melt accident even though no probabilistic fault tree assessment is available for such a case. It is therefore necessary to apply the LBB concept approach or integrity assessment to steam and feedwater lines as well to primary piping and components.

TABLE I. PRESCRIBED CHEMICAL COMPOSITION OF THE BASE MATERIALS AND HETEROGENEOUS JOINTS — WELDING MATERIALS IN THE PIPING SYSTEMS OF WWER-440

Type (denotation) of the materials	Application	Chemical composition in %										
		C	Mn	Si	Ni	Cr	Mo	S	P	V	Ti	Comment
08X18N12T	Pipeline of primary circuit DN500	max. 0.08	max. 2.0	max. 0.8	11-13	17-19	-	max. 0.020	max. 0.035	-	5 x C max. 0.6	
08X18N10T	Pipeline DN200 to pressurizer	max. 0.08	max. 2.0	max. 0.8	9-11	17-19	-	max. 0.020	max. 0.035	-	5 x C max. 0.7	
22K	Pressurizer P	0.19-0.26	0.75-1.00	0.20-0.40	max. 0.30	max. 0.40	-	max. 0.025	max. 0.025	-	-	Cu max. 0.30
ST20	Pipeline for steam (V-1)	0.17-0.24	0.35-0.65	0.17-0.37	max. 0.25	max. 0.25	-	max. 0.040	max. 0.035	-	-	
12022.1 (CSN)	Pipeline for steam and feed water	0.15-0.22	0.50-0.80	0.17-0.37	max. 0.25	max. 0.25	-	max. 0.040	max. 0.040	-	-	Cu max. 0.25
15CH.2MFA	Reactor pressure vessel (RPV)	0.15-0.18	0.30-0.60	0.17-0.39	1.0-1.5	1.8-2.3	0.5-0.7	max. 0.012	max. 0.010	-	-	Cu max. 0.08
SV.10CH16N25AM6 (electrodes)	Safe-end RPV Transient weld joint with mater.15CH.2MFA	0.05-0.12	0.8-2.0	1.0	23-27	14-17	5-7	max. 0.020	max. 0.030	-	-	
SV.04CH19N11M3 (welding wire)	Welding joint with reducer 0CH18N10T	max. 0.06	1.0-2.0	max. 0.60	10.0-12.0	18.0-20.0	2.0-3.0	max. 0.018	max. 0.025	-	-	
EA 395/9 (electrodes)	Pressurizer Safe-end Transient weld joint material P -22K	max. 0.12	1.0-2.2	0.3-0.7	22.0 27.0	13.5-17.0	4.5-7.0	max. 0.018	max. 0.025	-	-	N 0.10-0.15
EA 405/10T or SV 04CH19N11M3 (welding wire, electrodes)	Welding joint with reducer 0CH18N10T	max. 0.10	1.15-3.10	max. 0.60	9.0-12.0	16.8-19.0	2.0-3.5	max. 0.025	max. 0.025	-	-	

Preliminary results of assessments provide an indication of the critical sections on which non-destructive testing has to be concentrated. On completion of full integrity assessments the critical sections will be more closely defined.

Main circulating piping

WWER-440 model 230 primary circuit and ECCS was not designed in accordance with the DEGB principle and the containment does not meet western requirements. Stations were intended to be constructed exclusively on low seismicity sites in accordance with regulations in the former USSR. The safety concept of these reactors was based on accident prevention through design in the late 1960s, i.e. through high quality fabrication and control with only limited design basis accident scope and accident mitigation measures.

The primary circuit with inner diameter 500 mm consists of six loops. Each loop has a main isolation valve. The loop is longer than those of western design, with several bends and elbows. The austenitic steel type 0Ch18N10T and 0Ch18N12T is used (Ti stabilized, 18% Cr, 10 and 12% Ni). The chemical composition as well as the mechanical properties are summarized in Tables I and II. All piping components were forged except the cast main valve gate and main circulating pump body. The elbows were longitudinally welded with thickness much larger than its design requirement. Original design is without antiseismic measures, pipe whip protection or jet impingements shields. The primary piping layout is shown in Figs 1(a) and 1(b).

There are 78 circumferential welds which have been made on site. Their schematic position within each loop is seen in Fig. 7. These welds, together with longitudinal welds of elbows, are of great concern with respect to the LBB status.

Heavy components supports are on the RPV, the steam generator, the MCP and the main isolation valve. With respect to the stability due to seismic loading the main concern is the RPV support. The schematic view is shown in Fig. 8.

There is a dissimilar weld between the RPV nozzle and the 'safe end'. The schematic view of this weld is shown in Fig. 9. Note the two layers of cladding and weld.

In order to determine the most critical areas within the primary circuit, it is necessary to consider the combination of material and stress state. The aforementioned welds and their position within circuit provides the material aspect. The stress state aspect could be recognized from Fig. 10 where results from the assessment for NOC and SSE of 5° MSK-64 are shown.

The fatigue damage assessment demonstrated that the most critical locations are the primary piping to surge lines nozzles and elbow to pipe connections.

In general it is possible to conclude that data on NOC assessment including fatigue damage are now available [36–39]. All the piping components have been assessed analytically and in detail using three dimensional finite element analysis. The stress state in all NOC can be described for all components with a high degree of confidence. The assessment including NOC, seismic loading, water hammer, etc. was not carried out until recently. Data from the former USSR are not available but some work was carried out [32–34].

Results shown on Fig. 10 are from the Czechoslovak LBB programme [39], where all safety related piping (main circulating pipes, surge lines, steam and feed water lines) were

TABLE II. MECHANICAL PROPERTIES OF BASE AND WELDING MATERIALS WITH CHEMICAL COMPOSITION ACCORDING TO TABLE I

Type (denotation) of the materials	Temperature 20° C				Temperature 300° C			
	Ultimate strength σ_U	Yield point σ_y	Ductility A5	Contraction z	Ultimate strength σ_u	Yield point σ_y	Ductility A5	Contraction
	MPa	Mpa	%	%	MPa	MPa	%	%
08X18N12T	min. 50	min. 22	-	min. 55	min. 38	min. 18	-	min. 45 (50)
08X18N10T	min. 50 (49)	min. 20 (19)	min. 38 (35)	min. 40	min. 36 (tepl.	min. 18 (17) 350° C)	min. 25	min. 40
22K	44 - 60	min. 22	min. 21 (18)	min. 45 (40)	min. 36	min. 19	18	40
ST20	min. 42	min. 25	-	min. 55	min. 30	min. 20 (17)	-	-
12022.1	45 - 58	min. 26	min. 21	-	-	min. 20.2 (226° C)	-	-
15CH 2MFA	59.2	44.8	-	50	51.4	40.7	-	-
SV04CH19N11M3	56	29.5	-	45	40.8	18.3	-	-
EA 400 T	69	47	-	40	51.4	41.7	-	-
EA 395/9	60	37	13	15	50	35	-	-

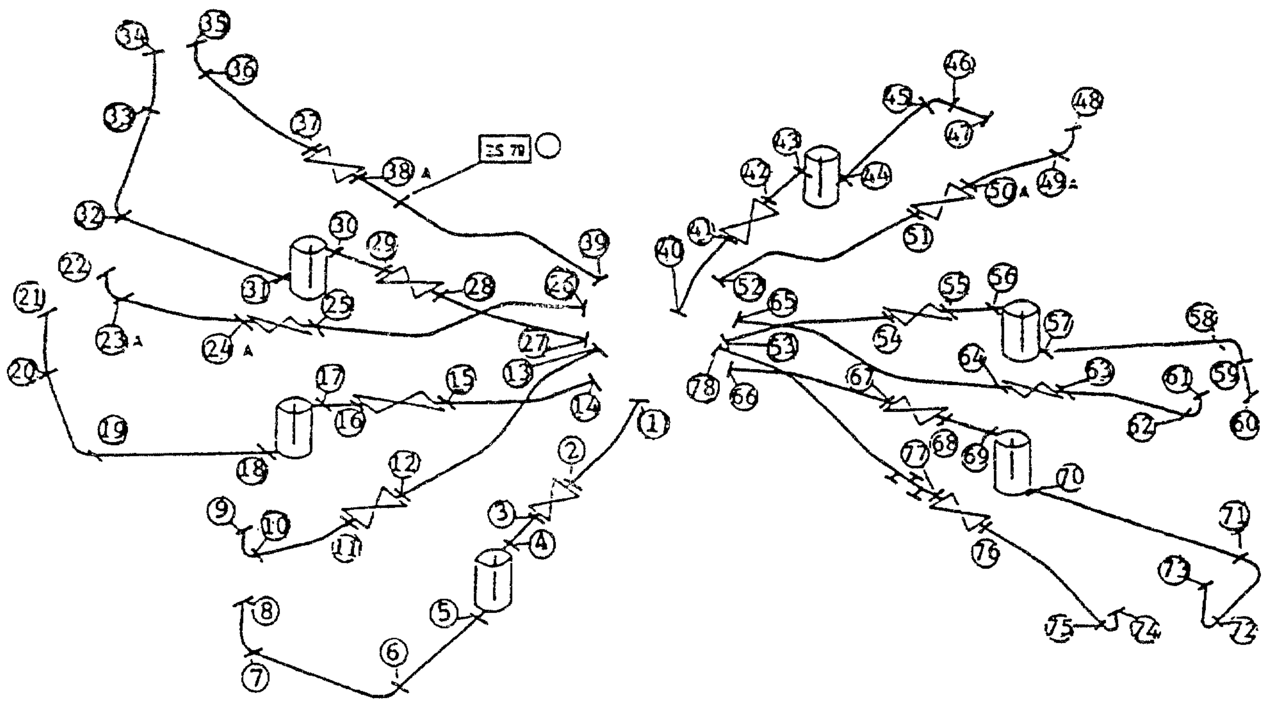


FIG.7. Schematic view of the 78 circumferential welds made on-site for the WWER-440 model 230 reactor.

analysed as complete system. It is important to note that all results derive from piping systems in their present state which is without antiseismic measures. The assessment with proposed antiseismic measures is now being carried out and will be completed by the beginning of 1993.

For the assessment of defects including fatigue crack growth, it is necessary to have details of NDE and defects classification. Due to technical problems and lack of most advanced techniques for the NDE of austenitic base and weld metal piping there is very few volumetric data available. Most of the data are from the outside surface inspections. Only recently modified inspections techniques have been used to inspect internal surfaces of critical sections. No systematic database is available for the inside surface and 100% volumetric control. The database of all defects left in circumferential welds made on site (all of them under the allowable limit) will be available for the Jaslovske Bohunice plant and will also be used for the LBB assessment.

Pressurizer surge lines

There are three pressurizer surge lines, two of which are connected to the bottom side of the pressurizer, and the third of which goes on the top. The layout and dimensions are seen in Fig. 11, the material is again austenitic steel 08Ch18N10T. The maximum design parameters are inside pressure 14 MPa and 335°C temperature and the principal dimensions of the pipe is 245 mm × 18 mm. All piping components were manufactured in the USSR and there is very little information on wall thickness tolerances, out of roundness, etc. The calculations performed used the maximum code allowable values. Data from the USSR on the assessment of these lines are not available. Most comprehensive data available are from the

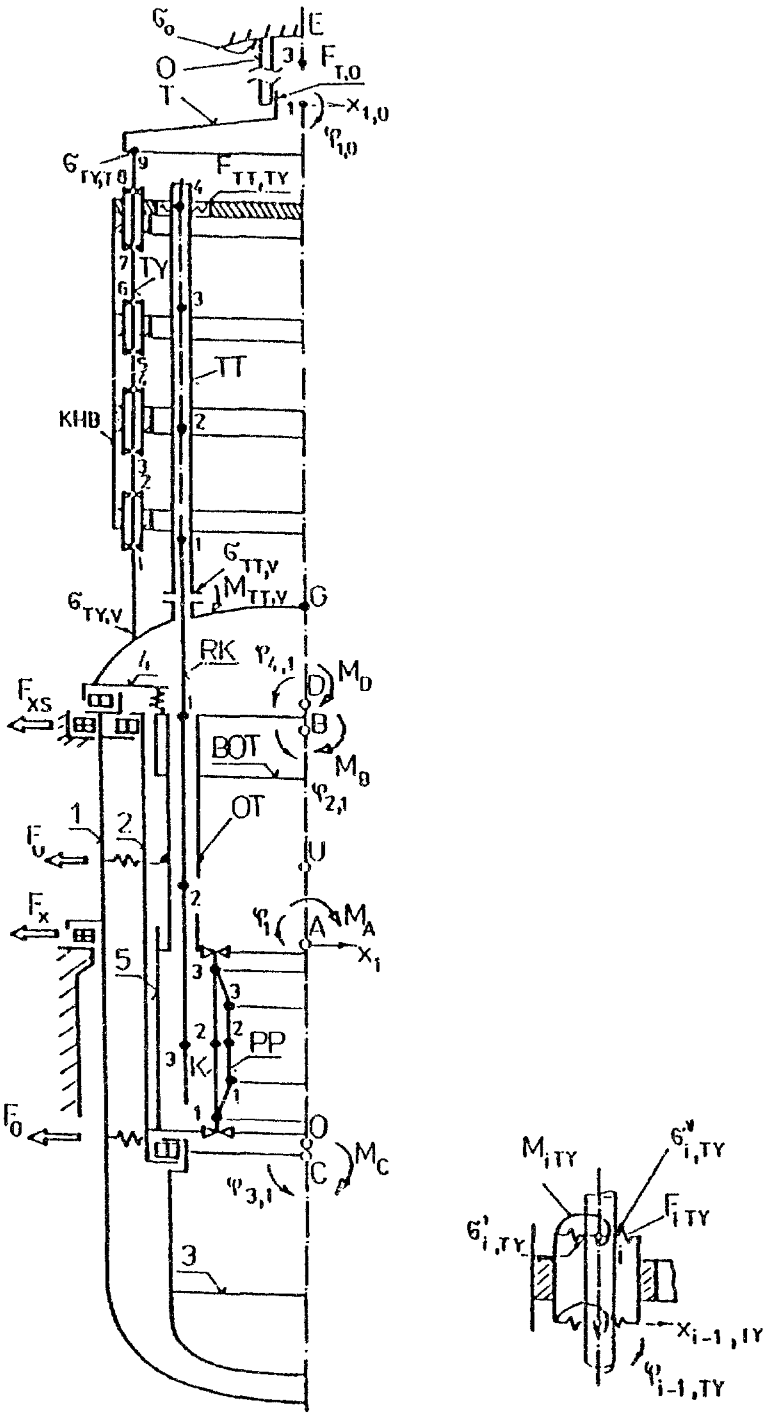
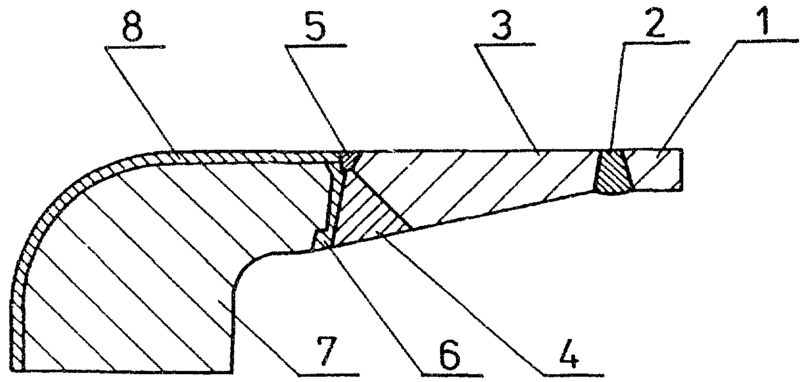


FIG. 8. Schematic view of the RPV with the principal loading scheme and support arrangement.

Czechoslovak LBB programme. Figures 12(a)–12(c) show all three pressurizer surge lines with high stressed cross-sections for the NOC and NOC plus SSE (5° MSK-64). There is an attempt to prepare a database of defects left in on-site made welds and defects found by ISI.

Sections of the pressurizer surge lines could be influenced by temperature stratification. Data from Loviisa NPP and other WWER plants have been collected and a conservative assessment method will be used to include available results in any LBB assessment.



Part	Material	Name of part
1	OCH18N10T	Base metal, main coolant pipe (MCP)
2	EA 400/10T	On site weld with MCP
3	OCH18N10T	Safe end
4	sv04CH19N11M3	Factory made safe end weld
5	sv04CH19N11M3	with RPV nozzle
6	sv04CH19N11M3 sv10CH16N25AM6	Austenitic cladding with semilayer
7	15CH2MFA	RPV nozzle
8	sv04CH19N11M3 sv10CH16N25AM6	Austenitic cladding with semilayer

FIG. 9. The RPV nozzle of internal diameter 500 mm with dissimilar weld to the safe end with specification of materials used.

There is also a dissimilar weld between the pressurizer nozzle made of 22K steel and austenitic 'safe end'. A schematic view of the nozzle and dissimilar weld is given in Fig. 13 (see also Tables I and II). The assessment of this weld is also part of the Czechoslovak LBB programme.

Main steam line

The design parameters are 5.9 MPa pressure and 258°C temperature. The dimensions of the pipe are 456 mm × 16 mm. The negative tolerance in wall thickness is 15%, the minimum wall thickness of bends and elbows is 10 mm and the permissible out of roundness 8%. The material used is ST 20 (see Tables I and II).

The principal layout is shown in Fig. 14 where also the highly stressed cross-sections for the NOC and NOC plus SSE (5° MSK-64) are given. Results from the former USSR are not available. These results are part of the Czechoslovak LBB assessment.

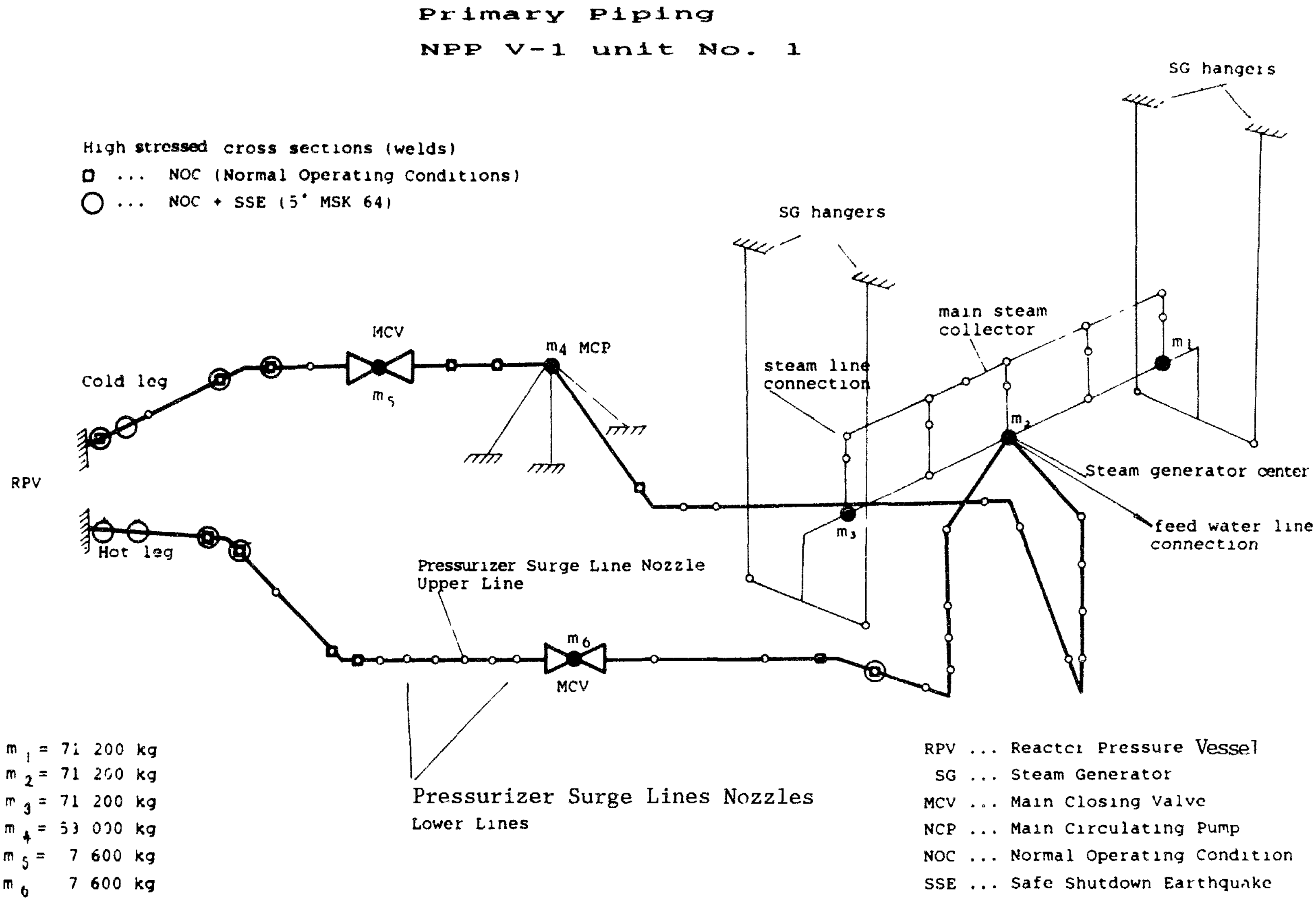


FIG. 10. Primary piping with high stressed cross-sections for the NOC and NOC plus SSE (5° MSK 64).

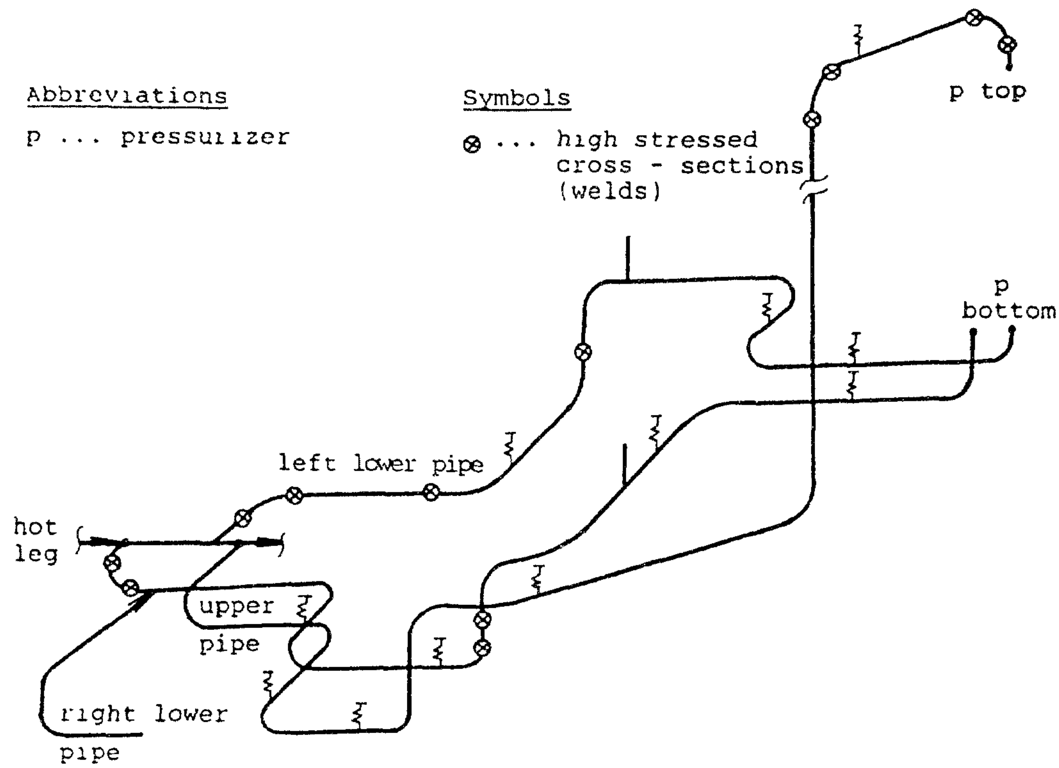


FIG. 11. Schematic view of the pressurizer surge line layout.

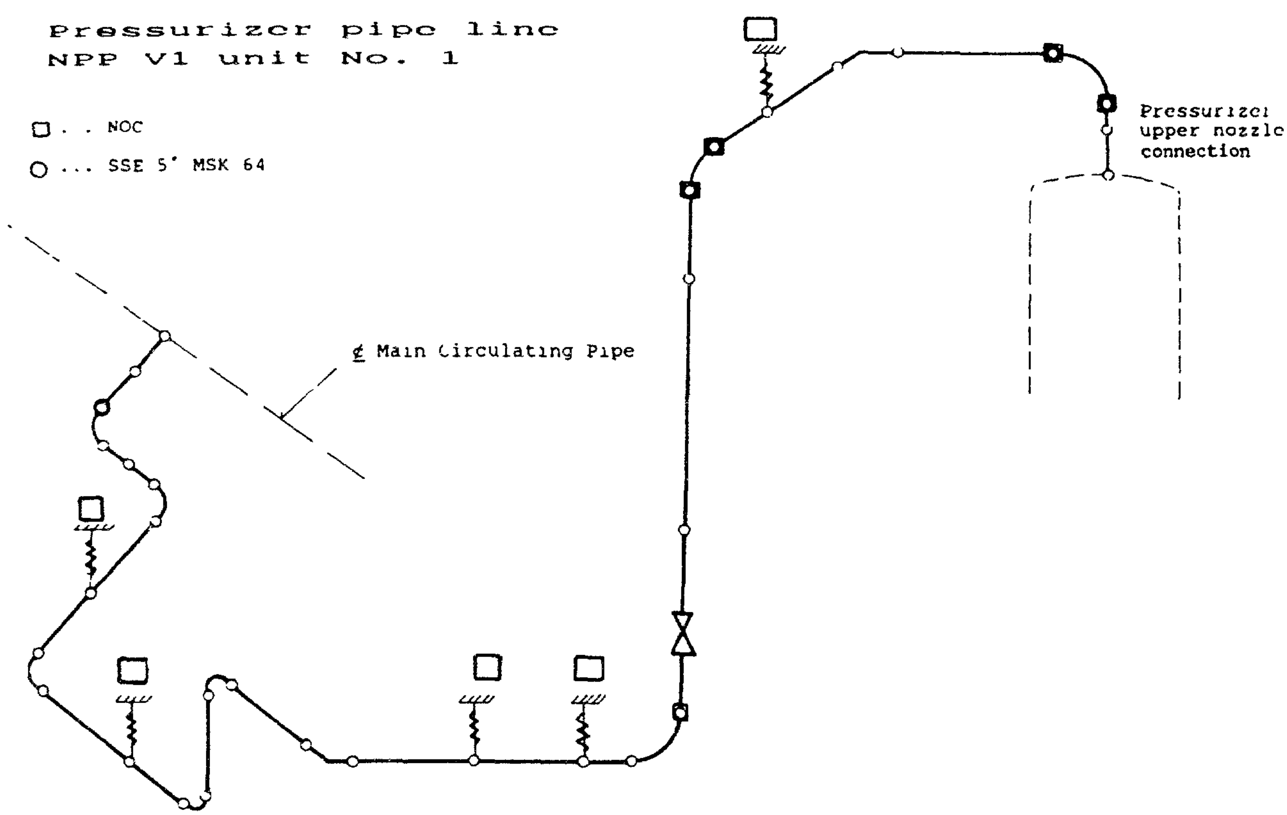


FIG. 12(a). Pressurizer surge lines with high stressed cross-sections for the NOC and NOC plus SSE (5° MSK 64).

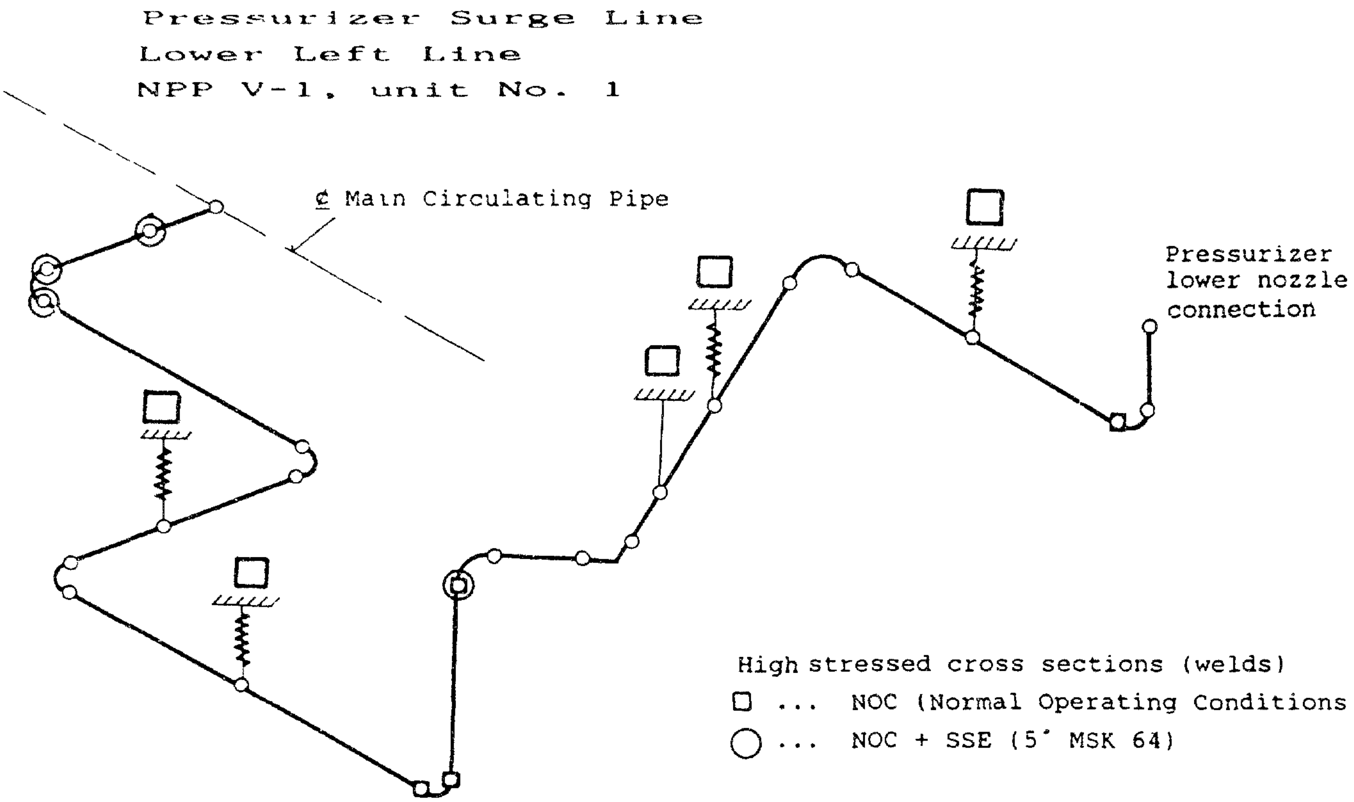


FIG. 12(b). Pressurizer surge lines with high stressed cross-sections for the NOC and NOC plus SSE (5° MSK 64).

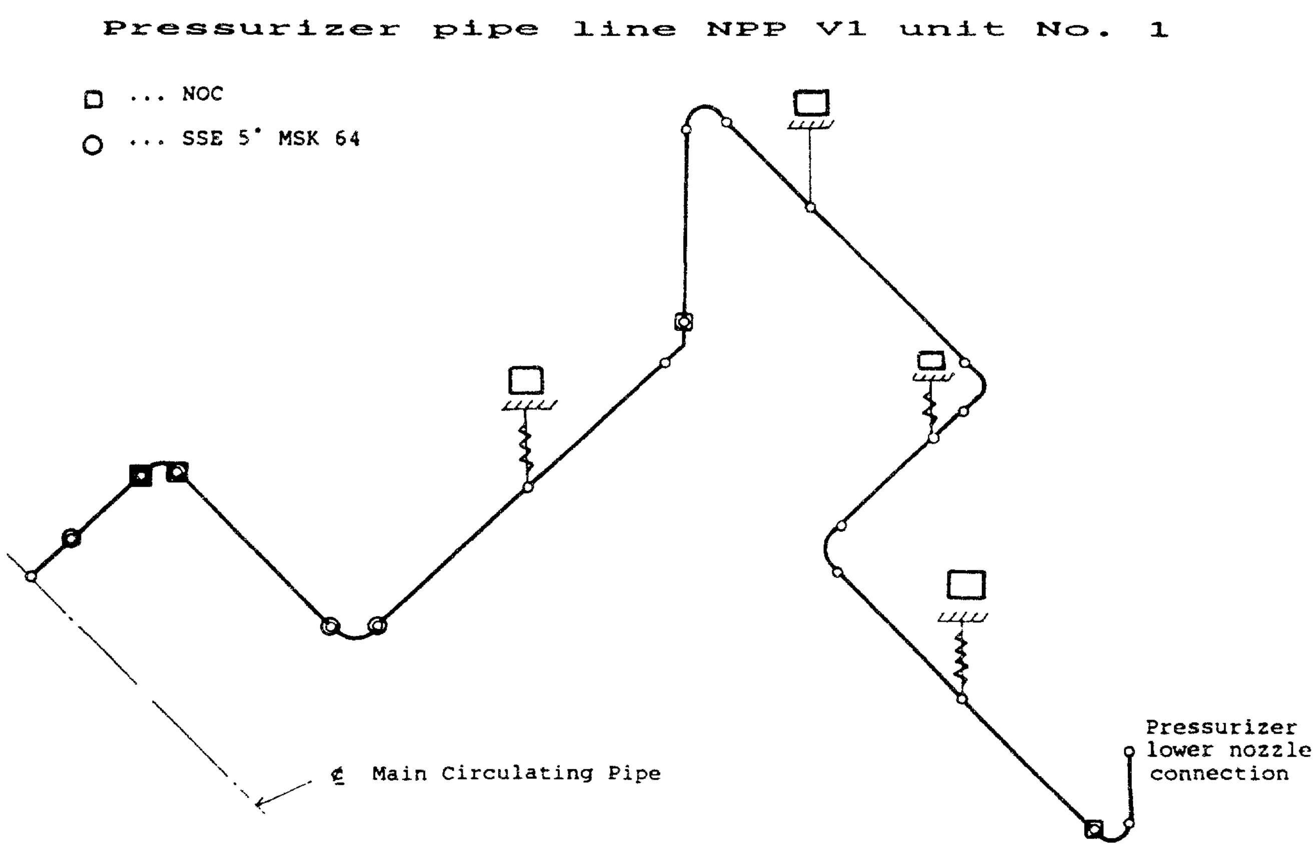


FIG. 12(c). Pressurizer surge lines with high stressed cross-sections for the NOC and NOC plus SSE (5° MSK 64).

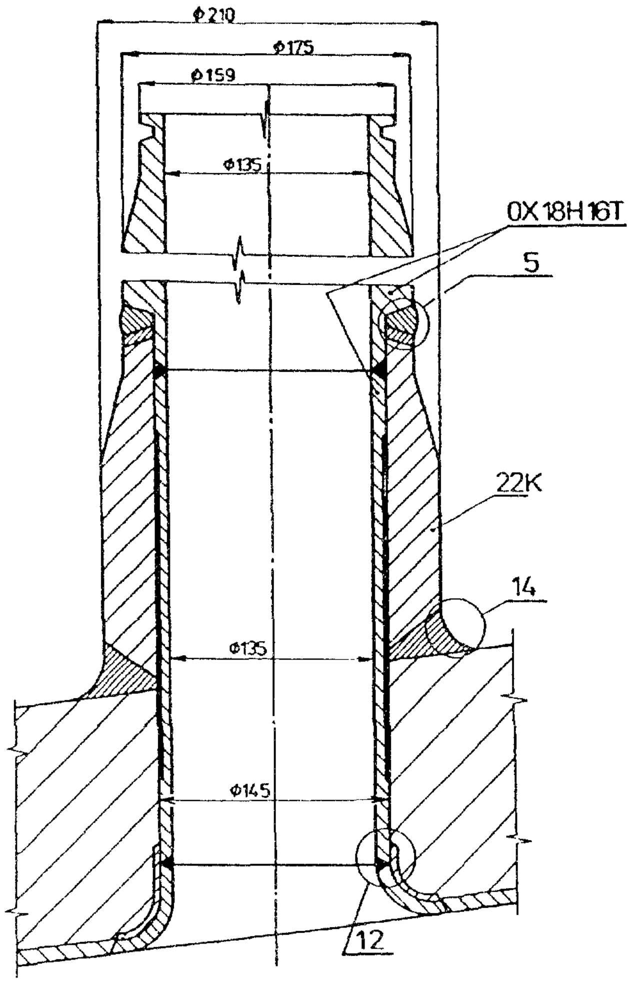


FIG. 13. Pressurizer dissimilar weld with austenitic safe end (22K/08X18N10T).

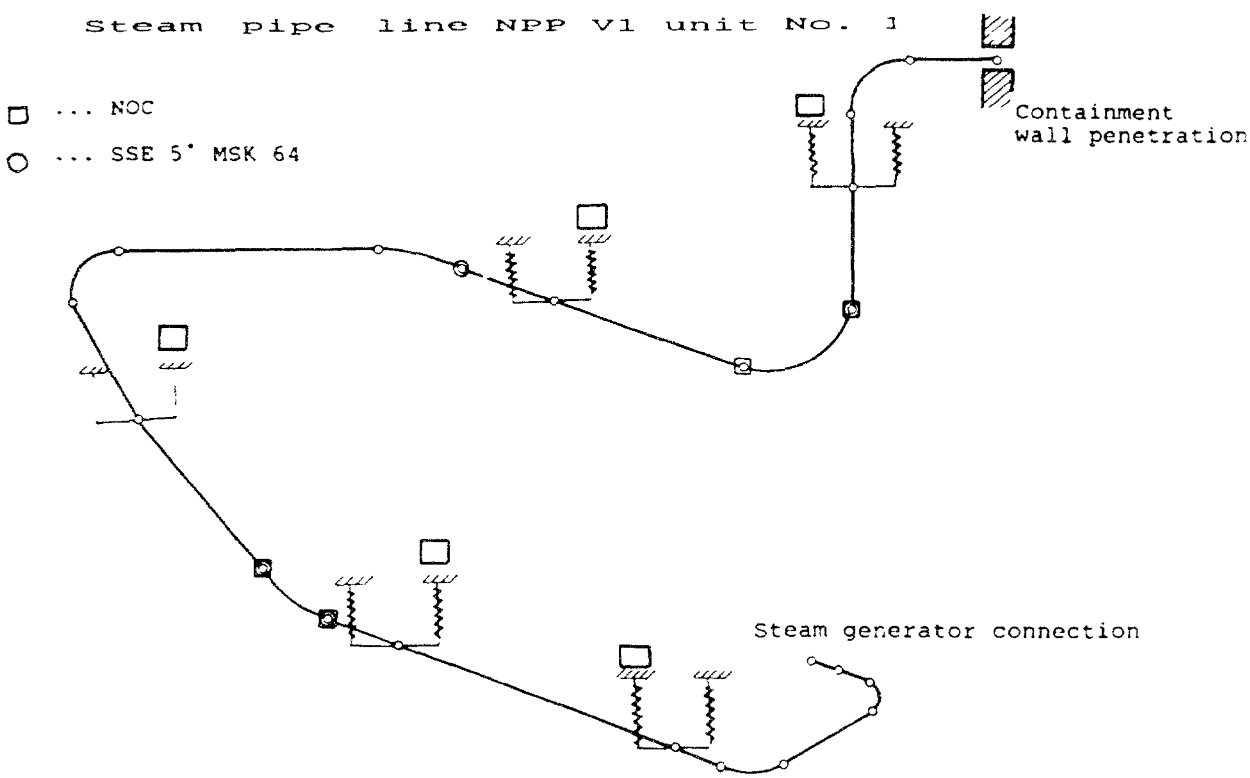


FIG. 14. Steam line with high stressed cross-sections for the NOC and NOC plus SSE (5° MSK 64).

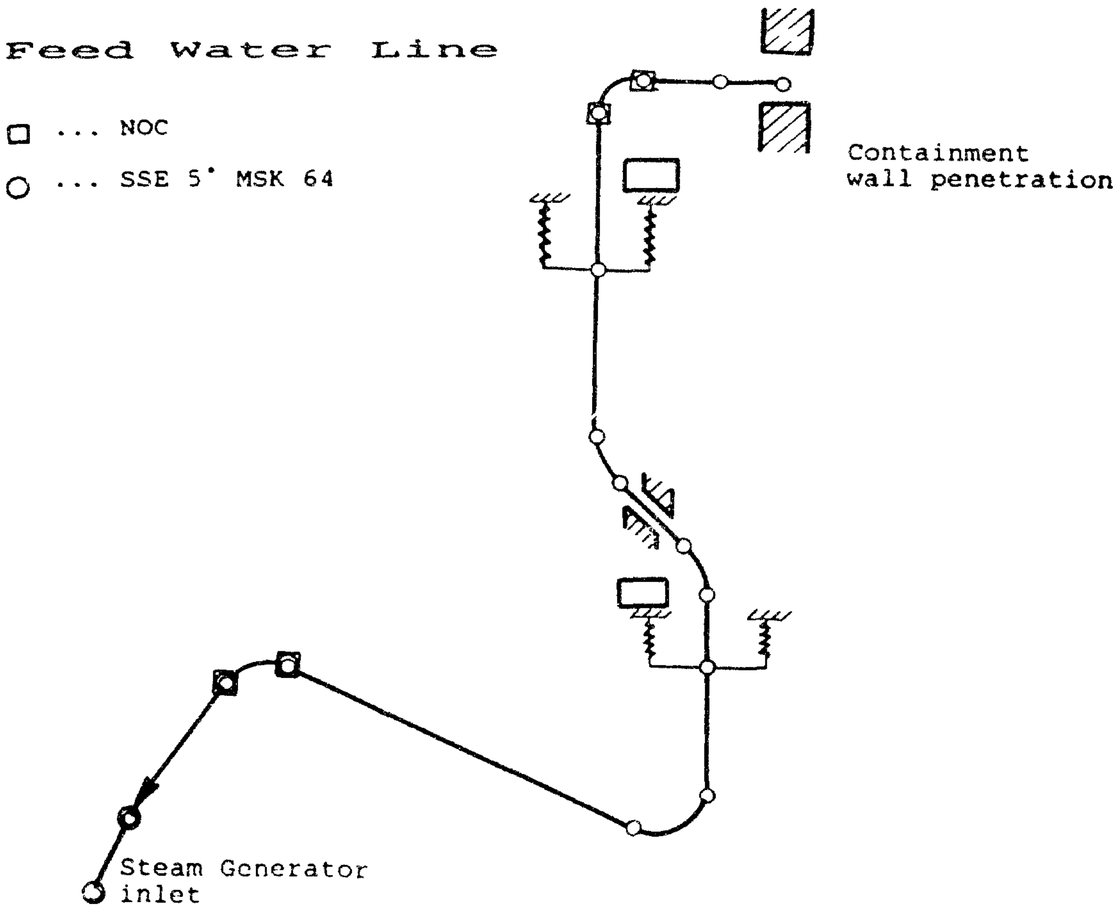


FIG. 15. Feedwater line with high stressed cross-sections for the NOC and NOC plus SSE (5° MSK 64).

For the defect assessment a similar database will be prepared as for the pressurizer surge lines.

Feedwater line

The design parameters are 6.7 MPa pressure and 226°C temperature. The dimensions of the pipe are 273 mm × 16 mm. The negative tolerance is 12.5%, minimum wall thickness of bends and elbows is 10 mm and permissible out of roundness is 8%. The material used is ST 20 or CSN 12022.1. Inside the hermetic zone only the NOC feedwater line is located; the emergency feedwater system is outside the hermetic zone and is connected directly to NOC feedwater line.

The principal layout is shown in Fig. 15 where there are also highly stressed cross-sections for the NOC and NOC plus SSE cases (5° MSK-64). Results from the former USSR are not available. These results are part of the Czechoslovak LBB programme.

For the defect assessment, a similar database will be prepared as for the other pipe lines.

REFERENCES

[1] CLOUD, R.L., et al., Residual Life Assessment of Major Light Water Reactor Components: Overview, Vol. 1 (SHAH, V.N., MACDONALD, P.E., Eds), Rep. NUREG/CR-4731, USNRC, Washington, DC (1987).

[2] GOODISON, D., "The acceptability of the leak before break criterion", paper presented at Steering Committee Mtg under the IAEA Extrabudgetary Programme on the Safety of WWER-440 Model 230 NPPs, Vienna, 1991.

[3] SWAMY, S.A., PALUSAMY, S.S., ADAMONIS, D.C., Criteria and Requirements to Design for Leak Before Break, PVP Vol. 210-1, Codes and Standards and Applications for Design and Analysis of Pressure Vessel and Piping Components, American Society of Mechanical Engineers, New York (1991).

[4] STURM, D., STOPPLER, W., Versuche zum Versagens-und Bruchverhalten von Rohren mit Langs-und Umfangsfehlern, Staatliche Materialprüfungsanstalt (MPA) Stuttgart, Abschlussbericht, MPA, Auftrags-Nr. 8180000, Stuttgart (1989).

[5] Modification of General Design Criterion 4: Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures, US Federal Register, Vol. 51, No. 70, Office of the Federal Register, Washington, DC (1986) 12502–5.

[6] BARTHOLOMÉ, G., BIESELT, R.W., ERVE, M., LBB for Kraftwerk Union plants, Nucl. Eng. Des. **111** (1989) 3–10.

[7] CLAYTON, A.M., Multi-legged Safety Cases for Vessels Using Structural Integrity Diagrams. Innovative Approaches to Irradiation Damage and Failure Analysis, ASME PVP Vol. 170, New York (1989).

[8] Pozadavky pro sestaveni a obsah bezpecnostnich zprav a jejich dodatku. Postup pro stanoveni prukazu "unik pred roztrzenim". Detekcni systemy uniku z tlakoveho chladiciho okruhu reaktoru. Ustav Jadernych Informaci c.1 (1991).

[9] MILNE, I., et al., Assessment of the Integrity of Structures Containing Defects, CEGB Rep. R/H/R6 Rev. 3, Central Electricity Generating Board, London (1990).

[10] KUPPERMAN, D.S., et al., Leak Detection Technology for Reactor Primary Systems, Nucl. Saf. **28** 2 (1987).

[11] NUCLEAR REGULATORY COMMISSION, Reactor Coolant Pressure Boundary Leakage Detection Systems, Guide 1.45, NRC, Washington, DC (1983).

[12] BARTHOLOMÉ, G., KASTNER, W., "Requirements on an acoustic leak monitoring system for NPPs with respect to mechanical and thermodynamic analysis", paper presented at 19th European Working Group Mtg on Acoustic Emission, Erlangen, 1990.

[13] BARTHOLOMÉ, G., et al., "Calibration of a leakage detection system by thermohydraulic and fracture mechanics analysis", paper presented at 17th MPA Sem., Stuttgart, 1991.

[14] Safety Evaluation of Westinghouse Topical Report Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops, Generic Letter 84-04, USNRC, Washington, DC (1984).

[15] NUCLEAR REGULATORY COMMISSION, Evaluation of Potential for Pipe Breaks, Rep. NUREG-1061, Vol. 3, NRC, Washington, DC (November 1984).

[16] Modification of General Design Criterion 4: Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures, US Federal Register, Vol. 50, No. 126, Office of the Federal Register, Washington, DC (1985) 27006–9.

[17] Modification of General Design Criterion 4: Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures, US Federal Register, Vol. 51, No. 141, Office of the Federal Register, Washington, DC (1986) 26393–9.

[18] Modification of General Design Criterion 4: Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures, US Federal Register, Vol. 52, No. 207, Office of the Federal Register, Washington, DC (1987) 41288–95.
[19] Standard Review Plan: Public Comment Solicited, Federal Register, Vol. 52, No. 167, Office of the Federal Register, Washington, DC (1987) 32626–33.
[20] WICHMAN, K., LEE, S., Development of USNRC standard review plan 3.6.3 for Leak Before Break Application to Nuclear Power Plants, Int. J. Press. Vessels Piping **43** 1–3 (1990) 57–65.
[21] Leak Before Break in Nuclear Reactor Piping (Proc. CSNI Specialists Mtg., Monterey, CA, 1983), Rep. NUREG/CP-0051, CSNI Report No. 82, Nuclear Regulatory Commission, Washington, DC (1984).
[22] Leak Before Break: International Policies and Supporting Research (Proc. Sem. Columbus OH, 1985), Rep. NUREG/CP-0077, Nuclear Regulatory Commission, Washington, DC (1986).
[23] WILKOWSKI, G.M., CHAO, K.S. (Eds), Leak Before Break: Further Developments in Regulatory Policies and Supporting Research, (Proc. Sem. Taipei, 1989), Rep. NUREG/CP-0109, Nuclear Regulatory Commission, Washington, DC (1990).
[24] OECD/NEA COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS, Leak Before Break in Water Reactor Piping and Vessels (COLEMAN, C.E. Ed.), CANDU Owners Group Specialists Mtg., Int. J. Press. Vessels Piping **43** 1–3 (1990).
[25] SMITH, E., et al., Directed Discussion, OECD/NEA COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS, Leak Before Break in Water Reactor Piping and Vessels (COLEMAN, C.E., Ed.), CANDU Owners Group Specialists Mtg, Int. J. Press. Vessels Piping **43** 1–3 (1990).
[26] Osnovnyje Polozhenija po Jadernoj Bezopasnosti. OPB-82 Atomenergoizdat (1982).
[27] Pravila Ustrojstva i Bezopasnoj Ekspluataciji Oborudovanija Atomnych Elektrostancij. Atomenergoizdat (1985).
[28] Normy Rasceta na Procnost Oborudovanija i Truboprovodov Atomnych Energeticeskich Ustanovok. PINAEG–7–002–86 (1987).
[29] Pravila Kontrolja Ocnovnovo Metala, Svarnych Soedinenij i Naplavok Oborudovanija i Truboprovodov AEC. PK–1514.
[30] Osnovnyje Polozhenija po Svarke i Naplavke. OP–1512.
[31] IAEA Experts Conclusion on Integrity of Primary Circuit of the WWER-440 Model 230 Reactor, Vienna, 1991.
[32] Design Analysis of an Accident Involving Probability of the WWER-440 Main Pipeline Rupture, Rep. All-Union Scientific Research Institute for Nuclear Power Plant Operation (VNIIAES), Moscow (1990).
[33] Analysis of 38–200 mm Piping Condition and Assessment of the Primary Piping Reliability as a Whole, Rep. All-Union Scientific Research Institute for Nuclear Power Plant Operation (VNIIAES), Moscow (1990).
[34] Tentative Technical Specifications (procedure) for the Operational Monitoring of the Base Metal and Welding Joints of the V-179 and V-230 Plants Primary Piping at Novovoronezh and Kola NPPs, Rep. All-Union Scientific Research Institute for Nuclear Power Plant Operation (VNIIAES), Moscow (1990).
[35] TUTNOV, A.A., TKACHEV, V.V., "Evaluation of failure probability under the condition of undetermined parameters which effect the brittle strength", paper presented at Soviet–American Conf. on NPP Safety, Moscow, 1990.
[36] ZDAREK, J., et al., Influence of the Loading Conditions on the NPP V1 Primary Piping Life Assessment, Rep. 1378–77–ZVZ–74, Sigma Research Institute (1977) (in Czech).
[37] ZDAREK, J., et al., "Leak before break application on the primary piping of the WWER type nuclear power plants", paper presented at 17th MPA Sem., Stuttgart, 1991.

[38] ZDAREK, J., PECINKA, L., Full Scale Experiments to Support the Czechoslovak LBB Programme of the WWER NPPs, Internal Rep. No. 9541-T, Nuclear Research Institute, Řež (1991).

[39] ZDAREK, J., et al., Leak Before Break Project: Application on the Primary Piping of the WWER Type Nuclear Power Plants, internal report, Nuclear Research Institute, Řež, 1991.

[40] Protokol Obsuzdenija Vypolnenija Raboty po Rascetno–Eksperimentalnomy Isledovaniju Avarii s Razryvom Truboprovodov I Kontura Reaktora VVER-440 s Obosnovaniem Meroprijatij, Snizajuscich Verojatnost Polnovo Razryva do Dopustimovo Urovnja. MAE SSSR, VNIIAES Moskva (1989).

[41] BARTHOLOMÉ, G., SCHULZ, G., "Basic safety and break preclusion", paper presented at 2nd Soviet–German Sem., 1989, Moscow 3.1–3.24.

[42] BARTHOLOMÉ, G., ERVE, M., Leck-vor-Bruch-Verhalten von Rohrleitungen, 3. Fachgespräch Werkstofftechnik, Greifswald, 1989.

[43] BARTHOLOMÉ, G., ERVE, M., "Evaluation of operating plants with respect to structural integrity of components and systems", paper presented at 1st Int. Conf. on Problems of Material Science in Manufacture and Service of NPPs, Leningrad, 1991.

[44] RSK Guidelines for Pressurized Water Reactors, 2nd Edn, 24 January 1979, incl. Appendices to Section 4.2; 3rd Edn, 14 October 1981; later changes of RSK Guidelines Sections 21.1 and 21.2, 1983, Gesellschaft für Reaktorsicherheit mbH, Köln.

[45] Comments on Revision of Section 21.1 of the RSK Guidelines for Pressurized Water Reactors, Appendix 1 to Draft Report on Results of 173rd Session of Reactor Safety Commission, 17 February 1982, Gesellschaft für Reaktorsicherheit mbH, Köln.

[46] KUSSMAUL, K., German basic safety concept rules out possibility of catastrophic failure, Nucl. Eng. Int. (1984) 41–46.

[47] BARTHOLOMÉ, G., STEINBUCH, R., WELLEIN, R., "Preclusion of double ended circumferential rupture of the main coolant line", paper presented at 7th MPA Sem., 1981, 1–26; and NEAD 72 1 (1982) 97–105.

[48] BARTHOLOMÉ, G., KASTNER, W., KEIM, E., "Calculation of leakage area and leakage rate for the design of leakage detection systems", paper presented at SMIRT 11, Tokyo, 1991, 159–164.

[49] BARTHOLOMÉ, G., KASTNER, W., "Requirements on an acoustic leak monitoring system to NPPs with respect to mechanical and thermohydraulic analysis", European Working Group on Acoustic Emission, 19th EWGAE Meeting, Erlangen, 1990, 1–17.

[50] IIDA, K., "Recent issues and researches on nuclear pressure vessel and piping", 7th Int. Conf. on Pressure Vessel Technology, Düsseldorf, 1992.

LIST OF ABBREVIATIONS

AE	Acoustic emission
ASME	American Society of Mechanical Engineers
BWR	Boiling water reactor
DEGB	Double ended guillotine break
ECCS	Emergency core cooling system
FCG	Fatigue crack growth
GDC	General design criterion
HCLPF	High confidence of low probability of failure
ISI	In-service inspection
J–R curve	J–resistance curve
LBB	Leak before break
LOCA	Loss of coolant accident
MCL	Main coolant line
MSK-64	Medvedev–Sponheuer–Karnik macroseismic intensity scale
NDE	Non-destructive examination
NOC	Normal operating condition
NPP	Nuclear power plant
OD	Outside diameter
OECD	Organisation for Economic Co-operation and Development
PSA	Probabilistic safety assessment
PSI	Preservice inspection
PWR	Pressurized water reactor
R	Radius
RPV	Reactor pressure vessel
SG	Steam generator
SSE	Safe shutdown earthquake
t	Wall thickness
TWC	'Through wall' crack
USI	Unresolved safety issue
USNRC	United States Nuclear Regulatory Commission
VNIIAES	All-Union Scientific Research Institute for Nuclear Power Plant Operation (Russian Federation)

HOW TO ORDER IAEA PUBLICATIONS

An exclusive sales agent for IAEA publications, to whom all orders and inquiries should be addressed, has been appointed for the following countries:

CANADA	
UNITED STATES OF AMERICA	UNIPUB, 4611-F Assembly Drive, Lanham, MD 20706-4391, USA

In the following countries IAEA publications may be purchased from the sales agents or booksellers listed or through major local booksellers. Payment can be made in local currency or with UNESCO coupons.

ARGENTINA	Comisión Nacional de Energía Atomica, Avenida del Libertador 8250, RA-1429 Buenos Aires
AUSTRALIA	Hunter Publications, 58 A Gipps Street, Collingwood, Victoria 3066
BELGIUM	Service Courrier UNESCO, 202, Avenue du Roi, B-1060 Brussels
CHILE	Comisión Chilena de Energía Nuclear, Venta de Publicaciones, Amunategui 95, Casilla 188-D, Santiago
CHINA	IAEA Publications in Chinese China Nuclear Energy Industry Corporation, Translation Section, P O Box 2103, Beijing IAEA Publications other than in Chinese China National Publications Import & Export Corporation, Deutsche Abteilung, P O Box 88, Beijing
FRANCE	Office International de Documentation et Librairie, 48, rue Gay-Lussac, F-75240 Paris Cedex 05
HUNGARY	Librotrade Ltd , Book Import, P O Box 126, H-1656 Budapest
INDIA	Oxford Book and Stationery Co , 17, Park Street, Calcutta-700 016 Oxford Book and Stationery Co , Scindia House, New Delhi-110 001
ISRAEL	YOZMOT Literature Ltd., P O Box 56055, IL-61560 Tel Aviv
ITALY	Libreria Scientifica Dott Lucio di Biasio "AEIOU", Via Coronelli 6, I-20146 Milan
JAPAN	Maruzen Company, Ltd, P O Box 5050, 100-31 Tokyo International
PAKISTAN	Mirza Book Agency, 65, Shahrah Quaid-e-Azam, P O Box 729, Lahore 3
POLAND	Ars Polona, Foreign Trade Enterprise, Krakowskie Przedmieście 7, PL-00-068 Warsaw
ROMANIA	Ilexim, P O Box 136-137, Bucharest
RUSSIAN FEDERATION	Mezhdunarodnaya Kniga, Sovinkniga-EA, Dimitrova 39, SU-113 095 Moscow
SLOVAK REPUBLIC	Alfa, Publishers, Hurbanovo námestie 3, 815 89 Bratislava
SOUTH AFRICA	Van Schaik Bookstore (Pty) Ltd, P O. Box 724, Pretoria 0001
SPAIN	Diaz de Santos, Lagasca 95, E-28006 Madrid Díaz de Santos, Balmes 417, E-08022 Barcelona
SWEDEN	AB Fritzes Kungl. Hovbokhandel, Fredsgatan 2, P O Box 16356, S-103 27 Stockholm
UNITED KINGDOM	HMSO, Publications Centre, Agency Section, 51 Nine Elms Lane, London SW8 5DR
YUGOSLAVIA	Jugoslovenska Knjiga, Terazije 27, P O Box 36, YU-11001 Belgrade

Orders from countries where sales agents have not yet been appointed and requests for information should be addressed directly to:

Division of Publications
International Atomic Energy Agency
Wagramerstrasse 5, P.O. Box 100, A-1400 Vienna, Austria