SKI Report 95:60

Reliability of Piping System Components

Volume 3: A Bibliography of Technical Papers and Reports Related to Piping Reliability

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A Bibliography of Technical Papers and Reports Related to Piping Reliability

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SUMMARY

1. Background

Reflecting on older analysis practices, passive components failures seldom receive explicit treatment in PSA. to expand the usefulness of PSA and to raise the realism in plant and system models the Swedish Nuclear power Inspectorate has undertaken a multi-year research project to establish a comprehensive passive components database, validate failure rate parameter estimates and model framework for enhancement of integration passive components failures in existing PSAs. Phase 1 of the project (completed in Spring 1995) produced a relational data base on worldwide piping system failure events in nuclear and chemical industries. Approximately 2300 failure events allowed for data exploration in Phase 2 to develop a sound basis for PSA treatment of piping system failure. In addition, a comprehensive review of the current consideration of LOCA in PSA and a comprehensive review of all available literature in this area was undertaken.

2. Implementation

Available public and proprietary database and information sources on piping system failures were searched for relevant information. Specific utilities were asked to contribute their own experience with piping components. Using a relational database to identify groupings of piping failure modes and failure mechanisms, together with insights form extensive reviews of published PSAs, the project team attempt to determine how and why piping fail, and what is the expected frequency of failure.

3. Results

This Phase 2 report contains a comprehensive selection of literature devoted to the piping reliability. Both general and specific topics are covered. More than 800 entries were identified in major bibliographical sources dealing with the subject mater. In addition several dozens of reports conference papers and other material which was identified by the project team were included in the data base.

4 Conclusions

The objective of this report is to summarize for the purpose of further research and development as much as possible material which is important for topics related to the reliability of piping components and reactor pressure boundary related issues. This report is a self standing document, in a sense that it provides the information which are useful for any research on the topic. However, its purpose is primarily to serve as the condensed bibliographical reference source on this topic.

ACKNOWLEDGMENT

The results of the Phase 2 of the project SKI's Reliability of piping system components represents a joint effort between SKI and two contractors ENCONET Consulting , Vienna, Austria and RSA Technologies, San Diego, USA. Volume 1 and 4 were written by Mr. Bengt Lydell of RSA Technologies with the assistance of the project team members. Volume 2 and 3 were written by the ENCONET Consulting team (Mr. B. Tomic, Mr. H. Wimmer, and Mr. P. Boneham) with the assistance of the project team members form SKI and RSA technologies.

The overall project manager who also made a significant contribution to all 4 volumes is Mr. Ralph Nyman of the SKI's Department of Plant Safety Assessment.

The project team greatly acknowledges the encouragement and support from the following individuals and organizations: Mr. Kalle Jänkälä (IVO International Ltd., Finland) for providing pipe failure information from Loviisa Power Plant; Dr. Yovan Lukic (Arizona Public Service, Phoenix, AZ) for providing work order information on leak events at Palo Verde Nuclear Generating Station; Mr. Vic Chapman (Rolls Royce and Associates Ltd., UK) for providing technical papers on risk-based in-service inspection of piping system components; Mr. Jerry Phillips (TENERA Idaho Falls, ID) for introducing us to the work by "ASME Research Task Force on Risk-Based Inspection"; our colleagues at the Nuclear Research Institute, Div. of Integrity and Materials (Rez, Czech Republic) for information on their research on leak-before-break concepts. Authors of this report are specifically grateful to Mr. Mario van der Borst (KCB, the Netherlands), Mr. J. Fossion for information on Belgian PSAs, Mr. J. Munoz for Spanish perspective and Mr. P. Ross for his support.

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1. INTRODUCTION

1.1 Overview of the SKI project on Reliability of piping

The Swedish Nuclear Power Inspectorate (SKI) in 1994 commissioned a multi-year, four-phase research project in piping system component reliability. That is, determination of reliability of passive components, such as pipe (elbow, straight, tee), tube, joint (weld), flange, valve body, pump casing, from operating experience data using statistical analysis methods compatible with today's probabilistic safety assessment (PSA) methodology. Directed at expanding the capability of PSA practices, the project scope includes development of a comprehensive pipe failure event data base, a structure for data interpretation and failure rate estimation, and an analysis structure to enhance existing PSA models to explicitly address piping system component failures.

Phase 1 of the research consisted of development a relational, worldwide database on piping failure events. This technical report documents Phase 2 results. *Interim piping failure data analysis insights are presented together with key piping reliability analysis considerations*. Phase 3 will be directed at detailed statistical evaluations of operating experience data, and development of a practical analysis guideline for the integration of passive component failures in PSA. Finally, Phase 4 will include pilot applications.

A fundamental aspect of PSA is access to validated, plant-specific data and models, and analysis insights on which to base safety management decisions. As an example, in 6,300 reactor-years of operating experience large loss-of-coolant accident (LOCA) has been experienced. Interpretation and analysis of the available operating experience indicates the large LOCA frequency to be about $1.0 \cdot 10^{-4}$/year. Several probabilistic fracture mechanics studies indicate the large LOCA frequency to be $1.0 \cdot 10^{-8}$/year.

Decision makers should be able to confidently rely on PSA. By definition, PSA uses applicable operating experience and predictive techniques to identify event scenarios challenging the engineered safety barriers. *The usefulness of PSA is a function of how well operating experience (including actual failures and incident precursor information) is acknowledged during model (i.e., event tree and fault tree) development*.

The past twenty years have seen significant advances in PSA data, methodology, and application. *An inherent feature of PSA is systems and plant model development in presence of incomplete data*. The statistical theory of reliability includes methods that account for incompleteness of data. Expert judgment approaches are frequently (and successfully) applied in PSA. Legitimacy of expert judgment methods rests on validation of results by referring to the "best available" operating experience. Despite advances in PSA methodology, it remains a constant challenge to ensure models and results accurately reflect on what is currently known about component and system failures and their effects on plant response.

One technical aspect of PSA that has seen only modest R&D-activity is the integrated treatment of passive component failures. Most PSA projects have relied on data analysis and modeling concepts presented well over twenty years ago in WASH-1400. Piping failure rate estimates used by WASH-1400 to determine frequency of loss of coolant accidents (LOCAs) from pipe breaks were based on approximately 150 US reactor-years of operating experience combined with insights from reviews of pipe break experience in US fossil power plants.

In this context, the SKI-project is directed at enhancing the PSA "tool kit" through a structure for piping failure data interpretation and analysis. Phase 2 results are documented in four volumes:

- Volume 1 (SKI Report 95:58). Reliability of Piping System Components. Piping Reliability - A Resource Document for PSA Applications. This is a summary of piping reliability analysis topics, including PSA perspectives on passive component failures. Some fundamental data analysis considerations are addressed together with preliminary insights from exploring piping failure information contained in a relational data base developed by the project team. A conceptual structure is introduced for deeper analysis of passive component failures and their potential impacts on plant safety.

- Volume 2 (SKI Report 95:59). Review of Methods for LOCA Evaluation and PSA LOCA Data Base. The scope of the review included about 100 PSA studies. Unique deviations from the WASH-1400 practice of categorizing LOCAs and estimating their frequencies are presented. This volume gives a detailed overview of LOCA categories and the passive component failures contributing to these categories. The report provides a unique perspective on treatment of LOCA in PSAs but also discuss the issues related to various LOCA categories.

- Volume 3 (SKI Report 95:60) This Report

- Volume 4 (SKI Report 95:61), SLAP-SKI’s Worldwide piping Failure Event Data Base. Includes printouts of failure reports classified as ‘public domain” information not undergoing additional investigation. A large portion of event reports remains subject to interpretation and classification by the project team. The report include graphical presentation of the worldwide operating experience with piping system components. The report also include an overview of fundamental data analysis considerations.

1.2 Need to Address Piping Failures in PSA

Plant risk is highly dynamic. Results from plant-specific PSAs change with advances in data, modeling, operating experience, and changes in system design. The significance of risk contributions from passive component failures tends to become more pronounced by each living PSA program iteration. Shifts in risk topography are caused by strengthened defense-in-depth and decreasing transient initiating event frequencies. As the relative worth of risk contributions from transient initiating events decreases, the relative worth of LOCAs caused by passive component failures increases. The relative contributions from LOCAs and transients identified by early PSA studies (i.e., 1975-1985) may no longer be universally applicable.

Directed at PSA practitioners, this project provides a consolidated perspective on passive component failures. This volume of the Phase 2 reports addresses fundamental issues related to the treatment of LOCA initiators in PSAs, by reviewing the historical development and explaining the logic behind the LOCA categorization and determination of frequency.

An important aspect of the Swedish Nuclear Power Inspectorate's Research project on piping reliability is the consideration of the treatment of LOCAs in PSA studies. Since the time of first comprehensive PSA (WASH-1400, published in 1975), a tremendous amount of work was devoted to probabilistic approaches worldwide. Among other methodological issues, approaches to LOCA definition and determination of LOCA frequencies were often addressed.

One of the main aims of the SKI research project is to enhance the capability of PSA practices through assessment of operational practices and other insights. To enable the application of the collected knowledge directly in PSAs, an assessment of how PSAs have treated LOCAs was performed. An assessment of up to 100 PSA studies, including all the major international projects is documented in this report. At present, significant efforts are placed on determining the failure probabilities and related failure mechanisms on stainless steel and intergranular stress corrosions cracking, and not so much on the other frequent failure mechanisms like corrosion/erosion and similar. This is the other reason why this project stresses the "passive components" issues and the PSA categorization and treatment of those.

1.3 Structure of this report

The purpose of SKI-sponsored project on "Reliability of High Energy Pipework " is to make a step forward in establishing LOCA frequencies on the basis on the wealth of information on operating experience and events which have affected the integrity of pipework at nuclear and industrial facilities worldwide. Many of the PSAs performed nowadays are still referring to WASH 1400 LOCA frequencies which have been established on the basis of expert opinion, nuclear and non-nuclear experience available in early seventies. The SKI project is aiming in filling the gap with both operating experience and scientific advances which have been accumulated since that time.

The major activities under the SKI project on "Reliability of High Energy Piping " are to collect and process the data on actual operational experience of nuclear and non nuclear facilities. In order to select appropriate approaches and to qualify the results which would be generated through the analysis of the data collected form the operational experience of the nuclear power plants internationally, a comprehensive review of literature was performed. The literature review aimed at identifying the sources of information, new methods and approaches as well as results of those which are relevant for the project. The literature review was based on identification of selected keywords in titles and abstracts and included the search of books, reports, conference proceedings, papers and presentations relevant for piping reliability in nuclear and non nuclear industries.

This Appendix presents the data sources used to identify the relevant information, selection process used, and contains the listing of the entire database on LITERATURE with almost 1000 data records.

2. SELECTION OF DATA SOURCES

High energy piping reliability is an area of importance and interest for nuclear and selected non-nuclear industries alike. In nuclear field, research and development relevant for reliability of piping has been on-going since the advent of nuclear facilities, both military and civilian. Piping related activities are included in programs of nuclear laboratories and material laboratories in many countries worldwide. While specific findings like new material compositions etc. may not be freely available, numerous other findings and experience relevant for reliability of high energy pipework is being published internationally.

Used Literature Sources

- International Nuclear Information System **(INIS)**
- UN International Labor Occupational Safety and Health Information database **(CISDOC)**
- US National Institute for Occupational Safety and Health database **(NIOSHTIC)**
- UK Health and Safety Executive's Library and Information database **(HSELINE)**

The most comprehensive literature sources collection system devoted to use for nuclear energy is the International Nuclear Information System (**INIS**) maintained since early sixties by the International Atomic Energy Agency. This information system which contains millions of entries was selected for the comprehensive search of literature relevant for nuclear piping.

As piping reliability issues are relevant for non-nuclear industries too, other international literature sources were searched to identify entries describing either event or methods and approaches which are relevant. In non-nuclear industries, a comprehensive and all inclusive source like INIS does not exist. Entries relevant for piping reliability, operating experience, material properties etc. could be found in virtually hundreds of different databases. As the SKI's project on "Reliability of High Energy Piping" focuses on safety related issues, data sources for non nuclear industries which collect safety relevant literature citations were selected. To enable a broad worldwide search, three major international safety and health databases were selected and thoroughly searched. Those were: **CISDOC**, the UN International Labor Organization's Occupational Safety and Health Information Service's database, **NIOSHTIC** US' National Institute for Occupational Safety and Health (NIOSH) database and **HSELINE**, UK Health and Safety Executive's Library and Information Services database. The details of each of those is provided in section 3.

3. DETAILS OF DATA SOURCES SELECTED

3.1 INIS

INIS, the International Nuclear Information System, is an international bibliographic database. It is produced by the International Atomic Energy Agency (IAEA), in collaboration with participating countries and international organizations. The INIS Secretariat at the IAEA is responsible for the central processing of the database.

The subject scope of INIS is all aspects of the peaceful uses of nuclear science and technology, with emphasis on engineering, energy, safety, and life sciences. In 1992 INIS began covering the economic and environmental aspects of all energy sources. The literature covered includes not only conventional documents, such as journal articles, books, published conference proceedings, etc., but also non-conventional material, such as scientific and technical reports, theses, conference papers, etc., which are not readily available through normal commercial channels. Non-conventional materials constitute about 30 percent of the database.

INIS is compiled from data submitted by 88 national INIS centers and 17 co-operating international organizations. Each center is responsible for cataloguing and indexing all documents within the INIS subject scope which are published within its borders. Input from all centers are collected by the INIS Secretariat and merged into the database. Non-English documents include an English translation of the title. English abstracts are included for 85 percent of the database.

INIS contains millions of entries covering the whole spectrum of nuclear related applications. To limit the information to most recent findings, only those INIS database records which were younger than 1988 were selected. That was accomplished by selecting INIS database available on two CD ROM discs, namely disks 1990-December 1992 and the most recent one 1993 to March 1995. In order to assure that the information which is lost because of cut off in the year 1988, a search of complete INIS library was performed at the IAEA mainframe. The full INIS database search was performed for keywords “PIPE FAILURES” and “PIPE RUPTURES”. While it was confirmed that there are numerous entries older than 1988, the review of titles supported the conclusion that the most relevant documents (for a pipe reliability study performed in the year 1995) would be contained in recent years.

3.2 CISDOC

The CISDOC database, a product of the International Occupational Safety and Health Information Center of the International Labor Organization in Geneva, contains references from over 35 countries to key literature on safety and health at work. Subject areas include:

- *industrial hygiene*
- *occupational medicine*
- *ergonomics*
- *toxicology*
- *safety engineering*
- *environmental stress*
- *accident prevention*
- *physiology*

CIS was formed as the main center within the UN system for collecting and disseminating safety and health information worldwide. It is supported by a network of 50 National Information Centers throughout the world which collect and evaluate relevant literature.

The full collection of entries in CISDOC was searched for relevant entries.

3.3 NIOSHTIC

NIOSHTIC, published by the National Institute for Occupational Safety and Health of US (NIOSH), is a bibliographic database containing references to workplace safety and health literature. The subject areas covered include toxicology, industrial hygiene, occupational medicine, behavioral sciences, epidemiology, ergonomics, pathology, hazardous wastes, physiology, chemistry, and engineering control technology.

The information sources for NIOSHTIC include 150 English-language technical journals, NIOSH publications and reports, references from CIS (the International Labor Organization’s occupational safety and health database), conference proceedings and symposia, English translations of non-English documents acquired by NIOSH, and the personal collections of occupational safety and health researchers.

The full collection of entries in NIOSHTIC was searched for relevant entries.

3.4 HSELINE

Since 1977, the Library and Information Services of the Health and Safety Executive (HSE) has accumulated in a computer database documents relevant to health and safety at work. The database contains citations to HSE and Health and Safety Commission (HSC) publications, together with documents, journal articles, conference proceedings, and legislation in the following areas:

- *Manufacturing Industries*
- *Agriculture*
- *Production*
- *Occupational Hygiene*
- *Explosives*
- *Engineering*
- *Mining*
- *Nuclear Technology*
- *Industrial Pollution*

HSE, the government body responsible for health and safety at work in Great Britain, is the working arm of the HSC, formulated under the Health and Safety at Work etc. Act 1974.

The full collection of entries in HSELINE was searched for relevant entries.

3.5 Retrieval of Records from Data Sources

The retrieval of records from all four selected databases was made through keywords search from both titles and abstracts. The guiding idea for searches was to identify and retrieve as many as possible diversified information on actual operational experience with piping, including studies of operational experience as well as both methods and approaches for determining reliability of pipework. The interest in all kinds of piping damages actually focused selection of search keywords. After review of entries, indexes and thesauruses available, the keywords finally selected for the search in all four databases were as follows:

- *PIPE DAMAGE*
- *PIPE BREAK*
- *PIPE LEAK*

- *PIPE FAILURE*
- *PIPE RUPTURE*
- *PIPE CRACK*

Those keywords were truncated to allow selection of entries even if a keyword would not appear in the exact form. The selection was thoroughly checked against application of keywords like PIPE RELIABILITY or PIPE FRACTURE. Actual number of records selected from every source using the above keywords is summarized in Table 1.

Table 1: Number of records matching specific keywords in sources evaluated

KEYWORDS	*INIS*	*CISDOC*	*NIOSHTIC*	*HSELINE*	*TOTAL*
Pipe damage	114	4	22	52	192
Pipe break	303	12	14	52	381
Pipe leak	333	20	72	153	578
Pipe failure	263	7	23	116	409
Pipe rupture	198	8	17	92	315
Pipe crack	413	3	8	115	539
TOTAL	1624	54	156	580	2414

There have been numerous overlapping entries in the data sources as far as those were identified to contain the same entries. Those were removed through a careful review. In addition, numerous entries which contained one or more selected keywords were devoted to an entirely different subject. Those were also removed. Remaining entries were entered into a custom designed database "**LITERATURE**" which was developed using Microsoft ACCESS database manager. The number of entries from each source is summarized in Table 2

Table 2: Initial number of entries into the database "**LITERATUR**E"

DATA SOURCE	*INIS*	*CISDOC*	*NIOSHTIC*	*HSELINE*	*TOTAL*
NUMBER OF ENTRIES	715	28	42	229	1014

Further search and comparative evaluation, including removal of entries which were of limited interest, resulted in a database having a total of 786 relevant entries. The size of the compacted version of the database is about 1.4 MB.

4. ORGANIZATION OF "LITERATURE" DATABASE

Entries in the **LITERATURE** database were fully reproduced from original sources. For every record there are nine fields, seven of them taken from the original data sources, one assigned by the database (counter) and another one (category) designated during the data entry.

Seven fields in every record selected from the original data source include:

- ***TITLE** of the entry (paper, book, report);*
- ***AUTHOR**(s) name and affiliation as available;*
- ***CORPORATE AUTHOR/CONFERENCE** where the work was presented;*
- ***SOURCE** from where the paper/report could be obtained;*
- ***PUBLICATION YEAR** of the entry;*
- ***LANGUAGE** in which the original entry is prepared;*
- ***ABSTRACT** prepared by the author or a database manager in English language.*

The database assigned field COUNTER is a consecutive numbering system from 1 to 786. There is no relevance nor any special structure in the way the numbers have been assigned.

The "**CATEGORY**" field has been added to every record. The purpose of this additional field is to enable grouping and/or sorting records in accordance to the specific information contained/discussed in specific record. Total of 11 categories were defined. Those are:

DAMAGE PROBABILITY, containing all the records where actual probability of pipe damage is discussed or estimated, including discussion of methods and approaches used for such estimates. A total of 54 records fall into this category

EXPERIENCE/EVENTS is a category containing either description of events where pipe damage has occurred (or have lead to) or the analysis of operational experience of piping system in general. Records on specific processes like corrosion/erosion or similar which would ultimately affect the reliability of piping or operational parameters (like thermal stratification) which have induced piping damage as well as discussion of aging are also contained here. A total of 145 record fall into this category.

RESEARCH/THEORETICAL is a category which contain the records describing various research and development activities, but exclude those where test rigs or similar were used to confirm theoretical results. This category is primarily meant to group sources where new methods or refinement of the approaches are discussed. Several computer codes designed to support fracture mechanics or other approaches are also listed here. A total of 85 records were grouped into this category.

TEST/ANALYSIS is a category meant to group the records where development of methods and approaches using tests are discussed. These include studies of crack growth and behavior toughness of metal structures and similar. A total of 150 records fall into this category.

METHODS/DESIGN/COMPARISON is a category containing records discussing various methods from those used to determine piping reliability (without actually doing so) to those used to model flow and growth rates of cracks. Methods and approaches to improve design or operation as well as comparison of different methods are also grouped within this category. A total of 92 records fall into this category.

ANALYSIS OF BREAK EFFECTS is a category grouping primarily the records where a complex thermal hydraulic analysis was undertaken to determine the effects of a break onto the rest of a facility. While many records describing standard RELAP type calculations have been excluded from the database, some of which are found to be of interest were retained. In this category , some of the records are labeled CRITERIA as those discuss establishment of specific criteria for i.e. acceptable crack sizes etc. A total of 66 records were placed into this category.

LBB JUSTIFICATION is a category specifically designed to group the records related to LBB issues. Entries in this category range from policy making papers to specific testing and research required for the acceptance of LBB. There is some overlap between this and some other categories, but all entries specifically relevant for LBB were included in this specific category. A total of 51 records fall into this category.

INSPECTION METHODS is a category specifically designed to group the methods and approaches for inspection of piping and other components including both the theoretical methods (like risk based inspection prioritization) and technological approaches (like new ISI probes). The purpose of this category is to retain records which are of interest in establishing an improved ISI programme which could positively impact piping reliability. 63 record were grouped in this category.

PRESSURE RIPPLE/WATER HAMMER is a small category meant to group a set of records dealing with this specific failure mode. A total of 9 records belong to this category.

OTHER is a category grouping all the records which were found to be of interest and at the same time could not be logically grouped in any of the other categories mentioned above. Some of the records contained here are those which would simultaneously fit into several other categories, events related to steam generators and similar. Although efforts were made to minimize the number of entries in this category, 71 record were grouped here.

5. PRESENTATION OF THE DATABASE

All the records from the database **LITERATURE** are presented in two distinctive sets.

The first, termed APPENDIX 1, presents the titles and the record numbers grouped by category. To enable an easy selection of records, titles of entries in every of 11 categories discussed in previous section are listed separately. Within a category, the titles are sorted by the publication year in descending order (the most recent ones first). The categories are presented in the following order:

- *Damage probability*
- *Experience/events*
- *Analysis of break effects, Criteria*
- *Inspection methods*
- *LBB justification*
- *Methods/design/comparison*
- *Other*
- *Pressure ripple/water hammer*
- *Research/theoretical*
- *Test/analysis*

APPENDIX 2 contains a full listing of the database. Here the records are presented in category groups following the same order as in the APPENDIX 1. All the information available in a record is presented. If some of the entries are missing (like in several cases Corporate author/conference) those were not available in the original source. Records within a category are sorted in accordance with the original language, publication year (descending) and the alphabetical order of titles. This means that entries with English as the publication language, published in 1994 (the most recent year) and with the title starting with “A” are at top of the list.

APPENDIX 1
"REVIEW OF RECENT LITERATURE-TITLES

CATEGORY
Damage probability
Experience/events
Analysis of break effects, Criteria
Inspection methods
LBB justification
Methods/design/comparison
Other
Pressure ripple/water hammer
Research/theoretical
Test/analysis

APPENDIX 2
"REVIEW OF RECENT LITERATURE-ABSTRACTS"

Title: Fracture Criterion of Japanese Large-Diameter Carbon Steel Cracked Pipe.

Author: Noda, H. et al **Corp. Author:** JAERI

Source: Shibata,-Heki (Ed.), 1991. Trans. 11th SMIRT Conference. Tokyo (Japan). pp 383-394. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan, ISBN 4-89047-060-3.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Criteria / LBB **ID:** 1

Abstract: Our goal is to develop the fracture criterion of the large-diameter carbon steel pipe with a circumferential crack in the leak-before-break (LBB) evaluation. The paper presents the results of Japanese large diameter carbon steel cracked pipe tests, the predictions of the failure load using the various simplified analysis methods and finite element analysis, and these comparisons. The comparisons of the test results with the predictions demonstrated that plastic collapse dominated the fracture of the Japanese large outer-diameter cracked carbon steel pipes. (author).

Title: On modelling, simulation and measurement of fluid power pumps and pipelines.

Author: Weddfelt,-K. **Corp. Author:** LiTH, Dept. Mech. Eng.

Source: Linkoeping Univ. Dissertations (1992), 243 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Water hammer **ID:** 2

Abstract: Pressure ripple in fluid power systems can cause functional problems, incl. fatigue and breakdown of pipes and connections. To examine this problem both the sources of pressure ripple and its transmission properties must be considered. A major source of pressure ripple in fluid power systems is positive displacement pumps which can be modeled as a flow source with an internal source impedance. Special measurement techniques must be developed to determine these source properties experimentally. Pressure and flow ripple propagate through the pipeline as waves. When impedance of system changes, part of the energy in the wave is being transmitted while the remaining part is reflected. Therefore, the mechanism for standing waves to occur is present, causing resonances and possibly large pressure pulsations at certain frequencies. Destructive interference between these waves can be used to design reactive attenuators, which can be used to accoustically separate the source of flow ripple from the rest of the fluid system. A mathematical model of wave transmission is of importance when modelling and measuring ource characteristics of pumps. Such a mathematical model must include transmission and reflection of waves as well as frequency-dependent losses from viscous friction in the fluid. (au).

Title: Multiple reactor pressure tubes rupture probabilistic analysis under operation and seismic loads for RBMK-type reacto

Author: Butorin,-S.L.; Shiverskiy,-E.A. **Corp. Author:**

Source: Shibata,-Heki (Ed.), 1991. Trans. 11th SMiRT Conference. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. M-SD0 p. 127-131.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Damage probability **ID:** 3

Abstract: A series of studies is being conducted with the aim of assessing the probability of damage to a serviceable pressure tube (PT) with regard to the dynamic loads arising with a break of a neighbouring tube (dependent events). This work has already yielded a tentative forecast of the probability of multiple PT damage, allowing for the dynamics of interaction between the broken tube and the surrounding structures. The forecast results are given in Table 3. These data will be verified in the course of further research. In conclusion, it appears necessary to add the following. The above results should be regarded as an integral part of a whole package of work which is expected to yield a fairly reliable forecast of multiple PT damage in RBMK reactors, i.e., an accident brought with the most disastrous effects for this type of reactors. The top-priority research objectives, in our opinion, include the probabilistic assessment of PT stresses, allowing for the dynamic loads, with the possible breaks in the pipelines of the recirculation circuit, water hammers and falls of handling equipment. (author).

Title: Fracture toughness and fatigue crack growth of PWR materials in Japan.

Author: Kansaki, H.; Funada, T.; Morinaka, I.; Koizumi, K. **Corp. Author:**

Source: Japan Society of Mechanical Engineers, Tokyo (Japan). The 1st JSME/ASME Joint Int. Conf. on Nuclear Engineering. Tokyo (Japan). Japan Society of Mechanical Engineers. 1991. 1273 p. v. 1 p. 527-531.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 4

Abstract: Fracture toughness and fatigue crack propagation rate of Japanese PWR primary and secondary piping materials were obtained, in the course to establish Leak Before Break (LBB) criteria based on fracture mechanics. Centrifugally cast stainless steel pipings, a statically cast stainless steel elbow and a forged stainless steel safe end were tested as PWR primary main coolant piping materials. Weld joints by Tungsten Inert Gas Welding and Shield Metal Arc Welding were also tested. Carbon steel pipings were used as PWR secondary main steam and main feedwater piping testing materials. Weld joints by Submerged Arc Welding, Metal Inert Gas Welding and Shield Metal Arc Welding were also tested. Fracture toughness tests were conducted at room temperature and at 310 approx 325degC to obtain J sub I sub C and J-R curves. Fatigue crack propagation rate tests were conducted in air and in simulated PWR primary or secondary water at approx. 310-325 deg C. (author).

Title: Structural and fracture mechanics study of a pipe with a circumferential crack under blowdown-induced loading.

Author: Brosi,-S.; Wanner,-R.; Reichlin,-K.; Schrammel,-D.; Kobes,-E. **Corp. Author:** Paul Scherer Institute (PSI)

Source: Shibata,-Heki (Ed.), 1991. Trans. 11th SMiRT Conference. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. F p. 219-224. ISBN 4-89047-060-3.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Research/theoretical **ID:** 5

Abstract: For an unflawed piping the linear global structural dynamic response after pipe break and undamped closure of a check valve was studied; furthermore with a local pipe model containing a circumferential internal surface crack the influence of bending direction on the stress intensity was investigated for both a uniform bending moment and for the loading as measured in the experiment. Up to ca. 93 ms the bending moments of calculation and experiment agree very well; the agreement for the pipe deflection however is good in quality only. Quantitatively the discrepancy between measured and calculated displacement is surprisingly large. After 93 ms better results can be expected from a calculation with a nonlinear material law which is able to include the high level of plastification occurred in the experiment. But before performing this expensive calculation, the reason for the mentioned displacement discrepancy should be found. From the comparison of the effective stress with the design limit 3 S sub m it could be shown that the KTA design rule is conservative even for the considered flawed piping. Due to asymmetrical loading one crack half is substantially less stressed than the other one.

Title: Short cracks in piping and piping welds. Semiannual report, October 1990--March 1991: Volume 1, No. 2.

Author: Wilkowski,-G.M.; Brust,-F.; Francini,-R.; Ghadiali,-N.; Kilinski,-T.; Krishnaswamy,-P.; Landow,-M.; Marschall,-C.W.; Rahman,-S.; Scott,-P. **Corp. Author:** Battelle Columbus Labs.

Source: Apr 1992. 203 p. Nuclear Regulatory Commission, Washington, DC (United States). Div. of Engineering.Battelle, Columbus (OH).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Research/theoretical **ID:** 6

Abstract: This is the 2nd semiannual report of NRCs Short Cracks in Piping and Piping Welds research program. The program began in 1990 and will extend into 1994. The intent is to verify and improve fracture analyses for circumferentially cracked large-diameter nuclear piping with crack sizes typically used in LBB analyses or in-service flaw evaluations. Only quasi-static loading rates are evaluated since the NRC's International Piping Integrity Research Group (IPIRG) program is evaluating the effects of seismic loading rates on cracked piping systems. Progress for through-wall-cracked pipe involved (1) conducting a 28-inch diameter stainless steel SAW and 4-inch diameter French TP316 experiments, (2) conducting a matrix of FEM analyses to determine GE/EPRI functions for short TWC pipe, (3) comparison of uncracked pipe maximum moments to various analyses and FEM solutions, (4) development of a J-estimation scheme that includes the strength of both the weld and base metals. Progress for surface-cracked pipe involved (1) conducting two experiments on 6-inch diameter pipe with d/t = 0.5 and THETA/pi = 0.25 cracks, (2) comparisons of the pipe experiments to Net-Section-Collapse predictions, and (3) modification of the SC.TNP and SC.TKP J-estimation schemes to include external surface cracks.

Title: Double-ended break probability estimate for the 304 stainless steel main circulation piping production reactor.

Author: Mehta,-H.S.; Daugherty,-W.L.; Awadalla,-N.G.; Sindelar,-R.L. **Corp. Author:** General Electric

Source: Shibata,-Heki (Ed.), 1991. Trans. 11th SMiRT Conference. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. M-SD0 p. 277-282., ISBN 4-89047-060-3

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Damage probability **ID:** 7

Abstract: The SRS production reactors use relatively thin-walled piping for the primary coolant system, a result of a low operating temperature and pressure. The material of construction for the primary pressure boundary is Type 304 stainless steel. These reactors were built in the 1950's. The objective of this paper is to present the methodology and results of a probabilistic evaluation for the direct failure of the primary coolant piping. This evaluation supports the ongoing PRA effort and complements analyses regarding the credibility of a Double-Ended Guillotine Break (DEGB). (author).

Title: Dynamic analysis of reactor internals for the tributary pipe breaks.

Author: Jhung,-M.J.; Choi,-S.; Song,-H.G.; Park,-K.B.; Shon,-G.H. **Corp. Author:** Korea Atomic Energy Research

Source: Shibata,-Heki (Ed.), 1991. Trans. 11th SMiRT Conference. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. J p. 19-24.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Analysis of break effects **ID:** 8

Abstract: This paper investigates the lateral responses of the reactor internals to a 14 inch safety injection nozzle break which is expected to generate the largest loads among the branch line pipe breaks postulated. The effects of two forcing terms, reactor vessel motions and internals hydraulic loads, are examined and a new procedure for the tributary pipe break analysis is suggested. The result confirms the applicability of the proposed procedure to tributary pipe break analyses. This paper also considers the lateral responses of the reactor internals to the 3 inch pressurizer spray line break, main steam line break and economizer feedwater line break. Pressurizer spray line break is expected to remain as the only design basis pipe break in the primary side after leak-before-break (LBB) evaluation is extended to smaller size pipes in the near future. The results are compared with the internals responses to the safe shutdown earthquake (SSE). The comparative evaluation shows that, when the LBB concept is applied to the primary side piping systems with a diameter of 10 inches or over, SSE loads with a conservative margin can be used for the pipe break loads in the preliminary faulted condition design. (author).

Title: Experimental and numerical study of circumferentially through-wall cracked pipe under bending including ductile cra

Author: Le-Delliou,-P.; Crouzet,-D. **Corp. Author:** EDF

Source: ASME, 1990. Fatigue, Degradation, and Fracture 1990. PVP-Volume 195; MPC-Volume 30. New York (NY). 205 p. p. 85-92.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 9

Abstract: In 1986, EDF started a program on fracture of carbon and stainless steel cracked pipes. The purpose of the program was to develop a better understanding of pipe fracture behavior in order to evaluate the leak-before-break (LBB) approach and improve in service flaw assessments. Until now, fifteen pipe experiments have been performed on 6 inch and 16 inch diameter pipes containing circumferential through-wall cracks with total angles between 30 degrees and 120 degrees. The range of experiments include studies of crack growth and pipe ovalization, base and weld metal comparison and cyclic loading effects. This paper reports that the main results are: large crack propagation between initiation and maximum load, with crack turning out from the original crack plane; base and weld metal tests show almost the same load-displacement behavior; limited amount of ovalization in the crack plane. The analytical studies include limit-load analyses, conventional fracture mechanics (GE-EPRI and R6 methods) and finite element analyses of several tests. Accuracy of FEM analysis is about 10%, with a tendency to underestimate the maximum load for small crack angles (30 degrees).

Title: A leak-before-break assessment of BWR recirculation piping.

Author: Mehta,-H.S.; Chexal,-B. **Corp. Author:** GE Nuclear Energy

Source: ASME, 1991. Pressure Vessel Integrity 1991. PVP-Volume 213; MPC-Volume 32. New York (NY). 290 p. p. 223-228.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: LBB justification **ID:** *10*

Abstract: Postulation of a sudden DEGB in the high energy piping of LWRs has led to the installation of protective devices such as pipe whip restraints and jet impingement barriers. However, in many cases, these devices impede the regular in-service inspection and maintenance, which in turn, leads to increased personnel exposure and adverse effects on plant safety. Through a recent modification of General Design Criterion 4 of 10CFR50, Appendix A, the NRC has recognized the leak-before-break (LBB) approach as an alternate to the DEGB postulation. The objective of the LBB analysis is to demonstrate that the detection of flaws either by in-service inspection or by leakage monitoring systems is assured long before the flaws can grow to critical or unstable sizes which can lead to a DEGB. This paper is based on the results of an EPRI-sponsored study on the application of LBB approach to Boiling Water Reactor (BWR) piping. Recirculation piping system of a typical BWR/4 plant was selected for the LBB assessment. The piping system stress report was reviewed to determine the limiting stress locations for each of the four pipe sizes involved.

Title: Predicting the life of high-temperature structural components in power plants.

Author: Liaw,-P.K.; Saxena,-A.; Schaefer,-J. **Corp. Author:** Westinghouse

Source: JOM.-Journal-of-the-Minerals,-Metals-and-Materials-Society. (Feb 1992). v. 44(2) p. 43-48.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Inspection methods **ID:** *11*

Abstract: This paper reports on the concept of time-dependent fracture mechanics that has been used to develop the quantitative life-prediction methodology and inspection criteria for high-temperature structural components. As an example, the methodology was applied to steam pipes. Leak-before-break analyses were utilized to determine the flaw inspection criteria of steam pipes. Both static and cyclic loading conditions were included in the life-prediction analyses. Increasing the frequency of shut-downs was found to decrease the remaining life. The effects of operating pressures and temperatures and material properties on the life of steam pipes were quantified.

Title: Service water system issues and containment response transient analysis for nuclear power plant applications.

Author: Smith,-L.C.; Jakub,-R.M. **Corp. Author:** Westinghouse

Source: Transient thermal hydraulics and resulting loads on vessel and piping systems 1990. PVP-Volume 190. New York, NY (United States). American Society of Mechanical Engineers. 1990. 70 p. p. 21-28.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: SWS Operating Experience **ID:** *12*

Abstract: Service Water Systems (SWSs) perform an important safety function by reducing the magnitude of the containment response to a design basis pipe rupture by cooling many safety systems including the reactor containment fan coolers, which are utilized to directly cool the containment atmosphere and the recirculation mode heat exchangers, which are used to directly and indirectly cool the inside containment fluid used for vital emergency safeguards equipment during recirculation cooling mode. Operating experience shows that SWS failures and degradations can occur due to a variety of causes, including sediment deposition, biofouling and corrosion, and because of the high safety significance associated with SWS's, ultimate heat sink cooling systems must be evaluated for the effects of degradation on the containment response and performance of safety-related systems. Degradations in SWS have potential to significantly impact the performance of the emergency safeguards equipment. SWS problem areas, operating experience, the U.S. NRC position and typical SWS's are discussed.

Title: Atucha I PHWR (pressurized heavy water reactors) Power Plant. System event tree analysis for loss of coolant acciden

Author: Layral,-S.I. **Corp. Author:** CNEA

Source: 1989. 7 p. 17. Annual meeting of the Argentine Association of Nuclear Technology. Buenos Aires (Argentina). 4-7 Dec 1989. 17. Reunion anual de la Asociacion Argentina de Tecnologia Nuclear.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** Spanish

Category: Failure probability **ID:** 13

Abstract: This study is part of a Probabilistic Safety Assessment performed for Atucha I PHWR Power Plant. The objective of this report was to develop a system event tree analysis for two cases selected for this plant, as representative of the family of loss of coolant accidents (LOCA). Probabilistic assessment is focussed to identification and quantification of the most significant accidental sequences contributing to the core melt frequency. In a former stage - sup S election of Initiating Events sup - , two events were selected as representative for the LOCA family: a) Guillotine break of a reactor coolant pipe, between pressure vessel and circulating pump (large LOCA); b) Guillotine break of a moderator connecting pipe to the reactor coolant system, used for shutdown cooling (small LOCA). Core melt frequencies obtained through the use of event tree and safety system unavailability models are respectively, 1.3 x 10 sup - sup 6 /y for large LOCA AND 1.1 X 10 sup - sup 5 /y for small LOCA. In both cases the major contributions are: failure of Moderator System to conmute to shutdown cooling mode, and failure of Low Pressure Emergency Core Cooling Injection. These results are considered acceptable from safety point of view. (Author).

Title: UPTF test results with regard to loop flow dependant reactor safety issues.

Author: Zipper,-R. **Corp. Author:** GRS

Source: 18th Water Reactor Safety Information Meeting. Proceedings: Volume 1. Apr 1991. 672 p. p. 381-427.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 14

Abstract: For ten years the BMFT in Germany, the Japan Atomic Energy Research Institute (JAERI) and the USNRC performed a coordinated experimental and analytical study on multidimensional coolant behavior in the primary system of a PWR during LOCA, known as the 2D/3D project. In the FRG the Upper Plenum Test Facility (UPTF) was constructed and operated as part of the German contribution to the 2D/3D project. The UPTF simulates all relevant parts of a four loop PWR primary coolant system in 1:1 scale except the core, the steam generators and the main coolant pumps which are replaced by simulators. One of the loops is equipped with quick opening gate valves to simulate the break of a pipe. The controlled pressure boundary at the break is formed by a pressure suppression system called containment simulator. The objectives of the UPTF test program were to perform integral tests simulating the low pressure phases of a large break LOCA in US, Japanese, and German reactor and ECCS system design, to perform separate effects tests investigating multidimensional flow phenomena, and to investigate small break LOCA phenomena to improve and assess computer code models. Test results and their evidence to reactor safety issues related to loop flow behavior are presented.

Title: Frequencies of Leaks and Breaks in Safety Related Piping of PWR-Plants a Initiating Events for LOCAs.

Author: Beliczey,-S. (Gesellschaft fuer Reaktorsicherheit mbH (GRS), Koeln (Germany)) **Corp. Author:** OECD/BMU

Source: Hauptmanns,-U. (comp.). Gesellschaft fuer Reaktorsicherheit mbH (GRS), Koeln (Germany). Proceedings of the OECD/BMU-workshop on special issues of level 1 PSA. Jul 1991. 407 p. p. 364-380.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Failure probability **ID:** 15

Abstract: The analysis of the effects of LOCA events shows, that there are various ranges of leak rates that are to be distinguished corresponding to the capabilities of systems that are directed to assure the safe condition of the plant. The actuation and subsequent operation of these systems is a further barrier to prevent core damage. Other ranges apply for the steam generators. The frequency of some leak rates will be dominated by inadvertent or faulty opening actions of valves. Some LOCA-relevant leak rates however are mainly caused by wall-penetrating cracks or a break of a pipe. These damages in the walls of the primary coolant retaining system and their frequencies will be discussed here. (orig.).

Title: Water-Hammer in the Cold Leg During an SBLOCA Due to Cold ECCS Injection.

Author: Ortiz,-M.G.; Ghan,-L.S. **Corp. Author:** EG&G Idaho, Inc.

Source: [1991]. 4 p. Westinghouse Savannah River Co., Aiken, SC (USA).American Society of Mechanical Engineers (ASME) pressure vessels and piping conference. San Diego, CA (USA). 23-27 Jun 1991.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Pressure ripple/water hammer **ID:** *16*

Abstract: Water-hammer might occur in the cold leg of pressurized water reactors (PWR) during small break loss-of-coolant accidents (SBLOCA's), when cold emergency core cooling system (ECCS) water is injected into a pipe that may be partially filled with saturated steam. The water may mix with the steam and cause it to condense abruptly. Depending on the flow regime present, slugs of liquid may then be accelerated towards each other or against the piping structure. The possibility of this phenomenon is of concern to us because it may become a dominant phenomenon and change the character of the transient. In performing the code scaling, applicability, and uncertainty study (CSAU) on a SBLOCA scenario, we had to examine the possibility that the transient being analyzed could experience water-hammer and thus depart from the scope of the study. Two criteria for water-hammer initiation were investigated and tested using a RELAP5/MOD3 simulation of the transient. Our results indicated a very low likelihood of occurrence of the phenomenon. 8 refs., 6 figs.

Title: Crack opening area of pressurized pipe for leak-before-break evaluation.

Author: Hasegawa,-Kunio; Okamoto,-Asao; Yokota,-Hiroshi; Yamamoto,-Yoshio; Shibata,-Katsuyuki; Oshibe,-Toshihiro; Matsumura,-Kazuhiro **Corp. Author:**

Source: JSME-International-Journal.-Series-1,-Solid-Mechanics-and-Strength-of-Materials. (Jul 1991). v. 34(3) p. 332-338.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: LBB methodology **ID:** *17*

Abstract: The prediction method for analyzing the crack opening area of a pipe is essential for leak-before-break evaluation. Several theoretical approaches are proposed for predicting crack opening areas. One approach is the Tada and Paris formula, developed based on the linear elastic fracture mechanics. Another is the engineering approach developed by German and Kumar. Round-robin analyses for crack opening areas are performed using these two methods. The pipe analyzed is a 6-inch-diameter Type 304 stainless steel pipe with a circumferential through-wall crack. The applied load is bending moment. The crack opening areas calculated by Tada and Paris method coincided among the four participants. However, the areas using German and Kumar method were quite different. It is concluded that in the present situation, Tada and Paris method is suitable for a leak-before-break standard to predict the crack opening area. In addition, material properties used in the calculation of the standard are discussed compared with the results of the pipe bending experiment. (author).

Title: On the validity of fracture assessment methods for flawed large-scale pressure vessels.

Author: Rintamaa,-R.; Keinaenen,-H.; Talja,-H.; Wallin,-K. (Technical Research Centre of Finland, Helsinki (Finland)) **Corp. Author:**

Source: Proceedings of the seminar on assessment of fracture prediction technology: Piping and pressure vessels. Feb 1991. 329 p., pp 3.3-3.35.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Fracture mechanics **ID:** *18*

Abstract: Cracking and subsequent catastrophic failure in pressure vessels and piping systems has a significant impact on NPP safety and reliability. To assure structural integrity of pressurized components, reliable knowledge of the relevant material properties must be available. To improve the accuracy and validity of experimental and computational fracture assessment methods, a four year Nordic research program was initiated 1985. The aim of the program was to clarify how catastrophic failure can be prevented in pressure vessels and piping systems by developing the necessary elastic-plastic fracture mechanics analyses and by providing appropriate experimental data for their validation. The engineering fracture assessment methods (Battelle's limit load method) that were applied gave reliable and conservative estimates for rupture pressure and leak-before-break considerations in case of flawed thin-walled pipes. Fracture behavior of the large pressure vessels was simulated more precisely by both elastic-plastic and geometrically nonlinear analyses based on the finite element method. The calculated strains and stresses from the 3-D analysis agreed well with the experimental findings. This project has produced new insights into the structural integrity assessment of flawed pressurized components.

Title: Prediction of the failure stress from Japanese carbon steel pipe fracture experiments.

Author: Kashima,-K.; Matsubara,-M.; Miura,-N. **Corp. Author:**

Source: Hiser,-A.L. Jr.; Mayfield,-M.E. (eds.). Nuclear Regulatory Commission, Washington, DC (United States). Office of Nuclear Regulatory Research. Proceedings of the seminar on assessment of fracture prediction technology: Piping and pressure vessels. Feb 1991. 329 p. p. 2.27-2.50.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: LBB justification **ID:** 19

Abstract: Extensive research programs on leak-before-break have been organized and are in progress in many countries to evaluate structural integrity of nuclear piping systems. This paper describes the prediction of failure loads of Japanese carbon steel pipes. Three analytical approaches, three-dimensional finite element method, two-criteria approach and Japanese G-factor approach, were applied to estimate the failure loads of circumferentially cracked pipes under bending load. Analytical solutions were compared with the results from the fracture tests of 6-inch and 30-inch diameter pipes. Good agreement was obtained between the fracture loads from the pipe tests and the predictions by the finite element method and two-criteria approach. The G-factor approach predicted a conservative failure load. The finite element analysis showed a higher J-integral resistance in large-diameter pipe than the compact tension specimens. From the analytical results, it was found that plastic collapse was a dominant fracture criterion in both 6-inch and 30-inch diameter Japanese carbon steel pipes.

Title: Comparisons between finite-element analysis predictions and pipe fracture experiments.

Author: Brust,-F.W.; Ahmad,-J.; Brickstad,-B.; Faidy,-C.; Gilles,-P. **Corp. Author:**

Source: Hiser,-A.L. Jr.; Mayfield,-M.E. (eds.). Nuclear Regulatory Commission, Washington, DC (United States). Office of Nuclear Regulatory Research. Proceedings of the seminar on assessment of fracture prediction technology: Piping and pressure vessels. Feb 1991. 329 p. p. 2.3-2.26.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: LBB justification **ID:** 20

Abstract: This paper presents the results of ten finite-element analyses of cracked pipe subjected to bending loads compared to the corresponding experimental results produced from full-scale tests. As part of the presentation, detailed results from two international round-robin problems are presented. In all, nine through-wall cracked pipe and one surface cracked pipe is considered. The cracked pipe includes stainless, carbon, and welded pipe. Most of the experimental results were developed during the course of the U.S. NRC's degraded piping program for LWR primary coolant circuit and pressure vessel leak-before-break studies.

Title: Large break frequency for the SRS (Savannah River Site) production reactor process water system.

Author: Daugherty,-W.L.; Awadalla,-N.G.; Sindelar,-R.L.; Bush,-S.H. **Corp. Author:**

Source: Lawrence Livermore National Lab., CA (United States). Second DOE natural phenomena hazards mitigation conference. [1989]. 436 p. p. 375-380.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Failure probability **ID:** 21

Abstract: The objective of this paper is to present the results and conclusions of an evaluation of the large break frequency for the process water system (primary coolant system), including the piping, reactor tank, heat exchangers, expansion joints and other process water system components. This evaluation was performed to support the ongoing PRA effort and to complement deterministic analyses addressing the credibility of a double-ended guillotine break. This evaluation encompasses three specific areas: the failure probability of large process water piping directly from imposed loads, the indirect failure probability of piping caused by the seismic-induced failure of surrounding structures, and the failure of all other process water components. The first two of these areas are discussed in detail in other papers. This paper primarily addresses the failure frequency of components other than piping, and includes the other two areas as contributions to the overall process water system break frequency.

Title: Probabilistic evaluation of main coolant pipe break indirectly induced by earthquakes Savannah River Project L and P

Author: Short,-S.A.; Wesley,-D.A.; Awadalla,-N.G.; Kennedy,-R.P. **Corp. Author:**

Source: Lawrence Livermore National Lab., CA (United States). Second DOE natural phenomena hazards mitigation conference. [1989]. 436 p. p. 365-374.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Failure probability **ID:** 22

Abstract: A probabilistic evaluation of seismically-induced indirect pipe break for the Savannah River Project (SRP) L- and P-Reactor main coolant (process water) piping has been conducted. Seismically-induced indirect pipe break can result primarily from: (1) failure of the anchorage of one or more of the components to which the pipe is anchored; or (2) failure of the pipe due to collapse of the structure. the potential for both types of seismically-induced indirect failures was identified during a seismic walkdown of the main coolant piping. This work involved: (1) identifying components or structures whose failure could result in pipe failure; (2) developing seismic capacities or fragilities of these components; (3) combining component fragilities to develop plant damage state fragilities; and (4) convolving the plant seismic fragilities with a probabilistic seismic hazard estimate for the site in order to obtain estimates of seismic risk in terms of annual probability of seismic-induced indirect pipe break.

Title: Failure probability estimate of type 304 stainless steel piping.

Author: Daugherty,-W.L.; Awadalla,-N.G.; Sindelar,-R.L.: Mehta,-H.S.; Ranganath,-S. **Corp. Author:** Westinghouse & General Electr

Source: Lawrence Livermore National Lab., CA (United States). Second DOE natural phenomena hazards mitigation conference. [1989]. 436 p. p. 129-134.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Failure probability **ID:** 23

Abstract: IGSCC has occurred in a limited number of weld heat affected zones in the SRS units. A model has been developed to estimate the pipe large-break frequency. It is based on the probability that an IGSCC crack will initiate, escape detection by ultrasonic (UT) examination, and grow to instability prior to extending through-wall and being detected by the sensitive leak detection system. These events are combined as the product of four factors: (1) the probability that a given HAZ contains IGSCC; (2) the conditional probability, given the presence of IGSCC, that the cracking will escape detection during UT examination; (3) the conditional probability, given a crack escapes detection by UT, that it will not grow through-wall and be detected by leakage; (4) the conditional probability, given a crack is not detected by leakage, that it grows to instability prior to the next UT exam. For the SRS production reactors, these factors produce an extremely low break frequency. The paper presents assumptions, methodology, results and conclusions of a probabilistic evaluation for the direct failure of the primary coolant piping resulting from normal operation and seismic loads. This evaluation was performed to support the ongoing PRA effort and to complement deterministic analyses addressing the credibility of a DEGB.

Title: Ductile fracture analysis of carbon steel pipe with a circumferential through-wall crack.

Author: Asano,-Masayuki; Fukakura,-Juichi; Kashiwaya,-Hideo; Saito,-Mashiro **Corp. Author:**

Source: Nuclear-Engineering-and-Design. (Jul 1991). v. 128(1) p. 1-7.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: LBB justification **ID:** 24

Abstract: It is necessary to make clear the pipe fracture conditions based on elastic-plastic fracture mechanics to assess the leak before break situation of carbon steel pipes for LWR plants. The aim of the present work is to discuss the effects of pipe size, initial crack length and fracture toughness on the estimated fracture load and mode of carbon steel pipes with a circumferential through-wall crack. As an analytical method, the R6-Rev.3 approach was applied to the pipe fracture analyses considering its simplicity in the application. The results indicate that the net-section collapse attainment becomes difficult with increasing diameter and decreasing thickness of carbon steel pipes. The degree of the net-section collapse attainment decreases with increasing crack length up to some critical size and then increases. The predicted fracture load is more sensitive to the material's J - R curve than to the elastic-plastic fracture toughness J sub I sub C. And a simple limit load analysis based on the yield stress is appropriate to evaluate the fracture load, as long as a proper margin was included. (orig.).

Title: Procedure for setting up the "leak before break" document.

Author: Ceskoslovenska Komise pro Atomovou Energii, Prague (Czechoslovakia). **Corp. Author:**

Source: Detekcni systemy uniku z tlakoveho chladiciho okruhu jaderneho reaktoru. 1991. 23 p. p. 1-16.ST: Pozadavky pro sestaveni a obsah bezpecnostnich zprav a jejich dodatku. (no.1).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** Czech

Category: Failure probability **ID:** 25

Abstract: This procedure serves to determine the basic requirements of the Czechoslovak Atomic Energy Commission put on the evaluation of the dynamic impacts of complete breakdown of the PWR primary coolant piping. The method makes it possible to give evidence of the extremely low probability of such an accident, reduction in the irradiation of personnel, as well as reduced building and maintenance costs. The Commission will apply this procedure to the evaluation, and if evidence is gained that leaks will appear prior to the piping breakdown, the body may approve changes in nuclear safety provisions. The objects of evaluation include water shocks, flow damage, erosion, corrosion, fatigue and external impacts. The requirements placed on the containment, stand-by coolant circuit and resistance of the electric and mechanical equipment are thereby not affected. (M.D.).

Title: Steam condensation and liquid hold-up in steam generator U-tubes during oscillatory natural circulation.

Author: De-Santi,-G.F.; Mayinger,-F. **Corp. Author:** American Nuclear Society (AN

Source: Transactions-of-the-American-Nuclear-Society. (1990). Vol. 62:695-696.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Analysis of break effects **ID:** 26

Abstract: In many accident scenarios, natural circulation is an important heat transport mechanism for long-term cooling of light water reactors. In the event of a small pipe break, with subsequent loss of primary cooling fluid loss-of-coolant accident (LOCA), or under abnormal operating conditions, early tripping of the main coolant pumps can be actuated. Primary fluid flow will then progress from forced to natural convection. Understanding of the flow regimes and heat-removal mechanisms in the steam generators during the entire transient is of primary importance to safety analysis. Flow oscillations during two-phase natural circulation experiments for pressurized water reactors (PWRs) with inverted U-tube steam generators occur at high pressure and at a primary inventory range between two-phase circulation and reflex heat removal. This paper deals with the oscillatory flow behavior that was observed in the LOBI-MOD2 facility during the transition period between two-phase natural circulation and reflex condensation.

Title: Reliability evaluation of the Savannah River reactor leak detection system.

Author: Daugherty,-W.L.; Sindelar,-R.L.; Wallace,-I.T. **Corp. Author:** Westinghouse

Source: [1991]. 6 p. Westinghouse Savannah River Co., Aiken, SC (USA). ASME PVP Conference. San Diego (CA). 23-27 Jun 1991.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: IGSCC / LBB **ID:** 27

Abstract: The Savannah River Reactors have been in operation since the mid-1950's. The primary degradation mode for the primary coolant loop piping is intergranular stress corrosion cracking. The leak-before-break (LBB) capability of the primary system piping has been demonstrated as part of an overall structural integrity evaluation. One element of the LBB analyses is a reliability evaluation of the leak detection system. The most sensitive element of the leak detection system is the airborne tritium monitors. The presence of small amounts of tritium in the heavy water coolant provide the basis for a very sensitive system of leak detection. The reliability of the tritium monitors to properly identify a crack leaking at a rate of either 50 or 300 lb/day (0.004 or 0.023 gpm, respectively) has been characterized. These leak rates correspond to action points for which specific operator actions are required. High reliability has been demonstrated using standard fault tree techniques. The probability of not detecting a leak within an assumed mission time of 24 hours is estimated to be approximately 5 x 10 sup - sup 5 per demand. This result is obtained for both leak rates considered. The methodology and assumptions used to obtain this result are described in this paper. 3 refs., 1 fig., 1 tab.

Title: Preliminary observations of upcoming Phase II gate valve flow interruption tests.

Author: Steele,-R. **Corp. Author:** INEL

Source: Weiss,-A.J.-Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research. Transactions of the 17th Water Reactor Safety Information Meeting. Oct 1989. 186 p. p. 7.15-7.16.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Other **ID:** 28

Abstract: A current research program at INEL is testing the ability of full-scale flexible wedge gate valves to close under design basis flow and pressure loadings. The purpose of this program is to provide technical information regarding Generic Issue 87, Failure of the HPCI Steamline Without Isolation. Phase I testing was completed in June 1988, and results are being analyzed. The objective of Phase II of the program is to expand the technical data base in determining whether isolation valves in high energy BWR piping systems will close against high flows in the event of a pipe break outside containment. Generic Issue 87 includes those BWR process lines that communicate with the primary system, pass through containment, and contain normally open isolation valves. Three process lines fall under this description: (1) the HPCI steam supply line, (2) the RCIC steam supply line, and (3) the RWCU supply line. Of the three, an unisolated break in the RWCU supply line was determined to have the greatest safety impact and was the subject of the Phase I test program. The Phase II test program will be configured to answer questions raised by results of the Phase I testing on the RWCU valves and will include steam flow interruption testing representative of the HPCI system.

Title: Results of gate valve flow interruption tests in the RWCU line environment.

Author: DeWall,-K.G. **Corp. Author:** INEL

Source: Weiss,-A.J.-Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research. Transactions of the 17th Water Reactor Safety Information Meeting. Oct 1989. 186 p. p. 5.5-5.6.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Other **ID:** 29

Abstract: For some NPP valves, the equations used by industry to size the valves do not conservatively calculate the thrust needed to close the valves under design basis loadings. Tests showed that results of in situ valve testing at lower loadings cannot be extrapolated to design basis loadings. An unisolated break in the BWR-RWCU supply line was selected since such a break would have the greatest safety impact. Two representative RWCU isolation valves were subjected to hydraulic qualification tests described in ANSI B16.41, and then to full flow RWCU pipe break flow interruption tests. In all, 14 flow interruption tests were performed. In the Valve A tests, the parametric study included varying both the degree of inlet water subcooling and the pressure. Break flows were maintained throughout the 30-second valve closure. Valve B tests were all performed at a normal BWR 10F subcooling, and inlet pressure only was varied. The valves were instrumented to determine valve response to flow, including a load cell installed in the valve stems to measure thrust. Test results show that the variables used by industry for determining valve thrust are not conservative, and internal valve design differences can result in large response differences and that prototypical testing may be necessary to determine actual valve performance.

Title: Experiences using three-dimensional finite element analysis for leak-before-break assessment. CANDU reactor piping.

Author: Vanderglas,-M.L. **Corp. Author:** Ontario Hydro

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 241-253.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: LBB methodology **ID:** 30

Abstract: Because of practical limitations, analytical problems in fracture mechanics have often been solved using simplified geometries (e.g. plane stress/strain, shell models). We have applied the Leak-Before-Break approach extensively to the large diameter piping of a new CANDU reactor plant. Various piping components such as elbows, tee and branch connections with postulated cracks were analyzed. Since no credible geometric simplification was possible, fully three-dimensional (3D) analytical models were found to be essential. The paper describes our experiences in performing 3D Finite Element (FE) analysis of these components. Included are comparisons of numerical and test results of compact specimens, material modeling considerations, handling of 3D effects, such as the variation of the J-integral along a crack front, and especially, the effects of plasticity. The overall intent of the paper is not simply to present specific numerical results, but rather to give some perspective on the effort required and results attainable. (author).

Title: Ontario Hydro's leak-before-break approach to Darlington NGS heat transport system piping.

Author: Nathwani,-J.S.; Stebbing,-J.D. **Corp. Author:** Ontario Hydro

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 113-127.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: LBB Justification **ID:** 31

Abstract: The primary objective in our Leak-Before-Break studies is to show how a rational and comprehensive approach can provide an adequate measure of confidence in the assessment of piping integrity such that provision of design features (viz. pipewhip restraints, jet impingement shields) to protect against the dynamic effects of pipe rupture is not necessary. This study is one component of the overall Leak-Before-Break approach adopted at Ontario Hydro. The results of a review undertaken to evaluate the system transients or events sequences which may subject the piping to a potentially significant increase in loadings are reported. The focus in this paper is to show the approach used in deriving loadings for use in the elastic-plastic fracture mechanics analyses required to demonstrate crack stability. (author).

Title: Comments on Probabilities of Leaks and Breaks of Safety-Related Piping in PWR Plants.

Author: Beliczey,-S.; Schulz,-H. **Corp. Author:** GRS

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 219-227.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Damage probability **ID:** 32

Abstract: Leaks or failures with a safety significance in Cl.1 or Cl.2 piping of NPPs in Germany are very rare events. This excellent record in operating experience is matched by many NPPs in other countries. The advances achieved in the understanding of fracture behaviour, in the methods of non-destructive testing and surveillance, together with operating experiences, can be used in the re-evaluation of piping systems that have been designed and manufactured to the standards given at the time of construction. Comments and examples are presented for determining the probability of leaks and breaks in the whole range of Cl.1 and Cl.2 piping systems. (author).

Title: Leak-Before-Break in Steam Generator Tubes.

Author: Flesch,-B; Cochet,-B. **Corp. Author:** EDF

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 165-179.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: LBB justification **ID:** 33

Abstract: The steam generator tubing in a pressurized water reactor constitutes one of the main barriers against the release of activity to the environment. The capacity of the tubing to withstand safely the loads exerted on it during normal operation and faulted conditions is therefore the most important factor in steam generator safety evaluation. Another important consideration in safety evaluation is the tendency of the tubes to leak at a significant but acceptable rate under normal operating conditions before there is a risk of rupture under accidental overpressure: the Leak-Before-Break (LBB) criterion. This paper presents the theoretical and experimental programme undertaken in France to assess the LBB criterion for PWR steam generator tubes. Criteria for instability of different types of defect have been deduced from experimental and numerical results. Leakage models have been derived from leak tests, as well as crack-opening measurements and calculations. (author).

Title: Leak-Before-Break in French Nuclear Power Plants.

Author: Faidy,-C.; Bhandari,-S. Jamet,-P. **Corp. Author:** EDF-SEPTIN, FRAMATOM,

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 151-163.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: LBB justification **ID:** 34

Abstract: Practical applications of the LBB concept at the present stage are quite limited in French NPPs. However, discussions with safety authorities have included LBB arguments for different types of reactors. At present the fracture mechanics part of the studies are complete for the following components: pipe in gas graphite reactors; primary and auxiliary lines and steam-generator tubes in PWRs; pipes and main vessels in liquid metal fast breeder reactors. The different approaches are consistent but some specific problems have to be taken into account, depending on the plant, such as the creep regime, thin shell components, in-service inspection or the issue of design safety. A large research and development program, realized in different cooperative agreements (national or international), completes the general approach. It comprises different topics, such as material properties, elastoplastic fracture mechanics, leak-area determination and leak-detection devices. The objective of this paper is to present the application of fracture-mechanics methodology used in France to demonstrate the LBB behavior of PWR components. The results presented represent a synthesis of the various studies conducted in a view of the applicability of this concept on French PWRs. (author).

Title: Development of criteria for protection against pipe breaks in LWR plants.

Author: Asada,-Y.; Takumi,-K.; Hata,-H.; Yamamoto,-Y. **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 95-111.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: LBB justification **ID:** 35

Abstract: A proving test for the structural integrity of safety-related carbon steel piping components in light water reactor plants was conducted in NUPEC as a four-year project, in which the applicability of the Leak-Before-Break (LBB) concept to protect against a postulated pipe break was reviewed in parallel with the clarification of fracture behavior. The comprehensive review of LBB applications consists of applicable piping systems, premise for evaluations, procedure and evaluation findings. The review concluded that present practice for design, fabrication, installation and operation can ensure structural integrity and moreover postulated that instantaneous pipe break as a basis for structural design is unrealistic if certain conditions are met. Fatigue is the only failure mechanisms to be considered to affect the piping system. (author).

Title: Application of leak-before-break to primary loop piping to eliminate pipe whip restraints in a Spanish nuclear power p

Author: Rodriguez,-M.; Esteban,-A. (Consejo de Seguridad Nuclear (CSN), Madrid (Spain)) **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 85-93.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: LBB justification **ID:** 36

Abstract: The Spanish plant described in this study is a 3-loop 982 MWe PWR plant with primary circuit of piping made from centrifugally-cast stainless steel SA351 CF8A. The licensee requested from Consejo de Seguridad Nuclear (CSN) an exemption from the general design criterion, GDC-4, so as to avoid the need to postulate a guillotine rupture of the primary loop piping. The request was based on the generic work performed for a US PWR plant group in order to have such an exemption. As the piping material in the Spanish plant is different from that in the plants included in the generic work, CSN performed a review of the applicability of the generic results to the Spanish plant. Also, aspects such as fatigue evaluation, net section collapse, crack growth and leak detection, specifically analyzed for the Spanish plant, were reviewed. CSN found that fracture toughness test results from generic work are applicable to the Spanish plant; sufficient margin exists against unstable crack extension, and adequate leak detection capability exists with the leakage detection systems available in the plant. Exemption from GDC-4 was approved and CSN authorized the licensee to remove protection devices against dynamic loads from guillotine breaks in the primary coolant loops. (author).

Title: Leak-before-break application in US light water reactor balance-of-plant piping.

Author: Beaudoin,-B.F.; Quinones,-D.F.; Hardin,-T.C. (Cloud (Robert L.) and Associates, Inc., Berkeley, CA (USA)) **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 67-83.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: LBB justification **ID:** 37

Abstract: This paper describes criteria and methodology for a LBB program for high energy BOP-piping. LBB can be applied to any operational or pre-operational LWR plant to minimize pipe rupture hardware and to discount pipe rupture dynamic effects. The general methodology described herein, encompasses applicable U.S. NRC requirements and incorporates experience gained in the licensing process of actual LBB programs. First, candidate piping systems must be carefully screened to verify that they are not subject to failure by phenomena that would adversely affect the accurate evaluation of flaws. Next, pipe stresses, material properties, and leak detection capabilities are gathered for the fracture mechanics and fluid mechanics analyses. At the piping locations which have the least favorable combination of material properties and stress, a crack is postulated which is of sufficient size that the resulting leakage will be detected by installed leak detection systems. Finally, LBB is demonstrated if the postulated crack remains stable even if a seismic event takes place before the crack is discovered and repaired. An LBB example is presented in this paper for a generic pressurizer surge line, and reflects the consideration of flow stratification on LBB analyses. (author).

Title: Measurement of leak-rate through fatigue-cracks in pipes under four-point bending and BWR conditions.

Author: Isozaki,-T.; Shibata,-K.; Shinokawa,-H.; Miyazono,-S. **Corp. Author:** JAERI

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 399-411.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 38

Abstract: Leak-rate tests were performed using 114 mm and 165 mm (4 and 6 in) diameter, schedule 80 pipes made of austenitic stainless steel SUS304 and carbon steel STS42. Each pipe contained a through-wall fatigue crack and was mounted on a four-point bending machine of 400 kN maximum loading. Tests were done under a pressure of 7 MPa, with a subcooling temperature. The leak rate was measured by a Venturi flow meter and a differential pressure transducer attached to the pressure vessel. Comparisons of the effect of pipe material, diameter and crack angle were made. This paper shows that from a Leak-Before-Break viewpoint, the stainless-steel pipe is superior to the carbon-steel one, and that the pipe with the larger diameter is better than the one with the smaller diameter. No unstable fracture was observed in the tests. (author).

Title: Failure probability of nuclear piping due to IGSCC.

Author: Nilsson,-F.; Brickstad,-B.; Skaanberg,-L. **Corp. Author:** RIT-Stockholm

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 205-217.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Failure probability **ID:** 39

Abstract: A simple model for the estimation of the pipe break probability due to intergranular stress corrosion cracking is developed and discussed. It is partly based on analytical procedures partly on service experiences from the Swedish boiling water reactor program. Some rough estimates of the resulting break probabilities indicate that further studies are urgently needed. A sensitivity study is performed and it is found that the uncertainties about the initial crack configuration are the most important contributors to the total uncertainty. The results of inservice inspection are studied and it is found that the inspection intervals need to be shortened if a significant reduction in the failure probabilities is to be obtained. (author).

Title: Leak-before-break experiments on heat-treated Zr-2.5 wt% Nb pressure tubes.

Author: Koike,-M.H.; Takahashi,-T.; Baba,-H. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center) **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 39-56.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: LBB justification **ID:** 40

Abstract: The pressure tubes of the Advanced Thermal Reactor (boiling-light-water-cooled, heavy-water-moderated, pressure tube-type reactor) in Japan are made of heat-treated Zr-2.5wt%Nb alloy and both ends are mechanically joined with stainless steel extension tubes. Sharp artificial cracks were introduced in the rolled joint region of pressure tube specimens. The cracks were propagated, and penetrated the tube wall due to fatigue and delayed hydride cracking in a high-temperature, high-pressure water loop. From the results, it was shown that the leak-before-break criteria were valid for the rolled joint region of the pressure tube under the reactor operating conditions and that the critical crack length was more than 50 mm. Calculations were performed for the subsequent leak rate, using critical flow data. (author).

Title: Directed discussion [on leak-before-break in water reactor piping vessels].

Author: Smith,-E.; Simpson,-L.A.; Coleman,-C.E. **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 425-432.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: LBB justification **ID:** 41

Abstract: A discussion directed towards eight key issues relating to the Leak-Before-Break (LBB) concept in water reactor piping and vessels is summarized. The key issues are: (1) the sensitivity and reliability of leak detection devices; (2) factors that affect leakage and make detection difficult; (3) the gradual development of a part-through crack, its shape and effect on instability; (4) correct consideration of weld properties; (5) application of LBB methodology to non-ideal regions (6) the probabilistic approach to LBB; (7) when should LBB be used? (8) the incorporation of LBB in safety codes. (UK).

Title: Leak-before-break verification test and evaluations of crack growth and fracture criterion for carbon steel piping.

Author: Asada,-Y.; Takumi,-K.; Gotoh,-N.; Umemoto,-T.; Kashima,-K. **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 379-397.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 42

Abstract: A proving test on the integrity of carbon steel piping in LWRs was planned by the Nuclear Power Engineering Test Center as a four-year verification test program; it was completed at the end of March 1989. The objective of this proving test was to demonstrate the validity of the Leak-Before-Break (LBB) concept for high quality carbon steel piping under actual plant conditions. (author).

Title: Strength behaviour of flawed pipes under internal pressure and external bending moment: comparison between experi

Author: Sturm,-D.; Stoppler,-W. **Corp. Author:** MPA

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 351-366.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 43

Abstract: Experimentally determined failure curves for pipes weakened by surface longitudinal or circumferential defects, were compared with results calculated with the aid of engineering approximation methods. Considering the scatter bands of the mechanical properties and the geometrical dimensions, then by use of the engineering approximation methods, one can make only rough estimates of the load bearing behaviour. (author).

Title: Recent results of fracture experiments on carbon steel welded pipes.

Author: Wilkowski,-G.M.; Guerrieri,-D.; Jones,-D.; Olson,-R.; Scott,-P. **Corp. Author:** Battelle Columbus Labs.

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 329-350.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 44

Abstract: Several pipe fracture experiments were conducted with circumferential cracks in the center of ferritic nuclear pipe welds. These experiments involved either submerged arc or shielded metal arc welds with either through-wall cracks or internal surface cracks. The pipe diameters varied from 940 mm (37 inches) to 152 mm (6 inches), and thickness from 10.9 mm (0.43 inches) to 86.6 mm (3.41 inches). Some of the through-wall and surface-cracked pipe experiments were conducted under constant internal pressure and four-point bending. The test temperature was 288 sup 0 C (550 sup 0 F). The results of these experiments are compared with limit-load analyses, the ASME, Section XI, article IWB-3650 criterion, and more elaborate elastic-plastic fracture mechanical analysis. (author).

Title: Fracture toughness of weld metals in steel piping for nuclear power plants.

Author: Yoshida,-K.; Kojima,-M.; Iida,-M.; Takahashi,-I. **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 273-284.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 45

Abstract: To determine the toughness behaviour of dissimilar welds in steel piping and obtain data to evaluate Leak-Before-Break for these welds, an experimental study on fracture toughness was carried out. This paper provides Charpy impact results and fracture toughness data for the base and weld metals of dissimilar welds in nuclear piping. (author).

Title: Development of USNRC Standard Review Plan 3.6.3 for leak-before-break applications to nuclear power plants.

Author: Wichman,-K.; Lee,-S. (Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Reactor Regulation) **Corp. Author:** U.S. NRC

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 57-65.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: LBB justification **ID:** 46

Abstract: In the United States, it is now permissible to eliminate the dynamic effects of postulated high energy pipe ruptures from the design basis of nuclear power plants using LBB technology. To provide review guidance for the implementation of LBB, a new Standard Review Plant (SRP) 3.6.3 was issued for public comment. Based upon public comments received and advances in fracture mechanics application, further development of SRP 3.6.3 is in progress. SRP 3.6.3 will outline the review procedures and acceptance criteria for LBB licensing applications. A deterministic fracture mechanics evaluation accounting for material toughness will be required. Margins on load, crack size, and leakage will be specified and the load combination methods and leakage detection sensitivity will be described. Piping particularly susceptible to failure from potential degradation mechanisms will be excluded from the application of LBB. The design basis of containment, emergency core cooling systems, and environmental qualification of equipment in the context of LBB applicability will be clarified. (author).

Title: A failure probability estimate of Type 304 stainless steel piping.

Author: Daugherty,-W.L.; Awadalla,-N.G.; Sindelar,-R.L.; Mehta,-H.S. **Corp. Author:** Westinghouse

Source: Westinghouse Savannah River Co., Aiken, SC (USA).International topical meeting on the safety, status, and future of non-commercial reactors and irradiation facilities. Boise, ID (USA). 4 Oct 1990. 7 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Failure probability **ID:** 47

Abstract: The large break frequency resulting from intergranular stress corrosion cracking (IGSCC) in the main circulation piping of the Savannah River Site (SRS) production reactors has been estimated. Four factors are developed to describe the likelihood that a crack exists that is not identified by ultrasonic inspection and that grows to instability prior to becoming through-wall and being detected by the ensuing leakage. The estimated large break frequency is 3.4 x 10 sup - sup 8 per reactor year. This result compares favorably to similar estimates made for commercial boiling water reactors. 9 refs., 8 figs.

Title: Evaluation of catastrophic failure risk in pressure vessels. Results of pressure vessel test with a large vessel (HC2-test).

Author: Keinaenen,-H.; Rintamaa,-R.; Oeberg,-T.; Sarkimo,-M.; Talja,-H.; Saarenheimo,-A. **Corp. Author:** VTT

Source: Sep 1990. 63 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Fracture mechanics **ID:** 48

Abstract: Within the Nordic countries a four-year research programme in the area of elastic-plastic fracture mechanics was initiated in 1985. Seven laboratories are participating in the programme. The main technical objective of the programme is to clarify how catastrophic fracture can be prevented in pressure vessels and piping by using the LBB concept. The major experimental effort of the programme is destructive pressurization of a large size pressure vessel up to rupture. The vessel has dimensions similar to a nuclear reactor pressure vessel and it has been in operation for 20 years in a Finnish oil refinery plant. The materials characterization of the vessel has been partially carried out within an extensive Nordic round-robin programme. Two pressure tests have been carried out. In both tests an artificial sharp axial surface flaw was made on the inner wall of the vessel. The experimental details of the last test including repair welding of the vessel, flaw prepration, instrumentation and material characterization are described in this report. The fracture behaviour as well as experimental results are reported. The failure pressure is compared to estimates of the analytical pre-test calculations.

Title: Probabilistic fracture analysis of carbon steel pipes. Pipe break probability depends on pipe diameter.

Author: Fujioka,-Terutaka; Kashima,-Koichi **Corp. Author:**

Source: Denryoku-Chuo-Kenkyusho-Hokoku. (May 1990). (no.T89056) p. 1-26.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** Japanese

Category: Failure probability **ID:** 49

Abstract: Based on the assumption of a large-scale break in a pipe, piping systems in Japanese LWRs are designed to withstand dynamic effects. However, it is now recognised that such breaks seldom or never occur without prior warning sings such as leakage. The relaxation of design requirements in the United States and the Federal Republic of Germany, permit exclusion of a large-scale break from hypothetical events. The deterministic evaluation of a leak-before-break, which can indirectly prove that the probability of a break is extremely low, is noted in the design basis. But such deterministic approaches cannot quantify the safety of pipes. This report presents the breakage probabilities of 15 carbon-steel pipes used in Japanese LWRs based on probabilistic fracture analysis. The results show that larger pipes break at lower probabilities. (author).

Title: Indirect failure probability of Type 304 stainless steel piping.

Author: Kennedy,-R.P.; Daugherty,-W.L.; Awadalla,-N.G.; Sindelar,-R.L.; Wesley,-D.A. **Corp. Author:**

Source: Trans. the 10th SMiRT Conference. Volume K1-K2. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1989. 967 p. p. 929-934.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Failure probability **ID:** 50

Abstract: The NRC has developed criteria for establishing LBB conditions in high energy piping. The Savannah River Plant production reactors operate at relatively low temperature and pressure, making them moderate energy systems. While these reactors are not under NRC jurisdiction, the NRC criteria of NUREG-1061 provide a useful and complete framework for demonstrating LBB. These criteria include demonstrating a low failure probability of piping form indirect cause (resulting from the failure of surrounding equipment and structures). This paper presents an evaluation of the seismic indirect failure probability for the primary coolant piping at Savannah River Plant.

Title: Development of leak analysis programs from through-wall-crack.

Author: Shinokawa,-Hidetoshi; Shibata,-Katsuyuki; Isozaki,-Toshikuni (Japan Atomic Energy Research Inst., Tokai, Ibaraki (Japan). Tokai Research Establishment) **Corp. Author:** JAERI

Source: Mar 1990. 118 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** Japanese

Category: Test/analysis **ID:** 51

Abstract: To introduce LBB concept into the piping design standard of the LWR pressure boundary piping, LBB research programs are actively conducted in many nuclear electricity countries. It is one of the most important items to evaluate leak rate through the pipe wall crack to shut down a nuclear power plant safely. At JAERI, a test on the leak rate from a cracked pipe under BWR or PWR operating condition has been carried out from 1987 till 1989. This test is planned to measure the leak flow through circumferential fatigue cracks in 4-, 6-and, 12-inch diameter pipes and through slit specimens. On the other hands, it is necessary to predict and analyse the leak flow through a crack for applying the result of the tests to the structural design standard including the LBB concept. This report describes the some computer programs that calculate crack-opening-area, the crack length, and the flow rate through SCC or fatigue cracks. In these programs, a crack-opening-displacement calculation is available based on modified Tada-Paris equation when pipe geometries and pipe stress conditions are given. The leakage rate calculation is based on Henry's homogeneous nonequilibrium critical flow model and Moody's slip model with several modifications to account for friction and fluid conditions. (author).

Title: Numerical evaluation of cracked pipes under dynamic loading.

Author: Petit,-M.; Jamet,-P. **Corp. Author:** CEA-CEN

Source: Proceedings ASME PVP Conference. Honolulu, HI (USA). 22-26 July 1989. 7 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: LBB justification **ID:** 52

Abstract: In order to apply the LBB concept to piping systems, the behavior of cracked pipes under dynamic, and especially seismic, loadings must be studied. A simple finite element model of a cracked pipe has been developed and implemented in the general purpose computer code CASTEM 2000. The model is a generalization of the approach proposed by Paris and Tada (1). Considered loads are bending moment and axial force (representing thermal expansion and internal pressure.) The elastic characteristics of the model are determined using the Zahoor formulae for the geometry-dependent factors. Owing to the material behabior plasticity must be taken into account. To represent the crack growth, the material is defined by two characteristic values: J sub 1 sub c which is the level of energy corresponding to crack initiation and the tearing modulus, T, which governs the length of propagation of the crack. For dynamic loads, unilateral conditions are imposed to represent crack closure. The model has been used for the design of dynamic tests to be conducted on shaking tables. Test principle is briefly described and numerical results are presented. Finally evaluation of margin, due to plasticity, in comparison with the standard design procedure is made.

Title: A spacial cracked pipe element for leak before break application.

Author: Brochard,-J.; Petit,-M.; Millard,-A. **Corp. Author:** CEA-CEN

Source: 10th SMiRT Conference. Anaheim (CA). 14-18 Aug 1989. 8 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: LBB methodology **ID:** 53

Abstract: In order to apply the LBB-concept on pipes, characterization of the stability of circumferential cracks, in case of accidental loadings, has to be done. The questions are the following: can the crack growth be initiated, and if yes how large is the growth. The loading can be static or dynamic, and plastification of the material must be taken into account. A simple finite element model has been developed and will be an industrial tool in leak before break applications.

Title: Leak-before-break analysis of type 304 stainless steel piping.

Author: Awadella,-N.G.; Sindelar,-R.L.; Daugherty,-W.L.; Mehta,-H.S.; Ranganath,-S. **Corp. Author:**

Source: Hadjian,-A.H. (Bechtel Power Corp., Los Angeles, CA (USA)). Transactions of the 10th SMiRT Conference. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1989. 375 p. p. 369-374.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: LBB justification **ID:** 54

Abstract: The nuclear materials production reactors at the Savannah River Plant (SRP) were designed and built in the 1950's and have operated successfully since that time. Unlike commercial power reactors, the production reactors are moderated and cooled by heavy water and are operated at moderately low temperatures and internal pressures. In addition, the entire primary coolant pressure boundary is constructed of Type 304 stainless steel or its cast equivalent, CF-8, except for seals, gaskets and other serviceable parts. This paper presents the leak-before-break demonstration of the SRP primary coolant piping.

Title: Progress in failure assessment of piping systems in PWR's.

Author: Heliot,-J.; Boneh,-B. **Corp. Author:**

Source: Trans. of the 10th SMiRT Conference. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1989. 375 p. p. 347-352.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Methods **ID:** 55

Abstract: The paper shows the application of new concepts in failure assessment of piping systems in P.W.R.: the elastic plastic fracture mechanics, the real time monitoring of damage, the LBB and the probabilistic failure analysis.

Title: An estimation of the probability of failure for BWR Piping in Sweden.

Author: Nilsson,-F.; Brickstad,-B. **Corp. Author:** Uppsala University

Source: Hadjian,-A.H. Transactions of the 10th SMiRT Conference. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1989. 199 p. p. 99-104.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: IGSCC failure probability **ID:** 56

Abstract: A simple model for the estimation of the pipe break probability due to IGSCC is given. It is partly based on analytical procedures partly on service experiences from the Swedish BWR program. Some rough estimates of the resulting break probabilities indicate that further studies are urgently needed.

Title: Numerical evaluation of cracked pipes under dynamic loadings using a special finite element.

Author: Petit,-M.; Jamet,-P. **Corp. Author:** CEA-CEN

Source: Hadjian,-A.H. (Bechtel Power Corp., Los Angeles, CA (USA)). Transactions of the 10th SMiRT Conference. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1989. 375 p. p. 314-346.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: LBB methodology **ID:** 57

Abstract: In order to apply the LBB concept to piping systems, the behavior of cracked pipes under dynamic, and especially seismic, loading must be studied. A simple finite element model of a cracked pipe has been developed and implemented in a general purpose computer code. This model has been used for the design of dynamic tests to be conducted on shaking tables. The influence of the frequency of excitation was studied. Evaluation of margin, due to plasticity, in comparison with the standard design procedure is made.

Title: Analysis of fatigue crack growth and unstable fracture in carbon steel piping.

Author: Kashima,-K.; Matsubara,-M.; Miura,-N.; Takumi,-K. **Corp. Author:**

Source: Hadjian,-A.H. (Bechtel Power Corp., Los Angeles, CA (USA)). Transactions of the 10th iSMiRT Conference. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1989. 375 p. p. 81-86.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: LBB methodology **ID:** 58

Abstract: Establishment of LBB concept for nuclear piping is important form the viewpoints of structural integrity and design rationalization in light water reactors. For this purpose, extensive research is in progress on the application of fracture mechanics approach to LBB evaluations. The objective of the present study is to analyze the fatigue crack growth behavior and unstable fracture conditions for Japanese carbon steel piping based on fracture mechanics approach. Analytical solutions are compared with experimental results obtained from the pipe tests in Japan.

Title: Development of a leak-before-break procedure for pressurized components.

Author: Langston,-D.B.; Haines,-N.F.; Wilson,-R. (CEGB, Berkeley Nuclear Lab., Berkeley (UK)) **Corp. Author:**

Source: Hadjian,-A.H. (Bechtel Power Corp., Los Angeles, CA (USA)). Trans.10th SMiRT Conference, Los Angeles (CA). American Association for Structural Mechanics in Reactor Technology. 1989. 375 p. p. 287-292.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: LBB justification **ID:** 59

Abstract: For pressurized components there is an increasing interest in the use of leak-before-break arguments to show that defects will behave in a failsafe manner by growing in such a way as to cause a detectable leak before a disruptive failure of the pressure boundary can occur. The authors' company operates a wide variety of plant and has recognized the need for a flexible leak-before-break procedure which can be applied in a variety of different situations rather than the more rigid code approach adopted for LWR piping for example in NUREG-1061. This paper describes the development of such a procedure and discusses some of the key aspects of the leak-before-break procedure.

Title: Experimental study on PORV break LOCA in PWR plants.

Author: Kawanishi,-Kouhei; Nakamori,-Nobuo; Tsuge,-Ayao; Kodama,-Kenji; Kohriyama,-Tamio; Nagumo,-Hiroichi **Corp. Author:**

Source: Journal-of-Nuclear-Science-and-Technology-Tokyo. (Feb 1990). v. 27(2) p. 133-148.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Analysis of break effects **ID:** 60

Abstract: Small break LOCA tests simulating a PORV break LOCA were performed using the EOS (Emergency of System) test facility. The break sizes were 0.25 and 0.88% of a guillotine break of a primary piping. The following major conclusions were obtained and the useful data and information for the verification of a computer code were obtained: (1) The pressurizer was almost full of water due to flooding limitation in the surge line of the pressurizer, when no water was in the hot leg piping. (2) The core was kept completely covered with two-phase mixture during the small LOCA. (3) The core was sufficiently cooled down by reflux condensation in the steam generator even after the primary system natural circulation stopped. (4) After the depressurization of the primary system was stopped or when the depressurization rate of the primary system was small, the operator action to bleed the steam in the secondary system could steadily depressurize the primary system. (5) This operator action could resume the natural circulation in the primary system. (author).

Title: Regulatory experience in Canada on leak-before-break.

Author: Jarman,-B.Seminar on LBB: further developments in regulatory policies and supporting research. Taipei, Taiwan (China). 11-12 May 1989. **Corp. Author:**

Source: Wilkowski,-G.M.and Chao,-K.S. (eds.). Leak-Before-Break: Further developments in regulatory policies and supporting research. Feb 1990. 350 p. p. 179-210.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: LBB methodology **ID:** 61

Abstract: The paper discusses regulatory experiences in Canada on LBB. The paper also discusses the probability of failures in the large diameter (21-inch) heat transport piping. Several examples of cracked pipes and pipe components are given. It is concluded that they have a concern that leak-before-break may become a rationale for eliminating important and effective measures in the defense in depth concept (i.e., in-service inspection) along with poorly conceived measures such as pipe whip restraints.

Title: LBB application optimization must be our goal.

Author: Arlotto,-G.A. **Corp. Author:**

Source: Wilkowski,-G.M. et al, 1989. Leak-Before-Break: Further developments in regulatory policies and supporting research. Feb 1990. 350 p. p. 1-12.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: LBB justification **ID:** 62

Abstract: The paper addressed LBB as a goal for optimization. LBB applications were noted as being currently limited to exclusion of hardware for dynamic effects from a pipe break. The Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards recommended that to encourage a technological pursuit of evidence that could justify potential future LBB applications, an avenue for consideration of further extension of the LBB concept should exist. Future applications could be for containment design, Emergency Core Cooling System (ECCS) design, or equipment qualification.

Title: Application of leak-before-break justification approach to BWR piping.

Author: Mehta,-H.S.; Patel,-N.T.; Chexal,-B. (Electric Power Research Inst., Palo Alto, CA (USA)) **Corp. Author:**

Source: Proceedings-of-the-American-Power-Conference. (1988). v. 50 p. 617-622.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: LBB justification **ID:** 63

Abstract: The NRC has published initial guidelines for application of the leak-before-break (LBB) approach. This paper reports the results of a study to develop criteria for applying the LBB approach to boiling water reactor (BWR) piping systems and to determine the order in which high energy piping systems should receive LBB evaluations. The author identify major LBB related to the fracture mechanics technology in the application of the LBB approach. They demonstrate a typical LBB analysis.

Title: Analysis of the failure performance of internally pressurized piping with surface flaws.

Author: Iorio,-A.F; Crespi,-J.C. **Corp. Author:** CNEA

Source: Third Latin American colloquium on technological developments in failure analysis in Buenos Aires, 19-23 October, 1987, pp 79-88.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Criteria **ID:** 64

Abstract: Due to frequent failures an Atucha-1 PHWR moderator circuit branch piping, made of stainless steel type AISI 347 (DIN 1.4550), studies have been made, involving the application of several fracture mechanics criteria, in order to determine the conditions of leak-before-break (L.BB) and the critical crack length of the piping. These studies lead to the conclusions that, for a straight pipe of outer diameter of 219 mm and 16 mm wall thickness, with a circumferential flaw and the principal stress being in the bending, the L.BB criteria are satisfied, being the critical crack length of the order of 400 mm. A better mechanical finishing and heat treatment was suggested in order to improve the resistance to crack initiation. (Author).

Title: Degraded piping program - Phase II. Battelle Columbus Division.

Author: Ahmad,-J.; Barnes,-C.; Brust,-F.; Guerrieri,-D.; Kramer,-G.; Landow,-M.; Marschall,-C.; Nakagaki,-M.; Papaspyropoulos,-V.; Scott,-P. **Corp. Author:** U.S. NRC

Source: Nuclear Regulatory Commission, Washington, DC (USA). Div. of Engineering. Compilation of contract research for the Materials Engineering Branch, Division of Engineering. Annual report for FY 1987. pp. 107-131.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Methods **ID:** 65

Abstract: The overall objective of the Degraded Piping Program is to verify and improve simple estimation schemes to predict the fracture behavior of circumferentially cracked pipe. The program is limited to quasi-static fracture and cracks in straight pipe. There are a variety of materials, flaw geometries, pipe sizes, and loading conditions evaluated. In 1987, many topical reports were completed on the following work packages: leak-before-break analysis of cracked pipe; significance of results on in-service flaw acceptance criteria; and impact of material characterization evaluation and unusual fracture modes.

Title: Guillotine breaks indirectly caused by seismically-induced failures.

Author: Holman,-G.S.; Lo,-T. **Corp. Author:** LLNL

Source: Weiss,-A.J. (Ed.). Proc. 16th Water Reactor Safety Information Meeting, Vol. 3, Nuclear plant aging, structural and seismic engineering, mechanical research, environmental effects in primary systems. Mar 1989. pp 213-246.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Methods **ID:** *66*

Abstract: The LLNL has developed techniques for evaluating how piping support failures caused by earthquakes would contribute to the overall probability of piping system failure. These techniques have been applied to evaluate various reactor coolant piping systems in both PWR and BWR plants. These evaluations typically found that the likelihood of pipe break due to seismically-induced support failure is small, not only for the large, stiff piping found in PWR primary systems, but for more complex, more flexible piping systems as well. We have also applied these reliability assessments have also been applied to specific regulatory issues such as the safety significance of various support failure scenarios, identifying individual supports whose failure would most serious affect system integrity, and assessing system failure on the basis of realistic failure criteria. The usefulness of such evaluations in a regulatory context has been demonstrated through recent NRC rulemaking actions, which were based in large part on the results of LLNL piping reliability studies. 8 refs., 8 figs., 6 tabs.

Title: Probability of crack-induced failure in the BWR recirculation piping.

Author: Holman,-G.S. **Corp. Author:** LLNL

Source: Weiss,-A.J. (Ed.), 1989. Proc. 16th Water Reactor Safety Information Meeting. Vol. 3, Nuclear plant aging, structural and seismic engineering, mechanical research, environmental effects in primary systems. pp 185-212.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Failure probability **ID:** *67*

Abstract: The LLNL has estimated the probability of DEGB in RCS piping of Mark I BWR plants. Two causes of pipe break are considered: crack growth at welded joints and the seismically-induced failure of component supports. For the former a probabilistic fracture mechanics model is used, for the latter a probabilistic support reliability model. This paper describes a probabilistic model developed to account for effects of IGSCC. The IGSCC model, based on experimental and field data compiled from several sources, correlates times to crack initiation and crack growth rates for Types 304 and 316NG stainless steel against material-specific damage parameters which consolidate the separate effects of coolant environment (temperature, dissolved oxygen content, level of impurities), stress (including residual stress), and degree of sensitization. Application of this model to actual BWR recirculation piping shows that IGSCC clearly dominates the probability of failure in 304SS piping, mainly due to cracks that initiate within a few years after plant operation has begun. Replacing Type 304 piping with 316NG reduces failure probabilities by several orders of magnitude. 11 refs., 16 rigs., 1 tab.

Title: Experiments with the behaviour of safety valves in fluid-carrying systems.

Author: Benitz,-K. (ABB Reaktor GmbH, Mannheim (Germany)); Grams,-J. (ABB Kraftwerke AG, Mannheim (Germany)) **Corp. Author:**

Source: Bauer,-K.G. (ed.). Deutsches Atomforum e.V., Bonn (Germany). HIGH SERVE '90 - nuclear engineering services. HIGH SERVE '90 - Service fuer die Kerntechnik. Bonn (Germany). INFORUM Verl. 1991. 363 p. p. 205-210.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** German

Category: Pressure ripple/water hammer **ID:** *68*

Abstract: The dynamic loads on feed pipes together with impact-like pressure fluctuations caused damages to valves and pipes up to total failure. To handle this complex, ABB reactor has developed a comprehensive testing concept for safety valves in fluid-carrying systems. This concept centers on the mathematical checking of valves and feed pipes. From the results of such calculations specific improvement measures can be derived, if necessary. Thus global and, in view of recurrent tests, problematic backfitting measures can be foregone. (orig./DG).

Title: Seismic fragility analysis of buried steel piping at P, L, and K reactors.

Author: Wingo,-H.E. **Corp. Author:** Westinghouse

Source: Oct 1989. 22 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Research/theoretical **ID:** 69

Abstract: Analysis of seismic strength of buried cooling water piping in reactor areas is necessary to evaluate the risk of reactor operation because seismic events could damage these buried pipes and cause loss of coolant accidents. This report documents analysis of the ability of this piping to withstand the combined effects of the propagation of seismic waves, the possibility that the piping may not behave in a completely ductile fashion, and the distortions caused by relative displacements of structures connected to the piping.

Title: AECB staff review of Bruce NGS 'A' operation for the year 1989.

Author: **Corp. Author:** AECB

Source:

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Operating experience **ID:** 70

Abstract: The operation of the Bruce Nuclear Generating Station 'B' is monitored and licensing requirements are enforced by the Atomic Energy Control Board (AECB). This report records the conclusions of the AECB staff assessment of Bruce NGS 'A' during 1989 and the early part of 1990. Overall operation of the station met acceptable safety standards. Despite numerous problems and technical difficulties encountered, station management and supervisory personnel acted with due caution and made decisions in the interests of safety. There was evidence of improvement in a number of key areas, supported by pertinent indicators in the objective measures table. The extensive inspection and maintenance programs carried out during the year revealed the extent of component deterioration due to aging to be larger than expected. Hydrogen embrittlement of pressure tubes, erosion/corrosion of steam and feed water valves, heat exchanger tubes and piping, fouling of boilers and heat exchangers, and environmental damage of electrical equipment are examples. Continued aging of plant equipment and its potential for reducing the margins for safe operation must be taken into account by Ontario Hydro in establishing priorities and target dates for completion of actions to resolve identified problems at Bruce NGS 'A'. (2 tabs.).

Title: Control system for checking corrosion-erosion effects on the pipelines of underground gas reservoirs.

Author: Kigyos,-J. (East Hungarian Oil and Natural Gas Co., Hajduszoboszlo (Hungary). Hajuduszoboszlo Unit) **Corp. Author:** Symposium on developments a

Source: United Nations Economic Commission for Europe (ECE), Geneva (Switzerland). Underground storage of natural gas and LPG. Geneva (Switzerland). UN. 1990. 545 p. p. 449-473.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Erosion-corrosion experience **ID:** 71

Abstract: A procedure and required means to diagnose the damaging effects on the pipelines of underground gas reservoirs are described. They include: determination of the inner corrosion of pipelines; ensuring the mounting and dismounting of the rings without interruption of the pipeline operation, and ensuring the constant flow cross section of the test equipment; observation and determination of erosion effects with the extension of the function of the test equipment by installing erosion probes into already improved fitting; activating an alarm-interlocking system by the existing instrument with multi-channel continuous detection. 10 figs, 1 tab.

Title: Study on structural strength of carbon steel pipes in applying ice plugging procedure.

Author: Gotoh,-Nobuho; Ishiwata,-Masayuki; Kanno,-Satoshi (Hitachi Ltd., Tokyo (Japan)); Kanno,-Minoru **Corp. Author:**

Source: Shibata,-Heki (ed.). Trans. 11th SMiRT Conference. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. suppl. p. 55-60. ISBN 4-89047-060-3

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Maintenance/repair experience **ID:** 72

Abstract: In in-service plant pipings, a procedure called 'ice plugging' or 'freezing' is practical for purposes of maintenance or pressure testing. The concept of the procedure is the formation of internal ice plug to temporarily block water filled pipes. In this research, two types of test were conducted to assess the applicability of this procedure especially for carbon steel piping. In the material property tests, no differences of properties were found between the conditions as received and after low temperature holding in typical carbon steelpipe material, STPT42 (JIS G3456). In the mock-up tests, unnotched and notched pipes (STPT42, nominal dia. 40A) were subjected at -80degC, and reliable ice plugging performance and no permanent pipe damage were resultantly comfirmed. (author).

Title: A comparison of damage assessment techniques in the evaluation of Cr-Mo piping specimens.

Author: Melnick,-R.M.; Thomas,-R.D. Jr.; DeLong,-J.F. **Corp. Author:**

Source: Bamford,-W.H. (Ed.). Service experience in operating plants 1991. PVP-Volume 221. New York, NY (United States). American Society of Mechanical Engineers. 1991. 126 p. p. 73-90.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Experience/events **ID:** 73

Abstract: This paper reports on examination of pipe specimens from Cromby main steam (MS) lines and Eddystone Unit 1 reheat steam (RHS) lines using various metallurgical techniques. The Cromby MS lines had operated for close to 250,000 hours; the Eddystone RHS lines, for almost 160,000 hours. Stress-rupture and creep strain rate tests were conducted. The results show rupture times close to the minimum range for virgin Cr-Mo steels. Life assessment indicate a life consumption of less than 50% for the Cromby pipe and less than 25% for the Eddystone pipe. Metallographic techniques identified scattered creep voids, but only in the Cromby MS and the Eddystone RHS pipe. Identification of carbide types showed evidence of transformations offering a qualitative assessment of life consumption. Hardness in the bainitic constituent was found to decrease as interparticle spacing increased and carbide coarsening took place at elevated temperatures. These microstructural changes account for the modest loss in toughness as measured by the upward shift in the Charpy transition temperatures. Measurements of chromium and molybdenum diffusion are offered as a means of assessing service life; by these techniques Cromby pipes are the to have exhausted their useful life, while the Eddystone pipes have a life consumption of less than 26%.

Title: Stratification and the operational aspects of nuclear-power plants.

Author: Obadiah,-R.; Bain,-R.A.; Van-Duyne,-D.A. (Stone & Webster); Bankley,-A.V.; Dwivedy,-K.K. (Virginia Power, Innsbrook, VA) **Corp. Author:**

Source: Penfield,-S.R. Jr. (Ed.). Excellent and economic nuclear plant performance. NE-Volume 4. New York, NY (United States). American Society of Mechanical Engineers. pp 119-126.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Thermal stratification **ID:** 74

Abstract: Operating experience indicates that thermal stratification is a significant event unaccounted for in original nuclear-power-plant designs. With the aging of nuclear plants, the long-term effects of thermal stratification on several piping systems have been exhibited in through-wall cracks, damaged supports, and thermal fatigue. This paper describes a comprehensive program developed and implemented at Virginia Power's North Anna and Surry pressurized water reactor power stations to address thermal stratification of the pressurizer surge line. Field inspections, temperature and displacement measurements synchronized with various plant events, and analytical evaluations indicate that stratification can be satisfactorily accounted for without undue restrictions on plant operations and with only minor hardware modifications to the pipe supports and pipe whip restraints. The fatigue evaluation rigorously considered measured stratification profiles, specified operating conditions, striping, and additive thermal stratification cycles.

Title: Aging and low-flow degradation of auxiliary feedwater pumps.

Author: Adams,-M.L. (Case Western Reserve Univ., Cleveland, OH (United States). Dept. of Mechanical and Aerospace Engineering) **Corp. Author:**

Source: U.S. NRC. Aging Research Information Conference. Rockville, MD (United States). 24-27 Mar 1992.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Pressure ripple/water hammer **ID:** 75

Abstract: This paper documents the results of research done under the auspices of the Nuclear Regulatory Commission Nuclear Plant Aging Research Program. It examines the degradation imparted to safety Auxiliary Feedwater System pumps at nuclear plants due to the low flow operation. The Auxiliary Feedwater (AFW) System is normally a stand-by system. As such it is operated most often in the test mode. Since few plants are equipped with full flow test loops, most testing is accomplished at minimum flow conditions in pump by-pass lines. It is the vibration and hydraulic forces generated at low flow conditions that have been shown to be the major causes of AFW pump aging and degradation. The wear can be manifested in a number of ways, such as impeller or diffuser breakage, thrust bearing and/or balance device failure due to excessive loading, cavitation damage on such stage impellers, increase seal leakage or failure, sear injection piping failure, shaft or coupling breakage, and rotating element seizure.

Title: Elastic-plastic analyses of cracked straight and curved pipes under bending.

Author: Brochard,-J. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)); Chhu,-S.C.; Nedelec,-M. **Corp. Author:**

Source: Shibata,-Heki (ed.), Transactions of the 11th SMiRT Conference, Vol. G2, pp 219-224. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Damage probability **ID:** 76

Abstract: Pipe no.5 calculation validated 3D thin shell analyses for prediction of the elastic plastic and fracture behaviour of stainless steel cracked straight pipes. In case of carbon steel pipes, for which initiation would occur at low load level, several computations, with increasing crack angles, or a damage technique might be necessary to simulate the propagation. For the cracked elbow, numerical prediction is not as well as for the straight pipe, but is acceptable regarding the notable variations of wall thickness and material characteristics. (author).

Title: Behavior of complex loaded components in the creep range.

Author: Kussmaul,-K.; Maile,-K.; Eckert,-W. (Staatliche Materialpruefungsanstalt, Stuttgart (Germany)) **Corp. Author:**

Source: Bamford,-W. (Ed.). Fatigue, fracture, and risk 1991. PVP-Vol. 215. New York (NY). American Society of Mechanical Engineers, pp 141-146.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 77

Abstract: Components of power and other plants which operate in the elevated temperature range, where time-dependent creep deformation occurs are exposed to complex loading. Creep and fatigue loading but also corrosive influences constitute essential factors limiting the lifetime of these components. Typical loading situations are explained using examples of power plant components. In this paper selected research projects running in the Federal Republic of Germany and in particular at MPA Stuttgart and the results obtained to date are presented. The projects concentrate on the treatment of creep-fatigue on nozzles of piping and valve casings, the damage process in pipe bends under static creep loading, the life assessment of a dissimilar weldment in the watersteam circuit of the HTR plant and the failure and deformation behavior of a reactor tank wall of an LMFBR under creep loading with superimposed bending moment.

Title: Flow-induced damage in valves and piping: A regulatory perspective.

Author: Koscielny,-S.S. (Nuclear Regulatory Commission, Washington, DC (United States)) **Corp. Author:**

Source: Evans,-S.O. (Ed.). Proceedings: EPRI power plant valves symposium 3. Jun 1991, pp 3A.1-3A.12.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Experience/events **ID:** 78

Abstract: On 12/9/86, Virginia Power identified flow-induced damage to piping systems and components at Surry-2. The failure occurred in the 18" suction line to the 'A' main feed pump at a 90-degree elbow about 1' downstream of a tee from the 24" MFW-header. The catastrophic failure resulted in complete separation and dislocation of the suction line. The heated, pressurized water in the feed system flashed to steam causing injury to eight individuals in the turbine building. On 12/13/88, the CP&L identified valve body damage during an inspection at Brunswick-1. This damage resulted from significant but localized erosion on the internal surfaces of several cast carbon steel valves. The valves that were identified as having erosion damage are used during throttling conditions to control cooldown during shutdown conditions. On 3/23/90, piping downstream of a level control valve for the 'B' low-pressure heater drain pump failed at Surry-1. Initial information indicated that the failure mechanism was erosion/corrosion caused by the single-phase flow that had higher localized flow velocities immediately downstream of the flow control valve. These are only a select group of problems noted in valves and piping. The plant piping failures and the subsequent challenges to safe reactor operation are of concern to the NRC.

Title: Pump full-flow test valve cavitation problems and their solutions.

Author: Ozol,-J.; Horbaczewski,-M. (Commonwealth Edison, Downers Grove, IL (United States)) **Corp. Author:**

Source: Evans,-S.O. (Ed). Proceedings: EPRI power plant valves symposium 3. Jun 1991. 580 p. p. 3A.43-3A.75.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Other **ID:** 79

Abstract: Valve body erosion became an NRC concern in 12/88 when cavitation in pump full-flow test throttling valves were discovered to be the cause of severe valve body material pitting and erosion at BWR plants. To assess this valve cavitation erosion issue, NRC issued 'Information Notice No. 89-01: Valve Body Erosion'. Cavitation may limit the flow through the valve and thus the valve is undersized; cavitation may cause material damage to valve parts, trim, or valve body, or erodes downstream piping and thus the valve or piping leaks; and cavitation may cause noise and vibration, which may cause major damage or destruction to equipment such as valve positioners, actuators, pipe supports and sometimes to other downstream valves. The above problems produced by cavitation are by far the most common and present the biggest maintenance problem to the utilities. The purpose of this paper is to enhance the above information and explain many of the subtle and salient features of valve cavitation induced problems and their effects on the piping system. This paper discusses eight problems induced to the valve and to the piping system by the cavitation pump full-flow test valve, orifices, and pump recirculation valve; and provides recommendations for best solution to this problem.

Title: Industry survey on experience with main steam line thermal quenching.

Author: Mandke,-J.S.; Burghard,-H.C. (Southwest Research Inst., San Antonio, TX (United States)); Lamping,-G.A. (Karta Technology, Inc., San Antonio, TX (United States)); Campbell,-W.A. (SaskPower, Regina, Saskatchewan (Canada)) **Corp. Author:**

Source: Proceedings-of-the-American-Power-Conference. (1991). v. 53 p. 470-473.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Experience/events **ID:** 80

Abstract: A survey of the North American power industry was commissioned by the Canadian Electrical Association to gather information about the experiences in fossil fuel-fired power stations with thermal quenching of main steam piping. Thirty utility industry companies responded to a questionnaire which asked for details about thermal quenching incidents and mechanisms. Fifteen companies responded that thermal quenching occurrences had resulted in main steam line damage such as wall cracks, permanent pipe distortions, severe damage to pipe hangers and supports, and loss of pipe material structural properties. Information about the corrective actions taken was acquired, including pipe repairs, design modifications, operational changes, and new monitoring measures. The paper presents a compilation of the industry survey results.

Title: Improving check valve reliability through research regarding degradation of valve internals.

Author: Kalsi,-M.S.; Horst,-C.L.; Wang,-J.K. (Kalsi Engineering, Inc., Sugar Land, TX (USA)) **Corp. Author:**

Source: Weiss,-A.J. (Ed.). Proc. 17th Water Reactor Safety Information Meeting. Vol. 1. pp 27-37.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Other **ID:** 81

Abstract: Degradation of check valve internal parts during normal operating conditions has been responsible for a majority of the check valve failures at US NPPs. Even though the actual failure rates have been relatively low, failures have been responsible for extensive damage to piping systems and have raised concerns about the reliability of the safety systems. There has been a significant lack of reliable technical data that can be used to identify the problem installations and quantify the severity of expected degradation. This paper presents a summary of recent research performed in the last three years toward development of quantitative prediction models to fill this gap.

Title: Plant monitoring, an increasingly important NDT task; a practical example in earthquake experiments.

Author: Dobmann,-G.; Brinette,-R.; Weiss,-R. (Fraunhofer-Institut fuer Zerstoerungsfreie Pruefverfahren, Saarbruecken (Germany)) **Corp. Author:**

Source: Deutsche Gesellschaft fuer Zerstoerungsfreie Pruefung e.V., Berlin (Germany). Modern nondestructive testing. Analyses and forecasts. Proc. Moderne ZfP. Analysen und Prognosen. Vortraege und Plakatberichte. pp 256-262.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Methods **ID:** 82

Abstract: From the point of view of plant safety, a knowledge of the reaction of pipeline systems is of special interest if previously damaged components are additionally loaded with dynamic fault case loads, for example in an earthquake. The determination of load-bearing reserves of previously damaged components under these load conditions and the confirmation of the validity of leak - before break criteria are of interest. These are the reasons for carrying out experiments in the German reactor safety research programme. They were done in primary safety circuit pipeline systems of the hot steam reactor. Potential sensor processes were used to inspect crack growth. (orig./DG).

Title: Early detection of creep damage by ultrasonic and electromagnetic techniques.

Author: Willems,-H.; Dobmann,-G. (Fraunhofer-Institut fuer Zerstoerungsfreie Pruefverfahren (IzfP), Saarbruecken (Germany, F.R.)) **Corp. Author:**

Source: Nuclear-Engineering-and-Design. (Jul 1991). v. 128(1) p. 139-149.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Methods **ID:** 83

Abstract: Residual lifetime analysis of components of power plants requires information on the degree of damage in the material. In the case of creep damage in components such as pipe bends, it is necessary to detect damage at the stage of micropore formation in order to ensure safe operation. Based on the influence of porosity on physical material properties (density, elastic moduli, electrical resistivity, coercivity), the potential of several NDT techniques for the detection of creep cavities is discussed. Changes in density and elastic moduli can be traced by ultrasonic velocity measurements. Experimental results obtained so far under laboratory conditions show rather good agreement with theoretical estimations. The practical applicability of the techniques used has still to be demonstrated, which is the objective of further work. (orig.).

Title: Crack opening in a pre-damaged piping under high pressure surge strains due to rapid closure of valves.

Author: Kobes,-E.; Diem,-H., Brosi,-S., Schrammel,-D. **Corp. Author:** KFK, PSI

Source: Katzenmeier,-G. (Comp.). 14. Statusbericht des Projektes HDR-Sicherheitsprogramm des Kernforschungszentrums Karlsruhe. Arbeitsbericht 05.48/90. 1990. 425 p. p. 395-424.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Pressure ripple/water hammer **ID:** *84*

Abstract: The blowdown accident experiments aim at ensuring a plant-related, realistic leak-before-break criterion for pipework. In addition, the structure-dynamic reaction of the piping due to the transient course of the simulated blowdown accident is determined by means of linear and non-linear calculations. Marginal conditions for damaged piping components are defined for local, three-dimensional finite-element models by which the mechanics of their break behaviour can be studied. (DG).

Title: Consequences of pressure/calandria tube failure in a CANDU reactor core during full-power operation.

Author: Muzumdar,-A.P.; Frescura,-G.M. (Ontario Hydro, Toronto, ON (Canada)) **Corp. Author:**

Source: Canadian Nuclear Society, Toronto, ON (Canada). Proceedings of the Canadian Nuclear Society 8. Annual Conference. 1987. 483 p. p. 31-39.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Analysis of break effects **ID:** *85*

Abstract: The consequences of a hypothetical rupture of a fuel channel i.e., simultaneous failure of both the pressure and calandria tubes, are described generically for CANDU reactors. The transient hydrodynamic and impact loads, and the steady state jet forces resulting from such an accident are discussed for various rupture geometries. Various possible modes of damage to the in-core structures are evaluated. The possibility of fuel ejection is assessed to quantify the potential mechanical damage to the in-core structures due to projectiles. The pressure loading on the calandria vessel is shown to result only in elastic stresses within the vessel within the vessel wall. The adjacent fuel channels are shown to be well able to withstand the mechanical loadings imposed on them, so that channel failure propagation is precluded. Some guide tubes of the reactivity devices are likely to be damaged due to a combination of the hydrodynamic loads, impact by fuel projectiles, and pipe whip. The extent of damage to the shut-off rod guide tubes is quantified for various Ontario Hydro reactors. For each reactor, calculations of the reactivity depth of the available shut-off rods show that SDS1 acting alone is still capable of shutting down the reactor, and maintaining subcriticality with sufficient margin.

Title: Status - risk evaluation from aging of passive components.

Author: Phillips,-J.H.; Nguyen,-S.M.; Roesener,-W.S.; Magleby,-H.L. (Idaho National Engineering Lab., Idaho Falls (USA)) **Corp. Author:**

Source: Weiss,-A.J. (comp.). Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research. Trans. 18th Water Reactor Safety Information Meeting. Oct 1990. 211 p. p. 7.7-7.8.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Aging analysis **ID:** *86*

Abstract: The risk of core damage at all nuclear power plants is being determined using PRA techniques. These assessments do not consider aging and consider the failure of passive components to only a limited degree. As a result of the low failure probability of the relatively new passive components, the failure of passive components is often ignored or not given adequate consideration in PRAs. Because of the large number of passive elements (e.g., many feet of pipe, numerous valve bodies, and pump casing), the large consequence when these components fail, and the increasing failure probability due to aging, passive components should be considered in PRA calculation of core damage risk. The purpose of this project is to develop techniques to incorporate the effects of passive element failure into PRAs. The increased risk of core damage is calculated as a result of the aging of passive components. This effort will contribute to the U.S. Nuclear Regulatory Commission Nuclear Plant Aging Research Program.

Title: Inserting auxiliary equipment to the heat steam pipeline of the Paks Nuclear Power Plant, Hungary for reducing dama

Author: Kertai,-Pal (Paksi Atomeroemue Vallalat, Paks (Hungary)) **Corp. Author:**

Source: PAV-Koezlemenyek. (1990). (Mar 1991). (no.1) p. 52-54.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** Hungarian

Category: Other **ID:** 87

Abstract: Erosion and corrosion effects in nuclear power plants due to the effect of wet steam and their control thechniques are discussed. In the Paks Nuclear Power Plant units, water droplet separating devices are planned to be inserted into the stage 1 of the drop separator overheater (CSTH) unit, in order to trap water droplets driven by the steam by an angle tube. The expected erosion control of the inserted device of the heat steam pipe is explained. (R.P.) 8 figs.

Title: Damage and fracture mechanisms of an aged duplex stainless steel. Stainless steel CF8M.

Author: Joly,-P.; Pineau,-A. (Ecole des Mines de Paris, 91 - Evry (France)) **Corp. Author:**

Source: Institut National des Sciences et Techniques Nucleaires (INSTN) - Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Microstructural Aspects of Rupture. Aspects Microstructuraux de la Rupture. Paris-La-Defense (France). Revue de Metallurgie. 1990. 328 p. p. 49-58.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** French

Category: Test/analysis **ID:** 88

Abstract: Damage and fracture micromechanisms of an aged duplex stainless steel, containing 20% ferrite used for pipes, bends and pump shell of nuclear power plants, are investigated. Deformation modes are studied both with optical and electron scanning microscopy. The effect of stress triaxiality on ductility is investigated by using smooth and notched specimens. Mechanical parameters controlling nucleation of microcracks in the notch, are derived from finite element calculations. We present a model including cavity nucleation and growth. These cavities are nucleated from cleavage microcracks in the ferrite phase. The relationship between fracture toughness and crack tip opening displacement (CTOD) is established.

Title: Fission product transport in the reactor coolant system for a spectrum of interfacing system LOCA scenarios.

Author: Warman,-E.P.; Metcalf,-J.; Hessian,-R.; Donahue,-M. (Stone and Webster Engineering Corp., Boston, MA (USA)) **Corp. Author:**

Source: Rogers,-J.T. (Carleton Univ., Ottawa, Ontario (Canada)). Fission product transport processes in reactor accidents. New York, NY (USA). Hemisphere Publishing. 1990. 865 p. p. 329-338.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Other **ID:** 89

Abstract: One of the most important potential severe accident sequences for PWR) is an ISLOCA. As initially described in the reactor safety study WASH-1400, interfacing system LOCAs involved the failure of check valves inECCS but could also involve the RHRS. The check valves protect the low-pressure portions of these systems from the high pressures of the reactor coolant system (RCS) to which they are connected to provide cold leg injection. A consequent break in the low-pressure piping outside the containment may result in core damage and a direct pathway for fission products to be transported from the core, through the RCS and ECCS or RHR to the auxiliary building, from which they can escape to the environment. This paper addresses the retention and transport of fission products (specifically, CsI) in the RCS in V-sequence scenarios. It summarizes some of the major differences between models resulting from the latest version of the IDCOR-MAAP Computer Program, MAAP 3.0B. Discussed are the differences in: fission product transport and retention in small, medium, and large ECCS pipe breaks, as well as the effect of ECCS and AFWS operation and fission product retention in the various regions of the RCS as calculated by MAAP 3.0B and the STCP.

Title: Mechanical damage experience in major light water reactor systems.

Author: Ware,-A.G. (Idaho National Engineering Lab., EG and G Idaho, Inc., Idaho Falls, ID (USA)) **Corp. Author:**

Source: Chung,-H.H. et al. Advances in Dynamics of Piping and Structural Components. PVP-Volume 198. New York, NY (USA). American Society of Mechanical Engineers. 1990. 83 p. p. 7-14.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Experience/events **ID:** 90

Abstract: This paper describes the nuclear power industry's experience with mechanical (as opposed to thermal or electrochemical) damage in the major systems of light water reactor (LWR) plants. Almost all of the occurrences of damage were caused by mechanical vibration. The sources of vibration include flow-induced vibration, water-hammer events, and pump and valve vibration. However, the damage has sometimes been initiated or aggravated by other sources, such as stress corrosion cracking, loss of preload, or corrosion-fatigue. Mechanical vibration can also cause metal loss in the walls of thin tubes when they impact with their supports. Some of the components that have experienced mechanical damage are reactor coolant pump shafts, PWR and BWR reactor vessel internals, PWR instrument tubes, thermal sleeves in piping, and steam generator tubes. Various mitigation methods can e implemented to reduce or eliminate these problems.

Title: Advances in dynamics of piping and structural components. PVP-Volume 198.

Author: Chung,-H.H. (Argonne National Lab., Argonne, IL (USA)); Goodling,-E.C. Jr. (Gilbert/Commonwealth, Inc. (USA)); Mizra,-S. (Univ. of Ottawa, Ottawa (Canada)); Sinnappan,-J. (Sargent and Lundy (USA)) **Corp. Author:**

Source: New York, NY (USA). American Society of Mechanical Engineers. 1990. 83 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Operating experience **ID:** 91

Abstract: This book contains articles presented at the 1990 Pressure Vessels and Piping Conference. Included are the following chapters: Mechanical damage experience in major light water reactor systems, Damping considerations in CANDU feeder pipe design and analysis, Seismic testing of experimental pipeline loop.

Title: Fatigue evaluation of piping systems with limited vibration test data.

Author: Huang,-S.N. **Corp. Author:**

Source: American Society of Mechanical Engineers (ASME). 1991 Pressure Vessels and Piping Conference. San Diego, CA (USA). 23-27 Jun 1991.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Experience/events **ID:** 92

Abstract: The safety-related piping in a nuclear power plant may be subjected to pump- or fluid-induced vibrations that, in general, affect only local areas of the piping systems. Pump- or fluid-induced vibrations typically are characterized by low levels of amplitudes and a high number of cycles over the lifetime of plant operation. Thus, the resulting fatigue damage to the piping systems could be an important safety concern. In general, tests and/or analyses are used to evaluate and qualify the piping systems. Test data, however, may be limited because of lack of instrumentation in critical piping locations and/or because of difficulty in obtaining data in inaccessible areas. This paper describes and summarizes a method to use limited pipe vibration test data, along with analytical harmonic response results from finite-element analyses, to assess the fatigue damage of nuclear power plant safety-related piping systems. 5 refs., 2 figs., 11 tabs.

Title: Vibration control in piping system by dual dynamic absorber. Realization of piping systems with unresonant characteri

Author: Yamashita,-Shigeo; Sawatari,-Katsumi; Seto,-Kazuto (National Defence Academy, Yokosuka, Kanagawa (Japan)) **Corp. Author:**

Source: JSME-International-Journal.-Series-3,-Vibration,-Control-Engineering,-Engineering-for-Industry. (Dec 1990). v. 33(4) p. 488-494.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Methods **ID:** 93

Abstract: This paper shows the design method for constructing a piping system with well suppressed resonance peaks through a wide range of frequencies. The piping system is strongly influenced by sources of excitation for blade vibrations since it is usually made flexible and has a low damping property. Thus, many problems, like fatigue damage or noise caused by vibration, occur frequently in the piping system. In order to suppress the resonance peaks and obtain high damping, dual dynamic absorbers proposed in the previous paper are applied. In this paper, it is confirmed theoretically by the transfer matrix method that the piping system with seven resonance peaks within 100 Hz is well suppressed by using five dynamic absorbers. The effectiveness of the five optimally designed dual dynamic absorbers is also demonstrated experimentally. (author).

Title: Thermal stratification and fatigue of piping in nuclear-power plants.

Author: Van-Duyne,-D.A.; Obadiah,-R.; Bain,-R.A. (Stone and Webster Engineering Corp., Boston, MA (USA)); Bankley,-A.V. (Virginia Power, Glen Allen, VA (USA)); Mukherjee,-S. (Duquesne Light Co., Shippingport, PA (USA)) **Corp. Author:**

Source: Truong,-Q.N., Short,-W.E. II and Ezekoye,-L.I. (Eds.). Design and Analysis of Piping and Components 1990: Including Valve Testing and Applications. PVP-Volume 188. New York, NY (USA). ASME 1990. 94 p. p. 77-82.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Thermal stratification **ID:** 94

Abstract: Thermal stratification has been the cause of fatigue cracking and unexpected movement in a number of piping systems in nuclear-power plants. The term has been applied within the industry to three types of thermal-hydraulic phenomena: local stratification in mixing flows; general stratification in horizontal piping and interrelated striping phenomena; and stratification in secondary flow regions of branch piping. This paper focuses on the latter two types and describes a practical approach to assess their contribution to cumulative fatigue damage to the pipe.

Title: Assessment and Avoidance of Erosion-Corrosion Damage in PWR Feedpipework.

Author: Woolsey,-I.S. **Corp. Author:**

Source: Proc. . International Working Group on Reliability of Reactor Pressure Components. Corrosion and Erosion Aspects in Pressure Boundary Components of LWRs, IWG-RRPC-88-1 (1990), pp 60-66.

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Erosion-corrosion **ID:** 95

Abstract: Following the MFW pipe rupture at the Surry-2 in the US, the CEGB undertook an evaluation of the possibility of similar damage in the feedpipework of other PWRs including future UK designs. The assessment method was based on an extensive body of experimental erosion-corrosion data accumulated during investigations of possible single phase erosion-corrosion in the low temperature sections (100 to 200 deg. C) of UK AGR boilers. The analysis focussed on the materials specification required to avoid significant erosion-corrosion damage throughout the feedpipework, taking account of pipework configuration, flow rates, temperature and water chemistry. It allowed identification of locations which would be potentially vulnerable to unacceptable erosion-corrosion damage over the operational life of the plant. However, significant damage could be avoided by adopting a minimum chromium specification for the carbon steel pipework, and a sufficiently high operational feedwater pH. For the majority of feed pipework it should not be necessary to use a chromium alloy steel. By adopting these measures, it is considered that the UK PWR currently under construction at Sizewell with not suffer significant erosion-corrosion damage of the main feedpipework over the full period of its operational life. (author). 14 refs, 10 figs, 1 tab.

Title: Experience of Erosion and Erosion-Corrosion in Nuclear Steam Turbines.

Author: Hedstroem,-M. (SSPB, Sweden) **Corp. Author:** IAEA

Source: Proc. International Working Group on Reliability of Reactor Pressure Components. Corrosion and Erosion Aspects in Pressure Boundary Components of LWRs, IWG-RRPC-88-1 (1990), pp 66-69.

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Erosion-corrosion **ID:** 96

Abstract: The report covers erosion-corrosion in nuclear steam turbines altogether in 12 units, 3 PWR and 9 BWR. The turbine processes varies little as far as the parameters are concerned between the BWR and PWR installations. The paper reports corrosion and erosion damages observed in steam turbines, condensers and piping. The maintenance and repair methods are also presented. 6 figs, 3 tabs.

Title: Fundamental study on probabilistic assessment of torsional vibration of base isolated FBR structure, (2). On simplified

Author: Yabana,-Shuichi (Central Research Inst. of Electric Power Industry, Abiko, Chiba (Japan). Abiko Research Lab.) **Corp. Author:**

Source: Denryoku-Chuo-Kenkyusho-Hokoku. (Jun 1990). (no.U90008) p. 1-4, 1-48.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** Japanese

Category: Other **ID:** 97

Abstract: Torsional response of FBR structure by randomness of isolation pad's characteristics is evaluated using Monte Carlo simulation. The simplified estimation method of maximum torsional angle is proposed and validated using numerical model. Using the simplified estimation method, the amount of calculation is reduced. In spite of giving big randomeness that is considered as critical level, torsional response is small, so that the equipment in the building and piping between isolated building and non-isolated building may not suffer damage. (author).

Title: Damage to heating pipes in steam generators.

Author: Debnar,-A. (Asea Brown Boveri Reaktor GmbH, Mannheim (Germany, F.R.). Abt. Technische Dienstleistungen) **Corp. Author:**

Source: Materialpruefung. (Jan-Feb 1991). v. 33(1/2) p. 17-20.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** German

Category: Inspection methods **ID:** 98

Abstract: Heating pipes in steam generators are checked regularly during their service life and damaged pipes sealed off or repaired where necessary. The standard eddy current test technique is used to locate and evaluate the damage, as well as for quality checking after repairs to heating pipes. Depending on the type of damage, different analysis processes are used for more detailed defect analysis. This paper describes the analysis techniques with reference to special images of defects in the steam generator, like buckling and changes of pipe diameter, as well as the test technology for monitoring pipe damage after repairs to heating pipes. (orig.).

Title: The 1989 progress report: Solid-state Mechanics.

Author: Habib,-P. (Ecole Nationale du Genie Rural des Eaux et des Forets, 75 - Paris (France)) **Corp. Author:** Ecole Polytechnique

Source: 1989. 31 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** French

Category: Other **ID:** 99

Abstract: The 1989 progress report of the laboratory of Solid-state Mechanics of the Polytechnic School (France) is presented. The investigations are focused on the study of strain and failure of solids and structures. The results reported concern the fields of: stability and bifurcation of elastic or inelastic systems, damage and fatigue (resistance improvement, failure risks on pipe systems, crack propagation), the development of a computer code for soil strengthening by using linear inclusions, mechanical behavior of several rocks for the safety of underground works, expert systems. The published papers, the conferences and the Laboratory staff are listed.

Title: Value/impact assessment of jet impingement loads and pipe-to-pipe impact damage. Revised methods and criteria.

Author: Brown,-J.B. Jr.; Bampton,-M.C.C.; Alzheimer,-J.M. (Pacific Northwest Lab., Richland, WA (USA)) **Corp. Author:**

Source: Jun 1990. 62 p. Nuclear Regulatory Commission, Washington, DC (USA). Div. of Engineering.Pacific Northwest Lab., Richland, WA (USA).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Analysis of break effects **ID:** 100

Abstract: To account for effects that might result from a loss-of-coolant accidents (LOCA), nuclear power plant designers have been required to analyze the effects of double-ended guillotine breaks (DEGB) in high-energy piping. The US Nuclear Regulatory Commission (NRC), through its Standard Review Plan (SRP), requires that plant designers follow certain prescribed methods and criteria in the estimation of dynamic effects associated with the postulated rupture of piping. The work reported in this NUREG is intended to provide the basis for NRC decisions on adopting revisions to parts of the SRP 3.6.2 entitled "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping." The revisions considered in this work evaluated updated prescriptions for calculating jet impingement forces on critical systems and the requirement to consider pipe-whip damage to a new population of pipes. In accordance with the procedures documented in NUREG/CR-3586 entitled "A Handbook for Value-Impact Assessment," this report found indication that substantial costs and occupational radiation exposure would result from the proposed action without substantially reducing the risks to public health and safety. 21 refs., 2 figs., 18 tabs.

Title: Evaluation of creep damage in some major components of power generator.

Author: Matousek,-J.; Svarc,-M.; Valenta,-J. (National Research Inst. for Machine Design, Bechovice (Czechoslovakia)); Loebl,-K.; Bina,-V. **Corp. Author:**

Source: ISIJ-International. (Jun 1990). v. 30(6) p. 451-456.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Methods **ID:** 101

Abstract: Life extension of installed machinery and equipment has become a matter of strategic importance with significant economic implications especially in power plants. The paper presents a complex of methods aimed at assessment of the residual service lives of major components of power plant machinery (steam turbines, steam line systems) under creep conditions. The methods are based on a mathematical model of properties of creep-resistant steels based on the dislocation mechanism for which stochastic behavior of material in the course of the strain mechanism has been assumed. A method is proposed to evaluate the equivalent service load in actual service conditions by means of a computerized data acquisition. The method is applied to monitor creep damage of steam turbine pipelines. Creep damage of turbine rotors and casings and plastic deformation in their critical points were estimated numerically. The actual time span of reliable operation of a component is assessed by these method and compared with results obtained by diagnostic methods prepared in cooperation with manufactures of power generating machinery. (author).

Title: Calculation code for erosion corrosion induced wall thinning in piping systems.

Author: Kastner,-W.; Erve,-M.; Henzel,-N.; Stellwag,-B. (Siemens AG Unternehmensbereich KWU, Erlangen (Germany, F.R.)) **Corp. Author:**

Source: Nuclear-Engineering-and-Design. (May 1990). v. 119(2/3) p. 431-438.

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Erosion-corrosion **ID:** 102

Abstract: Extensive experimental and theoretical investigations have been performed to develop a calculation code for wall thinning due to erosion corrosion in power plant piping systems. The so-called WATHEC code can be applied to single-phase water flow as well as to two-phase water/steam flow. Only input data which are available to the design engineer or the operator of a plant are taken into consideration. Together with a continuously updated erosion corrosion data base containing results from experimental investigations and actual damage in power plants the calculation code forms one element of a weak point analysis for power plant piping systems which can be applied to minimize material loss due to erosion corrosion, reduce non-destructive testing and curtail monitoring programs for piping systems, recommend life-extending measures. (orig.).

Title: In-service lifetime monitoring of piping systems taking into account external forces and moments besides internal press

Author: Bietenbeck,-F. (RW TUV, Essen (Germany, F.R.)); Rohler,-K. (Deutsche Babcock AG, Oberhausen (Germany, F.R.)) **Corp. Author:**

Source: Hadjian,-A.H. (Bechtel Power Corp., Los Angeles, CA (USA)). Trans. 10th SMiRT Conference. Volume D. Los Angeles (CA). American Association for Structural Mechanics in Reactor Technology. 1989. 259 p. p. 31-36.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Inspection methods **ID:** 103

Abstract: The continuous registration of the cumulative creep and fatigue damage of highly loaded components of high pressure piping in the creep range by means of lifetime monitoring systems contributes to maintaining the reliability and availability of the plant and, with preventive maintenance, delivers useful information about the actual state of material damage. In addition, the records of lifetime monitoring systems permit to derive parameters for an optimized operation and better utilization of the service life of the plant. Lifetime monitoring systems presently installed in high pressure piping in the creep range normally record pressures and fluid temperatures as well as the through wall temperature gradients in thick-walled components such as fittings and valves. On the basis of these data, the increase in creep and fatigue damage of the monitored components is determined through routines implemented in the central processing unit of a personal computer.

Title: Research needs for fatigue damage assessment in PWR surge and spray piping.

Author: Shah,-V.N.; Conley,-D.A. (Idaho National Engineering Lab., Idaho Falls, ID (USA)) **Corp. Author:**

Source: Hadjian,-A.H. (Bechtel Power Corp., Los Angeles, CA (USA)). Trans. 10th SMiRT Conference. Volume F. Los Angeles (CA). American Association for Structural Mechanics in Reactor Technology. 1989. 257 p. p. 87-92.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Experience/events **ID:** 104

Abstract: This paper identifies the research necessary for revising regulator requirements so that fatigue damage in PWR surge and spray piping can be realistically assessed, for developing and modifying ASME codes and standards so that fatigue damage can be detected and monitored, and for revising design practices and operating procedures so that the fatigue damage can be reduced. The authors discuss operating conditions leading to stratified flows and thermal striping, high- and low-cycle fatigue damage, estimation and detection of fatigue damage and conclusions and recommendations.

Title: Behaviour of an undamaged and a pre-damaged piping system under earthquake-like loads.

Author: Diem,-H., Malcher,-L.; Schrammel,-D. Projektbereich Heissdampfreaktor - Sicherheitsprogramm/Handhabungstechnik) **Corp. Author:**

Source: Deutsches Atomforum e.V., Kerntechnische Gesellschaft e.V. Jahrestagung Kerntechnik '90. Tagungsbericht. Bonn (Germany, F.R.). INFORUM Verl. May 1990. 710 p. p. 151-154.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Other **ID:** 105

Abstract: Published in summary form only.

Title: PWR secondary system pipe thinning.

Author: Shor,-S.W.W.; Osbourne,-M.R. (Bechtel Western Power Corp., San Francisco, CA (USA)); Wilzbach,-J.H.; Freid,-S.H. (Bechtel Power Corp., Los Angeles, CA (USA)) **Corp. Author:**

Source: Proceedings-of-the-American-Power-Conference. (1988). v. 50 p. 647-654.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Erosion-corrosion **ID:** *106*

Abstract: NPPs have experienced significant thinning of pipe walls from wet steam at least since 1962, when a leak occurred in an extraction line at Dresden-2. Many plants have had valves and piping dow-stream of valves damaged by flashing water. However, it was not until Surry-2 experienced a dramatic pipe rupture in December 1986 at the suction of a MFW pump that thinning in high energy lines carrying only liquid water attracted widespread attention, although a similar failure had occurred in a pipe on the discharge side of a heater drain pump at Trojan about 20 months earlier.Seven months after the Surry incident the NRC issued a bulletin (IEB 87-01) requiring utilities to report their programs to identify and control erosion-corrosion. The NRC also sent out a questionnaire to collect information on the secondary water chemistry of PWRs. Their responses indicate that not only is erosion-corrosion widespread but that there is need for an easy way to understand its causes in a particular plant, evaluate alternative actions for its correction and arrive at practical, cost-effective programs to control it. The paper suggests how to stop or nearly stop the progress of wall thinning and provide convincing evidence that it has been arrested. Specifically, it identifies water chemistry changes as the most cost-effective way to arrest widespread erosion-corrosion.

Title: Corrosion problems in the WWER-440 secondary cooling circuit.

Author: Koehler,-S. (Betrieb des VE Kombinat Kernkraftwerke Bruno Leuschner, Leipzig (German Democratic Republic). Inst. fuer Energetik) **Corp. Author:**

Source: Kernenergie. (Apr 1990). v. 33(4) p. 161-165.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Corrosion experience **ID:** *107*

Abstract: Corrosion behaviour of secondary coolant circuit components in pressurized water nuclear power plants is essentially determined by the chemical mode of operation. The uncorrected mode used earlier in the WWER-440 type units led to corrosion damages in the brass pipes of low-pressure feed heaters and in the pipes of the steam generators. The hydrazine mode now preferably applied in practice is characterized by insufficient corrosion behaviour of the carbon steel components. The corrosion phenomena occurring in the two modes and their causes are studied. (author).

Title: Avoidance of erosion corrosion damage in water and wet steam carrying pipes.

Author: Kastner,-W. (Siemens AG Unternehmensbereich KWU, Erlangen (Germany, F.R.)) **Corp. Author:**

Source: Deutsche Gesellschaft fuer Chemisches Apparatewesen, Chemische Technik und Biotechnologie e.V. (DECHEMA). Sicherheitsfragen in der Rohrleitungstechnik. Kurzfassungen. 1990. 4 p. p. 3.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Erosion-corrosion experience **ID:** *108*

Abstract: Published in summary form only.

Title: Flow-assisted corrosion-consequences for piping.

Author: Remy,-F.N.; Bouchacourt,-M.; Bellon,-M. **Corp. Author:**

Source: Transactions-of-the-American-Nuclear-Society. (1988). v. 57 p. 249.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Erosion-corrosion experience **ID:** 109

Abstract: For 2 yr, power plant owners have been aware that damage caused by flow-assisted corrosion (FAC) can be overcome: on the one hand with a proper choice of water chemistry and on the other hand the use of prediction methods that allow limiting the inspection areas and providing a sure and accurate fitting replacement and pipe surveillance program. To explain the phenomena, Electricite de France uses an analytical approach, made possible by 10 yr of loop tests and the feedback from many operating plants. If the FAC is well known when there is monophasic water flow, with the specific influence of each parameter, it is possible with this development to include the influence of the two-phase flow in the analysis. To predict the lifetime of the piping, it is necessary to take into account all loads that concern the piping's stress field, i.e., internal pressure, weight, and external load. A way is presented to integrate all consequences of the thinning and avoid unpleasant surprises when the lifetime is given with only a pressure stress analysis.

Title: Calculation code for erosion-corrosion induced wall thinning in piping systems.

Author: Henzel,-N.; Kastner,-W.; Stellwag,-B.; Erve,-M. (Siemens AG Unternehmensbereich KWU, Erlangen (Germany, F.R.)) **Corp. Author:** MPA-Stuttgart

Source: MPA Stuttgart (1988). 14. MPA-Seminar. Sicherheit und Verfuegbarkeit in der Anlagentechnik mit dem Schwerpunkt 'Langzeitintegritaet der druckfuehrenden Bauteile von Kernkraftwerken'. Bd. 1 und 2. Bd. 1: Anlagentechnik, Thermoschock, strahleninduzierte Versproedung, Korrosion/Verschleiss, zerstoerungsfreie Pruefung. Bd. 2: Zeitstandverhalten/Kriechvorgaenge, Behaelter- und Komponenten-Integritaet, Rohrleitungsverhalten. 1988. 1003 p. p. 17.1-17.21. Published in 2 separate volumes.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** German

Category: Erosion-corrosion monitoring **ID:** 110

Abstract: There was great material erosion mainly in consequence of an extremely unfavourable geometry at the damaged place in Surry-2. The pipeline sections affected in Trojan were in the area of action of great sources of turbulence, i.e.: less than 10 pipe diameters from junctions, elbows etc. Because of the many parameters which determine the amount of material removal by erosion-corrosion, the analysis of such damage is only possible using a computer program. The main purpose of such a PC code called WATHEC developed by Siemens/KWU is not the subsequent confirmation of damage which has occurred, but its application for preventive diagnosis in pipeline systems. (orig./DG).

Title: Validation of pressure boundary structural analyses at the HDR LWR plant to confirm calculation processes and the tr

Author: Katzenmeier,-G., Kussmaul,-K.; Roos,-E.; Diem,-H. **Corp. Author:** MPA-Stuttgart

Source: 14. MPA-Seminar. Sicherheit und Verfuegbarkeit in der Anlagentechnik mit dem Schwerpunkt 'Langzeitintegritaet der druckfuehrenden Bauteile von Kernkraftwerken'. Bd. 1 und 2. Bd. 1: Anlagentechnik, Thermoschock, strahleninduzierte Versproedung, Korrosion/Verschleiss, zerstoerungsfreie Pruefung. Bd. 2: Zeitstandverhalten/Kriechvorgaenge, Behaelter- und Komponenten-Integritaet, Rohrleitungsverhalten. 1988. 1003 p. p. 43.1-43.24. Published in 2 separate volumes.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** German

Category: Test/analysis **ID:** 111

Abstract: The pressure vessel of the HDR and various pipeline systems similar to the plant were loaded with static and transient loads both thermally and mechanically, until damage and crack growth (RDB) or through cracks (RL) occurred. In parallel with the component experiments in the system, laboratory tests were done on samples in autoclave conditions and comprehensive calculations were carried out for the experiments. The measured and calculated results were evaluated in order to answer the question on the transferability from samples to components. Information is summarized on the individual phases of cracks starting, crack growth, crack initiation, stable crack growth and leakage before fracture behaviour. (orig./DG).

Title: Role of damage tolerance and fatigue crack growth in the power generation industry.

Author: Coffin,-L.F. (General Electric Co., Schenectady, NY (USA)) **Corp. Author:**

Source: Cruse,-T.A. (Southwest Research Inst., San Antonio, TX (USA)). Fracture mechanics. Philadelphia, PA (USA). ASTM. 1988. 939 p. p. 235-259.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: IGSCC monitoring techniques **ID:** 112

Abstract: The problem of intergranular stress-corrosion cracking (IGSCC) in boiling water reactor (BWR) piping is discussed and the body of work undertaken in the author's laboratory to solve that problem is described. Particular attention is given to the development of electrical potential crack monitoring techniques and their application to surface crack growth, particularly under conditions approaching those found in service. The important role of water chemistry and its control is described in this context. The concept and description of sensors to monitor in situ the degree of damage containment from intergranular stress-corrosion cracking is then described, with reference to use in piping components and other types of monitoring. Finally, a concept for the life management of structures is described where damage processes are identified and monitored in situ using appropriate sensors to measure the damage rate continuously.

Title: Underground pipe leak detection system.

Author: Thompson,-G.M. **Corp. Author:** Tracer Research Corp.

Source: 10 Sep 1991; 26 Jan 1989. vp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Other **ID:** 113

Abstract: This patent describes an apparatus for detecting a leak from at least one of subsurface fluid pipes containing fluids therein and surrounded by a backfill material. It comprises volatile liquid phase tracer means for providing a gas phase detectable component in a fluid leak, a quantity of the volatile liquid phase tracer means being mixed with the fluid in the at least one subsurface fluid pipes; at least one gas permeable tubular members disposed in the backfill material above at least a portion of the at least one subsurface fluid pipes, wherein the at least one gas permeable tubular members further comprises a sintered rubber hose having an air permeability of about 6.9 liters +- 0.7 liters per minute per meter at 10.7 cm Hg pressure differential at 27 degrees C.; and access means disposed in the backfill material for accessing at least one end of the at least one gas permeable tubular members.

Title: Component external leakage and rupture frequency estimates.

Author: Eide,-S.A.; Khericha,-S.T.; Calley,-M.B.; Johnson,-D.A.; Marteeny,-M.L. **Corp. Author:** EG&G Idaho, Inc.

Source: EGG-SSRE--9639 (de92012357); 102 PAGES

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Experience/events **ID:** 114

Abstract: To perform detailed internal flooding risk analyses of nuclear power plants, external leakage and rupture frequencies are needed for various types of components - piping, valves, pumps, flanges, and others. However, there appears to be no up-to-date, comprehensive source for such frequency estimates. This report attempts to fill that void. Based on a comprehensive search of Licensee Event Reports (LERs) contained in Nuclear Power Experience (NPE), and estimates of component populations and exposure times, component external leakage and rupture frequencies were generated. The remainder of this report covers the specifies of the NPE search for external leakage and rupture events, analysis of the data, a comparison with frequency estimates from other sources, and a discussion of the results.

Title: Experiments on crack opening and leak rate behaviour of small piping components at the HDR facility.

Author: Hunger,-H.; Katzenmeier,-G. (Kernforschungszentrum Karlsruhe GmbH (Germany)); Grebner,-H. **Corp. Author:**

Source: Shibata,-Heki (Ed.). Trans. 11th SMiRT Conference. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. F p. 225-230. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3 for 17 vols.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 115

Abstract: Experiments were carried out on small bore austenitic piping components (under DN100) at the HDR (hot steam reactor) test installation for the purpose of examining crack opening and fluid discharge (leak) behaviour. The pipes at elevated internal pressure and temperature were, in addition, subjected to externally applied bending moments. The applied load, the resulting temperatures, strains, crack opening and fluid leak rates were measured. A few representive measurements on straight pipes (DN80) with circumferential flaws were selected and are presented here. (author).

Title: A cracked pipe element coupling plasticity and crack growth for leak before break applications.

Author: Brochard,-J.; Combescure,-A.; Jamet,-Ph. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)) **Corp. Author:**

Source: Shibata,-Heki (Ed.). Trans. 11th SMiRT Conference. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. G2 p. 225-230. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: LBB justification **ID:** 116

Abstract: In its actual version, the cracked pipe element is proved to be an efficient tool for Leak Before Break assessment of cracked piping system, subjected to static or dynamic loads. (Petit el al., 1989). But the precision of results obtained using this finite element depends on the accuracy of the moment-rotation data. The accuracy of (M, phi) relation could be affected by a bad prediction of the additional plastic flexibility due to the crack and also by a bad extrapolation of the CT specimen resistance curve. For plastic flexibility prediction, a new engineering method, based on experimental data, has been developed. First applications on DPII experiments are encouraging. (author).

Title: Pipe fracture behavior under high-rate (seismic) loading - The IPIRG Program.

Author: Schmidt,-R.A.; Wilkowski,-G.M.; Scott,-P.M.; Olson,-R.J. (Battelle, Columbus, OH (United States)) **Corp. Author:** U.S. NRC

Source: Weiss,-A.J. (Comp.). Trans. 19th Water Reactor Safety Information Meeting. Oct 1991. 220 p. p. 2.3-2.4.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 117

Abstract: The International Piping Integrity Research Group (IPIRG) Program was an international group program managed by the USNRC and funded by a consortium of organizations from nine nations: Canada, France, Italy, Japan, Sweden, Switzerland, Taiwan, the UK, and the US. The five-year program was conducted at Battelle in Columbus, Ohio, and was completed in July 1991. The objective of the program was to develop data that are needed to verify engineering methods for assessing the integrity of nuclear power plant piping that contains circumferential defects. The program encompassed numerous tasks including material characterization studies, updates of a pipe fracture data base, seminars and workshops, and a leak-rate investigation that involves experiments, analysis, and computer code development, but the primary focus was an experimental task designed to investigate the behavior of circumferentially flawed piping and piping systems subjected to high rate loading typical of seismic events. The behavior of flawed piping and piping systems subjected to high rate loading was investigated by conducting both separate effects experiments on simple pipe specimens and full-scale experiments on a large-diameter piping system tested at PWR conditions. Key conclusions are noted.

Title: Stratification issues and experience in operating power plants.

Author: Strauch,-P.L.; Roarby,-D.H.; Palusamy,-S.S. (Westinghouse Electric Corp., Pittsburgh, PA (United States). Nuclear and Advanced Technology Div.) **Corp. Author:**

Source: Zamrik,-S.Y. , Perez,-E.H. (Eds.), 1990. High Pressure Technology, Fracture Mechanics, and Service Experience in Operating Power Plants. PVP-Volume 192. New York, NY (United States). ASME, 104 p. p. 85-92.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Thermal fatigue **ID:** 118

Abstract: This paper provides a background of recent pipe cracks which have occurred in commercial nuclear power plants and have led to the issuance of NRC Bulletin 88-08. Specifically, these cracks occurred in unisolable sections of piping and were attributed to fatigue resulting from thermal stratification and cycling, which in tern resulted from valve leakage. Utility response to the bulletin is summarized, and details of an evaluation which addresses the issue are provided.

Title: Reactor Process Water (PW) Piping Inspections, 1984--1990.

Author: Ehrhart,-W.S.; Elder,-J.B.; Sprayberry,-R.E.; Vande-Kamp,-R.W. **Corp. Author:**

Source: Westinghouse Savannah River Co., Aiken (SC). 32. Meeting of the Weapons Agencies Nondestructive Testing Organization (WANTO). 27-29 Nov 1990.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Inspection methods **ID:** 119

Abstract: In July 1983, the NRC ordered shutdown of five BWRs because of concerns about reliability of UT-examination for detecting IGSCC. These concerns arose because of leaking piping at Nine Mile Point which was attributed to IGSCC. The leaks were detected shortly after completion of UT-examinations. At that time, investigations at Savannah River reactors determined that all conditions believed necessary for the initiation and propagation of IGSCC in austenitic stainless steel exist in SR reactor PW-systems. Sensitized, high carbon, austenitic stainless steel, a high purity water system with high levels of dissolved oxygen, and the residual stresses associated with welding during construction combine to provide the necessary conditions. A periodic UT inspection program is now in place to monitor the condition of the reactor PW-systems. Welds in upgraded or replacement piping are examined on a standard schedule (at least every five years) while welds with evidence of IGSCC, evaluated as acceptable for service, are inspected at every extended outage (15 to 18 months). This includes all welds in PW systems 3"-diameter and above. Welds are replaced when IGSCC exceeds the replacement criteria of more than 20% of pipe circumference of 50% of through-wall depth.

Title: Benchmark calculation for leak before break evaluation of nuclear plant piping.

Author: Khant,-L.H.; Ayres,-D.J. (Nuclear Power, ABB Combustion Engineering, Windsor, CT (United States)) **Corp. Author:**

Source: Mirza,-S. et al (Eds.). Piping Components Analysis: Piping and Structural Dynamics, PVP-Volume 218. New York (NY). ASME, 160 p. pp 13-18.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: LBB methodology **ID:** 120

Abstract: This paper reports on an analysis of a large scale cracked pipe test that was performed in order to verify the methods used to determine crack stability in leak before break evaluations of nuclear plant piping. The test specimen was a thirty-eight foot long section of thirty-six inch diameter cracked pipe which was tested and reported by Battelle Columbus. In this test, a section of pressurized water reactor main loop piping containing a partial circumferential through wall crack was loaded in bending until significant crack extension occurred. The analysis of the experiment used essentially the same finite element models, calculation steps, and material data interpretation and extrapolation as has been used in actual plant piping LBB evaluations. The excellent agreement between the analysis predictions and the experimental results confirms the appropriateness of the methods used for actual plant LBB evaluations.

Title: Dynamic experiments on cracked pipes.

Author: Petit,-M.; Brunet,-G.; Buland,-P. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. d'Etudes Mecaniques et Thermiques) **Corp. Author:**

Source: Shibata,-Heki (Ed.). Trans. 11th SMiRT Conference. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. G2 p. 243-252. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: LBB methodology **ID:** 121

Abstract: In order to apply the leak before concept to piping systems, the behavior of cracked pipes under dynamic, and especially seismic loading must be studied. In a first phase, an experimental program on cracked stainless steel pipes under quasi-static monotonic loading has been conducted. In this paper, the dynamic tests on the same pipe geometry are described. These tests have been performed on a shaking table with a mono frequency input signal. The main parameter of the tests is the frequency of excitation versus the frequency of the system. (author).

Title: Comparison of methods for detecting leakages in pipelines.

Author: Jedner,-U.; Voss,-U.; Schmitt,-K. (Bayer AG, Krefeld-Uerdingen (Germany). Zentrales Ingenieurwesen/Prozessleittechnik); Unbehauen,-H. (Bochum Univ. (Germany). Lehrstuhl fuer Elektrische Steuerung und Regelung) **Corp. Author:**

Source: TM.-Technisches-Messen. (Nov 1991). v. 58(11) p. 446-451.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** German

Category: Methods **ID:** 122

Abstract: The two methods that proved to be efficient are mass flux recording with measurement data filtering, and evaluation of the mass flow balance by a t-test. Both methods achieve safe detection of leakage down to 0.62% of the pipe volumetric flow, with leakages smaller than this not being detected with the same reliability. Comparing the two methods, statistical evaluation is found to be slightly better than empirical evaluation, as the alarm threshold is not defined in the empirical approach and the method therefore automatically adjusts itself to the variance of measuring errors. The third method, based on a correlation analysis, proved to have inherent flaws. It strongly depends on the distribution of the flow resistances ahead of and behind the leak, i.e. at the leak. (orig.).

Title: Evaluation of crack opening times and leakage areas for longitudinal cracks in a pressurized pipe.

Author: Bhandari,-S. (Societe Franco-Americaine de Constructions Atomiques (FRAMATOME), 92 - Paris-La-Defense (France)); Leroux,-J.C. **Corp. Author:**

Source: Shibata,-Heki (Ed.). Trans. 11th SMiRT Conference. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. G2 p. 147-158. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Methods **ID:** 123

Abstract: This study presents a method of evaluating the minimum time to crack opening as well as the maximum leakage area in the case of longitudinal through-wall cracks in a cylinder with internal pressure. The objective is to arrive at a realistic enveloping hypothesis for the conventional longitudinal break of an entry elbow of a steam generator through the application of the proposed method on the hot leg of French PWRs. The fracture mechanics theory permits to evaluate an upper bound to the leakage area for cracks in piping. The synthesis of the recent studies on fracture dynamics allows to determine the minimum crack opening time. This study is composed of four steps: the proposal of a computational model to evaluate the upper bound, the validation of the model, the application of the model to the entry elbow of a steam generator of a PWR designed by FRAMATOME, and the synthesis of the studies on fracture dynamics to evaluate the maximum crack opening velocity. The linear elastic theory, the effect of plasticity, the B-K method of evaluating the upper bound leakage area, and the above steps are reported. (K.I.).

Title: Leaking underground hydrocarbon storage tanks: A worldwide problem requiring site-specific solutions.

Author: Adams,-R.B.; Hayman,-J.W.; Eisenbach,-R.L.; Dove,-T.E. (C and E Engineering, Inc., Baton Rouge, LA (United States)) **Corp. Author:**

Source: Moore,-J.E.; Kanivetsky,-R.A.; Rosenshein,-J.S.; Zenone,-C.; Csallany,-S.C. (eds.). First USA/USSR joint conference on environmental hydrology and hydrogeology. Dubuque (IA). Kendall/Hunt Publishing Co., 463 p. p. 2-10.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Other **ID:** 124

Abstract: The problem of hydrocarbons leaking from underground storage tanks (USTs) is a very serious concern in the US where there are an estimated 1,800,000 USTs, of which about 1,200,000 contain petroleum (hydrocarbon) products. Leaks and spills occur from tanks, product piping, connections, dispenser pumps, and overfilling. The issue of leaking USTs is not confined to the US, it is a worldwide problem. The USSR, a leading industrial nation, faces the same potential impact upon the environment, particularly drinking water supplies. Proper assessment of the environmental effects resulting from UST leaks and spills is essential so that responsible parties can determine what action is needed. Assessment procedures and techniques have been well refined in the US where much emphasis has been placed on remediating problems caused by leaking USTs. This paper presents (1) considerations applicable to most UST leak/spill assessment scenarios and (2) the elements of the UST leak/spill assessment process. The UST assessment process encompasses the range of assessment tasks from discovery of a problem to design of a remedial response.

Title: Qualification of dissimilar metal welds for HTR pipework. Final report.

Author: Asea Brown Boveri AG, Mannheim (Germany). Geschaeftsbereich Kraftwerke.Hochtemperatur-Reaktorbau GmbH, Mannheim (Germany). **Corp. Author:**

Source: 17 Apr 1989. 17 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** German

Category: Research/theoretical **ID:** 125

Abstract: In addition to the operating loads due to the high operating temperatures (535deg C), dissimilar metal welds in HTR components are exposed to substantial thermomechanical stresses due to various physical properties. The project work reported was to show that a welded joint consisting of X20 CrMoV 12 1 and Incoloy 800 is reliable under the operating conditions of the HTR-500 to the extent that rupture or leakage of pipes can be excluded. This contribution to the final report by HRB deals with the definition of operating loads, fracture-mechanical analyses, and further development of calculation methods. (MM).

Title: Crack initiation and crack propagation of an elbow DN 400 subjected to repeated high in-plane bending. 15 NiCuMo

Author: Diehm,-H.; Blind,-D.; Kobes,-E., Hunger,-H. **Corp. Author:** MPA-Stuttgart

Source: 16. MPA-Seminar: Sicherheit und Verfuegbarkeit in der Anlagentechnik mit dem Schwerpunkt 'Kerntechnik'. Bd. 1 und 2. Bd. 1: Bruchmechanik, Zeitstandverhalten/Kriechvorgaenge, zerstoerungsfreie Pruefung. - Bd. 2: Behaelter- und Komponentenintegritaet, Rohrleitungsverhalten, strahleninduzierte Versproedung, Waermewechsel- und Thermoschockbeanspruchung. 1990. 784 p. p. 34.1-34.28.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Test/analysis **ID:** 126

Abstract: The pipe failure test carried out in phase II of the HDR safety engineering programme with an elbow with longitudinal defects subjected to cylcic loading in an oxygen containing water environment at high temperatures resulted in pipe cracking and leakage. For this test, incipient cracks were introduced into the inner surface at the elbow flanks of the pipe section serving as a test specimen. The cyclic bending stress applied was extremely high and its maximum was 3 times the bending moment at which calculations predict local maximum yield strength and failure. Some test phases performed with a slow loading cycle (e approx = 10 sup - sup 6 1/s) were analysed for their effects by fractographic examination and did not show a significant influence of the corrosive environment on crack propagation. (orig./DG).

Title: Methods for leak detection for KWU pressurized and boiling water reactors.

Author: Fischer,-K.; Preusser,-G. (Siemens AG Unternehmensbereich KWU, Erlangen (Germany, F.R.)) **Corp. Author:**

Source: Nuclear-Engineering-and-Design. (Jul 1991). v. 128(1) p. 43-49.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Inspection methods **ID:** 127

Abstract: Leakage monitoring is an essential criterion to rule out the possibility of double ended pipe rupture in the primary coolant system. Subcritical cracks can be detected with a considerable margin before they extend to critical crack lengths resulting in spontaneous failure. In those KWU PWRs which went into operation recently, a Leakage Monitoring System was installed that is based on thermodynamic analysis. It utilizes the following measured parameters: dew point temperature, accumulated condensate inside aircoolers, air temperature, sump water level, gully monitoring. In KWU's BWRs although the measurement concept has to be slightly changed because of a different approaches design of buildings and components, the same instrumentation will be used. Besides this installed monitoring system, different like acoustic leak detection systems or the application of moisture sensitive instrumentation have been considered. Both systems have been successfully tested. (orig.).

Title: Short cracks in piping and piping welds. Semiannual report, March--September 1990: Volume 1, No. 1.

Author: Wilkowski,-G.M.; Ahmad,-J.; Brust,-F.; Ghadiali,-N.; Krishnaswamy,-P.; Landow,-M.; Marschall,-C.W.; Scott,-P.; Vieth,-P. (Battelle, Columbus, OH (USA)) **Corp. Author:**

Source: May 1991. 125 p. Nuclear Regulatory Commission, Washington, DC (USA). Div. of Engineering.Battelle, Columbus, OH (USA).FUNDING ORGANIZATION: Nuclear Regulatory Commission, Washington, DC (USA).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Research/theoretical **ID:** 128

Abstract: This is the 1st semiannual report of NRCs "Short Cracks in Piping and Piping Welds" research program. The program began in March 1990 and will extend for 4 years. The intent of this program is to verify and improve fracture analyses for circumferentially cracked large-diameter nuclear piping with crack sizes typically used in leak-before-break analyses or in-service flaw evaluations. Only quasi-static loading rates are evaluated since the NRC's International Piping Integrity Research Group (IPIRG) program is evaluating the effects of seismic loading rates on cracked piping systems. Additional efforts involve investigating phenomena discovered during the course of conducting the Degraded Piping program. These include the evaluation of the occurrence of unstable crack jumps in ferritic steels at LWR temperatures, and the occurrence of anisotropic fracture properties causing helical crack growth. Both of these phenomena may affect the safety margins implicit in LBB analyses. Other investigations deal with the fracture behavior of bi-metallic welds, and improvements in crack opening area analyses used in LBB. Since much of the work in this program was just beginning during this first reporting period and progress is limited, a complete statement of work for the whole program is provided in this report. 42 refs., 14 figs., 11 tabs.

Title: Load tests with a pipe bend DN 425, applying slowly changing bending loads up to occurrence of leak.

Author: Uhlmann,-D., Hunger,-H. **Corp. Author:** KFK

Source: Katzenmeier,-G. (Comp.). 14. Statusbericht des Projektes HDR-Sicherheitsprogramm des Kernforschungszentrums Karlsruhe. Arbeitsbericht 05.48/90. 1990. 425 p. p. 129-175.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Test/analysis **ID:** 129

Abstract: The experimental program deals with the formation of incipient cracks and subsequent crack growth of axially oriented cracks at a pipe bend with a nominal width of DN 425. The pipe bend consists of the ferritic material 20MnMoNi55. The numerical experiments by means of 3 D-FE analyses concentrate on determining the influence of the asymmetric crack depths at the two bend halves, and of the multiple crack fields, on the effective crack strain. (DG).

Title: Evaluation and refinement of leak-rate estimation models.

Author: Paul,-D.D.; Ahmad,-J.; Scott,-P.M.; Flanigan,-L.F.; Wilkowski,-G.M. (Battelle Columbus Labs., OH (USA)) **Corp. Author:**

Source: U.S. NRC Report, Apr 1991. 94 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: LBB methodology **ID:** 130

Abstract: Leak-rate estimation models are important elements in developing a leak-before-break methodology in piping integrity and safety analyses. Existing thermal-hydraulic and crack-opening-area models used in current leak-rate estimations have been incorporated into a single computer code for leak-rate estimation. The code is called SQUIRT, which stands for Seepage Quantification of Upsets In Reactor Tubes. The SQUIRT program has been validated by comparing its thermal-hydraulic predictions with the limited experimental data that have been published on two-phase flow through slits and cracks, and by comparing its crack-opening-area predictions with data from the Degraded Piping Program. In addition, leak-rate experiments were conducted to obtain validation data for a circumferential fatigue crack in a carbon steel pipe girth weld. 56 refs., 30 figs., 4 tabs.

Title: The effect of pipe bends on the elastic flexibility of a piping system.

Author: Smith,-E. (Manchester Univ. (UK). Inst. of Science and Technology) **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1991). v. 45(1) p. 121-129.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Research/theoretical **ID:** 131

Abstract: The elastic flexibility of a piping system containing a circumferential crack is an important parameter with regard to crack stability and leak-before-break considerations. In determining a system's elastic flexibility, the analysis is considerably simplified if the piping segments are assumed to be rigidly linked at pipe bends. The present paper's theoretical analysis shows that this assumption is justified provided that the pipe-run lengths adjacent to a bend are greater than about five times the bend diameter. (author).

Title: Plugging inaccessible leaks in cooling water pipework in nuclear power plants.

Author: Powell,-A.B. (Ontario Hydro, Toronto, ON (Canada)); May,-R.; Down,-M.G. (National Nuclear Corp. Ltd., Knutsford (UK)) **Corp. Author:**

Source: Anon.-Proceedings of the topical meeting on nuclear power plant life extension. Volume 2. La Grange Park, IL (USA). American Nuclear Society. 1988. 645 p. p. 367-370.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Analysis of break effects **ID:** 132

Abstract: The manifestation of initially small leaks in ancilliary reactor cooling water systems is not an unusual event. Often these leaks are in virtually inaccessible locations - for example, buried in thick concrete shielding or situated in cramped and highly radioactive vaults. Such leaks may ultimately prejudice the availability of the entire nuclear system. Continued operation without repair can result in the leak becoming larger, and the leaking water can cause further corrosion problems and interfere with instrumentation. In addition, the water may increase the volume of radwaste. In short, initially trivial leaks may cause significant operating problems. This paper describes the sealing of such leaks in the biological shield cooling system of Ontario Hydro's Pickering nuclear generating station CANDU reactors.

Title: Pipeline leak detection method and control device therefor.

Author: Bell,-D.A. **Corp. Author:**

Source: Interprovincial Steel and Pipe Corp. Ltd., Regina, SK (Canada). 30 Aug 1983; 6 Feb 1981. 30 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1981 **Language:** English

Category: Inspection methods **ID:** 133

Abstract: Leaks may be located in a pipeline by introducing into the pipeline an assembly that includes a pipe-sealing packer unit, a control unit, and a radioactive source shielded from the control unit. The control unit includes a gamma ray detector that controls the sealing and unsealing of the pipe by the packer in response to the detection of radiation exceeding a preset threshold - a detection event. The assembly is pushed through the pipeline by a relatively low fluid pressure behind it. The progress of the assembly through the pipeline may be monitored externally by a gamma ray detector.

Title: Ductile fracture properties for assessing leak-before-break issues in ferritic weldments.

Author: Lepik,-O.E.; Mukherjee,-B. (Ontario Hydro, Toronto, ON (Canada). Research Center) **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 285-300.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: LBB justification **ID:** 134

Abstract: A Leak-Before-Break (LBB) approach is being used by Ontario Hydro's Darlington nuclear generating station as a design alternative to pipe rupture restraint hardware on the large diameter piping of the primary heat transport system. The J-resistance curves of four different ferritic weldments, fabricated by either the submerged arc weld or shielded metal arc weld process, were determined as part of this program. Results indicated that the as-welded and post-weld heat treated (PWHT) welds were susceptible to varying degrees of static or dynamic strain aging at 200 and 250 sup 0 C. Dynamic strain aging effects were most significant for as-welded welds, as evidenced by sudden load drops on the load-displacement curves and ductile crack jumping. The effect of loading displacement rate and PWHT on toughness was assessed and related to the weld's tensile properties and susceptibility to dynamic strain aging. The implications of strain aging for LBB assessments are discussed. (author).

Title: A probabilistic approach to leak-before-break in CANDU pressure tubes.

Author: Walker,-J.R. (Atomic Energy of Canada Ltd., Pinawa, MB (Canada). Whiteshell Nuclear Research Establishment) **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 229-239.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Methods **ID:** 135

Abstract: In the CANDU reactor, the coolant passes through the core in zirconium alloy pressure tubes. A few of these pressure tubes have leaked at cracks near the rolled joint where the pressure tube is attached to the end fitting. A probabilistic methodology, and associated computer code (called MARATHON), has been developed to calculate the time from first leakage to unstable fracture in a probabilistic format. The methodology explicitly uses material property distributions, and allows the risk associated with leak-before-break to be estimated. A model of the leak detection system is included to calculate the time between leak detection and unstable fracture. The sensitivity of the risk to changing reactor conditions allows the optimization of reactor management after leak detection. Preliminary material property distributions show the probability of unstable fracture is very low, and that ample time is available to shut down the unit and locate the leaking tube. (author).

Title: A plugging criterion for steam generator tubes based on leak-before-break.

Author: Esteban,-A.; Bolanos,-M.F.; Figueras,-J.M. (Consejo de Seguridad Nuclear (CSN), Madrid (Spain)) **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 181-186.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Criteria **ID:** 136

Abstract: Degradation of steam generator tubes is occurring in Spanish pressurized water reactors. The causes differ and give rise to leaks. The General Criteria of Design, technical specifications and availability are concepts which must be harmonized. The historic behaviour of certain tube defects, together with experiments and studies made by the Spanish Owner Group of Pressurized Water Reactors and approved by Consejo de Seguridad Nuclear, have led to a new plugging criterion based on the Leak-Before-Break concept and a new limit for leakages is in operation in some plants. (author).

Title: A leak-before-break assessment method for pressure vessels and some current unresolved issues.

Author: Sharples,-J.K.; Clayton,-A.M. (AEA Petroleum Services, Winfrith (UK). Petroleum Engineering) **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 317-327.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: LBB methodology **ID:** 137

Abstract: The structural integrity diagram, a plot of crack depth against crack length, can be used to investigate a wide range of safety arguments for flawed pressure vessels, including Leak-Before-Break. It enables clear margins to be shown for defects which might exist in the vessels and indicates crack sizes and loadings where the Leak-Before-Break case is valid. The use of this diagram requires a model of crack shape development as a crack grows through the wall of the vessel up to the stage at which the deepest part of the crack breaks through the well, and this is considered for a number of growth mechanisms. Uncertainties exist, however, in the understanding of crack behaviour relevant to this issue. These uncertainties are reviewed and work programmes underway in the UK aimed at resolving some of them are outlined. (author).

Title: Determination of J-integral values and leakage areas for circumferential cracks in pipes under bending loads.

Author: Grebner,-H.; Diekmann,-P. (Gesellschaft fuer Reaktorsicherheit mbH (GRS), Koeln (Germany, F.R.)) **Corp. Author:**

Source: Deutscher Verband fuer Materialforschung und -pruefung e.V., Berlin (Germany, F.R.). 20 Jahre DVM-Arbeitskreis Bruchvorgaenge. Bruchmechanische Kennwerte fuer die Bauteilbewertung. 1989. 550 p. p. 87-97.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** German

Category: Research/theoretical **ID:** 138

Abstract: Comparisons between FE calculation and two simplified methods are illustrated by COD, leakage area, and J-Integral values. While FE calculation for COD and leakage area furnishes values for the internal and external sides of the tube, the two methods of approximation indicate only average values. COD and/or J-integral values show a satisfactory consistency. Deviations of the results of the simplified methods from the FE mean values amount to a maximum of about 20% for COD values and to about 30% for J-integral comparisons. (orig./DG).

Title: Leakage before fracture behaviour of pipe systems. Comparison of experiments and calculations.

Author: Kussmaul,-K. (VGB Technische Vereinigung der Grosskraftwerksbetreiber e.V., Essen (Germany, F.R.)); Blind,-D.; Roos,-E.; Sturm,-D. (Stuttgart Univ. (Germany, F.R.). Staatliche Materialpruefungsanstalt)

Corp. Author:

Source: VGB-Kraftwerkstechnik. (Jul 1990). v. 70(7) p. 553-565.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Test/analysis **ID:** 139

Abstract: For accidents such as design earthquakes, aircraft crash, safe shut-down earthquakes and postulated pipe fracture it is shown that major fractures or consequential fractures can be excluded. Comprehensive experimental investigations have been undertaken in order to assess the relative stresses affecting individual components and the computer procedure taking into account the crack formation assumed. The results of tests on pipes with longitudinal and circumferential faults under internal pressure and partly superimposed bending are described. (orig.).

Title: Automatic shutdown of the pressurized water reactor without control rod drop in case of small leaks in the primary co

Author: Karner,-H.; Wegner,-R. (Siemens AG Unternehmensbereich KWU, Erlangen (Germany, F.R.))

Corp. Author:

Source: Deutsches Atomforum e.V., Bonn (Germany, F.R.); Kerntechnische Gesellschaft e.V., Bonn (Germany, F.R.). Jahrestagung Kerntechnik '90. Tagungsbericht. Bonn (Germany, F.R.). INFORUM Verl. May 1990. 710 p. p. 127-130.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Analysis of break effects **ID:** 140

Abstract: Published in summary form only.

Title: Investigations of leak opening and outflow behaviour on straight pipes with circumferential cracks with internal pressu

Author: Grebner,-H.; Hoefler,-A., Hunger,-H.

Corp. Author: KFK-Germany

Source: Katzenmeier,-G. (comp.). 13. Statusbericht des Projektes HDR-Sicherheitsprogramm des Kernforschungszentrums Karlsruhe. Arbeitsbericht 05.46/89. 1989. 404 p. p. 193-249.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** German

Category: Test/analysis **ID:** 141

Abstract: The experiments carried out so far on straight pipes with circumferential cracks and results of subsequent calculations from this experiment are introduced. The subsequent calculations are not yet completed at all points. From the experiments one can record that the selected crack sizes and stresses have guaranteed stable crack behaviour in all cases. The comparison of experimental and calculated crack openings shows satisfactory agreement (difference about 20% for experiment E22.03. The compared leak rates show differences of up to about 50% (in isolated cases even more). For small leak rates (0.01 to 0.1 kg/sec), one can expect a difference of about 100% between calculation and experiment. For medium leak rates, we regard a maximum difference of about 30% as achievable. For large leak rates, the achievable accuracy plays no part for the detectability. (orig./DG).

Title: Elastic plastic analyses of a cracked piping system.

Author: Grebner,-H.; Hofler,-A.; Haber,-O. (GRS) **Corp. Author:**

Source: Hadjian,-A.H. (Ed.). Trans. 10th SMiRT Conference, Los Angeles (CA). American Association for Structural Mechanics in Reactor Technology. 1989. 375 p. p. 317-322.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Test/analysis **ID:** 142

Abstract: The paper presents post-calculations of GRS to an experiment performed in the frame of the HDR-(overheated steam reactor) safety-program. In the experiment the failure of a large scale piping system (inner diameter 400 mm) loaded by steady internal pressure (10.5 MPa) and an increasing opening inplane bending moment, under conditions similar to those of a nuclear pressurized water reactor, was studied. The piping system failed with leakage through a 400 mm long crack in the crown of a 90 degree pipe elbow.

Title: Research programs in piping fracture behaviour in Italy.

Author: Milella,-P. **Corp. Author:** Seminar on leak-before-break:

Source: Wilkowski,-G.M., Chao,-K.S. (Eds.). Leak-Before-Break: Further Developments in Regulatory Policies and Supporting Research. Feb 1990. 350 p. p. 211-230.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: LBB justification **ID:** 143

Abstract: The paper summarized the LBB research efforts in Italy from 1981 to 1987. Some of the results are: (1) the net-section-collapse method seems to under estimate the maximum failure loads of carbon steel pipes while it accurately predicts that of stainless steel pipes, (2) the GE/EPRI method is valid method to predict the crack opening displacement and maximum moment for pipes, (3) the Tada-Paris method seems to overestimate the actual leak area in pipes particularly when at bending moments where plasticity occurs, (4) A106 carbon steel pipe can experience severe toughness loss, probably from dynamic strain aging at 280 C, and (5) leak areas remain well below 10 percent of the pipe cross section with through-wall cracks smaller than 160 degrees and load within the ASME Section III normal operating stress loads. An on-going experimental program to verify pipe fracture under inertial loading was also summarized. This program will continue from 1988 to 1992.

Title: ANSPipe: An IBM-PC interactive code for pipe-break assessment.

Author: Fullwood,-R.R.; Harrington,-M. (Brookhaven National Lab., Upton, NY (USA)) **Corp. Author:**

Source: Transactions-of-the-American-Nuclear-Society. (1988). v. 57 p. 148-149.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Methods **ID:** 144

Abstract: The advanced neutron source (ANS) being designed at Oak Ridge National Laboratory will be the world's highest flux neutron source and best facility for associated basic and applied research. The ANSPipe code was written as an aid for the piping configuration and material selection to enhance safety and availability. The primary calculation is based on the Thomas mode which models pipe leak or break probabilities as proportional to the length of the segment and diameter and the inverse square of the wall thickness. This scaling, based on experience, is adjusted for radiation effects, using the Regulatory Guide 1.99 model, and for cyclic fatigue, stress corrosion, and inspection, using adaptations form the PRAISE-B code. The key to an ANSPipe analysis is the definition of the pipe segments. A pipe segment is defined as a length of pipe in which all the parameters affecting the pipe are constant or reasonably so. Thus, a segment would be a length of pipe of constant diameter, thickness, material type, internal pressure, flux distribution, stress, and submergence or nonsubmergence.

Title: Operation of valves and pipelines at NPPs.

Author: Rumyantsev,-V.V. (Comp.) **Corp. Author:**

Source: Atomnaya-Tekhnika-za-Rubezhom. (May 1989). (no.5) p. 27-31.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** Russian

Category: Inspection methods **ID:** 145

Abstract: The diagnostic equipment for the detection of concealed defects in valves and forecasting their development as well as the methods to restore worm-out valve elements and the mthods of leak potting using suspensions supplied to the pipelines, are described.

Title: ECCS used in DIDO and PLUTO.

Author: Panter,-R. (UKAEA Harwell Lab. (United Kingdom)) **Corp. Author:**

Source: International Atomic Energy Agency, Vienna (Austria). Research reactor core conversion guidebook. V.2: Analysis (Appendices A-F). Apr 1992. 386 p. p. 149-151.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Other **ID:** 146

Abstract: DIDO and PLUTO are heavy water tank-type reactors with power levels of 25.5 MW. Measures are described that protect against failure of the smaller pipes in the primary system, and more importantly. against weld failure at the junction to a large pipe rather than in the small pipe itself. (author). 1 ref., 2 figs.

Title: Theoretical and user's manual for pc-PRAISE: A probabilistic fracture mechanics computer code for piping reliability

Author: Harris,-D.O.; Dedhia,-D.D. (Failure Analysis Associates, Inc., Menlo Park, CA (United States)); Lu,-S.C. (Lawrence Livermore National Lab. (CA) **Corp. Author:**

Source: NRC-Report 317 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: PFM methods **ID:** 147

Abstract: PC-PRAISE is the most recent version of the code, which is a PFM-code that has recently been modified to run on an IBM-PC to evaluate the reliability of welds in NPP piping systems. PC-PRAISE was adapted from the PRAISE Computer Code, which was originally developed in 1980-81 by LLNL and funded by U.S. NRC for assessment of the influence of seismic events on the failure probability of piping in pressurized water reactors. PRAISE has been significantly expanded in recent years to allow consideration of both crack initiation and growth in a variety of piping materials in pressurized and boiling water reactors. PRAISE has a deterministic basis in fracture mechanics. Some of the inputs, such as initial crack size and inspection detection probability, are considered to be random variables, and failure probability versus time for a given weldment is evaluated by Monte Carlo simulation. Complex realistic stress histories can be treated by the code, and sets of random material properties for representative piping materials are built into the code. This document provides a summary of the deterministic basis of the code, along with description of statistical distributions of random variables. Code inputs are described and an extensive set of sample problems is provided along with descriptions of representative outputs.

Title: Fracture assessment of Savannah River Reactor carbon steel piping. Revision 1.

Author: Mertz,-G.E.; Stoner,-K.J.; Caskey,-G.R. (Westinghouse Savannah River Co., Aiken, SC (United States)); Begley,-J.A. (Westinghouse Electric Corp., Pittsburgh, PA (United States))

Corp. Author:

Source: Westinghouse Savannah River Co., Aiken, SC (United States).American Society of Mechanical Engineers pressure vessel and piping conference. New Orleans, LA (United States). 21-25 Jun 1992.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 148

Abstract: The SRS production reactors have been in operation since the mid-1950's. One postulated failure mechanism for the reactor piping is brittle fracture of the original A285 and A53 carbon steel piping. Material testing of archival piping determined (1) the static and dynamic tensile properties; (2) Charpy impact toughness; and (3) the static and dynamic compact tension fracture toughness properties. The NDT temperature, determined by Charpy impact test, is above the minimum operating temperature for some of the piping materials. A fracture assessment was performed to demonstrate that potential flaws are stable under upset loading conditions and minimum operating temperatures. A review of potential degradation mechanisms and plant operating history identified weld defects as the most likely crack initiation site for brittle fracture. Piping weld defects and low fracture toughness material properties were postulated at high stress locations in the piping. Normal operating loads, upset loads, and residual stresses were assumed to act on the postulated flaws. Calculated allowable flaw lengths exceed the size of observed weld defects, indicating adequate margins of safety against brittle fracture. Thus, a detailed fracture assessment was able to demonstrate that the piping systems will not fail by brittle fracture, even though the NDTT for some of the piping is above the minimum system operating temperature.

Title: Subsidence strains.

Author: Kiefner,-J.F. (Kiefner and Associates, Inc., Worthington, OH (United States))

Corp. Author:

Source: McKetta,-J.J. (University of Texas at Austin, Austin, TX (United States)). Piping design handbook. New York, NY (United States). Marcel Dekker Inc. 1992. 1198 p. p. 1060-1078.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Other **ID:** 149

Abstract: Adequate monitoring and proper intervention can significantly increase a pipeline's chances of surviving the strains of soil subsidence in an area of longwall mining. The first part of this article on the effects of longwall mining on underground pipelines presents a technique for monitoring those effects. The concluding part examines intervention options and discusses the benefits of exposing pipelines in longwall mining areas. Longwall mining can constitute a threat to the integrity of a pipeline by way of surface subsidence and soil strains. The usual effects on a pipeline of mining-induced subsidence are increased axial and flexural strains affecting its longitudinal strength. In the presence of severe circumferentially oriented defects and added tensile strain, a rupture is possible. In the presence of added compressive strain, buckling of the pipe may occur. Pipeline operators can use available predictive methods and geophysical data to estimate the potential effects of longwall mining, and they can monitor their pipelines and intervene if necessary to prevent a pipeline failure due to subsidence.

Title: Relation between sensitization and failures of welded joints at furnaces of Cienfuegos refinery.

Author: Dominguez,-H.; Menendez,-C.M.; Sendoya,-F.A. (Centro de Estudios Aplicados al Desarrollo Nuclear (CEADEN), La Habana (Cuba))

Corp. Author:

Source: Nucleus-Havana. (1992). (no.12) p. 2-5.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** Spanish

Category: Experience/events **ID:** 150

Abstract: This work is concerned about the possible relation between sensitization and failures of welded joints at furnaces of Cienfuegos Refinery. This failures were detected in austenitic pipes by hydraulic testing. For determined the tendency to sensitization of heat affected zones (HAZ) of welded joints and piping, have been used standardized test methods AM and AMU (GOST 6032-89). In addition, the Electrochemical Potentiokinetic Reactivation (EPR) test was employed to quantity the tendency to intergranular corrosion. It was found that degree of sensitization was higher at HAZ and as a possible explanation is proposed the overheating during welding.

Title: Assumed process of piping failure in nuclear power plants under destructive earthquake conditions.

Author: Shibata,-H. (Tokyo Univ. (Japan). Inst. of Industrial Science) **Corp. Author:**

Source: Journal-of-Pressure-Vessel-Technology. (May 1991). v. 113(2) p. 268-272.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Research/theoretical **ID:** 151

Abstract: This paper deals with an assumed process of piping failure in nuclear power plants which may cause a catastrophic accident during a destructive earthquake conditions. The type of failure discussed is the so-called double-ended guillotine break, DEGB. As a safety problems, we are going to eliminate this type of failure by LBB, and we have assumed that this would then not occur by an earthquake. The author tries to clarify the possibility of failure during earthquakes. He reviews his related papers since 1973, and discusses zipping failure of snubbers and supporting devices. He shows a procedure to simulate the zipping failure of a piping system supported by snubbers.

Title: Fatigue and failure behaviour of a mechanically loaded ferritic pipe bend in high temperature water with elevated oxy

Author: Kussmaul,-K.; Diem,-H.; Uhlmann,-D. (Stuttgart Univ. (Germany)); Hunger,-H. **Corp. Author:**

Source: Shibata,-Heki (Ed.). Trans. 11th SMiRT Conference. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. F p. 213-218. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 152

Abstract: During the first experiment of the experimental group E21 - behaviour of crack growth of piping components at operating pressure and cyclic bending load in a corrosive medium - performed within phase III of the HDR Safety Program a pipe bend made of ferritic material 20 MnMoNi 5 5 was the object investigated. During the test incipient cracks were generated by cyclic bending on the inner surface around the bend flanks. In various phases of the test characterized by sinoidal and sawtooth modes of loading and different load frequencies (1 cycle per minute and 1 cycle per 15 minutes) the cracks were further extended. At the end of the phase cyclic testing the maximum crack depth of the macrocrack embedded in a crack field was approx. 21 mm. In the final load test with monotonously rising bending moment the pipe bend failed in the form of a leak. (author).

Title: Assessment of thermal fatigue crack propagation in safety injection PWR lines.

Author: Simos,-N.; Reich,-M.; Costantino,-C.J. (Brookhaven National Lab., Upton, NY (United States). Nuclear Energy Dept.) **Corp. Author:**

Source: Lin,-C.W., Tseng,-W.S. (Eds.). System Interaction With Linear and Nonlinear Characteristics. PVP-Vol. 187. New York, NY (United States). American Society of Mechanical Engineers. 94 p. p. 65-72.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Thermal stratification **ID:** 153

Abstract: Cyclic thermal stratification resulting in alternating thermal stresses in pipe cross sections has been identified as the primary cause of high cycle thermal fatigue failure. A number of piping lines in operating plants around the world, susceptible to thermal stratification, have experienced circumferential cracking as a result of high levels of alternating bending stresses. This paper addresses the mechanisms of crack initiation and crack growth and provides estimates of fatigue cycles to failure for a typical safety injection line with such cyclic load history. Utilizing a 3-D finite element analysis, the temperature profile and the corresponding thermal stress field of a complete thermal cycle in a safety injection line consisting of a horizontal pipe section and an elbow, is obtained. Since the observed cracking occurred in the region of the elbow-to-horizontal pipe weld, the analysis performed assessed the impact of the level of local geometric discontinuities on the initiation of an inside surface flaw is greatest and the number of thermal cycles required to drive a small surface crack through the pipe wall.

Title: PRA-based guidance for piping inservice inspection.

Author: Vo,-T.V.; Gore,-B.F.; Eschbach,-E.J.; Simonen,-F.A. (Pacific Northwest Lab., Richland, WA (United States)) **Corp. Author:**

Source: Anon.-Probability, reliability, and safety assessment PSA '89. La Grange Park, IL (United States). American Nuclear Society. 1989. 1300 p. p. 1060-1067.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Inspection methods **ID:** 154

Abstract: This paper reports that one of the goals of the Nondestructive Evaluation (NDE) Reliability Program sponsored by the Nuclear Regulatory Commission (NRC) at Battelle, Pacific Northwest Laboratories (PNL) is to assess current inspection requirements of all pressure boundary systems and components, determine if improvements to the requirements are needed and if necessary, develop recommendations for revising the ASME Code and Regulatory requirements. Part of the work performed in addressing this goal was the development and demonstration of a method to establish in service inspection priorities through the use of probabilistic risk assessment (PRA) results. The Oconee-3 PRA and the observed weld failure data of the United states operating plants were used to identify the prioritize the most risk-important systems for inspection. Failure Modes and Effects Analysis (FMEA) methodology was then used to identify and prioritize the most risk-important piping sections of the Oconee-3 Emergency Feedwater (EFW) system. Based on the results of this study, the method has been demonstrated to be a useful tool for identifying systems and piping sections or welds that need to be inspected.

Title: A ductile fracture mechanics methodology for predicting pressure vessel and piping failure.

Author: Landes,-J.D.; Zhou,-Z. (Univ. of Tennessee, Knoxville, TN (United States)) **Corp. Author:**

Source: Bhandari,-S. et al (Eds). Pressure Vessel Integrity 1991. PVP-Volume 213; MPC-Volume 32. New York (NY). ASME. 290 p. p. 207-216.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Research/theoretical **ID:** 155

Abstract: This paper reports on a ductile fracture methodology based on one used more generally for the prediction of fracture behavior that was applied to the prediction of fracture behavior in pressure vessel and piping components. The model uses the load versus displacement record from a fracture toughness test to develop inputs for predicting the behavior of the structural component. The principle of load separation is used to convert the test record into two pieces of information, calibration functions which describe the structural deformation behavior and fracture toughness which describes the response of a crack-like flaw to the loading. These calibration functions and fracture toughness values which relate to the test specimen are then transformed to those appropriate to the structure. Often in this step computation procedures could be used but are not always necessary. The calibration functions and fracture for the structure are recombined to predict a load versus displacement behavior for the structure. The input for the model was generated from tests of compact specimen geometries; this geometry is often used for fracture toughness testing. The predictions were done for five model structures.

Title: A vibrational test and analysis of vessel-piping system. Part 2. Response of vessel.

Author: Minowa,-C.; Ogawa,-N. (National Research Center for Disaster Prevention, Science and Technology Agency, Ibaraki (Japan)) **Corp. Author:**

Source: Ma,-D.C.; Chen,-S.S., Tani,-J. (Eds.). Flow-structure Vibration and Sloshing--1990. PVP-Volume 191. New York (NY). ASME. 1990. 164 p. p. 105-112.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 156

Abstract: In this paper, a shaking table test of thick wall vessel with pipe systems is reported. The purposes of the test were to investigate the earthquake failure characteristics of the vessels and to study the effects of piping systems on the vessel response.

Title: A method to assign failure rates for piping reliability assessments.

Author: Gamble,-R.M. (NOVETECH Corp., Rockville (MD)); Tagart,-S.W. Jr. (Electric Power Research Inst., Palo Alto (CA))

Corp. Author:

Source: Bamford,-W. (Ed.) . Fatigue, Fracture, and Risk 1991. PVP-Volume 215. New York (NY). ASME. 210 p. p. 3-12.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Failure rate estimation **ID:** 157

Abstract: This paper reports on a simplified method that has been developed to assign failure rates that can be used in reliability and risk studies of piping. The method can be applied on a line-by-line basis by identifying line and location specific attributes that can lead to piping unreliability from in-service degradation mechanisms and random events. A survey of service experience for nuclear piping reliability also was performed. The data from this survey provides a basis for identifying in-service failure attributes and assigning failure rates for risk and reliability studies.

Title: Probabilistic fracture mechanics analysis code CANIS-P.

Author: Watashi,-Katsumi; Furuhashi,-Ichiro (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center)

Corp. Author: 11. international conference on

Source: Shibata,-Heki (Ed.): Trans. of the 11th SMiRT Conference. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. G2 p. 343-348. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3 for 17 vols. bound in 15 (A through SD2).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Research/theoretical **ID:** 158

Abstract: This paper describes a function of a computer code CANIS-P and its application to parametric sensitivity analysis of pipings in pressurized water reactors and in fast breeder reactors. CANIS-P can calculate the failure probability of the pipings in nuclear plants and in other engineering plants, especially in FBRs operated at creep temperature range under several assumptions. (author).

Title: Backfitting requirements in nuclear power plants in Eastern Europe.

Author: Hoehn,-J.; Niehaus,-F. (Internationale Atomenergie-Organisation, Vienna (Austria))

Corp. Author:

Source: Atw.-Atomwirtschaft,-Atomtechnik. (Apr 1992). v. 37(4) p. 178-184.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** German

Category: Other **ID:** 159

Abstract: The IAEA has analyzed the safety status of reactors of Soviet design and made proposals about improving it. The safety features were contrasted with those of Western reactor lines. Despite many deficits in safety, especially the WWER-440 reactors have a number of advantages as well. Thus, e.g., the low fuel temperature ensures good retention of fissile materials in the fuel. The use of high-grade materials was felt to obviate the need for provisions against failure of the primary system pipes in the WWER line of reactors. Instead, the safety concept was aimed at avoiding initiating events. With the participation of Wano, an EC crash program of assistance to Bulgaria has been launched. At present, an international project of assessing the safety RBMK reactors is being considered. Similar programs for other reactor lines are under discussion. (orig./HP).

Title: Failure resistance evaluation for pipings of NPP with BWR.

Author: Timofeev,-B.T.; Vinogradov,-R.P.; Generalova,-S.P.; Chernaenko,-T.A. (Central Research Inst. of Structural Materials 'Pronetey', Leningrad (USSR))

Corp. Author: 11. international conference on

Source: Shibata,-Heki (Ed.): Trans. of the 11th SMiRT Conference, Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. G2 p. 231-236. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3 for 17 vols. bound in 15 (A through SD2).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 160

Abstract: In the present investigation the fracture resistance test results of modern structural steels of the grades 22K (USSR), Creselso 330E (France), 19MN5 (Japan) and their weldments have been extensively studied under the test conditions simulating the service operation of Dy752 pipings of NPP with reactors of BWR type. The materials low cycle fatigue resistance and crack growth kinetics have been evaluated in the operation temperature range of 20-350degC, and the fracture toughness values - with the use of brittle crack initiation criterium. It has been shown that all the characteristics of the investigated materials reflecting the various stages of fracture process are very close in their values and to predict the service life of pipings the correspondent correlations of the expanded in the USSR standard PNAE G-7-002-86 may be used. (author).

Title: The conservatism of the net-section stress criterion for the failure of cracked stainless steel piping.

Author: Smith,-E. (Manchester Univ. (United Kingdom))

Corp. Author: 11. international conference on

Source: Shibata,-Heki (Ed.): Trans. of the 11th SMiRT Conference. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. G1 p. 59-64. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3 for 17 vols. bound in 15 (A through SD2).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Methods **ID:** 161

Abstract: The failure of cracked stainless steel piping can be predicted by assuming that failure conforms to a net-section stress criterion, using as input an appropriate value for the critical net-section stress together with a knowledge of the anticipated loadings. The stresses at the cracked section are usually calculated via a purely elastic analysis based on the piping being uncracked. However because the piping is built-in at its ends into a larger component, this limits the amount of elastic follow-up and, consequently, use of the net-section stress approach in this manner can lead to conservative failure predictions. This paper quantifies the extent of this conservatism, and shows that it can be quite marked. There is an additional measure of conservatism due to the fact that unstable failure need not necessarily be associated with the onset of crack extension. A key parameter with regard to both these conservatisms is L sub E sub F sub F , a length parameter which is a measure of the degree of elastic follow-up in the system. (author).

Title: Piping engineering for nuclear power plant.

Author: Curto,-N.; Schmidt,-H.; Muller,-R.

Corp. Author: Empresa Nuclear Argentina de

Source: 1988. 7 p. Scientific meeting of the Argentine Association of Nuclear Technology. Mendoza (Argentina). 7-11 Nov 1988. 16. Reunion cientifica de la Asociacion Argentina de Tecnologia Nuclear.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** Spanish

Category: Methods/design **ID:** 162

Abstract: In order to develop piping engineering, an adequate dimensioning and correct selection of materials must be secured. A correct selection of materials together with calculations and stress analysis must be carried out with a view to minimizing or avoiding possible failures or damages in piping assembling, which could be caused by internal pressure, weight, temperature, oscillation, etc. The piping project for a nuclear power plant is divided into the following three phases. Phase I: Basic piping design. Phase II: Final piping design. Phase III: Detail engineering. (Author).

Title: A statistical approach for predicting volume of oil spill during pipeline operations.

Author: Kim,-B.I.; Sharma,-M.P.; Harris,-H.G. (Univ. of Wyoming, WY (United States)) **Corp. Author:**

Source: Proc. 1991 SPE Annual Technical Conference & Exhibition. PI Production Operations and Engineering. Part 1. Richardson (TX). Society of Petroleum Engineers. 1991. 573 p. p. 475-482. Technical Paper SPE 22807.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Other **ID:** 163

Abstract: This paper presents statistical models for predicting volume of oil spills in pipeline failures. It is based on performance and failure data of cross-country oil pipelines. The two-parameter Weibull distribution was used to predict the size (volume) of the oil spill. The goodness-of-fit for the models has been evaluated using Chi-square criterion. The two-parameter Weibull distribution proved to be an effective means of statistically predicting oil spill size. The statistical models developed make it possible to assess both the probability of volume of oil spillage due to all causes of pipeline failures, and of individual causes as well.

Title: Containments for new PWR-reactors.

Author: Eibl,-J.; Schlueter,-F.H.; Cueppers,-H. (Karlsruhe Univ. (T.H.) (Germany)); Hennies,-H.H.; Kessler,-G. **Corp. Author:** 11. international conference on

Source: Shibata,-Heki (Ed.): Trans. of the 11th SMiRT Conference. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. SD1-SD2 p. 381-386. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3 for 17 vols. bound in 15 (A through SD2).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Methods/design **ID:** 164

Abstract: Considering the tremendous amount of energy required for the fast growing world population, the increased use of nuclear energy will eventually become necessary, accordingly, measures must be taken to regain public acceptance. Aiming at this, the authors have begun the investigation to limit the consequence of severe accidents to a certain reasonable and acceptable level. Preliminary design possibilities have been studied in relation to the static and dynamic internal overpressure caused by hydrogen explosion or detonation, the failure of a pressure vessel under high system pressure or steam explosion, the retention of molten core to prevent basemat erosion, the removal of decay heat, and the passive closure of all pipes and locks penetrating containment. An initial proposal of a containmnet design based on the relevant design requirements is shown. The design criteria for the reactor pressure vessel environment resulting from the RPV failure in a low or high pressure path including a steam explosion case are demonstrated. A high pressure-resistant core catcher system is presented. A composite concrete-steel wall system, a filter system, the method of closing a large lock and so on are shown. (K.I.).

Title: Recent advances in analysis of PWR containment bypass accidents.

Author: Warman,-E.A.; Metcalf,-J.E.; Donahue,-M.L. (Stone and Webster Engineering Corp., Boston, MA (United States)) **Corp. Author:** Annual meeting of the America

Source: Transactions-of-the-American-Nuclear-Society. (1991). v. 63 p. 265-266.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Methods **ID:** 165

Abstract: WASH-1400 identified and quantified the contribution to off-site radiological risks of accident sequences at PWRs in which the release of fission products may be released by bypassing the containment building; i.e., ISLOCA events. Containment bypass sequence risks constitute a large fraction of the total pressurized water reactor (PWR) in NUREG-1150 in large part because estimates of competing risks from early containment failures have been greatly reduced since WASH-1400. Rigorous analyses of both SGTR and V sequence bypass sequences result in reductions in fission product release to such an extent that in-containment sequences are expected to dominate PWR risks at levels substantially lower than reported in NUREG-1150. It is important that these findings be confirmed by other investigators, particularly in light of the NRC's ongoing study of the frequency of occurrence of interfacing systems. LOCAs based on extensive investigations at operating plants. Progress in this latter effort should be matched by progress in the knowledge and understanding of the progression of bypass sequences once initiated.

Title: Mechanistic understanding of irradiation-induced corrosion of zirconium alloys in nuclear power plants: Stimuli, statu

Author: Johnson,-A.B. Jr.; Ishigure,-K.; Nechaev,-A.F.; Reznichenko,-E.A.; Cox,-B.; Lemaignan,-C.; Petrik,-N.G. **Corp. Author:** Pacific Northwest Lab., Richla

Source: May 1990. 26 p. International conference on radiation materials science. Alushta (Ukraine). 22-25 May 1990.FUNDING ORGANIZATION: USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Methods **ID:** 166

Abstract: Failures in the basic materials used in nuclear power plants continue to be costly and insidious, despite increasing industry vigilance to catch failures before they degrade safety. For instance, the overall costs to the US industry from materials problems could amount to as much as $10 billion annually. Moreover, estimates indicate that the cost of a pipe failure in a nuclear plant is one hundred times greater than the cost of a similar failure in a coal-fired plant. There are important practical stimuli and much scope for further understanding of the effects of irradiation on Zr-alloys (and other materials used in nuclear installations) by careful experimentation. Moreover, these studies need to address the effect of irradiation on all components of heterogeneous systems: the metal, the oxide and the environment, and especially those processes recurring at the interphases between these components. The present paper is aimed at providing specialists with some systematic information on the subject and with important considerations on the key items for further experimentation.

Title: External events assessment for an LMFBR plant.

Author: Aizawa,-K.; Nakai,-R.; Yamaguchi,-A. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)) **Corp. Author:** OECD/BMU-workshop on spe

Source: Hauptmanns,-U. (comp.). Gesellschaft fuer Reaktorsicherheit mbH (GRS), Koeln (Germany). Proceedings of the OECD/BMU-workshop on special issues of level 1 PSA. Jul 1991. 407 p. p. 304-317.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Methods **ID:** 167

Abstract: The quantitative screening analyses which identify dominant sequences on the following location-dependent failures were conducted: leak of water/steam/freon, leak of sodium, inadvertent actuation of water sprinkler system, high energy line break causing pipe whip, HVAC fan missile, and fire. The result, which is obtained from conservative evaluation under the assumption that the susceptible components fail, indicates the effect of fire is the largest among those external events. The quantitative seismic event analysis has also been conducted. Seismic hazard curves and spectral shape have been evaluated using the seismic activity data around the LMFBR site. The design analysis and the testing data for design basis seismic events were used to quantify seismic fragilities of the structures and components. Generic fragility curves were also evaluated based on the fragilities which were used in the precedent seismic probabilistic safety assessments (PSAs). (orig.).

Title: Automated system for the KSB complex thermal-physical bench of RBMK reactor safety and technological parameter

Author: Grigor'ev,-A.S.; Kurbatov,-V.P.; Melkov,-E.S.; Naryshkin,-V.S.; Proklov,-V.B.; Rybin,-S.G. **Corp. Author:** Gosudarstvennyj Komitet po Is

Source: 1989. 8 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** Russian

Category: Analysis of break effects **ID:** 168

Abstract: The construction principles, architecture and ways for developing an on-line system for the complex thermal-physical safety bench, as well as the results of the first experiments, are considered. The bench is a geometrically reduced model of multiple forced circulation circuit of the RBMK type reactors. The thermal capacity scale is 1:3000. The basic bench technological equipment includes four experimental channels, the circulation circuit and the system of fast-response valves allowing one to imitate different variants of emergency situations with the main pipeline ruptures or equipment failures. Experimental start-ups, a cycle of significant experiments on studying the character of heat transfer and hydrodynamics changes under the imitation of the LOCA type accidents, in particular, the process of nonhomogeneous dryout of the RBMK technological channel under the rupture of coolant-feeding pipeline were conducted in 1986-1988 in the KSB bench using the KSB technological parameter on-line control system. 1 tab.

Title: Studies on fission product retention by HTR containments according to the 'vented confinement' concept. Technical re

Author: Holzbauer,-H.; Schimetschka,-E. **Corp. Author:** Battelle-Institut e.V., Frankfurt

Source: Nov 1990. 103 p. Bundesministerium fuer Forschung und Technologie, Bonn (Germany).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Other **ID:** 169

Abstract: The capacity of the reactor building and offer components of the HTR module for retention of radioactivity released from the primary loop is analysed. The following pressure relief gradients are considered: 1. Pipe rupture in the gas cleaning unit (F-33 cm sup 2) and shut-off valve failure; 2. repture of the fuel element discharge tube (cross section equivalent-33 cm sup 2); 3. rupture of a steam generator heating tube, followed by pressure relief after failure of the pressure relief system (2F equivalent to 2x3 cm sup 3); 4. rupture of a steam generator heating tube, followed by pressure relief after failure of the live steam shut-off valve. The analysis of these four scenarios necessitated the further development of the PKL model for sufficiently accurate thermodynamic calculation also of slow flow leckage transients with acceptable computing times. This was necessary because there were no mass and enthalpy flow rates as a function of time for the four leakage flow transients in investigated. (orig./DG).

Title: Load bearing behaviour of large pipes with circumferential flaws under tensile loading - comparison of experiments wi

Author: Stadtmueller,-W.; Eisele,-U.; Julisch,-P.; Sturm,-D. **Corp. Author:** Materialpruefungsanstalt (MP

Source: 16. MPA-seminar: Safety und reliability of plant technology - long-term integrity of components of machines and systems of nuclear power plants against the background of mechanical, thermal, and corrosive loads as well as irradiation embrittlement. Stuttgart (Germany). 4-5 Oct 1990.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Test/analysis **ID:** 170

Abstract: For the tensile tests reported, performed by quasistatic tensile loading of pipe specimens, the predictions of established engineering approaches relating to maximum forces and, if possible, to crack initiating forces have been verified. The approximation methods of the engineering approaches examined, which assume idealised pipe conditions and apply flat tensile specimens, proved to be not suited for assessment of the maximum forces. The approaches especially developed for pipe geometries yielded, depending on the basic failure theory applied, in some cases results which agreed very well with the experimental data. The fracture mechanics approximation methods (R6-curve and R-curve) yielded results for the maximum forces which generally underestimate the forces in comparison to the experimental results, in some cases even by 40%, while the R6-curves in some cases overestimate the crack initiating forces. (orig./DG).

Title: MORIS. An experimental demonstrative plant for increased passive PWR safety.

Author: Avitabile,-M.; Calabro,-A. (ENEA, Casaccia (Italy). Centro Ricerche Energia) **Corp. Author:**

Source: Energia-Nucleare-Rome. (1989). v. 6(2) p. 44-50.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 171

Abstract: MORIS is an ENEA experimental facility completely made of transparent perspex and intended to show in a simple way the plant operation during an accident situation typical of a new generation of nuclear reactors. The core residual heat removal is obtained by a passive system based on natural circulation. Some typical accident sequences are applied to MORIS: TE sequence (station black-out, reactor trip, pump coastdown and loss of auxiliary feedwater), V sequence (failure of gate valve between primary system and suction of RHR pump), LOCA (loss of cooling through a break in the primary pipe). Videotape shows the MORIS behaviour during the tests. Natural circulation with numerical code results in order to analyze the flow rate and temperature established at the new stationary conditions. Optimizing studies were made on a special component, the emergency loop passive intervent check valve that stops flow in the emergency loop during normal operation. Other valve configurations were tested also. It was possible to verify the cooling level given by heat exchangers, in case of primary pump run-down. The level difference between vessel and exchangers assures also in the primary loops a natural circulation condition.

Title: On the failure probability of pipings.

Author: Schueller,-G.I.; Nienstedt,-J. (Innsbruck Univ. (Austria). Inst. of Engineering Mechanics); Tsurui,-A. (Kyoto Univ. (Japan). Dept. of Applied Mathematics and Physics) **Corp. Author:** 10. biennial international confe

Source: Nuclear-Engineering-and-Design. (Jul 1991). v. 128(2) p. 201-206.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Methods **ID:** 172

Abstract: Various methods for determining the structural reliability analysis of piping systems of NPP's are discussed in view of their accuracy, efficiency and possibility of practical application. Ultimate load as well as fatigue failure modes are considered in the analysis. The time variant reliability problem, e.g. due to fatigue and/or corrosion is solved by utilizing advanced simulation procedures. (orig.).

Title: Probabilistic fracture mechanics applied to high temperature reliability.

Author: Riesch-Oppermann,-H.; Brueckner-Foit,-A. (Karlsruhe Univ. (T.H.) (Germany, F.R.). Inst. fuer Zuverlaessigkeit und Schadenskunde im Maschinenbau) **Corp. Author:**

Source: Nuclear-Engineering-and-Design. Vol. 128:193-200.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Research/theoretical **ID:** 173

Abstract: An example is used to demonstrate the applicability of Probabilistic Fracture Mechanics (PFM) methods in high temperature reliability assessment. The failure probability of a pipe under pure bending at a temperature of 973 K is calculated using both Monte Carlo simulation and the First Order Reliability Method. The advantages and the accuracy of approximative methods for calculating failure probabilities are demonstrated. Additionally, probabilistic and deterministic methods for reliability assessment are compared with each other. It is shown that a deterministic reliability assessment becomes inadequate in cases where the failure probability is determined by equally significant contributions of several random variables. (orig.).

Title: Approximate fracture methods for pipes. Pt. 2. Applications.

Author: Gilles,-P. (Societe Franco-Americaine de Constructions Atomiques (FRAMATOME), 75 - Paris (France)); Chao,-K.S. (Taiwan Power Co. (Taiwan)); Brust,-F.W. (Battelle, Columbus, OH (USA). Structures and Mechanics Dept.) **Corp. Author:**

Source: Nuclear-Engineering-and-Design. (May 1991). v. 127(1) p. 19-31.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Methods/comparison **ID:** 174

Abstract: In the part I paper entitled 'Approximate fracture methods for pipes - Part I, Theory', five different J-estimation schemes for through-wall cracked pipes were presented. The (i) GE.EPRI method utilizes a compilation of finite-element solutions. The (ii) Paris/Tada and (iii) LBB.NRC methods utilize an interpolation between the linear elastic and rigid plastic solutions, (iv) the LBB.GE method also uses numerical solutions, and (v) the LBB.ENG uses an equivalent area method to estimate J. All five methods are very simple to use and all five give reasonable predictions of crack growth and failure in pipes. The present paper provides a comparison of some of the methods to full-scale finite-element analyses. In addition, predictions for actual pipe experiments compared to experimental data are also provided. (orig.).

Title: Program to justify life extension of older nuclear piping systems.

Author: Burr,-T.K.; Dwight,-J.E. Jr.; Morton,-D.K. **Corp. Author:** EG and G Idaho, Inc., Idaho Fa

Source: [1991]. 6 p. American Society of Mechanical Engineers (ASME) pressure vessels and piping conference. San Diego, CA (USA). 23-27 Jun 1991.FUNDING ORGANIZATION: USDOE, Washington, DC (USA).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 175

Abstract: Life extension evaluations of INEL's ATR have been initiated. Of particular importance are the associated high temperature, high pressure loop piping system supporting in--reactor experiments. Failure of this piping could challenge core safety margins. Since regulatory rules for nuclear power plant life extension are only in the formulation stage, the current technical guidance on this subject provided by the Department of Energy (DOE) or the commercial nuclear industry is incomplete. In the interim, order to assure continued safe operation of this piping beyond its initial design life, a program has been developed to provide the necessary technical justification for life extension. This paper describes a program that establishes Section 11 of the ASME Boiler and Pressure Vessel Code as the governing criteria document, retains B31.1 as the Code of record for Section 11 activities, specifies additional inservice inspection requirements more strict than Section 11, and relies heavily on flaw detection and fracture mechanics evaluations. 18 refs., 2 figs.

Title: Erosion-corrosion in secondary circuits the mastery of the damage.

Author: Bouchacourt,-M.; Lenormand,-A.; Remy,-F.N. (Electricite de France (EDF), 75 - Paris (France)) **Corp. Author:** International Colloquium on C

Source: Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France). Contribution of Materials Investigation to the Resolution of problems encountered in PWR Plants. Volume 2. Contribution des Expertises sur materiaux a la Resolution des problemes rencontres dans les REP. Volume 2. Paris (France). Societe Francaise d'Energie Nucleaire. 1990. 305 p. p. 517-525.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** French; ; English

Category: Experience/events **ID:** 176

Abstract: Before 1987, a lot of people thought that the damages due to erosion-corrosion appears randomly and concerned unlucky companies. After the Surry pipe failure, all utilities considered that it was necessary to take into account erosion-corrosion wear in all cases, for an immediate corrective action, or about life time study. This presentation contains three parts as follow: - The EDF efforts in research allow to explore the influence of parameters not often studied before now: oxygen level less than 15 ppb, chromium contents between 0 and 0,25%. Tests are performed on tubes where the mass transfer is well known. - To limit the erosion-corrosion wear kinetik, it is necessary to optimize water chemistry. We present the EDF choice for the secondary water chemistry, and the management of the inspection results to consider whether the modifications are enough to avoid future problems. - After the Surry failure, predictive methods have been used to confirm the previous choices and light inspection are performed to verify the theoretical analysis results. All the means used to master this problem are brought together in order to solve all the raised questions.

Title: Steam generator tube failure monitoring and break accident of PWR power plants.

Author: Ding-Xunshen (Southwest Inst. of Nuclear Reactor Engineering, Sichuan, SC (China)) **Corp. Author:**

Source: Nuclear-Power-Engineering. (Apr 1990). v. 11(2) p. 55-59.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** Chinese

Category: Inspection methods **ID:** 177

Abstract: The SG blowdown sampling analysis and sup 1 sup 6 N monitoring outside main steam pipeline are major monitoring means for SG tube failure. The accident process and the treatment measures after occuring SG tube failure are also described. Additionally, the methods for reviewing about SG tube break accident is also introduced.

Title: An analysis of molten-corium-induced failure of drain pipes in BWR Mark 2 containments.

Author: Taleyarkhan,-R.P. (Oak Ridge National Lab., TN (USA)); Podowski,-M.Z. (Rensselaer Polytechnic Inst., Troy, NY (USA)) **Corp. Author:** Oak Ridge National Lab., TN (

Source: [1991]. 12 p. .ASME/AIChE/ANS national heat transfer conference. Minneapolis, MN (USA). 26-31 Jul 1991.FUNDING ORGANIZATION: USDOE, Washington, DC (USA); Nuclear Regulatory Commission, Washington, DC (USA).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Research/theoretical **ID:** 178

Abstract: This study has focused on mechanistic simulation and analysis of potential failure modes for inpedestal drywell drain pipes in the Limerick boiling water reactor (BWR) Mark 2 containment. Physical phenomena related to surface tension breakdown, heatup, melting, ablation, crust formation and failure, and core material relocation into drain pipes with simultaneous melting of pipe walls were modeled and analyzed. The results of analysis have been used to assess the possibility of drain pipe failure and the resultant loss of pressure-suppression capability. Estimates have been made for the timing and amount of molten corium released to the wetwell. The study has revealed that significantly different melt progression sequences can result depending upon the failure characteristics of the frozen metallic crust which forms over the drain cover during the initial stages of debris pour. Another important result is that it can take several days for the molten fuel to ablate the frozen metallic debris layer -- if the frozen layer has cooled below 1100 K before fuel attack. 10 refs., 3 figs., 4 tabs.

Title: Analysis of parameter sensitivity of the probabilistic model of brittle fracture initiation in WWER pressure vessels.

Author: Horacek,-L. (Skoda, Plzen (Czechoslovakia). Zavod Energeticke Strojirenstvi) **Corp. Author:** Zavodni Pobocka Ceske Vedec

Source: . National conference on brittle fracture of materials and structures. Celostatni konference "Krehky lom materialu a konstrukci". Dec 1990. 166 p. p. 46-50.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** Czech

Category: Methods **ID:** 179

Abstract: A simplified reliability model designed for the probabilistic assessment of resistance of WWER pressure vessels to the initiation of brittle fracture is briefly described. The model is applicable particularly within the limits of validity of linear elastic fracture mechanics. Its use is demonstrated on examples of evaluation of accident regimes of the temperature shock type (failure of piping 20 and 32 mm in diameter at coolant water temperature 55 degC) for actual input data specific of units 1 and 2 of the V-1 nuclear power plant in Jaslovske Bohunice. (Z.M.). 2 figs., 5 refs.

Title: The effect of compressive loads on the integrity of a cracked piping system.

Author: Smith,-E. (Manchester Univ. (UK). Inst. of Science and Technology) **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1991). v. 46(2) p. 125-132.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Research/theoretical **ID:** 180

Abstract: The paper examines the integrity of a cracked piping system due to the effect of applied loadings, which give rise to axial compressive loads. A theoretical analysis for a simple model system defines the conditions for which the deformed, as distinct from the undeformed, piping configuration should be used when determining the failure criterion for a piping system. (author).

Title: Determination of creep conditions prior to rupture of WWER vessels and pipes.

Author: Brumovsky,-M. (Skoda, Plzen (Czechoslovakia). Zavod Energeticke Strojirenstvi); Zdarek,-I. (Ustav Jaderneho Vyzkumu CSKAE, Rez (Czechoslovakia)); Anikovskij,-V.V.; Karzov,-G.P. (TsNIIKM Prometej, Leningrad (USSR)); Dragunov,-Yu.G.; Getmanchuk,-A.V. (Opytno-Konstruktorskoe Byuro Gidropress, Podol'sk (USSR)); Rivkin,-E.Yu. (Nauchno-Issledovatel'skij i Konstruktorskij Inst. Ehnergotekhniki, Moscow (USSR))

Corp. Author: Zavodni Pobocka Ceske Vedec

Source: National conference on brittle fracture of materials and structures. Celostatni konference "Krehky lom materialu a konstrukci". Dec 1990. 166 p. p. 24-27.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** Russian

Category: Research/theoretical **ID:** *181*

Abstract: Elastic-plastic analysis was applied to calculate the limiting load of materials for WWER pressure vessels and pipelines for the case of disturbance of the transition zone beneath a surface defect or the case of complete failure of the vessel at the site of a deep crack. The results are presented in the tabular form. Relations suggested by various authors for the determination of the limiting load are also briefly characterized. (Z.M.). 1 tab., 1 ref.

Title: Rupture detection device for pipeline in reactor.

Author: Murakoshi,-Toshinori (Toshiba Corp., Kawasaki, Kanagawa (Japan)); Kanamori,-Shigeru; Shirasawa,-Hirofumi

Corp. Author: Toshiba Corp., Kawasaki, Kan

Source: 1 Feb 1991; 21 Jun 1989. 7 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** Japanese

Category: Inspection methods **ID:** *182*

Abstract: A difference between each of the pressures in a plurality of pipelines disposed in a shroud a reactor container and a pressure outside of the shroud is detected, thereby enabling safety and reliable detection even for simultaneous rapture and leakage of the pipelines. That is, a difference between the pressure of a steam phase outside of the shroud and a pressure in each of a plurality of low pressure injection pipelines in an emergency core cooling system opened to the inside of the shroud in the reactor container is detected by a difference pressure detector for each of them. Then, an average value for each of the pressure difference is determined, which is compared with the difference pressure obtained from each of the detectors in a comparator. Then, if openings should be caused by rupture, leakage or the like in any of the pipelines, the pressure in that pipeline is lowered to a vicinity of an atmospheric pressure and at the vapor phase pressure at the lowest. If the pressure is compared with the average value by the comparator, a negative difference is caused. Accordingly, an alarming unit generates an alarm based on the pressure difference signal, thereby enabling to specify the failed pipeline and provide an announce of the failure. (I.S.).

Title: Estimating the Relative Probability of Piping Severance by Fault Cause

Author: Wilson, S.A.

Corp. Author: General Electric, San Jose (CA

Source: GEAP-20615 (AEC Research and Development Report)

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1974 **Language:** English

Category: Failure probability **ID:** *183*

Abstract: The objectives of the work described in this report are to prepare a comprehensive list of fault causes and to estimate the relative contribution made by each to the probability of severance in reactor primary piping. Severance is regarded as the sudden failure of a pipe without prior detectable leakage, with either circumferential or axial opening of substantial area. A fault cause is an event in the course of design, fabrication or operation of the piping system which ultimately proves to be the cause of some fault contributing to piping severance. These faults are principally an increased actual stress, decreased critical stress for severance due to unfavorable fracture properties of the material, increased crack frequency or growth rate, and decreased crack detection capability.

Title: Pre-test analysis of a pipe system for high-level vibration response and failure.

Author: Weiner,-E.O.; Severud,-L.K. **Corp. Author:** Westinghouse Hanford Co., Ri

Source: Mar 1991. 8 p. American Society of Mechanical Engineers (ASME) pressure vessels and piping conference. San Diego, CA (USA). 23-27 Jun 1991.FUNDING ORGANIZATION: USDOE, Washington, DC (USA).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 184

Abstract: Simplified elastic and inelastic analyses for high level vibration response and cyclic failure capacity of a prototypic light-water reactor pipe system were carried out in a pretest environment. The system consists of a steam generator and a circulating pump with associated piping that has been tested on a shake table. Five analyses, ranging from standard linear elastic to detailed inelastic transient analysis, are compared in terms of response. With the inelastic analysis, subsequent failure analysis indicated that strain in the 3% to 4% range can be expected if the planned inputs are realized. Possible cyclic failure was predicted by through-wall cracking and leaking in 20 to 40 cycles of maximum strain range, caused by ratchet-fatigue in the pressurized system. 20 refs., 7 figs.

Title: Effects of aging on failure rates of pipes.

Author: Jamali,-K.M. (Atrek Corp., 12046 Montrose Village Terrace, Rockville, MD (USA)); Dube,-D.A. (Northeast Utilities Service Co., Hartford, CT (USA)) **Corp. Author:** American Nuclear Society (AN

Source: Anon.-Proceedings of the topical meeting on nuclear power plant life extension. Volume 2. La Grange Park, IL (USA). American Nuclear Society. 1988. 645 p. p. 599-607.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Experience/events **ID:** 185

Abstract: The behavior of time-dependent failure rates of piping in LWR's is analyzed. Quantification is based on the incidents of leakage and rupture failures reported in LER's. Results are presented in terms of a multiplicative time-factor including uncertainty bounds for leakage and rupture events in PWR and in BWR's, and rupture events in all light water reactors combined. The characteristics of time dependence are markedly different for the various categories. For example, the generic PWR failure rate is decreasing for the first 10 years of operation, and remains nearly constant thereafter; while, BWR generic failure rates display a periodic behavior with a period of about 10 to 15 years.

Title: Cooling system for heat dissipation type reactor container.

Author: Takahashi,-Hideaki **Corp. Author:** Toshiba Corp., Kawasaki, Kan

Source: 9 Oct 1990; 27 Mar 1989. 4 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** Japanese

Category: Other **ID:** 186

Abstract: The present invention provides a cooling system for spontaneous heat dissipation type reactor container suitable to the cooling of a BWR type reactor container upon loss of coolant accident (LOCA). That is, the system comprises an upper pool disposed in the upper portion of a reactor container, a cooling water recycling type cooling device disposed in a dry well and a circulation pipeline connecting the cooling water inlet/outlet of the cooling device and the upper pool. As a result, in case if high pressure pipe lines in the reactor primary coolant circuits should be failed to jet out coolants in the dry well as in the case of LOCA, or upon occurrence of an accident in which steams at high temperature and high pressure should be leaked from the pipelines of main steams to the inside of the reactor container, heat of the leaked steams is dissipated by the heat conduction pipes of the cooling device in the dry well into the upper pool. Further, if the pressure of the leaked steams is reduced, heat can be dissipated efficiently by way of the heat conduction pipes to the upper pool. Accordingly, cooling can be conducted rapidly after LOCA. (I.S.).

Title: An evaluation of the impact of inservice inspection on stress corrosion cracking of BRW piping.

Author: Simonen,-F.A. (Battelle Pacific Northwest Lab., Richland, WA (USA)) **Corp. Author:** 1990 pressure vessels and pipin

Source: Sammataro,-R.F. (General Dynamics Corporation (USA)). Codes and standards and applications for design and analysis of pressure vessel and piping components 1990. PVP-Volume 186. NDE-Volume 7. New York, NY (USA). American Society of Mechanical Engineers. 1990. 203 p. p. 187-194.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Research/theoretical **ID:** 187

Abstract: This paper describes probabilistic fracture mechanics calculations that evaluate the potential impact of inservice inspection (ISI) in reducing the occurrence of failures in boiling water reactor (BWR) piping due to intergranular stress corrosion cracking (IGSCC). The probabilistic model simulates the detection of cracks with extended periods of incubation and slow growth followed by a final period of relatively rapid growth to through-wall depths. This semi-empirical model was calibrated first with laboratory measurements of growth rates for stress corrosion cracks in stainless steel piping, and then with occurrence frequencies for weld cracking from reactor operating experience. The relative benefits of alternative ISI scenarios are addressed. Each scenario consisted of a specific inspection interval and a prescribed level of nondestructive evaluation (NDE) sensitivity as characterized by data from piping inspection round robins. Calculations show that significant improvements in piping system reliability can be achieved by frequent, high quality inspections.

Title: Tests on the failure of a main refrigerant pipe due to creep fracture at high system pressure. Final report.

Author: Obst,-V.; Klenk,-A.; Julisch,-P. **Corp. Author:** Stuttgart Univ. (Germany, F.R.

Source: May 1988. 190 p. Bundesministerium fuer Forschung und Technologie, Bonn (Germany, F.R.).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Test/analysis **ID:** 188

Abstract: For a better understanding of the fracture of failure behaviour of internal pressure loaded pipes with dimensions of the main refrigerant pipe of pressurized water reactors, at first tests on hot drawing, stress-rupture and heating have been carried out on small specimens made of the steels 20 MnMoNi 55 and 22 NiMoCr 37 of different material conditions. Then a structural test has been performed on a tubular tank made of the material 20 MnMoNi 55. The test tank has been heated in two heating phases to about 700deg C (heating gradients 4 or 7 K/min) at a constant internal pressure of p=163 bar (air as the pressure medium), and during a stop phase the temperature was kept constant up to the failure of the tank. (MM).

Title: Current and Emerging Pressure Boundary Issues - A US Regulatory Perspective.

Author: Shewmon,-P. **Corp. Author:**

Source: Nuclear-Engineering-and-Design, Vol. 124:17-21.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Experience/events **ID:** 189

Abstract: RPV problems stem from exposure to fast neutrons which changes the NDT and the elevated temperature fracture energy of some vessels. The predicted shift in NDT has increased over the last decade as more has been learned about the effect of impurities (copper) and the synergism between nickel and copper. In PWRs this has led to concern about PTS. In BWRs one cannot have PTS events, but the more rapid than expected rise in NDT due to irradiation is impacting operations. In another set of PWRs the upper shelf energy of the welds was initially low due to the use of a slag which led to many small inclusions in the weld. Radiation has lowered the Charpy fracture energy of these welds to below the 50 ft lb level at which there is concern that the vessel may undergo low energy ductile failure even if cleavage does not occur. Problems in pressure boundary piping has stemmed primarily from corrosion; i.e., IGSCC in BWR recirculation piping, and S/G tube failures in PWRs. These have made a large contribution to downtime and occupational exposure, but are not seen as significant contributors to risk. There has been some concern about the aging (loss of toughness) of cast stainless components with significant ferrite content, especially because inspection by UT is difficult. (orig.).

Title: Validation of Experimental and Computational Fracture Assessment Methods for Flawed Pressure Components.

Author: Rintamaa,-R.; Keinaenen,-H.; Wallin,-K.; Talja,-H.; Saarenheimo,-A.; Ikonen,-K.

Corp. Author:

Source: Nuclear-Engineering-and-Design, Vol. 124:193-216.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 190

Abstract: To improve the accuracy and validity of the experimental and computational fracture assessment methods, a 4-year Nordic research program under the auspices of NKA was initiated in 1985. The main technical objective of the program was to clarify how catastrophic failure can be prevented in RPVs and piping. Experiments with small fracture mechanics specimens and pressure vessels were performed to validate the computational fracture assessment analysis. Two tests were conducted on a decommissioned full-scale chemical RPV from an oil refinery plant, and were extensively instrumented, e.g. by utilizing a 64-channel acoustic emission monitoring system. The scattering of their material property values were determined by numerous fracture mechanics samples. In addition, as a part of the experimental work, the reactor pressure vessel was repaired by welding after the first test. The repair was done without postweld heat treatment and welding was done by applying the temper-bead technique. Residual stresses were measured during and after welding. Different fracture assessment methods were developed and subsequently applied to the tested components. Inter-laboratory round robin programmes with the participation of several laboratories were arranged to examine elastic-plastic finite element calculations and fracture mechanics testing.

Title: Safety analysis for pipe rupture accidents of primary cooling system for HTTR.

Author: Maruyama,-So; Okamoto,-Futoshi; Nakagawa,-Shigeaki; Shindo,-Masami (Japan Atomic Energy Research Inst., Oarai, Ibaraki (Japan). Oarai Research Establishment)

Corp. Author: Japan Atomic Energy Research

Source: Oct 1990. 38 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** Japanese

Category: Analysis of break effects **ID:** 191

Abstract: In order to evaluate the magnitude of the effects on the integrity of the reactor facility due to the pipe rupture accidents of the primary cooling system, which are important in the safety evaluation of the High Temperature Engineering Test Reactor (HTTR), safety evaluation was performed concerning with the area of rupture for the failure of the co-axial double pipe and the inner pipe of the primary cooling system. It was found through the present work that a double-ended rupture of the co-axial double pipe was the severest from a viewpoint of the core damage in the failure of the co-axial double pipe and a double-ended rupture of the inner pipe was the severest from a viewpoint of the effect on the reactor vessel (reactor vessel temperature) in the failure of the inner pipe, respectively. (author).

Title: Selected results of analysis of small and medium primary coolant leaks.

Author: Misak,-J. (Vyskumny Ustav Jadrovych Elektrarni, Trnava (Czechoslovakia))

Corp. Author: Operation and maintenance of

Source: Jaderna Elektrarna, Dukovany (Czechoslovakia). Operation and maintenance of nuclear power plant. National conference proceedings. Provozovani a udrzba jaderne elektrarny. Sbornik prednasek celostatni konference. Apr 1990. 246 p. p. 81-88.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** Slovak

Category: Research/theoretical **ID:** 192

Abstract: Problems are discussed of leaks through holes with equivalent diameters of less than 200-300 mm which can be repaired without the use of water reservoirs. During such failures, occlusions form in the hot and cold water branches of the main circulating pipe, preventing free flow of steam from the core. The primary circuit depressurization becomes a long-term problem. Processes arising during the pressure reduction and mechanisms of core overheating are discussed. The temperature and pressure changes are slowed down and the primary circuit depressurization is complicated by the coolant temperatures in the primary and the secondary circuits approaching each other. Improved secondary circuit cooling has a beneficial temperature and hydraulic effect when coping with an accident. (M.D.). 3 figs.

Title: Piping dynamic reliability and code rule change recommendations.

Author: Tagart,-S.W. Jr.; Tang,-Y.K. (Electric Power Research Inst., Inc., Palo Alto, CA (USA)); Guzy,-D.J. (Nuclear Regulatory Commission, Washington, DC (USA)); Ranganath,-S. (General Electric Co., San Jose, CA (USA))

Corp. Author: 2. symposium on current issues

Source: Nuclear-Engineering-and-Design. (Oct 1990). v. 123(2/3) p. 373-385.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Methods/design **ID:** 193

Abstract: The conservative nuclear piping design criteria for seismic and dynamic loads have led to piping systems with excessive numbers of snubbers. To improve this undesirable situation, a Piping and Fitting Dynamic Reliability Program was initiated by EPRI in 1985 with cooperation from the NRC. The objective of the program is to develop improved, realistic, and defensible ASME design rules by taking advantage of the inherent dynamic margins in the nuclear piping system. The research results have demonstrated that piping systems have large reserve dynamic capacity and the dynamic failure mode is due to fatigue or fatigue-ratcheting rather than plastic collapse. Based on such physical evidence, a set of code rule change recommendations is suggested in its preliminary form. (orig.).

Title: Pressure-Dependent Fragilities for Piping Components: Pilot Study on Davis-Besse Nuclear Power Station.

Author: Wesley,-D.A.; Nakaki,-D.K.; Hadidi-Tamjed,-H. (ABB Impell Corp., Mission Viejo, CA (USA)); Kipp,-T.R. (EQE, Inc., Costa Mesa, CA (USA))

Corp. Author: Nuclear Regulatory Commissio

Source: Oct 1990. 110 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Analysis of break effects **ID:** 194

Abstract: The capacities of four, low-pressure fluid systems to withstand pressures and temperatures above the design levels were established for Davis-Besse. The results will be used in evaluating the probability of plant damage from ISLOCAs as part of the Davis-Besse PRA undertaken by EGG Idaho, Inc. Included in this evaluation are the tanks, heat exchangers, filters, pumps, valves, and flanged connections for each system. The probabilities of failure, as a function of internal pressure, are evaluated as well as the variabilities associated with them. Leak rates or leak areas are estimated for the controlling modes of failure. The pressure capacities for the pipes and vessels are evaluated using limit-state analyses for the various failure modes considered. The capacities are dependent on several factors, including the material properties, modeling assumptions, and the postulated failure criteria. The failure modes for gasketed-flange connections, valves, and pumps do not lend themselves to evaluation by conventional structural mechanics techniques and evaluation must rely primarily on the results from ongoing gasket research test programs and available vendor information and test data. 21 refs., 7 figs., 52 tabs.

Title: Component wall thinning and a corrosion-erosion monitoring system.

Author: Bogard,-T.; Batt,-T.; Roarty,-D. (Westinghouse Electric Corp., Pittsburgh, PA (USA))

Corp. Author:

Source: Anon.-Power-gen 1989. Conference papers, Volumes V and VI. Houston, TX (USA). Power-Gen. 1989. 413 p. p. 977-990.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Inspection methods **ID:** 195

Abstract: Since a 1986 incident involving failure of a piping elbow due to erosion-corrosion, the utility industry has been actively developing technology for implementing long term programs to address erosion-corrosion. This paper describes a typical corrosion-erosion monitoring program, the types of NDEs performed on components, and the extensive NDE data obtained when the program is applied to components in a power plant. To facilitate evaluation of the NDE data on components, an automated NDE data manipulation and data display system is advisable and perhaps necessary due to the large amounts of NDE data typically obtained during a program. Such a comprehensive corrosion-erosion monitoring system (CEMS) needs to be integral with methods for selection of inspection locations and perform NDE data analysis to help in replace, repair, or run decisions. The structure for one CEMS software is described which addresses most data evaluation and decision making needs. CEMS features include automated input/output for typical NDE devices, database structuring, graphics outputs including color 2-D or 3-D contour plots of components, trending and predictive evaluations for future inspection planning, EC severity determination, integration of piping isometrics and component properties, and desktop publishing capabilities.

Title: Seismic reliability analysis of nuclear power plant components and piping systems, (2). Numerical studies on failure of

Author: Matsuura,-Shin-ichi; Hirata,-Kazuta; Nakamura,-Hideharu; Otomo,-Keizo; Hagiwara,-Yutaka (Central Research Inst. of Electric Power Industry, Abiko, Chiba (Japan). Abiko Research Lab.) **Corp. Author:**

Source: Denryoku-Chuo-Kenkyusho-Hokoku. (Mar 1990). (no.U89050) p. 1-4, 1-35.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** Japanese

Category: Methods/comparison **ID:** 196

Abstract: In a preceding report, which reviewed various methods of seismic reliability analysis, the authors pointed out the lack of fragility data and numerical results of seismic reliability analysis about nuclear power plant components, such as piping systems and important facilities. The main purpose of this report is to show the results of numerical seismic response analysis of a piping model shown in a NUREG fragility test report, and to confirm the effectiveness of the method. Material nonlinearity effects and ovalizations in pipe cross-sections are taken into account in the analysis. Comparing the numerical results with the experiments, following nonlinear dynamic characteristics of the piping system became clear: a) The piping system responds in a stable way even if some portions of the system cause plastic deformation and stiffness decreases locally. b) Estimated strain level and yielding positions by analyses are in good agreement with experiments. c) There are prospects of the fatigue failure estimation for piping systems by the numerical analyses. (author).Record 74 of 122 - INIS 1990 - 12/92

Title: The application of approximation methods to the calculation of the probability of failure of structures with cracks unde

Author: Riesch-Oppermann,-H. **Corp. Author:** Karlsruhe Univ. (T.H.) (Germa

Source: 16 Feb 1989. 70 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** German

Category: Research/theoretical **ID:** 197

Abstract: In the context of this work, the applicability of the first order method (FORM) for different areas of probabilistic fracture mechanics is examined. The main point was the consideration of failure at low temperatures due to static and alternating stresses, on the one hand, and the extension of the possible area of application of the first order method to high temperature failure of components under creep stress, on the other hand. The method is used for the calculation of the probability of failure of the safety containment of a pressurized water reactor and of a pipe elbow in the SNR 300 fast breeder reactor. The first order method can be used to calculate the probability of failure with a deviation of 10-20% from the numerically determined values or those from a Monte Carlo simulation. (MM).

Title: Using reliability techniques to investigate pipe breaks caused by seismically-induced support failures.

Author: Lo,-T.Y.; Holman,-G.S. (Lawrence Livermore National Lab., CA (USA)) **Corp. Author:** 10. international conference on

Source: Hadjian,-A.H. (Bechtel Power Corp., Los Angeles, CA (USA)). Transactions of the 10th international conference on structural mechanics in reactor technology. Volume K1-K2. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1989. 967 p. p. 935-940.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Methods **ID:** 198

Abstract: Based on the origin of failure, the study of piping reliability during an earthquake has been divided into two parts: the direct and the indirect pipe failures (either a leak or a double-ended guillotine break (DEGB)). Direct pipe failure is defined as pipe failure caused by the growth and instability of existing cracks in the piping system. Cracks grow during the lifetime of a piping system and may become unstable due to seismically-induced stresses. Indirect pipe failure is due to failure of other structures or components, which in turn cause the pipe to fail. One major indirect source is earthquake-generated missiles, such as falling objects. The other major source of indirect pipe failure is the failure of pipe and component supports. That is, and earthquake can cause the support to fail first, and this failure breaks the pipe. In reality, direct and indirect pipe failure can be closely related because the pipe can fail due to crack growth and instability (direct source) induced by the stress conditions caused by missile and/or support failure (indirect source). A formulation for comprehensive probabilistic analysis of the seismically induced pipe failure is presented.

Title: Load carrying behaviour of the primary system of PWRs for loads beyond the design limits. Pt. 2. Creep and failure be

Author: Maile,-K.; Klenk,-A.; Obst,-V.; Sturm,-D. (Stuttgart Univ. (Germany, F.R.). Staatliche Materialpruefungsanstalt) **Corp. Author:**

Source: Nuclear-Engineering-and-Design, Vol.. 119:131-137.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Research/theoretical **ID:** 199

Abstract: The material behaviour of components in the primary system of pressurized water reactors under conditions surpassing the design criteria, i.e. if temperature increases considerably and system pressure reaches a maximum level, was examined by means of a component test and small-scale specimen tests. The results of the tests with small-scale specimens regarding the creep behaviour at high temperature were compared with the material behaviour of a pipe section which had been exposed to internal pressure corresponding to real system pressure and a temperature of 700deg C. The components behaved as could be expected from the tests with small-scale specimens. (Steel 20 MnMoNi 55, St. 1-6310). (orig./GL).

Title: Resonant Excitation of a Pipe Section.

Author: Kerkhof,-K.; Stoppler,-W.; Sturm,-D.; Zirn,-R. (Stuttgart Univ. (Germany, F.R.). Staatliche Materialpruefungsanstalt) **Corp. Author:**

Source: Nuclear-Engineering-and-Design, Vol. 119:361-370.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 200

Abstract: During the recent phase of the project 'vessel failure' tests have been carried out on pipe sections under fast external cyclic bending. In connection to former examinations with slowly alternating bending loading now the resonance effect with high accelerations of masses together with energy dissipation due to material plastification was taken into account. In this contribution the design calculations and their results are described as well for the pretests with small dimensions as for the main tests with dimensions of outer pipe diameter 250 mm x 32 mm wall thickness. The main reason for performing pretests was to find out the right regulating technique for the load controlled excitation mechanism and to have a verification for the calculational model. A comparison between the predicted calculated system response and the measured one is pointed out. A good agreement between calculation and measurement was observed. An outlook to the kind of the main tests is given by calculation. (orig.).

Title: Pipe failure tests (RORV) at HDR facility. Experimental results.

Author: Hunger,-H.A. (Kernforschungszentrum Karlsruhe, Karlsruhe (Germany, F.R.)); Diem,-H. (MPA, Univ. Stuttgart, Stuttgart (Germany, F.R.)) **Corp. Author:** 10. international conference on

Source: Hadjian,-A.H. (Bechtel Power Corp., Los Angeles, CA (USA)). Transactions of the 10th international conference on structural mechanics in reactor technology. Volume F. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1989. 257 p. p. 141-146.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Test/analysis **ID:** 201

Abstract: This paper reports on pipe failure tests performed at the HDR test facility. Objects of the investigations were a straight pipe and a 90 degree pipe band each of diameter DN 400 both being parts of a 13/23 m long ferritic piping connected with the reactor pressure vessel. This paper emphasizes the final blowdown process, i.e. the crack breaking through the ligament and the effects of the escaping medium. Measured strains on the pipe surface, crack mouth opening displacement, and temperatures on the inside/outside of the cracked pipe are shown and commented.

Title: Crack initiation and crack propagation of an elbow under in-plane bending in high temperature water of elevated oxyg

Author: Diem,-H.; Blind,-D. (Stuttgart Univ. (Germany, F.R.)); Katzenmeier,-G.; Hunger,-H.A. (Kernforschungszentrum Karlsruhe GmbH (Germany, F.R.))

Corp. Author: 10. international conference on

Source: Hadjian,-A.H. (Bechtel Power Corp., Los Angeles, CA (USA)). Transactions of the 10th international conference on structural mechanics in reactor technology. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1989. 375 p. p. 65-76.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Test/analysis **ID:** 202

Abstract: This paper reports on a pipe bend failure experiment performed in a full size feedwater pipe system under operating conditions. The analysis of the fracture surface indicated that the crack propagation rate had increased as loading frequency had decreased. The final crack length in the leakage area reached 67% of the elbow center line. This macroscopically dominating crack was embedded in a multiple-crack field.

Title: LEFM of cracked pipes with P-version finite element modeling.

Author: Woo,-K.S.; Basu,-P.K. (Vanderbilt Univ., Nashville, TN (USA))

Corp. Author: 10. international conference on

Source: Hadjian,-A.H. (Bechtel Power Corp., Los Angeles, CA (USA)). Transactions of the 10th international conference on structural mechanics in reactor technology. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1989. 375 p. p. 363-368.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Research/theoretical **ID:** 203

Abstract: In the case of cylindrical pressure vessels and pipes used in chemical industries and power plants, it is important to ensure that no catastrophic failure caused by unstable crack growth can occur under both normal operating conditions and overload situations caused, for instance, by an accident or due to faulty conditions. Unstable crack growth is often initiated at existing flaws which may be in the form of laminations pit marks surface scars, unsound welds, etc. These flaws can be classified as surface flaws, embedded flaws, and through-wall flaws. If the material possesses low ductility, small scale crack-tip plasticity will occur and the crack growth will be K-controlled, so that LEFM will be applicable. The present study is concerned with the LEFM calculations in the presence of circumferential and longitudinal through-wall cracks in cylindrical shells.

Title: Application of degraded piping program results to leak-before-break and in-service flaw assessment criteria.

Author: Wilkowski,-G.M.; Ahmad,-J.; Kramer,-G; Marschall,-C.W. (Battelle Columbus Labs., OH (USA))

Corp. Author: 10. international conference on

Source: Hadjian,-A.H. Transactions of the 10th international conference on structural mechanics in reactor technology. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1988. 155 p. p. 105-116.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Research/theoretical **ID:** 204

Abstract: This paper summarizes the significance of the U.S. NRC's Degraded Piping Program - Phase II for pipe fracture evaluations. This was a 5 year program that ended in January of 1989. The intent of this program was to experimentally validate and enhance available analytical methods for evaluating the mechanical behavior of nuclear power plant piping containing circumferentially oriented defects. Included in this paper are discussions of: the significance of program results to LBB and in-service flaw acceptance criteria, the importance of material characterization and observations of failure modes in flaw evaluation procedures, and areas in which additional study is needed for improved piping.

Title: Rupture hardware minimization in pressurized water reactor piping.

Author: Mukherjee,-S.K.; Ski,-J.J.; Chexal,-V.; Norris,-D.M.; Goldstein,-N.A.; Beaudoin,-B.F.; Quinones,-D.F.; Server,-W.L. **Corp. Author:**

Source: ASME Journal-of-Pressure-Vessel-Technology, Vol. 111:64-71.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: LBB justification **ID:** 205

Abstract: For much of the high-energy piping in LWRs, fracture mechanics calculations can be used to assure pipe failure resistance, thus allowing the elimination of excessive rupture restraint hardware both inside and outside containment. These calculations use the LBB-concept and include part-through-wall flaw fatigue crack propagation, through-wall flaw detectable leakage, and through-wall flaw stability analyses. Performing these analyses not only reduces initial construction, future maintenance, and radiation exposure costs. This paper presents the LBB methodology applied a Beaver Valley-2; the application for two specific lines, one inside containment (stainless steel) and the other outside containment (ferrutic steel), is shown in a generic sense using a simple parametric matrix. The overall results indicate that pipe rupture hardware is not necessary for stainless steel lines inside containment greater than or equal to 6-in. (152-mm) nominal pipe size that have passed a screening criteria designed to eliminate potential problem systems (such as the feedwater system). Similarly, some ferritic steel line as small as 3-in. (76-mm) diameter (outside containment) can qualify for pipe rupture hardware elemination.

Title: Probabilistic risk assessment based guidance for piping in-service inspection.

Author: Vo,-T.V.; Gore,-B.F.; Eschbach,-E.J.; Simonen,-F.A. (Pacific Northwest Lab., Richland, WA (USA)) **Corp. Author:**

Source: Nuclear-Technology, Vol. 88(1):13-20.

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Inspection methods **ID:** 206

Abstract: Some of the goals of the Nondestructive Evaluation Reliability Program are to assess current inspection requirements for all pressure boundary systems and components, to determine whether improvements to the requirements are needed, and, if necessary, to develop recommendations for revising the ASME Boiler and Pressure Vessel Code and regulatory requirements. Part of the work performed in addressing this goal was the development and demonstration of a method to establish in-service inspection priorities through the use of PRA results. The Oconee-3 PRA and the observed weld failure data of the nuclear plants operating in the US are used to identify and prioritize the most risk-important systems for inspection. Failure modes and effects analysis methodology is then used to identify and prioritize the most risk important piping sections of the Oconee-3 emergency feedwater system. Based on the results of this study, this method is demonstrated to be a useful tool for identifying systems and piping sections or welds that need to be inspected.

Title: Erosion/corrosion experience in U.S. LWRs.

Author: McCracken,-C.E.; Wu,-P.C.S. (Chemical Engineering Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. (USA)) **Corp. Author:** American power conference. C

Source: Proceedings-of-the-American-Power-Conference. (1988). v. 50 p. 982-991.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Experience/events **ID:** 207

Abstract: In 1986, Unit 2 at the Surry Power Station experienced a catastrophic failure of a main feedwater pipe. Subsequent investigation of the accident and examination of data by the licensee, NRC, and others led to the conclusion that the piping failure was caused by erosion/corrosion of the carbon steel pipe. This incident was the first time that such a failure occurred in a large-diameter system containing high-purity water in a nuclear power plant. An informal NRC staff survey conducted during the first week of February 1987 demonstrated that the wall-thinning problem is widespread in two-phase lines at nuclear power plants, and most licensees either did not have a monitoring program for pipe wall-thinning or had an inadequate program. As a result of this finding, and NRC Bulletin 87-01 was issued on July 9, 1987. This bulletin required all licensees to provide information to the NRC on their erosion/corrosion (E/C) experience and monitoring programs for single-phase and two-phase high-energy carbon steel piping systems. This paper presents an overview of the responses to the NRC bulletin.

Title: Application of an expert system in the leak-before-break analysis.

Author: Sturm,-D.; Jovanovic,-A.; Stoppler,-W. (Stuttgart Univ. (Germany, F.R.). Staatliche Materialpruefungsanstalt); Hassler,-M. (Stuttgart Univ. (Germany, F.R.))

Corp. Author: 15. MPA-seminar on safety and

Source: Stuttgart Univ. (Germany, F.R.). Staatliche Materialpruefungsanstalt. Safety and reliability of plant technology with special emphasis on long-term integrity of pressure components of nuclear power plants. Vol. 1 and 2. Vol. 1: Integrity of vessels and components, irradiation embrittlement, nondestructive testing. Vol. 2: Fatigue/creep processes, integrity of line-pipes, fracture mechanics. Sicherheit und Verfuegbarkeit in der Anlagentechnik mit dem Schwerpunkt 'Langzeitintegritaet der druckfuehrenden Bauteile von Kernkraftwerken'. Bd. 1 und 2. Bd. 1: Behaelter- und Komponenten-Integritaet, strahleninduzierte Versproedung, zerstoerungsfreie Pruefung. Bd. 2: Zeitstandverhalten/Kriechvorgaenge, Rohrleitungsverhalten, Bruchmechanik. 1989. 784 p. p. 10.1-10.19. Published in 2 separate volumes.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** German

Category: LBB justification **ID:** 208

Abstract: The leak before break expert system was developed as a practical tool based on knowledge engineering. The prototype compares the failure curves determined in experiments on 20 MnMoNi 55 pipes for pipes weakened by surface longitudinal or circumferential faults with the load and leak before break curves calculated with the aid of an engineering approximation process. The practical application of the system should make the support and improvement of the structural safety analysis and the prediction of the service life of components under pressure possible. (DG).

Title: Crack initiation and experimental determination of J in bending for elbows and pipes in austenitic steel.

Author: Jamet,-P.; Moulin,-D.D.; Toubol,-F.; Lebey,-J.; Acker,-D.

Corp. Author: Seminar on leak-before-break:

Source: Wilkowski,-G.M. (Battelle, Columbus, OH (USA)); Chao,-K.S. (eds.) (Tawian Power Co., Taipei (Taiwan)). Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research; Battelle, Columbus, OH (USA). Leak-Before-Break: Further developments in regulatory policies and supporting research. Feb 1990. 350 p. p. 101-126.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Test/analysis **ID:** 209

Abstract: The paper describes a cooperative French research effort. The experiments were performed on cracked straight pipes and elbows with circumferential cracks. The obtained results showed that for shorter circumferential through-wall cracked pipe and elbows in bending, the failure loads were below those predicted by the net-section-collapse equation, unless the flow stress of the material is lowered.

Title: Acceptance criteria for structural evaluation of erosion-corrosion thinning in carbon steel piping.

Author: Norris,-D.et. al

Corp. Author: Seminar on leak-before-break:

Source: Wilkowski,-G.M. (Battelle, Columbus, OH (USA)); Chao,-K.S. (eds.) (Tawian Power Co., Taipei (Taiwan)). Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research; Battelle, Columbus, OH (USA). Leak-Before-Break: Further developments in regulatory policies and supporting research. Feb 1990. 350 p. p. 43-60.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Criteria **ID:** 210

Abstract: The paper presents an acceptance criterion for structural evaluation of erosion-corrosion thinning in carbon steel piping. This criterion is currently being considered for implementation into Section XI of the ASME Boiler and Pressure Vessel Code. This evaluation method was developed as a result of a failure of the Surry Unit 2 reactor feedwater piping, and subsequent NDE evaluations showing wall thinning in several other Pressurized Water Reactor (PWR) feedwater piping systems. 11 refs., 6 figs., 2 tabs.

Title: Fracture mechanical assessment of pipes under quasistatic and cyclic loading.

Author: Roos,-E.; Diem,-H.; Herter,-K.H.; Stumpfrock,-L. (Stuttgart Univ. (Germany, F.R.). Staatliche Materialpruefungsanstalt) **Corp. Author:**

Source: Steel-Research. (1990). v. 61(4) p. 181-187.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Methods **ID:** 211

Abstract: Methods will be outlined which allow the calculation of the load-bearing capacity of circumferentially cracked pipes. The reliability of the calculating procedures are checked by means of appropriate tests with pipes (DN 400) under internal pressure and with a superimposed bending moment. The loading conditions may be static as well as cyclic. The cyclic crack growth experiments were performed at the HDR test facility. Under quasistatic loading conditions the failure behaviour of pipes with through-wall cracks were calculated on the safe side. The cyclic experiments showed the decisive influence of the environmental conditions on the crack growth rate. (orig.).

Title: Primary Coolant Leak at Kola-2 NPP Due to Rupture of a Make-up Pipe

Author: **Corp. Author:** IAEA, Vienna (Austria)

Source: WWER-SC-112 Draft Report of a Consultants Meeting, Nov. 28 - Dec. 2, 1994

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** English

Category: Operating experience **ID:** 212

Abstract: The IAEA recently proposed, in the framework of the IRS activity, to have performed an in-depth analyses of a single selected event by international experts. In-dept study on "Primary system coolat leak event an NPP Kola-2" was conducted from 28 November to 2 December at the Agency's Headquarters. The specific objectives of this IRS meeting were (1) to discuss in detail information on the "Kola event", provided by the Russian experts; (2) to evaluate actions to prevent recurrence of similar events; and (3) to draw generic lessons for improving WWER safety.

Title: Principles of operation of CANDU multi-unit containment systems.

Author: Blahnik,-C.; McKean,-D.W.; Meneley,-D.A.; Skears,-J.; Yousef,-N. (Ontario Hydro, Toronto, ON (Canada)) **Corp. Author:** International conference on con

Source: Black,-R.K. (ed.). Canadian Nuclear Society, Toronto, ON (Canada). Proceedings of the Canadian Nuclear Society international conference on containment design. 1984. 227 p. p. 158-165.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: Analysis of break effects **ID:** 213

Abstract: Analysis has shown that the 'negative pressure' containment (NPC) concept is flexible and efficient in meeting challenges presented by a spectrum of initiating events ranging from operational upsets to failures of the largest piping in the heat transport system. Containment envelope ventilation is isolated promptly and the integrated overpressure period is minimal. A substantial holdup period is provided to remove the bulk of radiologically important fission products from the containment atmosphere. If and when controlled venting becomes necessary to avoid long term leakage, radiological consequences are minimized by treating the effluent stream and by providing the flexibility to interrupt the release during unfavourable weather conditions. The NPC concept incorporates "compartment venting" and "filtered atmospheric venting". These were recognized separately as the most effective means of reducing major accident consequences following LWR meltdown sequences. CANDU does not have a credible meltdown sequence; however, pressure and effluent control are important risk-reducing function in many accident sequences. This is particularly important when it is recognized that containment envelopes are not absolute containers, and so might be impaired at the time of an accident. The dual-failure requirements of the AECB Siting Guide properly recognize this reality. The NPC containment concept is well equipped to respond to the full range of accident sequences important to public safety. The actual consequences of major accidents are expected to be much lower than those calculated for the use in reactor licensing. The eventual objective is to bring nuclear safety equipment expenditures down, closer to a reasonable balance relative to other societal risk-mitigation expenditures.

Title: Prevention of catastrophic failure in pressure vessels and pipings.

Author: Rintamaa,-R.; Wallin,-K.; Ikonen,-K.; Toerroenen,-K.; Talja,-H.; Keinaenen,-H.; Saarenheimo,-A.; Nilsson,-F.; Sarkimo,-M.; Waestberg,-S.; Debel,-C. **Corp. Author:** Nordisk Kontaktorgan for Ato

Source: Nov 1989. 49 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Test/analysis **ID:** 214

Abstract: The fracture resistance and integrity of pressure-loaded components have been assessed in a Nordic research programme. Experiments were performed to validate the computational fracture assessment analysis. Two tests were also conducted on a large decommissioned pressure vessel from an oil refinery plant. Different fracture assessment methods were developed and subsequently applied to the tested components. Interlaboratory round robin programmes with the participation of several laboratories were arranged to examine elastic-plastic finit element calculations and fracture mechanics testing. The transferability of material parameters derived from small specimens with simple crack geometries to more realistic crack geometries in real components has been verified. (author).

Title: Some aspects of thermal fatigue in stainless steel.

Author: Iorio,-A.F.; Crespi,-J.C. (Comision Nacional de Energia Atomica, Buenos Aires (Argentina). Dept. de Materiales) **Corp. Author:** 3. Latin American colloquium

Source: Comision Nacional de Energia Atomica, Buenos Aires (Argentina). Dept. de Materiales. Third Latin American colloquium on technological developments in failure analysis in Buenos Aires, 19-23 October 1987. Tercer coloquio latinoamericano desarrollos tecnologicos en analisis de fallas en Buenos Aires, 19-23 de octubre de 1987. 1987. 152 p. p. 89-101.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Research/theoretical **ID:** 215

Abstract: This paper is concerned with the analysis of failures in a moderator circuit branch piping of the ATUCHA-I PHWR, made of austenitic steel to DIN 1.4550 specification (similar to AISI 347). These failures are considered to result from a thermal fatigue processes induced by fluctuations in a zone where stratified temperature layers occurred; the fluctuations being associated with variations in the heavy water flow. The first section evaluates the possibility of cracking due to thermal fatigue phenomena and concludes that under service conditions a crack may be initiated and growth through 7 mm of the wall thickness of the pipe. Laboratory thermal fatigue tests that simulated the thermo-mechanical conditions for such a component, showed that the number of cycles required to initiate a thermal fatigue crack in a notched modified standard fatigue specimen was about 10 sup 3. This value may be used to give a conservative prediction of the number of thermal cycles for crack initiation in actual station piping, including those who suffered a cold plug condition which is produced in some emergency shut-down and valve testing situations. It was also demonstrated that beyond a crack depth of 7 mm stress corrosion cracking has the main process in further crack propagation. The relevance of this prediction has been confirmed by microfractographic observations, since the brittle nature of the fracture surfaces under service conditions appears very different from the transgranular ductile striations found in both thermal and mechanical fatigue test specimens as a result of environmental effects. (Author).

Title: Statistical analysis of component failure reports of nuclear power plants.

Author: Kondo,-S. (Tokyo Univ. (Japan). Faculty of Engineering); Harima,-M. (Nuclear Power Safety Information Research Center, Tokyo (Japan). General Safety Center/Nuclear Power Engineering Test Center)

Corp. Author: International symposium on fee

Source: International Atomic Energy Agency, Vienna (Austria); Nuclear Energy Agency, Paris (France). Feedback of operational safety experience from nuclear power plants. Proceedings of an international symposium held in Paris, 16-20 May 1988. Vienna (Austria). IAEA. 1989. 695 p. p. 315-329.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Experience/events **ID:** 216

Abstract: Using the database composed of incident reports to the Government, (1) trends in piping system failure rates, and their causes, and (2) the availability of engineered safety system (ESS) functions at Japanese nuclear power plants (NPPs) have been studied with a view to assessing the present status of their safety and reliability. The study of piping system failures has revealed that the major causes of failures are fatigue, improper work and stress corrosion cracking and that effective countermeasures have been steadily implemented. The unavailability of the ESS function has been estimated using the reports of the detection of the inoperability of the ESS train during periodic tests. The study has indicated that the compulsory annual maintenance of the ESS, as practised in Japan, is quite effective in keeping its level of availability sufficiently high. These two studies have indicated that the incident reports to the Government have been effectively used for the validation of the safety and reliability of NPP operations. (author). 1 ref., 12 figs, 2 tabs.

Title: Piping and fitting dynamic reliability program.

Author: Guzy,-D.; Tagart,-S.; Tang,-Y.K.; English,-W.; Hwang,-H.; Merz,-K.; DeVita,-V.

Corp. Author: 16. water reactor safety inform

Source: Weiss,-A.J. (comp.). Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research; Brookhaven National Lab., Upton, NY (USA). Sixteenth water reactor safety information meeting. Proceedings: Volume 3, Nuclear plant aging, structural and seismic engineering, mechanical research, environmental effects in primary systems. Mar 1989. 562 p. p. 247-263.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Experience/events **ID:** 217

Abstract: In recent years, both industry and NRC have been concerned about the appropriateness of piping design rules for seismic and other dynamic loads. While experimental failure data was used to justify the ASME Code's piping stress criteria for static and fatigue loads, there was little available physical evidence of piping dynamic failure behavior when the current rules were written. The NRC Piping Review Committee recognized the need to obtain such data and recommended that the NRC support a test program in this area. This resulted in the NRC's cooperation with EPRI in the Piping and Fitting Dynamic Reliability Program (PFDRP). The PFDRP was initiated in 1985 with three main objectives: (1) to identify the failure mechanisms and failure levels of piping components and systems under dynamic loadings; (2) to provide a data base that will improve our prediction of piping system response and failure due to high level dynamic loads, and (3) to develop an improved and defensible set of piping design rules for inclusion into the ASME Code. All the experimental tasks of the PFDRP have been performed Forty-one piping components failure tests were completed by ANCO Engineers. Two piping system were ruptured by high seismic-like loads at ETEC, and one of these systems was retested. The Materials Characterization Laboratory has finished testing over 140 fatigue retchetting specimens. Also, piping system waterhammer tests have been performed by ANCO Engineers. General Electric of San Jose, the prime contractor for the PFDRP, has completed most of the data reduction and analysis associated with these tests. Recommendations for improved piping rules for dynamic loads have been under developed by General Electric and should be proposed formally to the ASME in the spring of 1989. The final reports for the PFDRP will be published by EPRI in 1989. 6 figs., 2 tabs.

Title: Evaluation and analysis of documents with a view to safety-relevant problems, and consideration of these problems in t

Author: Herter,-K.H. **Corp. Author:**

Source: Dec 1987. 225 p.Bundesministerium fuer Umwelt-, Naturschutz und Reaktorsicherheit, Bonn (Germany, F.R.).Stuttgart Univ. (Germany, F.R.). Staatliche Materialpruefungsanstalt.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** German

Category: Research/theoretical **ID:** 218

Abstract: The paper outlines the present status of the calculation of failure stress and behavior of pipes and containers with longitudinal and circumferential defects under internal pressure load and/or external bending momentum load. The experimental data of the research program of the Federal Ministry of Research and Technology on 'Phenomenological container bursting tests' phase 2 as well as data of tests carried out by Interatom were used for the comparison performed. The pipes used for these tests showed dimensions similar to those of the main coolant line of pressurized water reactors (PWR). The mathematical values were determined by the plastic critical load concepts as well as concepts of critical tension, since these calculation methods are, on the one hand, used for safety analyses and on the other hand included in the American set of rules as a criterion for the assessment of defects. (orig./DG).

Title: The criterion for the unstable failure of cracked stainless steel piping subject to a combination of applied loadings.

Author: Smith,-E. (Manchester Univ. (UK). Inst. of Science and Technology) **Corp. Author:**

Source: Engineering-Fracture-Mechanics. (1989). v. 34(5-6) p. 1139-1144.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Research/theoretical **ID:** 219

Abstract: For extreme accident conditions, the applied loadings on a cracked piping system are complex and can be a combination of loads and displacements applied to various parts of the system. In investigating the problem of crack instability for such conditions, this paper analyses the model where a straight pipe, containing a circumferential through-wall crack at its mid-length position, is subject to bending deformation as a result of rotations applied at its built-in ends through rotational springs and a transverse load applied at an intermediate position along the pipe, again through an appropriate spring system. It is thereby possible to examine the effect of a wide range of loading combinations on the crack instability criterion, as derived using the tearing modulus methodology. One important conclusion is the underscoring of the view that the loading characteristics at the pipe-ends have a very important effect on crack instability. (author).

Title: Effect of cyclic frequency on the fatigue life of ASME SA-106-B piping steel in PWR environments.

Author: Terrell,-J.B. (Materials Engineering Associates, Inc., Lanham, MD (USA)) **Corp. Author:**

Source: Journal-of-Materials-Engineering. (1988). v. 10(3) p. 193-204.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Research/theoretical **ID:** 220

Abstract: The author describes fatigue life tests in pressurized water reactor (PWR) environments performed on smooth and sharply notched specimens of ASME SA-106-B piping steel at cyclic frequencies of 1.0 Hz, 0.1 Hz, and 0.017 Hz. On the basis of these tests, it was concluded that no effect of cyclic frequency existed for smooth specimens whereas a frequency of 0.017 Hz proved to have the most detrimental effect on the cyclic life of the notched specimens. However, a reduction in fatigue strength in the low cycle fatigue regime and a fatigue strength enhancement in the high cycle regime was observed in both 288 sup 0 C (550 sup 0 F) air environment tests and PWR environment tests. This is believed to be due to dynamic strain aging processes. As a result, the current ASME Section III design curve for carbon steels is nonconservative in its positioning, which may decrease the presumed safety factor against fatigue failures in carbon steel piping components having structural discontinuities.

Title: Application of the R6-Rev. 3 approach to ductile fracture analysis of carbon steel pipe with a circumferential through-

Author: Asano,-Masayuki; Fukakura,-Juichi; Kashiwaya,-Hideo (Toshiba Corp., Yokohama (Japan). Heavy Apparatus Engineering Lab.); Saito,-Masahiro

Corp. Author:

Source: Nippon-Kikai-Gakkai-Ronbunshu,-A-Hen. (Nov 1989). v. 55(519) p. 2299-2306.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** Japanese

Category: Test/analysis **ID:** 221

Abstract: This study was performed to assess the validity of the R6-Rev. 3 approach to predict fracture behavior of carbon steel pipes for LWR plants. To verify the approach, a maximum load, predicted by analysis, was compared with an experimental load, obtained at JAERI. Analysis and experimentation were conducted on a STS 42 pipe(6B,sch. 80) with a circumferential through-wall crack (2 theta=180degC). The comparison of the results indicates that the R6-Rev. 3 approach gives conservative maximum load prediction with reasonable accuracy. In the next step, failure assessment curve (FAC) was discussed briefly, and sensitivity analysis was carried out to clarify the effects of initial crack length, pipe size, and toughness of the material on fracture load and the possibility of occurrence of net-section collapse. Although unstable fracture was predicted to occur by a mode other than net-section collapse in all analyses, fracture load was able to be evaluated by simple limit load analysis based on yield stress, so long as a proper margin was considered. (author).

Title: Rupture of a high pressure gas or steam pipe in a tunnel: A preliminary investigation of the jet thrust exerted on a tunn

Author: Baum,-M.R. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.)

Corp. Author:

Source: Nuclear-Engineering-and-Design. (Dec 1989). v. 117(3) p. 235-249.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Analysis of break effects **ID:** 222

Abstract: On power plant, if a high pressure pipe containing high temperature gas or steam were to rupture, sensitive equipment necessary for safety shutdown of the plant could possibly be incapacitated if exposed to the subsequent high temperature environment. In many plant configurations the high pressure pipework is contained in tunnels where it is possible to construct barriers which isolate one section of the plant from another, thereby restricting the spread of the high temperature fluid/air mixture. This paper describes a preliminary experimental investigation of the magnitude of the thrust likely to be exerted on such barriers by a gas jet issuing from the failed pipe. (orig.).

Title: The development of a validated leak-before-break methodology for application to fast reactor sodium boundary compo

Author: Tomkins,-B. (United Kingdom Atomic Energy Authority, Northern Research Labs., Risley, Warrington, Cheshire (UK))

Corp. Author: Fracture mechanics, creep and f

Source: Becht-IV,-C. (Becht Engineering Co., Inc., Liberty Corner, NJ (USA)); Bhandari,-S.K. (Framatome, Paris (France)); Tomkins,-B. (Northern Research Labs., Risley (UK)). Fracture mechanics, creep and fatigue analysis. New York, NY (USA). American Society of Mechanical Engineers. 1988. 86 p. p. 83-87.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: LBB justification **ID:** 223

Abstract: A major task in the European Fast Reactor Program in the Structural Integrity area is the establishment of the leak-before-break concept for application to sodium boundary components. Work is in hand within Collaborative R and D Program to develop the methodology for application to components, including secondary and primary circuit pipework and the primary vessel. All these are austenitic components with considerable resistance to tearing although the different scale of the components considered leads to some differences in approach. For example, for pipework, the methodology is developed from consideration of sub-critical growth of initial flaws to ligament failure whilst for the large primary vessel, the different stress, inspection and critical crack size circumstances dictate an approach to acceptable margins based on critical crack size considerations. The paper integrates the currently developed views from the four member countries and identifies the route for common methodology applicable to the range of components. The paper also includes some indication of the required connections to other technical area developments viz NDT, leak detection, materials properties along with structural tests and analysis to develop a validated and applicable methodology.

Title: Stable crack growth in large austenitic pipes under bending.

Author: Gruter,-L. (Interatom GmbH, Bergisch Gladbach (Germany, F.R.)); Debaene,-J.P. (Novatome, Lyon (France)); Faidy,-C. (Electricite de France, Villeurbanne (France))

Corp. Author: Fracture mechanics, creep and f

Source: Becht-IV,-C. (Becht Engineering Co., Inc., Liberty Corner, NJ (USA)); Bhandari,-S.K. (Framatome, Paris (France)); Tomkins,-B. (Northern Research Labs., Risley (UK)). Fracture mechanics, creep and fatigue analysis. New York, NY (USA). American Society of Mechanical Engineers. 1988. 86 p. p. 65-70.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Test/analysis **ID:** 224

Abstract: Results of current investigations on the evaluation of circumferential cracks in piping structures are presented, including bending experiments on straight pipes DN700 with a high ratio of pipe radius to wall-thickness made of austenitic stainless steel 316L, small specimen testing and analytical work. For crack extensions up to 600 mm, crack resistance curves are shown; parameters such as delta, COA/CTOA, J and J sub M are discussed. Failure of the present pipes is not necessarily due to plastic collapse. The screening criterion introduced by Battelle seems to be a useful approach. The extrapolation of the J.R-curves from small specimens to a full-size structures is reliable for crack initiation, but needs further work for stable crack growth. The classical engineering methods can be used for evaluation of the present pipes only if the calculation models are adapted to the given material-geometry-conditions and the relevant type of failure is considered.

Title: Engineered safety features against LOCA for the 'Democritos' reactor.

Author: Chrysochoides,-N.G. (Athens Univ. of Agricultural Sciences, Athens (Greece). Physics Lab.); Anoussis,-J.N.; Mitsonias,-C.A.; Papastergiou,-C.N. (National Research Centre for the Physical Sciences Democritos, Athens (Greece))

Corp. Author:

Source: International Atomic Energy Agency, Vienna (Austria). Research reactor core conversion guidebook. V.2: Analysis (Appendices A-F). Apr 1992. 386 p. p. 125-129.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Other **ID:** 225

Abstract: "Democritos" is a 5 MW swimming-pool type reactor. One safety concern of this type of reactor is a LOCA due to rupture either of a pipe of the primary cooling system or of an experimental beam tube. Existing engineered safety features against LOCA are described along with further solutions that are being considered. (author). 4 figs.

Title: The calculating analysis of fluid transients caused by LOCA event in primary loop.

Author: He-Feng; Wang-Xuefang; Ye-Hongkai (Qinghua Univ., Beijing, BJ (China))

Corp. Author: 11. international conference on

Source: Shibata,-Heki (ed.) (Tokyo Univ. (Japan). Inst. of Industrial Science). Atomic Energy Society of Japan, Tokyo (Japan). Transactions of the 11th international conference on structural mechanics in reactor technology. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. SD1-SD2 p. 457-462. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3 for 17 vols. bound in 15 (A through SD2).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Analysis of break effects **ID:** 226

Abstract: This paper deals with the rupture of pipe in primary loop in PWR. The ruptures involve the double-end of break, which results in that coolant jets simultaneously from the double end of break pipes, and the break opening of pipe which is a hole on the wall of pipe. The main loop links up with many large diameter pipes, while auxiliary systems linking up with main loop have thin piping system. A pipe of auxiliary system has rupture, there is still time to do something with emergency. When a main pipe breaks core may be uncovered because of a great deal of loss of coolant, the temperature of core rises so that the core may fuse. How about the flow of coolant ? Analysis is shown in this paper from a viewpoint of fluid mechanics. The research of fluid transients in LOCA event help us to know the flow condition of coolant and acquaint with the fluid force acting on pressure vessel, steam generator, support of pipes etc. (author).

Title: PWR type reactor.

Author: Abe,-Nobuaki **Corp. Author:** Toshiba Corp., Kawasaki, Kan

Source: 10 Jan 1992; 24 Apr 1990. 5 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Other **ID:** 227

Abstract: The reactor of the present invention suppresses a pressure difference between a cold leg pipeline and a hot leg pipeline upon occurrence of a small rupture of the cold leg pipeline, to prevent lowering of a reactor core water level. That is, a connection pipeline is disposed for connecting the cold leg pipeline and a hot leg pipeline. A valve is intervened to the connection pipeline. Then, a controller is disposed for opening the valve when the pressure in the hot leg pipeline is increased higher than that of the cold leg pipeline. With such a constitution, when a small rupture is caused to the cold leg pipeline, occurrence of the pressure difference between the hot leg pipeline and the cold leg pipeline can be prevented. Further, the lowering of the water level of the reactor core can be prevented. As a result, effective cooling for the reactor core can be ensured. (I.S.).

Title: Overview of piping reliability test program at the Japan Atomic Energy Research Institute.

Author: Isozaki,-Toshikuni; Shibata,-Katsuyuki; Suzuki,-Saburo; Ueda,-Shuzo; Kurihara,-Ryoichi **Corp. Author:**

Source: Shibata,-Heki (Ed.). Trans. 11th SMiRT Conference. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. M-SD0 p. 401-412. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 228

Abstract: This paper summarizes the piping reliability test program conducted at JAERI from 1975 to 1990. The tests have been performed to prove the integrity of the light water reactor piping, no possibility of unstable fracture during the service period of the plants and to prove the effectiveness of the protective devices such as jet shields or restraints against postulated pipe rupture events. (author).

Title: Qualification by analogy of the functional valving of French pressurized water nuclear power stations.

Author: Grenet,-M. (Electricite de France, 69 - Villeurbanne (France). Service Etudes et Projects Thermiques et Nucleaires) **Corp. Author:** 1. JSME/ASME joint internatio

Source: Japan Society of Mechanical Engineers, Tokyo (Japan). The 1st JSME/ASME joint international conference on nuclear engineering. Tokyo (Japan). Japan Society of Mechanical Engineers. 1991. 1273 p. v. 2 p. 505-509. Composed of two volumes.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Analysis of break effects **ID:** 229

Abstract: In certain postulated accidental conditions (loss of coolant accident or secondary pipe rupture, earthquake, high energy pipe rupture) plant valving is called on the important functions to bring the reactor to and maintain it at a safe shutdown condition. EDF has completed qualification tests of about forty valves to assure their operability. However, taking into account the costs and time required to obtain this qualification and the number of valves to be qualified, this method alone is not sufficient. For this reason, Electricite de France has developed the alternative qualification methodology by analogy for each postulated accidental situation. Feedback experience of these methods today is such that it can be they have achieved their objective; namely, to improve the safety of French pressurized water nuclear power stations, while at the same time avoiding the two dangers represented by excessive complexity resulting in unsatisfactory operation, and insufficient thoroughness not providing any real increase in safety. (author).

Title: Preliminary leak-before-break evaluation procedures for DOE's new production reactor-heavy water reactor.

Author: Gwaltney,-R.C. (Oak Ridge National Lab., TN (United States)) **Corp. Author:** 1991 American Society of Mec

Source: Sammataro,-R.F. (General Dynamics Corp. (United States)). Codes and standards and applications for design and analysis of pressure vessel and piping components 1991. PVP-Volume 2. New York, NY (United States). American Society of Mechanical Engineers. 1991. 206 p. p. 99-106.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: LBB justification **ID:** 230

Abstract: This paper discusses a comprehensive set of guidelines for the application of the leak-before-break (LBB) approach to new heavy water production reactors. The application of the LBB concept to the design of the new Production Reactor-Heavy Water Reactor (NPR-HWR) is not only to exempt the piping systems from the dynamic effects of pipe ruptures and the elimination of piping restraints as a design requirement but will also allow design of a more flexible (or less stiff) piping system. Such a system will accommodate the seismic design events much better than a stiff piping system and will also allow design of a more flexible system in that component replaceability can be a viable alternative. The LBB procedures for piping were extended to other components in the reactor system and these include pumps, valves, flange joints, curved sections of pipe, branch connections, pipe fittings, heat exchanger tubes, attachments, and vessels such as the reactor vessel and heat exchangers. These preliminary guidelines are based on the light water reactors to the NPR-HWR. These guidelines were extended to other components based on experience at Savannah River Laboratory in extending LBB to other components in the present reactors.

Title: Thermohydraulic behavior of the coolant in the initial phase of a loss-of-coolant accident.

Author: Suchanek,-M.; Bartak,-J. (National Research Inst. for Machine Design, 190 11 Prague (Czechoslovakia)) **Corp. Author:** 2. international symposium on

Source: Chen,-X.J. (Engineering Thermophysics Research Inst., Xi'an Jiaotong Univ., Xi'an, Shaanxi Province (China)); Veziroglu,-T.N. (Miami Univ., Coral Gables, FL (United States). Clean Energy Research Inst.); Tien,-C.L. (California Univ., Berkeley, CA (United States)). Proceedings of the second international symposium on multiphase flow and heat transfer. Volume 1 and 2. New York, NY (United States). Hemisphere Publishing. 1991. 1490 p. p. 929-938.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Analysis of break effects **ID:** 231

Abstract: Thermohydraulic phenomena related to the issue of nuclear reactor safety have been focusing attention of researchers and engineers for many years. Nevertheless, physical phenomena occurring after a large-break or small-break loss-of-coolant accident (LOCA) are still poorly understood. In the first instants after the primary circuit pipe rupture there is a rapid depressurization of the system followed by explosive vapor generation in the superheated coolant and by the discharge of the two-phase mixture. The propagating depressurization wave imposes severe loads on the internal structures of the reactor. The paper summarizes the results of experimental investigations carried out at the National Research Institute for Machine Design during the past few years and concentrates on three important issues: critical two-phase flow; depressurization wave propagation and vapor generation in super heated coolant; interaction of the depressurization wave with the internal structures of the reactor.

Title: Analysis of the loss of coolant accident 'double-ended guillotine break of one of the two surgelines' in the reactor plant

Author: Buntzen,-F.; Hrubisko,-M. **Corp. Author:** Gesellschaft fuer Reaktorsicher

Source: 1991. 120 p. Bundesministerium fuer Umwelt, Naturschutz und Reaktorsicherheit, Bonn (Germany)

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** German

Category: Analysis of break effects **ID:** 232

Abstract: Aim of the analysis was the investigation of the accident sequence during the rupture of the largest connection pipe in the primary system under best-estimate assumption for the emergency core cooling system (ECCS). It was found as a major result of the analysis that no core uncovery took place during the blowdown phase. The analysis was terminated when the leak mass flow rate was exceeded by the injected mass flow rate from the ECCS at about 6 bar, because no detoriation of the core cooling conditions had to be expected for the further accident sequence (refill of the primary system). It has been demonstrated with this analysis that the reactor plant possesses safety margins for beyond-design accidents. (orig./HP).

Title: Pipeline rupture detection device of after-heat removing facility.

Author: Yamamoto,-Yuji **Corp. Author:** Toshiba Corp., Kawasaki, Kan

Source: 27 Dec 1991; 17 Apr 1990. 3 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Inspection methods **ID:** 233

Abstract: The present invention concerns a pipeline rupture detection device which does not limit the size of a reactor container. A flow nozzle and an isolation value are disposed to a connection portion between a pressure vessel and suction pipeline introduced from the pressure vessel to the after-heat removing facility. If the pressure difference between the throat portion of the flow nozzle and the pressure vessel, the isolation value is controlled by the output. With such a constitution, since two elbow meters and straight pipelines disposed before and the back thereof are not necessary, the space occupying the reactor container is reduced, thereby enabling to minimze the device. (N.H.).

Title: New stresses for 1 and 11/4 Cr-Mo-Si alloys.

Author: Prager,-M. (Materials Properties Council, Inc., New York, NY (United States)); Gold,-M. (Babcock and Wilcox Co., Barberton, OH (United States)); Voorhees,-H.R. (Materials Technology Corp., Ann Arbor, MI (United States)) **Corp. Author:** 1990 pressure vessels and pipin

Source: Prager,-M.; Cantzler,-C. (Materials Properties Council, Inc., New York, NY (United States)). New alloys for pressure vessels and piping. PVP-Volume 201; MPC-Volume 31. New York, NY (United States). American Society of Mechanical Engineers. 1990. 203 p. p. 115-140.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Methods **ID:** 234

Abstract: This paper reports on the Subgroup on Strength-Ferrous Alloys (SG-SFA) of ASME which began a review of the allowable stresses for the Cr-Mo alloys following a request to reconsider the values for 1Cr-1/2Mo (T/P 12) and 1 1/4Cr-1/2Mo-Si (T/P 11) alloys [to take account of the change in the yield strength criterion of Section I, from 5/8 to 2/3]. While this effort was underway, the Mojave steam line rupture occurred in a P11 pipe, and the SG-SFA decided to broaden its concern to include a thorough review of the time-dependent properties of these similar grades. A Task Group of the Subgroup was appointed. With the subsequent, similar failure at the Monroe power plant, the activity was further broadened to review the values of 2 1/4Cr-1Mo alloy product forms. The latter activity continues at this date.

Title: Application of fail-safe structural design to piping system.

Author: Ibe,-Hidetoshi; Nakatogawa,-Tetsundo (Mitsubishi Atomic Power Industries, Inc., Tokyo (Japan)); Hisada,-Toshiaki; Noguchi,-Hirohisa; Murayama,-Osamu; Der-Kiureghian,-A. **Corp. Author:** 11. international conference on

Source: Shibata,-Heki (ed.) (Tokyo Univ. (Japan). Inst. of Industrial Science). Atomic Energy Society of Japan, Tokyo (Japan). Transactions of the 11th international conference on structural mechanics in reactor technology. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. M-SD0 p. 265-270. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3 for 17 vols. bound in 15 (A through SD2).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Analysis of break effects **ID:** 235

Abstract: Safety devices such as safety relief valves and rupture disks are sometimes installed in pipelines to make sure of preventing excessive pressure. If the structural members in a plant are designed with sufficient margin under the specified seismic design conditions, some difficulty is felt in assessing the safety margin exactly. In response to this situaiton, the fail-safe concept has been introduced, in which if a structure should fail, its overall safety is maintained because the failure is limited to an unimportant portion of the structure. This concept makes better use of the yielding or even breakage of structural members, and implies the overall improvement of the reliability and the cost of structures. The recognition and control of failure mode are very important when the fail-safe design concept is applied to a structure. As the first attempt, its application to the design of the aseismatic supports for piping systems is described. The sensitivity analysis on a typical piping system was performed using a time history of seismic acceleration by dynamic nonlinear FEM. The case study is reported. (K.I.).

Title: Studies on diffusion and natural convection of the two component gases.

Author: Takeda,-T.; Hishida,-M. (Heat Transfer Lab., High Temperature Engineering Div., Japan Atomic Energy Research Inst., Tokai-mura, Naka-gun, Ibaraki-ken 319-11 (Japan)) **Corp. Author:** American Nuclear Society (AN

Source: Anon.-The safety, status and future of non-commercial reactors and irradiation facilities. La Grange Park, IL (United States). American Nuclear Society. 1990. 830 p. p. 296-303.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 236

Abstract: This paper reports on a primary pipe rupture accident, one of the design-base accidents of a High Temperature Engineering Test Reactor (HTTR), which is being developed at JAERI. When the primary pipe ruptures, air is expected to enter into the reactor core from the breach by molecular diffusion and natural convection. In order to investigate the air ingress process during the early stage of the primary pipe rupture accident, experiment and analytical studies are performed on the conjugate phenomenon of the transient molecular diffusion and natural convection of two component gas mixtures in two test sections, a reverse U-shape tube and a test model simulating simply the reactor. One-dimensional basic equations for continuity and momentum conservation are numerically solved to obtain the concentration change of gas species in the reverse U-shape tube.

Title: Thermal transient analyses during a depressurization accident in the High Temperature Engineering Test Reactor (HT

Author: Kunitomi,-Kazuhiko; Nakagawa,-Shigeaki; Itakura,-Hirohumi (Japan Atomic Energy Research Inst., Oarai, Ibaraki (Japan). Oarai Research Establishment) **Corp. Author:** Japan Atomic Energy Research

Source: Oct 1991. 94 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Analysis of break effects **ID:** 237

Abstract: The behavior of the HTTR (High Temperature Engineering Test Reactor) during a depressurization accident which is caused by a primary pipe rupture was analyzed in a safety analysis. This paper describes analytical model, analytical condition and analytical results during the depressurization accident. The analytical results proved that thermal transient behavior during the depressurization accident is slower than that of the Light Water Reactor (LWR). It also proved that the maximum fuel temperature does not exceed the initial temperature (1495degC), and the maximum pressure vessel temperature would remain below its limit of 550degC determined for assuring its integrity. (author).

Title: Experimental study on air ingress during a primary pipe rupture accident with a graphite reactor core simulator.

Author: Takeda,-Tetsuaki; Hishida,-Makoto; Baba,-Shinichi (Japan Atomic Energy Research Inst., Tokai, Ibaraki (Japan). Tokai Research Establishment) **Corp. Author:** Japan Atomic Energy Research

Source: Nov 1991. 25 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Analysis of break effects **ID:** 238

Abstract: When a primary coolant pipe of a HTGR ruptures, helium gas in the reactor core blows out into the container, and the primary cooling system reduces the pressure. After the pressures are balanced between the reactor and the container, air is expected to enter into the reactor core from the breach. It seems to be probable that the graphite structures is oxidized by air. Hence, it is necessary to investigate the air ingress process and the behavior of the generating gases by the oxidation reactions. The previous experimental study is performed on the molecular diffusion and natural convection of the two component gas mixtures using a test model simulating simply the reactor. Objective of the study was to investigate the air ingress process during the early stage of the primary pipe rupture accident. However, since the model did not have any kind of graphite components, the reaction between graphite and oxygen was not simulated. The present model includes the reactor core and the high temperature plenum simulators made of graphite. The major results obtained in the present study are summarized in the followings: (1) The air ingress process with graphite oxidation reaction is similar to that without the reaction qualitatively. (2) When the reactor core simulator is maintained at low temperatures (lower than 450degC), the initiation time of the natural circulation of air is almost equal to that of the natural circulation of nitrogen. On the other hand, when the temperature of the reactor core simulator is high (more than 500degC), the initiation time of the natural circulation of air is earlier than that of nitrogen. (3) When the temperature of the reactor core simulator is higher than 600degC, oxygen is almost dissipated by the graphite structures. When the temperature of the reactor core simulator is below 700degC, carbon dioxide mainly is generated by the oxidation reactions. (author).

Title: Study on heat transfer and fluid flow in the stand pipe rupture accident. Buoyancy driven exchange flow behavior thro

Author: Fumizawa,-Motoo; Hishida,-Makoto (Japan Atomic Energy Research Inst., Tokai, Ibaraki (Japan). Tokai Research Establishment) **Corp. Author:** Japan Atomic Energy Research

Source: Sep 1991. 35 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Analysis of break effects **ID:** 239

Abstract: This paper deals with an experimental investigation of the buoyancy driven exchange flow which takes place through a narrow cylindrical channel, during the stand pipe rupture accident in a high temperature gas-cooled reactor (HTGR). The velocity distribution through the cylindrical channel is measured by a laser Doppler velocimeter, in order to evaluate the air ingress flow rate. The experiments are performed under atmospheric pressure with nitrogen as a working fluid. Rayleigh number ranges from 1.3 x 10 sup 7 to 7.0 x 10 sup 7. The following conclusions were obtained: (1) The laser Doppler velocimeter was found a good method for the measurement of the velocity of the exchange flow. (2) When the temperature of the hemisphere and the bottom heated plate, which simulate the top cover of the reactor, was kept uniform, the volumetric exchange flow rate agreed well with Epstein's result. (3) The exchange flow through a narrow cylindrical channel fluctuated irregularly with time and space. (author).

Title: Development of VVER-91 concept layout: a Finnish view.

Author: Maekelae,-K. (Imatran Voima Oy, Vantaa (Finland)) **Corp. Author:**

Source: Nuclear-Europe-Worldscan. (1991). v. 11(11-12) p. 9-11.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Other **ID:** 240

Abstract: The Soviet developed (1970-1980) original VVER-1000 NPP concept was designed to cope with simultaneous large diameter pipe rupture, design-basis earthquake and station blackout and prestressed containment for full accident pressure. There was a need to further the VVER-1000 to meet all today's requirements. A new concept, named VVER-91, using only proven design features is led by Atomenergoprojekt. The VVER-91 concept layout of VVER-1000 is described in detail. 2 figs.

Title: Measures for avoiding steam generator heating tube rupture in PWRs.

Author: Krosch,-G. **Corp. Author:** Technische Hochschule Aache

Source: 7 May 1990. 148 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Methods/design **ID:** 241

Abstract: Wastage corrosion is a specific phenomenon in PWR steam generator systems that can lead to rupture of the heating tubes made of Incoloy 800. It induces a reduction of tube wall thickness from the outer tube surface, eventually leading to pipe rupture. The countermeasures such as plugging taken so far in practical operation need to be replaced by a general, design-basis approach. The dissertation reports materials development and testing work for this purpose. A titanium-base alloy is presented, its alloying constituents and the testing work are explained, and the resulting heat-resistant material is compared to Incoloy 800 by means of experiments. As an additional measure, a modification of the design of the steam generator bottom plate is suggested, in order to improve the flow conditions over the bottom plate area, which is expected to delay fouling or the corrosive attack of salts on the tube surface. (orig./MM).

Title: Typical strain rates of piping systems in nuclear power plants for dynamic load cases.

Author: Charalambus,-B.; Loreck,-R. (Siemens AG Unternehmensbereich KWU, Erlangen (Germany)) **Corp. Author:** 16. MPA-seminar: Safety und r

Source: Stuttgart Univ. (Germany). Staatliche Materialpruefungsanstalt. Safety and reliability of plant technology with special emphasis on nuclear technology. Vol. 1 and 2. Vol. 1: Fracture mechanics, fatigue/creep processes, nondestructive testing. - Vol. 2: Integrity of vessels and components, integrity of line-pipes, irradiation embrittlement, thermal loading. Sicherheit und Verfuegbarkeit in der Anlagentechnik mit dem Schwerpunkt 'Kerntechnik'. Bd. 1 und 2. Bd. 1: Bruchmechanik, Zeitstandverhalten/Kriechvorgaenge, zerstoerungsfreie Pruefung. - Bd. 2: Behaelter- und Komponentenintegritaet, Rohrleitungsverhalten, strahleninduzierte Versproedung, Waermewechsel- und Thermoschockbeanspruchung. 1990. 784 p. p. 5.1-5.19.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Methods/design **ID:** 242

Abstract: Strain rates have been determined for piping systems under conditions of specified normal operation and for postulated conditions. This has been done on the basis of experimental and numerical results, which both show that is not the type of intensity of loads, but their frequencies that determine the piping's behaviour. The practical design of piping systems is oriented towards load conditions not affected by strain rates, i.e. these must not be considered in pipe design for load cases such as seismic effects, aircraft crash, explosion shock waves, bursts, or pipe rupture. (orig./DG).

Title: Nuclear regulation.

Author: **Corp. Author:** General Accounting Office, Wa

Source: 1988. 40 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Experience/events **ID:** 243

Abstract: In December 1986, a pipe rupture at Virginia Electric and Power Company's Surry Unit 2 nuclear power plant injured eight workers; four subsequently died. In July 1987, widespread pipe deterioration was discovered at General Electric's Trojan plant in Oregon. These events raise questions about the long-term safety of pipe systems in nuclear power plants. The Nuclear Regulatory Commision has now required utilities to provide information on the extend of known pipe deterioration at each plant. As of January 1988, NRC staff identified 34 new and mature plants with erosion/corrosion damage. It expects to gather additional information and use it to determine whether specific regulatory action is needed. In addition, a utility industry group has developed a program to help companies detect and repair pipe damage.

Title: Operation of Finnish nuclear power plants. Quarterly report, 2nd quarter, 1990.

Author: Tossavainen,-K. (ed.) **Corp. Author:** Finnish Centre for Radiation an

Source: Dec 1990. 29 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Experience/events **ID:** 244

Abstract: During the second quarter of 1990 the Finnish nuclear plant units Loviisa 1 and 2 and TVO and II were in commercial operation for most of the time. The feedwater pipe rupture at Loviisa 1 and the resulting inspections and repairs at both Loviisa plant units brought about an outage the overall duration of which was 32 days. The annual maintenance outages of the TVO plant units were arranged during the report period and their combined duration was 31.5 days. Nuclear electricity accounted for 35.3% of the total Finnish electricity production during this quarter. The load factor average of the nuclear power plant units was 83.0%. Three events occurred during the report period which are classified as Level 1 on the International Nuclear Event Scale: feedwater pipe rupture at Loviisa 1, control rod withdrawal at TVO I in a test during an outage when the hydraulic scram system was rendered inoperable and erroneous fuel bundle transfers during control rod drives maintenance at TVO II. Other events during this quarter are classified as Level Zero (Below Scale) on the International Nuclear Event Scale. Occupational radiation doses and external releases of radioactivity were considerably below authorised limits. Only small amounts of nuclides originating in nuclear power plants were detected in samples taken in the vicinity of nuclear power plants.

Title: Thickness measurements of pipes submitted to erosion and corrosion problems in the steam, feedwater and condensate

Author: Goffin,-J.P. (TRACTEBEL, Brussels (Belgium)) **Corp. Author:** Specialists' meeting on corrosio

Source: International Atomic Energy Agency, Vienna (Austria). International Working Group on Reliability of Reactor Pressure Components. Corrosion and erosion aspects in pressure boundary components of light water reactors. Apr 1990. 93 p. p. 43-49.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Operating experience **ID:** 245

Abstract: Although an in-service inspection program monitoring the pipe thinning in the steam, feedwater and condensate systems of Doel 1 and 2 plants, a pipe rupture occurred in September 1987 on an expansion piece. This fact proved that the control frequency was not sufficient and a comprehensive program was decided to check all sensitive pipe sections during the next outages of the plants. The paper presents the methodology and organisation of the inspection. Some global results for the first systematic measurements campaign of July 1988 are also given. 5 figs, 1 tab.

Title: Technology development by the U.S. industry to resolve erosion-corrosion.

Author: Chexal,-B.; Dietrich,-N.; Horowitz,-J.; Layman,-W.; Randall,-G.; Shevde,-V. (Electric Power Research Inst., Palo Alto, CA (USA)) **Corp. Author:** Specialists' meeting on corrosio

Source: International Atomic Energy Agency, Vienna (Austria). International Working Group on Reliability of Reactor Pressure Components. Corrosion and erosion aspects in pressure boundary components of light water reactors. Apr 1990. 93 p. p. 14-18.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Methods **ID:** 246

Abstract: Erosion-corrosion is a flow-accelerated corrosion process that leads to wall thinning (metal loss) of steel piping exposed to flowing water or wet steam. The rate of metal loss depends on a complex interplay of several parameters. These parameters include water chemistry, material composition, and hydrodynamics. Erosion-corrosion of plant piping can lead to costly outages and repairs, and can raise concerns about plant reliability and safety. Pipe wall degradation rates as high as 1.5 mm/year have occurred, resulting in pipe ruptures at both fossil and nuclear plants. The Nuclear Management and Resource Council (NUMARC) and EPRI have developed inspection planning methods and tools to help utilities identify areas of piping that might undergo erosion-corrosion. These tools provide utilities with the ability to predict wall thinning and to assess various remedial options. This allows utilities to plan and perform inspections, and to correct problems found during inspection. The U.S. electric power industry has developed the knowledge and the tools needed to protect against erosion-corrosion, and utilities have implemented erosion-corrosion monitoring programs. This paper describes EPRI's technical developments that support the utilities in determining where to inspect for erosion-corrosion. 15 refs, 7 figs.

Title: Studies on the primary pipe rupture accident of a high-temperature gas cooled reactor.

Author: Hishida,-M.; Ogawa,-M.; Takeda,-T.; Fumizawa,-M. (Japan Atomic Energy Research Inst., Tokai, Ibaraki (Japan). Heat Transfer Lab.) **Corp. Author:** 4. international topical meeting

Source: Mueller,-U.; Rehme,-K.; Rust,-K. (eds.). Fourth international topical meeting on nuclear reactor thermal-hydraulics (NURETH-4). Proceedings. Vol. 1. Karlsruhe (Germany, F.R.). Braun. 1989. 745 p. p. 163-169.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Analysis of break effects **ID:** 247

Abstract: In order to investigate the air ingress process during the early stage of the primary pipe rupture accident, experimental and analytical studies are performed on the conjugate phenomenon of transient molecular diffusion and natural convection of two component gas mixtures. The studies are carried out in two test sections, a reverse U-shape tube and a test model simulating simply the reactor of HTTR, which is being developed in Japan. The calculation is in good agreement with the experiment on gas concentration change and the initiation time of ordinary natural convection of pure N sub 2 gas in the reverse U-shape tube. Mass transfer between a high temperature graphite tube and a stream of He-O sub 2 gas mixture is experimentally studied in order to investigate the corrosion phenomenon of graphite structures in a high temperature regime during the later stage of the accident. (orig.).

Title: The effect of accident conditions on the condition of fuel pins of the VVER reactor.

Author: Bibilashvili,-Yu.K.; Sokolov,-N.B.; Dranenko,-N.B.; Kulikova,-V.V. (All Union Scientific Research Inst. for Inorganic Materials, Moscow (USSR)) **Corp. Author:** AEA Technology, Windscale (

Source: 1990. 25 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Analysis of break effects **ID:** 248

Abstract: At the present time, there has been a sufficiently detailed study of the list of events for nuclear power stations with water-cooled reactors which lead to accidents. This list includes events leading to a change in reactivity, disturbance to the coolant flow rate, a loss of coolant from the core etc. One of the most dangerous design basis accidents is an accident where the primary event is an instantaneous rupture in a large-diameter pipe (equivalent diameter for the VVER-1000 is 850 mm). This accident, which has been given the name "design basis accident" ("DBA"), concerns the class of accidents with a loss of coolant from the core. Accidents with an uncompensated leak from the primary circuit also relate to this class. Investigations into the behaviour of fuel pins in accident conditions are one of the main tasks for general analysis of the safety of nuclear power stations. (author).

Title: Analysis of loss-of-coolant accident for MURR 30-MW power-upgrade project using RELAP5/MOD2.

Author: Wang,-J.L. **Corp. Author:** Missouri Univ., Columbia, MO

Source: Columbia, MO (USA). Univ. of Missouri. 1987. 177 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Analysis of break effects **ID:** 249

Abstract: This study is part of the preliminary safety analysis for the new power expansion project on the University of Missouri Research Reactor (MURR). The loss of coolant accident (LOCA), which is initiated by hypothetical pipe ruptures at the most adverse positions (V507 A ampersand B) in both the hot and cold legs of the primary coolant loop, is analyzed with the thermohydraulic transient code RELAP5/MOD2. A complete MURR facility model developed in this work can be used for other transient analysis on the MURR. Results show, based on the RELAP5/MOD2 code predictions, that for the present 10 MW, film boiling never occurs with the postulated LOCA. For higher operating powers up to 30 MW, the peak fuel temperature is far below its melting temperature under the postulated LOCA, although part of the core will have experienced severe thermal hydraulic changes for a short period of five seconds following the pipe breaks. A number of suggestions are made for the future power-upgrade work to improve the reactor's response to abnormal accidents. Also, recommendations are made for facility transient tests to benchmark the RELAP/MURR model and for those users who will use this model in the future.

Title: Isolation valve control device for nuclear power plant.

Author: Yukinori,-Shigeru **Corp. Author:** Toshiba Corp., Kawasaki, Kan

Source: 16 Feb 1990; 10 Aug 1988. 4 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Inspection methods **ID:** 250

Abstract: The present invention provides an isolation valve control device for detecting pipeline rupture accidents in a BWR type nuclear power plant at an early stage to close an isolation valve thereby reducing the amout of radioactivity released to the circumstance. That is, isolation valves are disposed in the pipeline for each of the systems in the nuclear power plant and flow ratemeters are disposed to at least two positions in each of the pipelines. If a meaningful difference is shown for the measured values by these flow ratemeters, the isolation valve is closed. In this way, if pipeline rupture such as leak before break (LBB) is caused to a portion of a system pipelines, the measured value from the flow ratemeters at the downstream of the pipeline is lowered. Accordingly, when a meaningful difference is formed between the value of the flow ratemeters at the upstream and the downstream, occurrence of pipe rutpture between both of the flow ratemeters can be detected. As a result, the isolation valves of the system can be closed. According to the present invention, it is possible to detect the pipeline rupture at an early stage irrespective of the kind of the systems, diameter of the pipelines and the magnitude of the ruptured area, and the isolation valve can be closed. (I.S.).

Title: Pressure loadings of VVER release mitigation structures from large break LOCAs.

Author: Sienicki,-J.J. (Argonne National Lab., Argonne, IL (USA)); Horak,-W.C. (Brookhaven National Lab., Upton, NY (USA))

Corp. Author: 10. international conference on

Source: Hadjian,-A.H. (Bechtel Power Corp., Los Angeles, CA (USA)). Transactions of the 10th international conference on structural mechanics in reactor technology. Volume J. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1989. 337 p. p. 319-324.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Analysis of break effects **ID:** *251*

Abstract: This paper calculates the time dependent pressure loadings inside the accident localization or containment structures of VVER (Water cooled, water moderated energy reactor) reactors. Immediately following the double-ended guillotine rupture of a primary coolant pipe. The pressures are compared with the results of calculations of the response of the structures to overpressure.

Title: Consequences of pipe ruptures in metal fueled, liquid metal cooled reactors.

Author: Dunn,-F.E.

Corp. Author: Argonne National Lab., IL (US

Source: [1990]. 12 p. .International topical meeting on fast reactor safety. Snowbird, UT (USA). 12-16 Aug 1990.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Analysis of break effects **ID:** *252*

Abstract: The capability to simulate pipe ruptures has recently been added to the SASSYS-1 LMR systems analysis code. Using this capability, the consequences of severe pipe ruptures in both loop-type and pool-type reactors using metal fuel were investigated. With metal fuel, if the control rods scram then either type of reactor can easily survive a complete double-ended break of a single pipe; although, as might be expected, the consequences are less severe for a pool-type reactor. A pool-type reactor can even survive a protected simultaneous breaking of all of its inlet pipes without boiling of the coolant or melting of the fuel or cladding. 2 refs., 16 figs., 1 tab.

Title: Experimental basing of TECH'-M mathematical model.

Author: Zajtsev,-S.-I.; Logvinov,-S.A.; Spasskov,-V.P.; Sokolov,-A.S.; Khripachev,-Yu.B.

Corp. Author: Thermal physics 84. Thermal a

Source: Sovet Ehkonomicheskoj Vzaimopomoshchi, Moscow (USSR). Postoyannaya Komissiya po Ispol'zovaniyu Atomnoj Ehnergii v Mirnykh Tselyakh. Thermal physics 84. Thermal aspects of WWER nuclear reactor safety. V. 3. Collection of papers from CMEA seminar. Teplofizika 84. Teplotekhnicheskaya bezopasnost' yadernykh reaktorov VVEhR. Tom 3. Sbornik dokladov seminara SEhV, Varna, NRB, oktyabr' 1984 g. 1985. 250 p. p. 28-48.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1984 **Language:** Russian

Category: Analysis of break effects **ID:** *253*

Abstract: Experimental study results of thermal-hydraulic processes under modeled conditions of an accident with the main circulation pipeline rupture are presented. The investigations have been carried out on a test facility structurally similar to the WWER primary coolant loop. A description of the reactor model with the core simulated by a seven-rod cluster of fuel-rod-imitators with indirect heating is given. The rod cluster geometry is similar to that of WWER-1000. The initial parameters of experiments corresponded to actual ones in the WWER-1000 primary loop. Experimental results have been obtained on thermal-hydraulic parameter variations in the test facility loop and fuel cladding temperature regime during accidental coolant discharge. The maximum cladding temperature rise has been fixed in the experiment with 1.25 MW/m**2 specific heat load at the cluster center part. The cladding temperature level under these conditions did not exceed 900 C. Short description of TECH'-M mathematic model modified with respect to the test facility environment is given. The experimental results are compared with the calculated ones.

Title: Coolant leak detection device.

Author: Iwashita,-Tsuyoshi; Tamano,-Toyomi **Corp. Author:** Nippon Atomic Industry Group

Source: 5 Oct 1989; 31 Mar 1988. 4 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Inspection methods **ID:** 254

Abstract: The present invention concerns a device for detecting minor coolant leakages in a nuclear reactor using liquid metal sodium as coolants. That is, a coolant flow rate measuring device is disposed to a pipeway connecting a reactor vessel and a heat exchanger. Whether a flow rate signal measured by the flow rate measuring device is within a predetermined flow rate range or not is judged to rapidly detect a leakage. With such a constitution, since the leakage is detected by using the coolant flow rate measuring device disposed to each of the loops, depending on whether the flow rate of the coolants recycled in the loop is within an appropriate flow rate range or not, a loop causing leakage can be detected rapidly. The present invention has advantageous effects capable of rapidly detecting minor leakages due to small rapture of pipeways that has required much time for the detection and instantly specifying the ruptured loop in the case of a multi-loop structure. (I.S.).

Title: German standard problem (GSP) No. 9 'Dynamical behaviour of piping systems with a non-return valve under blowdo

Author: Firnhaber,-M.; Mueller,-W.C. **Corp. Author:** .Gesellschaft fuer Reaktorsiche

Source: Sep 1988. 251 p. Bundesministerium fuer Umwelt-, Naturschutz und Reaktorsicherheit, Bonn (Germany, F.R.)

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** German

Category: Analysis of break effects **ID:** 255

Abstract: In case of a FW-pipe rupture in a BWR, the FW non-return valves limit the outflow of water from the RPV, and thus the release of radioactivity. Upon closing of the valve, a pressure flush will pass through the piping system and cause considerable loads. The objective of the Standard Problem No. 9 is the investigation of the capability of computer codes, for piping systems. These are used in design and licensing by experts, to predict the behaviour of a large pipe. For the purpose of completeness all relevant loadings are considered: 1) static loads, 2) eigen values, 3) dynamic loads. For the comparison typical variables used are the following: 1) displacements, 2) moments, 3) stresses and strains. The comparison between experiment and calculation shows that the degree of agreement varies as well for the participants, the loadings as the variables selected for comparison. The analysis shows, that the greater part of the calculation differs from the experiment by not more than 10%, but no uniform tendency can be found. Neither the contributions by a single participant nor the total of all calculations for one variable lie completely inside the given 10% boundary. Specific parameters, which are known only approximately have been determined as sources of discrepancies. In the licensing procedure these uncertainties are covered by safety margins. Considering that the calculations are performed 'best-estimate', the static and modal results are adequate. The dynamic results are satisfactory for the first oscillation. (orig./HP).

Title: Experiments on rupture of a primary coolant pipe after creep fracture at high system pressure. Final report.

Author: Obst,-V.; Klenk,-A.; Julisch,-P. **Corp. Author:** Stuttgart Univ. (Germany, F.R.

Source: May 1988. 191 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** German

Category: Test/analysis **ID:** 256

Abstract: The hot tensile, creep fracture and heating-up tests with small specimens from the steels 20MnMoNi55 (1.6343) and 22NiMoCr 37 (1.6751) of different material conditions served the purpose of determining periodic break-down behaviour under temperature, stress and material conditions. The test specimens of the construction component test consisted of 20MnMoNi55. As regards pressure (p=163 bar) and temperature (350 to 700deg C), the testing conditions were oriented to the conditions of accident analysis by the GRS. The test results are summarized. (DG).

Title: Scanning and evaluation of documents with regard to safety engineering aspects, and consideration of results in the de

Author: Beisswaenger,-F. **Corp. Author:** Stuttgart Univ. (Germany, F.R.

Source: 1989. 31 p. Bundesministerium fuer Umwelt-, Naturschutz und Reaktorsicherheit, Bonn (Germany, F.R.).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** German

Category: Analysis of break effects **ID:** 257

Abstract: The results of blowdown experiments in the HDR experimental reactor have been compared with the computed design parameters of a PWR containment for the case of a rupture of the main coolant pipe, and good agreement has been found between computed and experimental data. Due to the condensation chambers of the BWR design, such direct comparison with internal pressure conditions in this reactor type cannot be done. The HDR blowdown experiments also permit an assessment of thermal stresses occurring in the containment in case of an accident. The results indicate that the stresses occurring can be judged according to current knowledge to remain within the limits defined in the regulatory guides. (orig.).

Title: Effect of interfacial transfer and wall heat transfer constitutive correlations in a model of PWR ECC bypass.

Author: Popov,-N.K. (Whiteshell Nuclear Research Establishment, Pinawa, Manitoba (Canada)); Rohatgi,-U.S. (Brookhaven National Lab., Upton, NY (USA)) **Corp. Author:** 5. Miami international symposi

Source: Veziroglu,-T.N. (Clean Energy Institute, Univ. of Miami, Coral Gables, FL (USA)). The 5th Miami international symposium on multi-phase transport and particulate phenomena (Condensed Papers). Coral Gables, FL (USA). Clean Energy Research Inst. University of Miami. 1988. 181 p. p. 76.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Analysis of break effects **ID:** 258

Abstract: The ECC bypass/refill process in a PWR downcomer, following a postulated large LOCA, is of importance to thermal-hydraulic safety. In the unlikely event of such accident, due to RPV rapid depressurization and blowdown at the break, coolant flashing and voiding of the reactor core occurs. To prevent fuel assembly overheating, the ECC subcooled water is injected into the reactor vessel. However, instead of penetrating the lower plenum, the ECC water, driven by the steam, flows azimuthally around the core barrel, bypasses through the downcomer and gets expelled out at the break. Mathematical modeling of such complex thermal-hydraulic phenomenon is accompanied with a difficult task of selecting an appropriate set of constitutive correlations. In this paper, using two-dimensional transient diabatic two-phase model of lower plenum ECC refilling and downcomer bypass flow, numerical calculations are performed to study the effect of interphase mass and momentum transfer, and wall heat transfer on lower plenum refilling initiation and rate. The results confirm that the interfacial momentum transfer by interfacial friction has dominant influence on the transient, and that the model is specially sensitive to annular interfacial friction correlation. Considerable difference in refilling predictions was obtained when various annular interfacial friction correlations were assessed in the model. It has been confirmed that with the Popov-Rohatgi correlation, the model refilling predictions are in very good agreement with the experimental data.

Title: Holographic testing of pipes and vessels.

Author: Ettemeyer,-A. **Corp. Author:** Autumn meeting of Deutsches

Source: Bauer,-K.G. (ed.). Deutsches Atomforum e.V., Bonn (Germany). HIGH SERVE '90 - nuclear engineering services. HIGH SERVE '90 - Service fuer die Kerntechnik. Bonn (Germany). INFORUM Verl. 1991. 363 p. p. 147-159.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Inspection methods **ID:** 259

Abstract: The examples demonstrating the use of holographic testing techniques in nuclear engineering refer to the deformation analysis of reactor components; dynamic measurements to determine expansion distributions, and materials testing. A holographic method has been developed in particular for materials testing at pressure pipes and vessels, which enables the testing of wall thickness weakening due to corrosion and crack formation. (DG).

Title: Investigations of crack formation and crack propagation on pipes made of austenitic steel AISI 316 L(N) under multi-

Author: Windelband,-B.; Munz,-D. (Karlsruhe Univ. (Germany). Inst. fuer Zuverlaessigkeit und Schadenskunde im Maschinenbau); Schinke,-B. (Kernforschungszentrum Karlsruhe GmbH (Germany). Inst. fuer Materialforschung 2)

Corp. Author: DFG final colloquium in the fr

Source: Deutscher Verband fuer Materialforschung und -pruefung e.V., Berlin (Germany). Early recognition of damage and course of damage on metal components. Schaedigungsfrueherkennung und Schadensablauf bei metallischen Bauteilen. 1992. 131 p. p. 101-108.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** German

Category: Test/analysis **ID:** 260

Abstract: This report introduces a test device which makes it possible to achieve any two-axial stress states on thin-walled pipe samples at room temperature. Using this plant, equi-biaxial stress states, which are typical of thermo-cycling loads, are simulated on the Austenitic steel AISI 316 L(N) in the LCF range. The load consists of a controlled axial alternating stress (tensile/compressive) and a controlled circumferential stress (inside/outside pressure). The formation and propagation of cracks were examined. Material data and results from single axis tests on pipe and solid samples and the first results from multi-axial tests are introduced. (orig./MM).

Title: Prevention of stress corrosion cracking in boiling water reactors.

Author: Jones,-R.L. (Electric Power Research Inst., Palo Alto, CA (United States))

Corp. Author:

Source: Materials-Performance. (Feb 1991). v. 30(2) p. 70-73.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Experience/events **ID:** 261

Abstract: Intergranular stress corrosion cracking (IGSCC) adjacent to girth welds in stainless steel piping systems has been a serious problem in boiling water reactor (BWR) plants in the United States for more than a decade. Recent observations suggest that IGSCC also may limit the service life of many reactor internals in BWRs. In this paper the pipe-cracking remedies in U.S. BWRs are described and adapting these remedies for protection of internals and attachments are presented.

Title: Effect of hydrogen water chemistry on ultrasonic response for intergranular stress corrosion cracking. Final report.

Author:

Corp. Author: Electric Power Research Inst.,

Source: May 1992. 123 p. . Electric Power Research Inst., Palo Alto, CA (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** 262

Abstract: Hydrogen water chemistry (HWC) is one of the approaches to control BWR water chemistry, which reduces the oxidizing power of the water to a level at which IGSCC (initiation and growth) is effectively suppressed. In this treatment, hydrogen gas is injected into the feedwater to lower the electrochemical corrosion potential (ECP) of stainless steel components. The objective of this work is to experimentally document the effect of HWC on IGSCC detectability. Two pipe samples were fabricated from a 12" Type 304 stainless steel pipe weldment containing a range of circumferential and axial cracks induced by the Creviced Piped Test. Initial characterization of IGSCC was performed for both pipes by UT and PT prior to application of HWC treatments. For each sample two separate UT methods were used. One was a manual technique that represents field practice, and the other was a laboratory technique that produced ultrasonic images of each crack. Both samples were subjected to a normal BWR water chemistry (NWC) for 168 hours before the HWC treatments. After NWC, one IGSCC sample B was treated for a period of 500 hours with a normal HWC condition (HWC-1) having electrochemical potential (ECP) value of about -0.60 volts (SHE) with Pt reference electrode, water dissolved oxygen content of less than 20 ppb, and water conductivity of less than 0.3 micro-S/cm. The other IGSCC sample C was treated for a period of 500 hours with an off-normal HWC condition (HWC-2) having ECP value of about -0.30 volts (SHE) with Pt reference electrode, and water conductivity of less than 0.3 micro-S/cm. (same as HWC-1). After the HWC treatments, the two IGSCC pipe samples were ultrasonically characterized in the exact manner that was done in the initial characterization to determine if there were any noticeable changes in the UT response of the cracks as indicated by their sizes and signal amplitudes.

Title: Full scale validation tests on the load bearing capacity of a degrated ferritic piping system when subjected to a blowdo

Author: Kussmaul,-K.; Diem,-H.; Kobes,-E. (Stuttgart Univ. (Germany)); Brosi,-S.; Schrammel,-D.

Corp. Author:

Source: Shibata,-Heki (Ed.). Trans. of the 11th SMiRT Conference. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. F p. 207-212. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 263

Abstract: A blowdown experiment followed by valve closure with water hammer load was performed at the HDR test facility on a piping system with close-to-reality isometry. The nominal width of the piping was DN 425 and the wall thickness was 25 mm. In the vicinity of the RPV nozzle a test pipe section precracked by an inner circumferential crack (a/t = 0.3, 2 alpha = 60deg) with 16 mm local wall thickness had been installed. Between 93 ms and 130 ms after the onset of blowdown the crack experienced maximum loading; the maximum bending moment at the crack cross-section was approx. 1300 kNm. In the experiment an increase in crack depth of 1.5 mm was detected. If one compares the results of the linear-elastic calculation used to limit the load according to ASME with the measured values, a limitation of the moment results for real loading of the test pipe component. Due to plastification in the test components, the loading in the pipe cannot be increased above a certain value. The linear-elastic post-test calculation yields a load in the crack cross-section which exceeds the allowable loading for a level D accident according to ASME. (author).

Title: Analytical evaluation method of creep-fatigue crack propagation for surface cracked pipe.

Author: Shimakawa,-T. (Kawasaki Heavy Industries Ltd., Kobe (Japan)); Takahashi,-H.; Doi,-H.; Watashi,-K.; Asada,-Y.

Corp. Author: 11. international conference on

Source: Shibata,-Heki (ed.) (Tokyo Univ. (Japan). Inst. of Industrial Science). Atomic Energy Society of Japan, Tokyo (Japan). Transactions of the 11th international conference on structural mechanics in reactor technology. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. L p. 205-210. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3 for 17 vols. bound in 15 (A through SD2).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 264

Abstract: This paper shows the estimated J-integral of surface cracked pipe and elbow under creep-fatigue conditions by 3-D FEM analyses. Predictions are compared with test data and the applicability of the analytical evaluation method is discussed. (author).

Title: A compound crack in a pipe under tension.

Author: Zahoor,-A. (Zenith Corp., Rockville, MD (United States))

Corp. Author:

Source: Nuclear-Engineering-and-Design. (Mar 1992). v. 133(2) p. 253-257.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Research/theoretical **ID:** 265

Abstract: Limit load and J-resistance curve solutions are developed for a compound crack in a pipe subjected to axial tension. The solutions are based on thick-walled cylinder assumption and the J solution can be applied with load-displacement data from one pipe test. The J-R solution can be used to assess the effect of loading type on the material's resistance to crack extension when used with previously published solution for bending moment loading. (orig.).

Title: Variation in fracture toughness of carbon steel due to test standards and its influence on fracture load prediction.

Author: Asano,-Masayuki; Fukakura,-Juichi; Kashiwaya,-Hideo; Saito,-Masahiro (Toshiba Corp., Kawasaki, Kanagawa (Japan))

Corp. Author: 11. international conference on

Source: Shibata,-Heki (ed.) (Tokyo Univ. (Japan). Inst. of Industrial Science). Atomic Energy Society of Japan, Tokyo (Japan). Transactions of the 11th international conference on structural mechanics in reactor technology. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. G1 p. 231-236. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3 for 17 vols. bound in 15 (A through SD2).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 266

Abstract: This paper describes apparent difference in fracture toughnesses obtained by JSME S001 and ASTM E813 test standards and its influence on fracture load prediction of carbon steel pipes. Fracture toughness tests were conducted in air at room temperature on 1CT specimens prepared from carbon steel pipe STS42 (20B, sch. 100) for LWR plants. And using these fracture toughnesses, R6-Rev. 3 approach was applied to estimate fracture load of a carbon steel pipe with a circumferential through-wall crack. It is found that predicted fracture loads for the two pipes with the same geometry are almost the same instead of large difference in apparent fracture toughness. (author).

Title: Three-dimensional thermoelastic analysis of a cylindrical pipe with an internal surface crack under convection cooling

Author: Chen-Wenhwa; Huang-Chincheng (Dept. of Power Mechanical Engineering, National Tsing Hua Univ., Hsinchu (Taiwan))

Corp. Author:

Source: Nuclear-Engineering-and-Design. (Dec 1991). v. 132(2) p. 143-151.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Research/theoretical **ID:** 267

Abstract: To predict thermoelastic fracture behaviors, the path-independent integral, which is physically the energy release rate per unit area of crack extension along the direction of crack growth, is computed by an accurate three-dimensional finite element model which provides both heat transfer and thermal stress analysis. The influence of realistic convection cooling on the inner surface of the cylindrical pipe on the computation of the temperature and the thermal stress intensity factor is evaluated. The variation of the thermal stress intensity factor along the crack front versus various configuration parameters and Biot numbers is also presented. This work is helpful to the safety evaluation of cylindrical pipes subjected to convection cooling. (orig./HP).

Title: Creep-fatigue crack behavior in surface cracked pipe.

Author: Takahashi,-H. (Toshiba Corp., Kawasaki, Kanagawa (Japan)); Mohri,-K.; Usami,-S.; Watashi,-K.; Asayama,-T.; Asada,-Y.

Corp. Author: 11. international conference on

Source: Shibata,-Heki (ed.) (Tokyo Univ. (Japan). Inst. of Industrial Science). Atomic Energy Society of Japan, Tokyo (Japan). Transactions of the 11th international conference on structural mechanics in reactor technology. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. L p. 193-198. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3 for 17 vols. bound in 15 (A through SD2).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 268

Abstract: The electrical potential method of predicting surface crack configuration has been established on the basis of the present tests. Valuable data about crack propagation and aspect ratio have been acquired. These are necessary data for verifying estimation methods based on analysis. (author).

Title: Short cracks in piping and piping welds.

Author: Wilkowski,-G.; Brust,-F.; Francini,-R.; Ghadiali,-N.; Kilinski,-T.; Krishnaswamy,-P.; Landow,-M.; Marschall,-C.; Rahman,-S.; Scott,-P. (Battelle, Columbus, OH (United States))

Corp. Author: 19. Nuclear Regulatory Comm

Source: Weiss,-A.J. (comp.). Nuclear Regulatory Commission, Washington, DC (United States). Office of Nuclear Regulatory Research. Transactions of the nineteenth water reactor safety information meeting. Oct 1991. 220 p. p. 2.5-2.6.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 269

Abstract: This program started on March 23, 1990, and has a duration of 4 years. The objective of the program is to develop and verify analyses by using existing and new experimental data for circumferentially cracked pipes, so modifications and improvements can be made to LBB and in-service flaw evaluation criteria. There are 7 technical tasks dealing, in general, with circumferentially cracked straight pipe under quasi-static loading. The tasks are as follows: short through wall cracked (TWC) pipe evaluations, short surface-cracked pipe evaluations, bi-metallic cracked pipe evaluations, dynamic strain aging and crack jump evaluations, anisotropic fracture evaluations, crack-opening-area evaluations, and NRCPIPE code improvements. There is also a separate task to develop international cooperation, interact with Section 11 of the ASME code, and perform program management functions.

Title: BWR internals problems and potential remedies.

Author: Jones,-R.L.

Corp. Author: 1989 workshop on LWR radiat

Source: Electric Power Research Inst., Palo Alto, CA (United States). Proceedings: 1989 workshop on LWR radiation water chemistry and its influence on in-core structural materials. Mar 1991. 531 p. p. 2.1-2.18.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Test/analysis **ID:** 270

Abstract: Many of the internal components of BWRs are potentially susceptible to SCC because the materials, construction methods and operating environments for internals are similar to those that have led to extensive stress corrosion cracking (SCC) of BWR coolant piping. Indeed, numerous replaceable components already have cracked, and concern is increasing about the possibility of corrosion damage to major internal structural components, such as the core support plate, core shroud and top guide, as well as to the welds that attached those components to the reactor vessel. A comprehensive research and development program has recently begun which addresses these concerns and has a structure similar to that of the recently completed pipe crack program. Because access to many of the locations that are potentially SCC-susceptible is very difficult, hydrogen water chemistry (HWC) appears to be the most attractive of the pipe crack remedies with regard to mitigation of SCC in major internal components and attachments. Work is underway to adapt HWC to prevent crack initiation and growth in internals, with emphasis on components located below the core, for which refurbishment, repair or replacement would be very difficult and expensive. This adaptation activity involves the predicting, on a plant-specific basis, the chemical and electrochemical conditions throughout the core and coolant circuit. A key aspect of the work is the generation of the substantial body of test reactor and in-plant data needed to benchmark this improved model. Results obtained to date are encouraging and suggest that, within a few years, it will be possible to define revised HWC guidelines that will substantially reduce the likelihood of SCC damage to most of the potentially susceptible pressure boundary and reactor internal components in US BWR plants.

Title: Application of simplified J-estimation methods to surface cracked structures under creep-fatigue loadings.

Author: Iwasaki,-Ryuichi (Babcock Hitachi K.K., Tokyo (Japan)); Shimakawa,-Takashi; Nakamura,-Kazuhiro; Takahashi,-Hiroyuki; Uno,-Tetsuro; Watashi,-Katsumi

Corp. Author: 11. international conference on

Source: Shibata,-Heki (ed.) (Tokyo Univ. (Japan). Inst. of Industrial Science). Atomic Energy Society of Japan, Tokyo (Japan). Transactions of the 11th international conference on structural mechanics in reactor technology. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. L p. 217-222. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3 for 17 vols. bound in 15 (A through SD2).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Research/theoretical **ID:** 271

Abstract: The crack extension analysis of surface cracked plates and a pipe under creep-fatigue loadings were performed by using four kinds of simplified J-estimation method, in order to investigate the applicability of them. As a result of benchmark analysis, we can conclude that, (1) Crack extension rates obtained by all simplified J-estimation methods correspond well to the results of experiments and BEM or FEM analysis. Hence, crack extension of surface cracked structures under creep-fatigue loadings can be estimated appropriately by simplified J-estimation methods. (2) The differences of results obtained by four kinds of simplified J-estimation methods were not so large and the results of Methods-2 and 3 were almost the same in all benchmark problems. (3) In the case when the thickness of a pipe is small compared to the diameter of a pipe, crack extension of a pipe can be estimated appropriately by using formulas for plates. (author).

Title: Growth of IGSC cracks in Type 304 stainless steel at 100 degrees C in an aqueous environment.

Author: Caskey,-G.R.; Stoner,-K.J.; Daugherty,-W.L.; Ondrejcin,-R.S.; Postles,-R.L.

Corp. Author: Westinghouse Savannah River

Source: [1990]. 18 p. 5. international symposium on environmental degradation of materials in nuclear power systems - water reactors. Monterey, CA (United States). 25-29 Aug 1991.FUNDING ORGANIZATION: USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Experience/events **ID:** 272

Abstract: IGSCC has been observed in the primary coolant system of the SRSs. Approximately 7% of the HAZs of pipe-to-pipe butt welds show indications of IGSCC during UT. Other piping and component areas, sensitized by flame washing or hot forming, have also developed cracks. The entire system was fabricated in the 1950's from Type 304 stainless steel. All joining was by the metal inert gas welding process. Crack growth rates have been measured on compact tension specimens under controlled environmental conditions. Growth rates were measured extending from less than 10 sup - sup 9 to approximately 10 sup - sup 5 millimeter per second. These growth rates bound the growth rates that have been inferred from a statistical analysis of UT indications. The UT data were collected since 1984 from HAZs in pipe-to-pipe butt welds in the SRS primary coolant piping. Chloride and sulfate anions, dissolved oxygen, and peroxide have been identified as the water impurities that influence IGSCC. A quantitative relationship has been established for susceptibility to IGSCC in terms of concentrations of these impurities and temperature. The heavy water reactor moderator and coolant is acidified with nitric acid to a pH of 4.7 to minimize corrosion of the aluminum cladding on the fuel elements.

Title: Chemistry and corrosion on steam generators in PWRs.

Author: Berge,-J.P.; Nordmann,-F. (Electricite de France (EDF), 93 - Saint-Denis (France). Groupe des Labs.)

Corp. Author: SVA further education course '

Source: SVA further education course 'Water chemistry in the nuclear power plant'. SVA-Vertiefungskurs 'Wasserchemie im Kernkraftwerk'. Bern (Switzerland). Schweizerische Vereinigung fuer Atomenergie (SVA). 1989. vp. p. C-2.1-C-2.32.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** German

Category: Experience/events **ID:** 273

Abstract: After a review of the objectives of primary coolant chemistry, the reasons are given for the 'decaying lithium' specification and for its development to decrease dose rates while avoiding increasing the risks of primary side cracking of steam generator pipes. For conditioning secondary coolant, the choice of volatile conditioning (ammoniac or morpholine) and its characteristics are specified. The different types of corrosion of steam generators are discussed, particularly cracking under stress corrosion on the primary side and intergranular attack of the pipes on the secondary side; the associated remedies and consequences are also discussed. 8 figs., 3 tabs., 5 refs.

Title: Numerical analysis of cracked pipe experiments within the IPIRG-program.

Author: Brickstad,-B. (Swedish Plant Inspectorate, Stockholm (Sweden)) **Corp. Author:** 11. international conference on

Source: Shibata,-Heki (ed.) (Tokyo Univ. (Japan). Inst. of Industrial Science). Atomic Energy Society of Japan, Tokyo (Japan). Transactions of the 11th international conference on structural mechanics in reactor technology. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. G2 p. 195-200. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3 for 17 vols. bound in 15 (A through SD2).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 274

Abstract: Numerical studies with non-linear FEM-analyses have been used for evaluation of a number of cracked pipe experiments conducted within the IPIRG program. Some verification tests are presented which demonstrate the capability of the ABAQUS-program to calculate different crack parameters for both surface cracks and through wall cracks in pipes. Numerical results are then compared with experiments for a number of IPIRG-experiments involving both monotonic and cyclic loading as well as quasi-static and dynamic loading. The numerical results confirm the experimental trends that dynamic loading will here degrade the fracture properties for carbon steel. They also indicate that the apparent J sub R curve evaluated for large cyclic loading at R=1 is not a unique material property but depend on the loading history. (author).

Title: A PC-based expert system for nondestructive testing.

Author: Shankar,-R.; Williams,-R.; Smith,-C.; Selby,-G. (EPRI NDE Center, J.A. Jones Applied Research Co., Charlotte, NC (United States)) **Corp. Author:** Expert systems applications for

Source: Naser,-J.A. (Electric Power Research Inst., Palo Alto, CA (United States)). Expert systems applications for the electric power industry. New York, NY (United States). Hemisphere Publishing. 1991. 1462 p. p. 573-592.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Methods **ID:** 275

Abstract: Rule-based decision logic which can emulate problem-solving expertise of humans is being explored for power plant nondestructive evaluation (NDE) applications. This paper describes an effort underway at the EPRI NDE Center to assist in the interpretation of NDE data acquired by automatic systems during ultrasonic weld examination of boiling-water reactors (BWRs). A personal computer (PC) -based expert system shell was used to encode rules and assemble knowledge to address the discrimination of intergranular stress corrosion cracking (IGSCC) from benign reflectors in the inspection of pipe-to-component welds. The rules attempt to factor in plant inspection history, ultrasonic examination data nd, if available, radiography testing data; a majority of them deal with specific ultrasonic signal temporal and spatial behavior during automatic scanning. The paper describes the efforts in the development of the expert system.

Title: Ductile fracture analysis of IPIRG cracked pipe experiments using strain energy density criterion.

Author: Shie-Jingjong; Ting-Kuen (Institute of Nuclear Energy Research, Lung-Tan (China)) **Corp. Author:** 11. international conference on

Source: Shibata,-Heki (ed.) (Tokyo Univ. (Japan). Inst. of Industrial Science). Atomic Energy Society of Japan, Tokyo (Japan). Transactions of the 11th international conference on structural mechanics in reactor technology. Tokyo (Japan). Atomic Energy Society of Japan. 1991. 6297 p. v. G2 p. 189-194. Distributed by Maruzen Co. Ltd. P.O. Box 5050, Tokyo Int'l, 100-31 Japan ISBN 4-89047-060-3 for 17 vols. bound in 15 (A through SD2).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Methods **ID:** 276

Abstract: The purpose of this work is to apply the strain energy density criterion, which has been used for studying the problem of ductile fracture, to predict the crack initiation and growth in the piping with circumferential through-wall cracks. The calculation of a strain energy density field was carried out by the finite element program 'ABAQUS' version 4.8. The incremental procedure considering the finite deformation and elastic-plastic behavior associated with von-Mises yield criterion, isotropic hardening and Prandtl-Reuss flow rule is employed. In order to demonstrate the accuracy and the validity of the failure criterion proposed in this study, the degraded piping experiment conducted by Battelle Laboratory is analyzed. (K.I.).

Title: Acceptance criteria of local wall thinning in carbon steel pipe subjected to axial force.

Author: Hasegawa,-Kunio; Kanno,-Satoshi; Hirano,-Akihiko; Ishiwata,-Masayuki; Gotoh,-Nobuho (Hitachi Ltd., Tokyo (Japan)) **Corp. Author:**

Source: Nippon-Kikai-Gakkai-Ronbunshu,-A-Hen. (Jul 1991). v. 57(539) p. 1470-1474.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** Japanese

Category: Criteria **ID:** 277

Abstract: Structural evaluation of local wall thinning caused by erosion is important for integrity of a high energy piping system. Acceptance criteria are required if pipe wall thinning is found or suspected. The purpose of this paper is to provide acceptance criteria for local erosion thinning in pipes. The pipe of interest is a STS 42 carbon steel pipe loaded with an internal pressure together with an axial force. The thinned region is characterized by the length, width and depth of wall loss. The length and width correspond to axial and circumferential crack lengths. Allowable combinations of length and width of wall thinning are determined from leak and break behavior of crack growth in pipes. The allowable depth of wall thinning is determined from the local membrane stress rule. Based on the crack growth behavior and the stress rule, the allowable extent and depth of local wall thinning are proposed for carbon steel pipes. Consequently, double-ended fracture and split fracture of the pipe are prevented when the local wall thinning is limited to within the allowable sizes. (author).

Title: Calculations accompanying the superheated steam reactor (HDR) leak rate tests at a pipe T junction with a crack in th

Author: Grebner,-H.; Hoefler,-A. (Gesellschaft fuer Reaktorsicherheit mbH (GRS), Koeln (Germany)) **Corp. Author:** Annual meeting on nuclear tec

Source: Deutsches Atomforum e.V., Bonn (Germany); Kerntechnische Gesellschaft e.V., Bonn (Germany). Annual meeting on nuclear technology '91. Proceedings. Jahrestagung Kerntechnik '91. Tagungsbericht. Bonn (Germany). INFORUM Verl. 1991. 630 p. p. 397-400.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** German

Category: Test/analysis **ID:** 278

Abstract: Published in summary form only.

Title: Crack growth in a pipe with incipient crack under pressure flush load due to valve closing.

Author: Kobes,-E.; Diem,-H.; Brosi,-S.; Schrammel,-D. **Corp. Author:**

Source: Deutsches Atomforum e.V., Bonn (Germany); Kerntechnische Gesellschaft e.V., Bonn (Germany). Jahrestagung Kerntechnik '91. Tagungsbericht. Bonn (Germany). INFORUM Verl. 1991. 630 p. p. 167-170.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** German

Category: Pressure ripple/water hammer **ID:** 279

Abstract: Published in summary form only.

Title: Crack growth tests on pipes with circumferential defects under internal pressure and superposed, alternating bending.

Author: Stoppler,-W.; Hippelein,-K.; Boer,-A.-de; Kerkhoff,-K.; Sommer,-H. (Stuttgart Univ. (Germany). Staatliche Materialpruefungsanstalt) **Corp. Author:** 16. MPA-seminar: Safety und r

Source: Stuttgart Univ. (Germany). Staatliche Materialpruefungsanstalt. Safety and reliability of plant technology with special emphasis on nuclear technology. Vol. 1 and 2. Vol. 1: Fracture mechanics, fatigue/creep processes, nondestructive testing. - Vol. 2: Integrity of vessels and components, integrity of line-pipes, irradiation embrittlement, thermal loading. Sicherheit und Verfuegbarkeit in der Anlagentechnik mit dem Schwerpunkt 'Kerntechnik'. Bd. 1 und 2. Bd. 1: Bruchmechanik, Zeitstandverhalten/Kriechvorgaenge, zerstoerungsfreie Pruefung. - Bd. 2: Behaelter- und Komponentenintegritaet, Rohrleitungsverhalten, strahleninduzierte Versproedung, Waermewechsel- und Thermoschockbeanspruchung. 1990. 784 p. p. 36.1-36.21.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Test/analysis **ID:** 280

Abstract: Pipes made of the steel 20 MnMoNi 55, with an outer diameter of 800 mm, wall thickness of 47 mm, and length of 5000 mm, were flawed by circumferential defects of a given length and depth and were loaded from the outside by a cyclic and a growing bending moment, with simultaneous effects. The pipes' deformation behaviour and the crack growth curves of propagation through the wall and in circumferential direction were measured and compared with calculated results. (orig.).

Title: FEM-analyses for fracture mechanics investigations on a tube with a circumferential flaw.

Author: Mueller,-W.; Noack,-H.D.; Veith,-H. (Bundesanstalt fuer Materialforschung und -pruefung, Berlin (Germany)) **Corp. Author:** 16. MPA-seminar: Safety und r

Source: Stuttgart Univ. (Germany). Staatliche Materialpruefungsanstalt. Safety and reliability of plant technology with special emphasis on nuclear technology. Vol. 1 and 2. Vol. 1: Fracture mechanics, fatigue/creep processes, nondestructive testing. - Vol. 2: Integrity of vessels and components, integrity of line-pipes, irradiation embrittlement, thermal loading. Sicherheit und Verfuegbarkeit in der Anlagentechnik mit dem Schwerpunkt 'Kerntechnik'. Bd. 1 und 2. Bd. 1: Bruchmechanik, Zeitstandverhalten/Kriechvorgaenge, zerstoerungsfreie Pruefung. - Bd. 2: Behaelter- und Komponentenintegritaet, Rohrleitungsverhalten, strahleninduzierte Versproedung, Waermewechsel- und Thermoschockbeanspruchung. 1990. 784 p. p. 1.1-1.22.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Test/analysis **ID:** 281

Abstract: Pipes were investigated by FEM for determining the steady crack growth of a 120deg surface crack under pure bending conditions, analysing the local load parameters J integral, crack tip opening, and strain deformation conditions near the crack tip. The pipes usually serve as primary coolant loop components, they have a cross-sectional area of 880 mm and a wall thickness of 40 mm. An elastic FEM analysis for the pipes under review with a four-point bending stress applied has shown that an asymmetrical bending stress distribution develops over the pipe's cross-sectional area, which is due to the latter's deformation to oval shape. (DG).

Title: Fracture margin of pipe with detectable crack by leakage.

Author: Hasegawa,-Kunio; Shimizu,-Tasuku; Matsumoto,-Koichi (Hitachi Ltd., Ibaraki (Japan). Mechanical Engineering Research Lab.); Gotoh,-Nobuho (Hitachi Ltd., Ibaraki (Japan). Hitachi Works) **Corp. Author:** 4. Japanese-German joint semi

Source: Nuclear-Engineering-and-Design. (Jul 1991). v. 128(1) p. 29-34.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Experience/events **ID:** 282

Abstract: This paper describes a theoretical method for calculating a detectable crack size by leak detection systems in BWR plants. Crack opening areas for carbon steel pipes of various diameters containing circumferential through-wall cracks are analyzed. It was shown that large diameter pipes have a much higher safety margin, and that the 0.1 A Criterion (10% of pipe cross-section) for postulated leak cross-sections gives a conservative estimate. (orig.).

Title: Ultrasonic inspection of the inner and outer surfaces of components in contact with liquids using horizontally polarized

Author: Salzburger,-H.J.; Huebschen,-G. **Corp. Author:** Fraunhofer-Institut fuer Zerstoe

Source: 16 Jul 1990. 81 p. .Bundesministerium fuer Forschung und Technologie, Bonn (Germany).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Inspection methods **ID:** 283

Abstract: Investigations and developments concerning UT for detection of cracklike and corrosive defects in vessels and pipes. Horizontally polarized shear (SH-)waves are used, which have advantageous features for these inspection tasks. Besides the complete corner reflection in the whole angle of incidence range and the grazing incidence along the near probe surface their propagation isn't influenced by liquids in contact with the surfaces. In thin walled components like plates and pipes (wall thickness <= 15 mm) these waves propagate as guided waves over large distances (up to 1 m) and are well suited for the inspection of large areas of these very commonly used components in the chemical industry. Free of couplant transmission and reception of these waves are realized by Electro-Magnetic Ultrasonic (EMUS)-transducers. The EMUS-technique is capable to perform inspections up to 300deg C without any cooling means by appropriate construction of the probes, so that the most components can be inspected at their temperatures of operation. Investigations have been performed concerning the detection of corrosive defects (wastage, pitting) in plates and tubes by guided SH-waves and cracklike defects - single cracks and crack-fields - by oblique incidence of bulk SH-waves. Laboratory measurements as well as trials in the field on real components have been carried out. (orig.).

Title: Approximate fracture methods for pipes. Pt. 1. Theory.

Author: Gilles,-P. (Societe Franco-Americaine de Constructions Atomiques (FRAMATOME), 75 - Paris (France)); Brust,-F.W. (Battelle, Columbus, OH (USA). Structures and Mechanics Dept.) **Corp. Author:**

Source: Nuclear-Engineering-and-Design. (May 1991). v. 127(1) p. 1-17.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Methods/comparison **ID:** 284

Abstract: Five simplified methods for predicting the fracture performance of circumferentially through-wall cracked pipes under pure bending are presented and discussed here, in this first-part paper. The theoretical foundations of the methods are examined in detail. In the second-part paper, moment-rotation predictions of cracked pipes experiencing stable crack growth are compared to experimental results and their capabilities checked with three-dimensional elasto-plastic finite element computations. In spite of their simplified theoretical foundations, these schemes give quite good predictions and are very easy to use. (orig.).

Title: Fracture behavior and acceptance size of flaw for pressurized piping.

Author: Hasegawa,-K.; Kanno,-S.; Shimizu,-T. (Mechanical Engineering Research Lab., Hitachi, Ltd., Hitachi-shi, Ibaraki-ken (Japan)); Gotoh,-N.; Saitoh,-T. (Hitachi Works, Hitachi, Ltd., Hitachi-shi, Ibaraki-ken (Japan)) **Corp. Author:** KSME/JSME joint conference

Source: Lee,-K.Y. (Yonsei Univ., Seoul (Korea, Republic of)); Takahashi,-Hideaki (Tohoku Univ., Sendai (Japan)) (eds.). Fracture and strength '90. Proceedings. Aedermannsdorf (Switzerland). Trans Tech Publ. 1991. 574 p. p. 555-560.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Research/theoretical **ID:** 285

Abstract: The safety margin for cracked piping must take into account the size of external load that would cause the leak and collapse of the piping. This study is concerned with the prediction of leak and collapse loads for cracked pipes under tensile and bending forces. The proposed methods gives allowable flaw sizes for pipes containing circumferential and axial cracks.

Title: An investigation on the behavior of longitudinally cracked pipe wall due to thermal striping.

Author: Yu,-Y.J.; Park,-S.H.; Sohn,-G.H. (Mechanical Design Dept., Korea Atomic Energy Research Inst., Daeduk-Danji, Taejeon (Korea, Republic of)) **Corp. Author:** KSME/JSME joint conference

Source: Lee,-K.Y. (Yonsei Univ., Seoul (Korea, Republic of)); Takahashi,-Hideaki (Tohoku Univ., Sendai (Japan)) (eds.). Fracture and strength '90. Proceedings. Aedermannsdorf (Switzerland). Trans Tech Publ. 1991. 574 p. p. 93-97.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Research/theoretical **ID:** 286

Abstract: An evaluation is performed on the behavior of longitudinally cracked pipe wall due to thermal striping, which is a phenomenon of high cycle fluid temperature fluctuation at hot/cold fluid boundary of stratified flow. The amplitude and frequency used as input values for this evaluation are determined from previous experimental data of open literatures. The finite element analyses were used to obtain temperature distribution and stress intensity factors at cracked pipe wall. The evaluation results show that a threat to the integrity of the piping systems due to thermal striping is not expected.

Title: Crack propagation in touch ductile materials. Phase II.

Author: Venter,-R.D.; Sinclair,-A.N.; McCammond,-D. (Toronto Univ., ON (Canada)) **Corp. Author:**

Source: Jun 1989. 176 p. Atomic Energy Control Board, Ottawa, ON (Canada).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Methods/comparison **ID:** 287

Abstract: The thrust of this work was to investigate published J material resistance and stress-strain data applicable to the understanding of crack propagation in tough ductile steels, particularly SA 106 Grade B pipe steel. This data has been assembled from PIFRAC, AECB report INFO-0254-1 and Ontario Hydro sources and has been uniformly formatted and presented to facilitate comparison and assessment. While the data is in many aspects incomplete it has enabled an evaluation of the influence of temperature, specimen thickness and specimen orientation to be made in the context of the experimental J-R curves so determined. Comparisons of the stress-strain data within the Ramburg-Osgood formulation are also considered. A further component of this report addresses the development of the required software to utilize what is referred to as the engineering approach to elasto-plastic analysis to investigate the load carrying capacity of selected cracked pipe geometries which are representative of applied crack propagation studies associated with piping systems in the nuclear industry. Three specific geometries and loading situations, identified as Condition A, B and C have been evaluated; the results are presented and illustrate the variation in applied load as a function of an initial and final crack extension leading to instability.

Title: The influences of mesh subdivision on nonlinear fracture analysis for surface cracked structures.

Author: Shimakawa,-T. (Kawasaki Heavy Industries, Kotoku, Tokyo (Japan). Dept. of Engineering) **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1991). v. 45(3) p. 327-349.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: LBB justification **ID:** 288

Abstract: The leak-before-break (LBB) concept can be expected to be applied not only to safety assessment, but also to the rationalization of nuclear power plants. The development of a method to evaluate fracture characteristics is required to establish this concept. The finite element method (FEM) is one of the most useful tools for this evaluation. However, the influence of various factors on the solution is not well understood and the reliability has not been fully verified. In this study, elastic-plastic 3D analyses are performed for two kinds of surface cracked structure, and the influence of mesh design is discussed. The first problem is surface crack growth in a carbon steel plate subjected to tension loading. A crack extension analysis is performed under a generation phase simulation using the crack release technique. Numerical instability of the J-integral solution is observed when the number of elements in the thickness direction of the ligament is reduced to three. The influence of mesh design in the ligament on the solution is discussed. The second problem is a circumferential part-through crack in a carbon steel pipe subjected to a bending moment. Two kinds of mesh design are employed, and a comparison between two sets of results shows that the number of elements on the crack surface also affects the solution as well as the number of elements in the ligament. (author).

Title: Study on crack opening area and coolant leak rates on pipe cracks.

Author: Matsumoto,-K.; Nakamura,-S.; Gotoh,-N. (Hitachi Ltd., Ibaraki (Japan). Hitachi Works); Narabayashi,-T.; Miyano,-H. (Toshiba Corp., Kawasaki, Kanagawa (Japan)); Furukawa,-S. (Toshiba Corp., Isogo-ku, Yokohama-shi (Japan)); Tanaka,-Y.; Horimizu,-Y. (Tokyo Electric Power Co., Inc. (Japan))

Corp. Author:

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1991). v. 46(1) p. 35-50.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** *289*

Abstract: This study was executed to support the establishment of Leak Before Break (LBB) standards for high energy piping, by examining crack opening shape on the pipe surface and crack opening area which may be used in the leak rate analysis. To decide the crack opening shape, a bending test was conducted by using 8in schedule 80 carbon steel pipe with an artificially produced circumferential through-wall crack. Crack opening displacement (COD) at some points along the crack were measured by clip gages. The results indicated that the crack opening shape was nearly elliptical. When the crack opening area changes from the pipe inner surface to the outer surface, it is also necessary to clarify which part of the crack opening area may be used in the analysis. Therefore expansion and reduction slits leak tests weret done. These results showed that the middle crack opening area between the inner and outer surfaces may be used in the analysis. Using the above results, the analytical leak rate calculated from the Tada-Paris equation and Moody's critical flow model was in good agreement with the measured one obtained from the leak test. (author).

Title: Recirculation piping replacement project at Dresden Unit 3.

Author: Brummit,-B. (ABB Impell Corp., Downers Grove, IL (USA))

Corp. Author: International conference on ene

Source: Hobbs,-B.F. (ed.). Energy in the 90's. New York, NY (USA). American Society of Civil Engineers. 1991. 380 p. p. 128-133.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: other **ID:** *290*

Abstract: During the late 1970's the incidence of stainless steel pipe weld flaws caused by intergranular stress corrosion cracking (IGSCC) had increased to the point that IGSCC became a generic concern to all operating boiling water reactor (BWR) plants. During the 1983 Dresden Unit 3 outage, inservice inspection revealed linear crack indications in 50% of the stainless steel piping welds in the drywell. The cause was attributed to intergranular stress corrosion cracking (IGSCC), which initiates and grows from the inside pipe surface under the combined influence of residual tensile stress, sensitized material and a supporting environment. By 1984, Commonwealth Edison Company (CECo) had performed various evaluations to determine the most cost effective remedy for resolving the IGSCC issue at CECo BWR stations. The conclusion was that total pipe replacement would be the most cost-effective and licensable solution for Dresden 3. This paper will discuss the Recirculation Piping Replacement Project performed at Dresden Unit 3 during 1985 and 1986. It will discuss several of the technical and administrative problems associated with performing such an extensive modification on an operating nuclear facility. This paper will also discuss the resolutions to these problems and some of the lessons learned during this project that are being implemented into the current modification process.

Title: Feature-based imaging system: The Peach Bottom field trials. Final report.

Author: Selby,-G.; Williams,-R.; Shankar,-R. (Jones (J.A.) Applied Research Co., Charlotte, NC (USA)) **Corp. Author:** Electric Power Research Inst.,

Source: May 1991. 230 p. .FUNDING ORGANIZATION: Electric Power Research Inst., Palo Alto, CA (USA).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Inspection methods **ID:** 291

Abstract: Feature-based systems that combine imaging and signal analysis capabilities may be useful for NDE. The report describes the field evaluation of an integrated, PC-based system at a plant site during a scheduled outage for pipe weld examination to discriminate IGSCC from benign, geometrical reflectors. The PC-based system capable of detailed analysis of UT signal data and an ISI-imaging system used in many commercial pipe examinations for IGSCC. The integrated system was trained to discriminate automatically IGSCC from other reflectors using EPRI NDE Center's inventory of field-removed samples. The methods and results of this training are described. Several classifiers were synthesized using mathematical features derived from signals that were collected to simulate field conditions. Data collection procedures were developed that required minimal operator training. Field data were collected and analyzed before and after pipe decontamination prior to pipe replacement. Automatic decision maps were generated for easier interpretation and comparison. The field trial was conducted during 10/87-1/88. Select pipe specimens were subjected to detailed metallurgical and dye penetrant analysis. A comparison between reflector calls from the UT-data and penetrant test results are provided in the final section. As a basis for comparison, the performance of the automated system was compared with manual calls. The automated technique results were better than manual; although both were well below acceptable standards as defined for IGSCC qualification. 2 refs., 174 figs., 10 tabs.

Title: Analyzing surface coatings in situ: High-temperature surface film analyzer developed.

Author: **Corp. Author:**

Source: USDOE Office of Energy Research, Washington, DC (USA). Technology '90. Accomplishments in technology transfer from DOE and its laboratories. Jan 1991. 192 p. p. 133-134.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Inspection methods **ID:** 292

Abstract: Scientists at ANL have devised a new instrument that can analyze surface coatings under operating conditions. The High-Temperature Surface Film Analyzer is the first such instrument to analyze the molecular composition and structure of surface coatings on metals and solids under conditions of high temperature and pressure in liquid environments. Corrosion layers, oxide coatings, polymers or paint films, or adsorbed molecules are examples of conditions that can be analyzed using this instrument. Film thicknesses may vary from a few molecular layers to several microns or thicker. The instrument was originally developed to study metal corrosion in aqueous solutions similar to the cooling water systems of light-water nuclear reactors. The instrument may have use for the nuclear power industry where coolant pipes degrade due to stress corrosion cracking, which often leads to plant shutdown. Key determinants in the occurrence of stress corrosion cracking are the properties and composition of corrosion scales that form inside pipes. The High-Temperature Surface Analyzer can analyze these coatings under laboratory conditions that simulate the same hostile environment of high temperature, pressure, and solution that exist during plant operations. The ability to analyze these scales in hostile liquid environments is unique to the instrument. Other applications include analyzing paint composition, corrosion of materials in geothermal power systems, integrity of canisters for radioactive waste storage, corrosion inhibitor films on piping and drilling systems, and surface scales on condenser tubes in industrial hot water heat exchangers. The device is not patented.

Title: Cracking in dissimilar steel points of superheater pipes in TP-100 boiler.

Author: Melekhov,-R.K.; Smiyan,-O.D.; Girnyj,-S.I.; Marchak,-I.I. (AN Ukrainskoj SSR, Lvov (Ukrainian SSR). Fiziko-Mekhanicheskij Inst.) **Corp. Author:**

Source: Fiziko-Khimicheskaya-Mekhanika-Materialov. (Jul-Aug 1990). v. 26(4) p. 105-106.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** Russian

Category: Research/theoretical **ID:** 293

Abstract: To clarify the reasons for cracking of welded joints of pipes manufactured of different steels (12KH1MF and 12KH18N10T), operating in high-pressure boiler superheaters, the, the concentration of residual hydrogen over all butt areas was determined. Alloying element disribution in the areas mentioned was analyzed. It is ascertained that heat-affected zone (HAZ) in pearlitic steel 12Kh1MF has the same composition as the basic metal. HAZ of austenitic steel 12KH18N10T is well stabilized and does not have a tendency for intercrystalline cracking under operation conditions (water vapour, 560 deg C). The assumption is made that hydrogen absorption in HAZ is realized during vapour decomposition in the points of mechanical damage of passivating film and in case of dissociation of other coolant components - hydrazine and ammonium.

Title: Fissuration by thermal fatigue of primary coolant circuit auxiliary pipes: analysis of encountering cases.

Author: Keroulas-de,-F.; Thomeret,-B. (Electricite de France, 75 - Paris (France). Service de la Production Thermique) **Corp. Author:** International Colloquium on C

Source: Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France). Contribution of Materials Investigation to the Resolution of problems encountered in PWR Plants. Volume 1. Contribution des Expertises sur materiaux a la Resolution des problemes rencontres dans les REP. Volume 1. Paris (France). Societe Francaise d'Energie Nucleaire. 1990. 312 p. p. 109-117.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** French

Category: Experience/events **ID:** 294

Abstract: Leaks due to thermal fatigue cracking have been detected on an elbow injection pipe from Tihange Unit 1, on a weld joint of a safety injection cold leg pipe, in United States and on a weld joint of a RHRS section line, in Japan. Destructive examinations results of elbow are compared to investigations on the 2 other cracked pipes. Origin of fatigue phenomenon and taken corrective actions are presented.

Title: Feature-enhanced-imaging field trials: Peach Bottom Unit 3. Interim report, Phase 2.

Author: Selby,-G.; Williams,-R.; Shankar,-R. (Jones (J.A.) Applied Research Co., Charlotte, NC (USA)) **Corp. Author:** Electric Power Research Inst.,

Source: June 1989. 250 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Inspection methods **ID:** 295

Abstract: Feature-based systems that combine imaging and signal analysis capabilities may be useful for nondestructive evaluation (NDE) of plant components. This report describes the field evaluation of an integrated system at a plant site during a scheduled outage for pipe weld examination to discriminate IGSCC from benign, geometrical reflectors. The integrated system consisted of a PC-based system capable of detailed analysis of UT-signal data and an in-service inspection (ISI) imaging system used in many commercial pipe examinations for IGSCC. The integrated system was trained to discriminate automatically IGSCC from other reflectors using EPRI NDE Center's inventory of field-removed samples. The methods and results of this training are described. Several classifiers were synthesized using mathematical features derived from signals that were collected to simulate field conditions. Data collection procedures were developed that required minimal operator training. Field data were collected and analyzed before and after pipe decontamination prior to pipe replacement. Automatic decision maps were generated for easier data interpretation and comparison. The field trial was conducted in October 1987--January 1988. Future activities will include collecting additional data after pipe removal to compare changes in baseline. These results will be presented in subsequent reports. 174 figs., 9 tabs.

Title: Investigation on field removed pipe sections in the PISC hot laboratories.

Author: Cambini,-M.; Crutzen,-S.; Jehenson,-P. (Commission of the European Communities, Ispra (Italy). Joint Research Centre); Edelmann,-X. (Sulzer Bros. Ltd., Leeds (UK)) **Corp. Author:**

Source: 1990. 21 p. Commission of the European Communities, Luxembourg (Luxembourg).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 296

Abstract: Action no. 1 of PISC III (Programme for the Inspection of Steel Components): Real Contaminated Structures (RCS), seeks to collect results from specific investigations and limited round robin tests on real service induced defects in materials and structures of the primary circuit of Light Water Reactors. The hot cell facilities at JRC-Ispra are fully equipped for non destructive and destructive work on a collaborative basis. Cracked austenitic steel pipes coming from the primary circuit of the Muehleberg reactor (Switzerland) have been inspected in order to demonstrate the validity of the facilities for the examination of these contaminated pieces.

Title: Crack initiation and growth behavior in dissimilar weld joint of SUS304 and 2.25Cr-1Mo steels subjected to cyclic the

Author: Ueda,-Masahiro (Japan Atomic Power Co., Tokyo (Japan)); Kano,-Takashi; Kanazawa,-Seiichi; Takani,-Satoru **Corp. Author:**

Source: JSME-International-Journal.-Series-1,-Solid-Mechanics-and-Strength-of-Materials. (Jan 1991). v. 34(1) p. 64-69.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 297

Abstract: Cyclic thermal transient fatigue tests of dissimilar weld joints of SUS304 and 2.25 Cr-1Mo steels with Inconel 82 buttering were performed in order to study the general behavior of crack initiation and growth, including crack locations, directions and densities under the respective loading conditions. The crack initiation sequence at different parts of the test models corresponded well with the result of analytical prediction using the finite element method. The results suggest that counter bore machining near the buttering boundary with 2.25 Cr-1Mo steel must be kept sufficiently apart from the joint area. The allowable number of thermal transient cycles according to the ASME CODE CASE N-47 and the elevated-temperature structural design guide developed by the PNC (Power Reactor and Nuclear Fuel Development Corporation) was shown to have a considerable safety margin compared with the crack initiation data of 1 mm in depth at all parts of the pipe models. (author).

Title: Short cracks in piping and piping wells.

Author: Wilkowski,-G.; Ahmad,-J.; Brust,-F.; Krishnaswamy,-P.; Landow,-M.; Marschall,-C.; Scott,-P.; Vieth,-P. (Battelle, Columbus, OH (USA)) **Corp. Author:**

Source: Weiss,-A.J. (comp.). Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research. Transactions of the eighteenth water reactor safety information meeting. Oct 1990. 211 p. p. 5.15-5.16.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 298

Abstract: This program started on March 23, 1990, and has a duration of 4 years. The objective of the program is to develop and verify analyses by using existing and new experimental data for circumferentially cracked pipes, so modifications and improvements can be made to LBB and in-service flaw evaluation criteria. There are 7 technical tasks with the following specific objectives. In general, they deal with circumferentially cracked straight pipe under quasi-static loading. The tasks described in the report are as follows: Task 1, short through-wall-cracked [TWC] pipe evaluations; Task 2, short surface-cracked [SC] pipe evaluations; Task 3, bi-metallic cracked pipe evaluations; Task 4, dynamic strain aging and crack jump evaluations; Task 5, anisotropic fracture evaluations; Task 6, crack-opening-area evaluations; and Task 7, NRCPIPE improvements. There is also a separate task to develop international cooperation, interact with Section XI of the ASME code, and perform program management functions. Cooperative efforts are underway with several international organizations (France, Italy, Japan, and West Germany) in exchanging analysis results and experimental data.

Title: Elastic-plastic fracture analysis of carbon steel piping using the latest CEGB R6 approach.

Author: Kanno,-S.; Hasegawa,-K.; Shimizu,-T. (Hitachi Ltd., Ibaraki (Japan). Mechanical Engineering Research Lab.); Kobayashi,-H.

Corp. Author:

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1991). v. 45(1) p. 89-99.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Methods **ID:** 299

Abstract: The elastic-plastic fracture of carbon steel piping having various pipe diameters and circumferential crack angles and subjected to a bending moment is analyzed using the latest United Kingdom Central Electricity Generating Board R6 approach. The elastic-plastic fracture criterion must be applied instead of the plastic collapse criterion with increase of the pipe diameter and the crack angle. A simplified elastic-plastic fracture analysis procedure based on the R6 approach is proposed. (author).

Title: BWR pressure vessel integrity. The good news.

Author: Herrera,-M.L.; Stancavage,-P.P. (GE Nuclear Energy, 175 Curtner Ave., MC747, San Jose, CA (USA))

Corp. Author: American Nuclear Society (AN

Source: Anon.-Proceedings of the topical meeting on nuclear power plant life extension. Volume 2. La Grange Park, IL (USA). American Nuclear Society. 1988. 645 p. p. 175-179.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Experience/events **ID:** 300

Abstract: The BWR pressure vessel shows the promise of a service life well beyond the 40 year licensed period. An evaluation of the materials, degradation mechanisms and filed experiences concludes that the BWR pressure vessels in the United States have structural margins to sustain operation for life extension. Thermal fatigue, intergranular stress corrosion cracking and, to a lesser extent, neutron embrittlement affect the life of vessel components. Fatigue has caused nozzle cracking and stress corrosion cracks have appeared in vessel pipe and safe ends. The susceptibility of materials and designs to these degradation mechanisms is well understood so that effective monitoring, preventive measures and durable refurbishments can assure long, safe and productive lives for BWR pressure vessels.

Title: Effect of induction heating stress improvement on ultrasonic response from intergranular stress corrosion cracking.

Author:

Corp. Author: Electric Power Research Inst.,

Source: Mar 1991. 326 p. Electric Power Research Inst., Palo Alto, CA (USA).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** 301

Abstract: Induction heating stress improvement (IHSI) is often performed for the BWR piping weldments in order to reduce the susceptibility of IGSCC. Recently, in an US BWR plant, IGSCC was detected where it was not expected because the plant had implemented IGSCC countermeasures, IHSI. The objective of this work is to experimentally document the effect of IHSI on IGSCC detectability. Two IGSCC pipe samples containing a range of circumferential and axial cracks were fabricated from two 12-inch Type 304 stainless steel pipe weldments by the Creviced Pipe Test. Each sample was documented in detail by UT and PT to establish its IGSCC characteristics prior to application of IHSI and I-(Inverse)IHSI. For each sample two separate UT methods were used. One pipe sample was subjected to normal IHSI, in which the pipe OD surface was heated while the ID surface was kept cool by circulating water. The other pipe was subjected to I-IHSI, in which the pipe was heated from the ID surface followed by rapid cooling from the OD surface with water jet spray. After IHSI treatments, the two IGSCC pipe samples were ultrasonically characterized in the exact manner that was done in the initial characterization to determine if there were any noticeable changes in the UT response of the cracks as indicated by their sizes and signal amplitudes. All of the maximum echo height data from the two samples were compared in terms of the effect of the IHSI treatments. There was no statistically significant difference in the echo height due to the treatments. For both cases, the crack sizes measured after the treatments were reported to be larger than those measured before the treatments. Ultrasonic imaging of cracks was carried out by a laboratory immersion technique coupled with an automated scan probe system. The results are expressed by imaging areas of different echo height levels and of depth.

Title: The assessment of creep crack growth in a welded pressure vessel.

Author: Jones,-M.R. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.); Coleman,-M.C. (Central Electricity Generating Board, Southampton (UK). Marchwood Engineering Labs.)

Corp. Author: 4. international conference on c

Source: Wilshire,-B.; Evans,-R.W. (University Coll. of Swansea (UK). Dept. of Materials Engineering) (eds.). Creep and fracture of engineering materials and structures. Proceedings of the conference held at Swansea (UK), 1-6 April 1990. London (UK). Institute of Metals. 1990. 1139 p. p. 605-619.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Methods **ID:** 302

Abstract: A procedure has been developed to assess the significance of defects in plant that operate at elevated temperatures. The procedure relates to plant subjected to steady loading and where creep is the dominant failure mode. Crack-like defects often occur in weldments and may grow by creep in service. Creep crack initiation and growth data have been produced from defects in heat-affected-zones (HAZs) of large welded ferritic pipes. The assessment procedure is used here, with modifications introduced to take account of the weldment heterogeneities, to predict the initiation, growth and final failure times at these defects. (author).

Title: Fracture behaviour of stainless steel pipes containing circumferential cracks at room temperature and 280 deg C.

Author: Maricchiolo,-C.; Milella,-P.P.; Pini,-A. (ENEA, Rome (Italy))

Corp. Author: Leak-before-break in water rea

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 367-377.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 303

Abstract: The paper presents the experimental results of a research programme on the fracture behaviour of austenitic stainless steel and TIG welds in pipes containing circumferential through-wall cracks at room temperature and 280 sup 0 C. The pipes were loaded in a pure bending mode using a four-point bend test method. The diameter of the pipes under investigation was 168 mm and 324 mm, with a thickness varying from 10 to 17 mm. As opposed to the behaviour of carbon steel pipes, it is found that the Net Section Collapse criterion predicts the moment of instability. Crack mouth opening displacements and collapse moments calculated using the GE-EPRI engineering approach show a rather high scatter with respect to the experimental results. (author).

Title: Analysis of two simplified methods (R6 and GE-EPRI) for circumferential crack stability in leak-before-break applicat

Author: Taupin,-Ph.; Gilles,-Ph.; Bhandari,-S. (Societe Franco-Americaine de Constructions Atomiques (FRAMATOME), 92 - Paris-La-Defense (France))

Corp. Author: Leak-before-break in water rea

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1990). v. 43(1-3) p. 129-149.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Methods/comparison **ID:** 304

Abstract: The Ainsworth method (R6 Option 1) and the GE-EPRI method are compared and analysed. These J-estimation schemes allow us to assess at low cost, crack stability in piping systems, which is an important step in Leak-Before-Break applications. The reliability, flexibility and self-consistency of the two methods are examined by focusing on the effect of geometry, nature of loading and superposition of tension and bending. The GE-EPRI method, based on Finite Element results, is used to check the validity of the hypothesis on which the simplifications in the R6 method for through-wall circumferentially cracked pipes rely. The R6 method appears to be easier to use for the treatment of secondary loads. It is shown that, the Ainsworth Failure Assessment Line, derived in the single load case, is still valid for combined proportional tension and bending loads, provided the appropriate limit load formula is chosen. (author).

Title: A microstructural investigation on the cracking of the Muhleberg reactor primary pipe CH1.

Author: Bottelier,-P.; Buscaglia,-G.; Cambini,-M.; Della-Rossa,-M. (ed.) **Corp. Author:**

Source: 1990. 37 p. Commission of the European Communities, Luxembourg (Luxembourg).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Experience/events **ID:** 305

Abstract: The cracking of a Primary Pipe of the Muhleberg Reactor, was investigated by means of Optical Microscopy, Scanning Electron Microscopy and Electron Probe Microanalysis. The circumferential crack, extending almost 360 sup 0 and 2-6 mm deep, started to propagate from the sharp edge of the counterbore. Since the crack initiation occurred at 45 sup 0 circumferential location in the heat-affected zone near the weld, where intergranular precipitation of carbibdes has been found, it is concluded that the crack initiated primarily by stress-corrosion, as a consequence of residual stresses and weld sensitization. Taking into account that at other circumferential locations the crack was found to be outside the heat affected zone, it cannot be excluded that the propagation of the circumferential crack might be due also to fatigue corrosion. However, given the fact that the deepest cracking occurs in the heat affected zone the stress-corrosion mechanism seems to be predominant. The corrosive environment was likely originated or enhanced by sulphide ions, formed by dissolution of manganese sulphide inclusions in the crevices, as also found in previous investigations on low alloyed steels.

Title: Crack initiation and growth behavior on SUS304 steel piping components during cyclic thermal transient strains.

Author: Ueda,-Masahiro; Kano,-Takashi; Kanazawa,-Seiichi; Takani,-Satoru (Japan Atomic Power Co., Tokyo (Japan)) **Corp. Author:**

Source: JSME-International-Journal.-Series-1,-Solid-Mechanics-and-Strength-of-Materials. (Oct 1990). v. 33(4) p. 514-519.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 306

Abstract: A series of thermal transient fatigue tests on five SUS304 stainless steel straight pipe models with circumferential weld joints and a nozzle model were performed to study the general behavior of crack initiation and growth including crack locations, directions and densities under respective loading conditions. The crack initiation sequence at different parts of models corresponded well with the result of analytical prediction using the finite element method. The potential margin on the allowable number of thermal transient cycles according to the ASME CODE CASE N-47 and the Elevated Temperature Structural Design Guide developed by the Power Reactor and Nuclear Fuel Development Corporation was evaluated, and the result showed a considerable safety margin compared with the data on initiation of cracks of 1 mm in depth at all parts of pipe models including circumferential weldings, notches, and other structural discontinuities. (author).

Title: Techniques for analyzing defect development in ASME Sec. XI.

Author: Kobayashi,-Hideo (Tokyo Inst. of Tech. (Japan). Faculty of Engineering) **Corp. Author:**

Source: Haikan-Gijutsu. (Jul 1990). v. 32(8) p. 97-102.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** Japanese

Category: Methods **ID:** 307

Abstract: For LWR machinery and equipment, the techniques of strength design and defect evaluation have been established. Generally ASME Boiler and Pressure Vessel Code is used. In Japan, Ministry of International Trade and Industry Notice No. 501 is equivalent to it. In this report, the outline of the ASME Boiler and Pressure Vessel Code, Sec.XI related to defect evaluation and the tendency of recent revision are explained. The Sec. XI is the stipulation for the inspection of machinery and equipment in nuclear reactor plants during service period, and the items in it are shown. The evaluation of the defects in low alloy steel pressure vessels is carried out in conformity with the stipulation, and it is explained. The outline of the evaluation of the defects in austenitic steel pipes is almost similar to the case of low alloy steel pressure vessels, therefore, only the different points are described. The critical crack dimensions for austenitic steel pipes were revised, and the main points are explained. Also the stipulation for ferritic steel piping was revised. (K.I.).

Title: Investigation on the behaviour of cracks in a pipe socket and in the cylindrical section of a reactor pressure vessel unde

Author: Klein,-M.; Neubrech,-G. (Kernforschungszentrum Karlsruhe GmbH (Germany, F.R.). Projektbereich Heissdampfreaktor - Sicherheitsprogramm/Handhabungstechnik); Roos,-E.; Stegmeyer,-R.; Diem,-H. (Stuttgart Univ. (Germany, F.R.). Staatliche Materialpruefungsanstalt)

Corp. Author: 21. lecture meeting of the DV

Source: Deutscher Verband fuer Materialforschung und -pruefung e.V., Berlin (Germany, F.R.). 20 years DVM Study Group 'Fracture Mechanisms'. Fracture-mechanical characteristics for structural components evaluation. 20 Jahre DVM-Arbeitskreis Bruchvorgaenge. Bruchmechanische Kennwerte fuer die Bauteilbewertung. 1989. 550 p. p. 99-110.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** German

Category: Test/analysis **ID:** 308

Abstract: Load application to postulated socket edge and circumferential defects in the cylindrical section by superimposed internal pressure and extended thermal shock due to flows of cold water is to be considered as the most unfavourable load condition. Such loads are to expected, for example, in case of a coolant loss accident with emergency coolant supply. The purpose of the investigations was to deepen the knowledge of flow processes and temperature gradients occurring in the medium; resulting structural stresses and crackloads as well as crack growth behaviour of the material during component tests; and, finally, to compare the results with computational models. (orig.).

Title: Mechanical behaviour of austenitic steel 316L mod. specimens with incipient cracks. Final report on a joint project of

Author: Schwalbe,-K.H.; Cornec,-A.; Kalinowski,-J. (GKSS-Forschungszentrum Geesthacht GmbH, Geesthacht-Tesperhude (Germany, F.R.). Inst. fuer Werkstofforschung); Huthmann,-H.; Gossmann,-O.; Grueter,-L. (Internationale Atomreaktorbau GmbH (INTERATOM), Bergisch Gladbach (Germany, F.R.))

Corp. Author:

Source: 1989. 207 p. GKSS-Forschungszentrum Geesthacht GmbH, Geesthacht-Tesperhude (Germany, F.R.).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** German

Category: Test/analysis **ID:** 309

Abstract: In order to see whether the Engineering Treatment Model (ETM) can be applied to predict the mechanical behaviour of notched specimens, or those with incipient cracks, of the austenitic steel 316 L mod. under monotonously increasing load, tests were carried out with notched laboratory specimens (partial report A), cracked compact specimens (partial report B), and straight pipes DN 700 with circumferential crack throughout the pipe wall (partial report C). (MM) With 56 figs., 13 tabs.

Title: Closeout of IE Bulletin 79-17: Pipe cracks in stagnant borated water systems at PWR [pressurized water reactors] pla

Author: Foley,-W.J.; Dean,-R.S.; Hennick,-A.

Corp. Author:

Source: Feb 1990. 30 p. Nuclear Regulatory Commission, Washington, DC (USA). Div. of Operational Assessment.Parameter,

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Experience/events **ID:** 310

Abstract: Documentation is provided in this report for the closeout of IE Bulletin 79-17 and its revision on the safety-related subject of pipe cracks in stagnant borated water systems at operating plants with pressurized water reactors (PWRs). Closeout is based on the implementation and verification of actions required by the bulletin. Evaluation of utility responses and NRC/Region inspection reports indicates that the bulletin is closed for all of the 41 operating PWRs to which it was issued for action. It is concluded that the concerns of the bulletin have been resolved. Background information is supplied in the Introduction and Appendix A.

Title: Initiation and instability behavior of cracked LMFBR piping: Comparison of different theoretical approaches and expe

Author: Bhandari,-S.; Nesa,-D. (Novatome Industries, 69 - Lyon (France)); Faidy,-C. (Electricite de France, 69 - Villeurbanne (France)); Grueter,-L. (Internationale Atomreaktorbau GmbH (INTERATOM), Bergisch Gladbach (Germany, F.R.))

Corp. Author: 14. MPA-seminar on safety and

Source: Nuclear-Engineering-and-Design. (May 1990). v. 119(2/3) p. 337-345.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 311

Abstract: Results of analytical studies of a cooperative joint fracture mechanics programme are presented. The programme is concerned with bending of original DN 700 straight pipes with circumferential through-wall cracks. Material is the austenitic stainless steel 316L SPH. This paper is following a previous publication on the experimental part. Details are given on studies using the finite element technique, the 'Screening Criterion' of Battelle, the double criteria approach (F.A.D.), the GE/EPRI handbook and the engineering treatment model (ETM). Most data are given for crack initiation; however, instability and corresponding crack extension are also considered. (orig.).

Title: Component testing at the HDR-facility for validating the calculation procedures and the transferability of test results fr

Author: Katzenmeier,-G. (Kernforschungszentrum Karlsruhe GmbH (Germany, F.R.)); Kussmaul,-K.; Roos,-E.; Diem,-H. (Stuttgart Univ. (Germany, F.R.). Staatliche Materialpruefungsanstalt)

Corp. Author: 14. MPA-seminar on safety and

Source: Nuclear-Engineering-and-Design. (May 1990). v. 119(2/3) p. 317-327.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 312

Abstract: The HDR RPV and various pipework systems were loaded by static and transient loads, both thermally and mechanically, until crack growth or breakoff occurred. Parallel to component testing laboratory-scale tests were carried out on specimens as well as calculations in support of the experiments. The results of measurements and calculations were compared. Incipient crack in the cladding of the reactor pressure vessel under cyclic thermoshock loading occurred after a number of load cycles which roughly correspond to those in the ASME design curve. For pipework components, in that case incipient crack in an elbow, good agreement was found between specimens and components. Crack propagation under cyclic thermoshock loading in the RPV wall at aggravated corrosive conditions is greatly overestimated in the calculation. The major causes of the deviation are the flow rate and the chemistry of the medium and differencies in the stress conditions of specimen and component. For pipework tested at constant temperature the transferability from specimen to component was found to be relatively good. The crack propagation had been only slightly overestimated in that case. The question of stable crack growth in the thermoshock tests performed at the RPV will be answered after fractographic analysis of the rupture surfaces. The LBB-behavior of pipework made from ductile material was confirmed in two tests performed under cyclic bending load and in two tests under seismic loading conditions. (orig.).

Title: Stress corrosion and thermal fatigue - experiences and countermeasures in austenitic SS pipes of Finnish BWR-plants.

Author: Hakala,-J.; Haenninen,-H.; Aaltonen,-P. (Industrial Power Co. Ltd., Olkiluoto (Finland))

Corp. Author: 14. MPA-seminar on safety and

Source: Nuclear-Engineering-and-Design. (May 1990). v. 119(2/3) p. 389-398.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Experience/events **ID:** 313

Abstract: A summary of the existence of pipe cracking in Finnish BWR plants is presented covering both thermal fatigue and IGSCC cases. Countermeasures against cracking are evaluated and the measures applied are sumamrized. Also the results of a research program to monitor ageing of the weld heat affected zones in a pipeline section of a shut-down cooling system are summarized. (orig.).

Title: Fracture mechanics study of a pipe bend with a longitudinal crack under static loading.

Author: Uhlmann,-D.; Diem,-H.; Brosi,-S. **Corp. Author:**

Source: Nuclear-Engineering-and-Design. Vol. 119:347-360.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Research/theoretical **ID:** 314

Abstract: The regulating norms for nuclear power plant component are based on the assumption that the components have no defects. Therefore it is of major interest to find what values the fracture mechanics parameters assume when a loading which produces in an unflawed elbow stresses which are acceptable according to the regulatory guides is applied to an elbow with cracks. It could be shown that with small flaws (1/3 of arc length, a/t = 0.25) calculations assuming linear-elastic behavior give results almost identical as when elastic-plastic behavior is assumed. If the cracks are small and the loading is according to the regulatory guides the initiation value J sub i for stable crack growth is not reached. (orig.).

Title: Evaluation of crack resistance of materials for reactor pipelines.

Author: Chizhik,-A.A.; Lanin,-A.A.; Ulizko,-Eh.P.; Anan'eva,-M.A.; Zelenin,-Yu.V. **Corp. Author:** 4. International conference on s

Source: AN SSSR, Moscow (USSR); Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Moscow (USSR); AN SSSR, Moscow (USSR). Inst. Metallurgii; Joint Inst. for Nuclear Research, Dubna (USSR). 4. International conference on study and design of thermonuclear reactor materials. Summaries of reports. 4. Mezhdunarodnaya konferentsiya po issledovaniyu i razrabotke konstruktsionnykh materialov dlya reaktorov termoyadernogo sinteza. Tezisy dokladov. 1990. 94 p. p. 12.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** Russian

Category: Research/theoretical **ID:** 315

Abstract: Short note.

Title: Strength and fracture behavior of pipes with circumferentially orientated cracks under monotonic bending loading.

Author: Stoppler,-W.; Sturm,-D.; Schiedermaier,-J.; Hippelein,-K. (Stuttgart Univ. (Germany, F.R.)) **Corp. Author:** 10. international conference on

Source: Hadjian,-A.H. Transactions of the 10th international conference on structural mechanics in reactor technology. Volume M. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1989. 223 p. p. 79-84.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Test/analysis **ID:** 316

Abstract: Pipes with the dimensions of the main coolant piping system of a pressurized water reactor (PWR) weakened by circumferential flaws and loaded by internal pressure as well as an external bending moment were tested to define the critical flaw length under service and upset conditions.

Title: Comparison of approximative Markov and Monte Carlo simulation methods for reliability assessment of crack contain

Author: Schmidt,-T.; Schomburg,-V. (Univ. of the Federal Armed Forces Hamburg, Hamburg (Germany, F.R.)) **Corp. Author:** 10. international conference on

Source: Hadjian,-A.H. Transactions of the 10th international conference on structural mechanics in reactor technology. Volume M. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1989. 223 p. p. 31-36.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Methods/comparison **ID:** 317

Abstract: Reliability assessments based on probabilistic fracture mechanics can give insight into the effects of changes in design parameters, operational conditions and maintenance schemes. Although they are often not capable of providing absolute reliability values, these methods at least allow the ranking of different solutions among alternatives. Due to the variety of possible solutions for design, operation and maintenance problems numerous probabilistic reliability assessments have to be carried out. This is a laborous task especially for crack containing welds of nuclear pipes subjected to fatigue. The objective of this paper is to compare the Monte Carlo simulation method and a newly developed approximative approach using the Markov process ansatz for this task.

Title: Surface crack configuration detection system by reversing DC potential drop method.

Author: Hayashi,-M.; Ontaka,-M.; Shimizu,-T.; Takaku,-K. (Hitachi Ltd., Ibaraki (Japan)) **Corp. Author:** 8. international conference on

Source: Stahl,-D. Proceedings of the 8th international conference on NDE in the nuclear industry. Metals Park, OH (USA). American Society for Metals. 1986. 683 p. p. 447-456.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Inspection methods **ID:** 318

Abstract: This paper reports on the development of a surface crack configuration detection system by a reversing DC potential drop method. The system employs a simplified method for determining surface crack configuration, which has been invented based on FEM analysis for variously deep surface cracks with different aspect ratios. Weld HAZ of 12 inch pipe with a surface crack can be inspected in 17 minutes by the system including measurements of the potential difference distributions, data processing and output of analyzed data. The crack configuration can be evaluated with the accuracy of 0.3 mm.

Title: Experimental evaluation of J in cracked straight and curved pipes under bending.

Author: Moulin,-D.; Touboul,-F.; Foucher,-N.; Lebey,-J.; Acker,-D. **Corp. Author:** CEA Centre d'Etudes Nucleaire

Source: 1989. 6 p. 10. international conference on Structural Mechanics in Reactor Technology (SMIRT). Anaheim, CA (USA). 14-18 Aug 1989.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: LBB justification **ID:** 319

Abstract: An experimental program is being carried out at the CEA Saclay in collaboration with FRAMATOME and IPSN with a view to validate analysis methods applicable for evaluation of leak before break behavior in P.W.R. piping. A large experimental work was already performed in USA, Germany and Japan and cracked pipes made of stainless steel material under bending. The methods of analysis got same validations for straight pipes. However applicability to elbows and comparison with toughness values obtained on small specimens like CT specimens was not completely dealt with.

Title: Wave formation in thermally stratified flow. surface cracks of feedwater pipelines.

Author: Haefner,-W. (Battelle-Institut e.V., Frankfurt am Main (Germany, F.R.)); Spurk,-J.H. (Technische Hochschule Darmstadt (Germany, F.R.))

Corp. Author: Annual meeting on nuclear tec

Source: Deutsches Atomforum e.V., Bonn (Germany, F.R.); Kerntechnische Gesellschaft e.V., Bonn (Germany, F.R.).INFORUM Verl. May 1990. 710 p. p. 105-108.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Research/theoretical **ID:** 320

Abstract: Published in summary form only.

Title: Subcritical-crack growth behavior for carbon steel in high-temperature pure water.

Author: Hasegawa,-Kunio (Hitachi Ltd., Tsuchiura, Ibaraki (Japan). Mechanical Engineering Research Lab.); Saito,-Takashi; Tanaka,-Nobuyuki; Kikuchi,-Masaaki; Suzuki,-Kazumi

Corp. Author:

Source: Nippon-Kikai-Gakkai-Ronbunshu,-A-Hen. (Mar 1990). v. 56(523) p. 474-481.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 321

Abstract: Subcritical-crack growth rates for carbon steels in high-temperature pure water are obtained from fatigue and constant load test. The test specimens, made of STS 42 and 49 carbon steel pipes are CT type and surface-cracked flat plate-type. The experimental environment is saturated pure water at 288degC and 7.8 MPa pressure. The subcritical-crack growth rates are influenced by several factors; the fatigue crack growth rate, da/dN in water is accelerated by low frequency and high stress ratio. The da/dN in water from a trapezoial load wave and for welded metal is less than from a triangular wave and for base metal. In addition, the da/dN in water at 150degC and in a steam environment is lower than in water at 288degC. The growth rate, da/dN, in the thickness and the width directions is almost equivalent to the results of the surface-cracked flat plate specimens. The curve of the crack growth rate, da/dt, is obtained from the constant load test. The threshold value of the stress intensity factor for stress corrosion cracking for STS base and welded metal of carbon steels is relatively large when compared with the sensitized Type 304 stainless steel and other carbon steels. (author).

Title: The instability of an asymmetric through-wall circumferential crack in a pipe subject to bending deformation.

Author: Smith,-E. (Manchester Univ. (UK))

Corp. Author: 10. international conference on

Source: Hadjian,-A.H. (Bechtel Power Corp., Los Angeles, CA (USA)). Transactions of the 10th international conference on structural mechanics in reactor technology. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1989. 375 p. p. 187-190.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Research/theoretical **ID:** 322

Abstract: This paper examines the fracture instability of a pipe, fabricated from a ductile material such as 304 stainless steel, in which a circumferential cross-section contains a through-wall crack, the pipe being subject to bending deformation. The crack is situated asymmetrically with regards to the bending axis, and the theoretical analysis, which is based on the tearing modulus procedure, demonstrates the extent to which crack asymmetry influences the fracture instability criterion.

Title: Fracture mechanics studies of a cracked pipe bend under in-plane loading.

Author: Wanner,-R.; Brosi,-S. (Paul Scherrer Inst. (PSI), Villigen (Switzerland)); Uhlmann,-D.; Diem,-H. (Stuttgart Univ. (Germany, F.R.))

Corp. Author:

Source: Hadjian,-A.H. (Ed.). Trans. of the 10th SMiRT Conference. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1989. 375 p. p. 19-24.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 323

Abstract: In cyclically loaded piping components, cracks may initiate and subsequently grow after a sufficient number of load cycles. This process has been studied within an experiment of the German HDR safety program. Under the operating conditions: internal pressure p sub i =10.6 MPa, temperature T = 240 degrees C and elevated oxygen content in the pressure medium (about 8 ppm), a fullsize bend (DN400) of the piping was loaded with an in-plane bending moment acting in opening mode by cyclic deflection of one pipe end.

Title: About two new efficient nonlinear shell elements.

Author: Yin,-J. (Xi'an Jiaotong Univ., Xi'an (China)); Suo,-X.Z.; Combescure,-A. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France))

Corp. Author: 10. international conference on

Source: Hadjian,-A.H. (Bechtel Power Corp., Los Angeles, CA (USA)). Transactions on the 10th international conference on structural mechanics in reactor technology. Los Angeles, CA (USA). American Association for Structural Mechanics in Reactor Technology. 1989. 419 p. p. 301-310.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Methods **ID:** 324

Abstract: The aim of the paper is to present the development of two shell elements for non linear analysis. The first one is an axisymetric curved shell element and it is developed for buckling analysis. The formulation is given, as well as some typical applications. The second one is an extension of the classical DKT element to large strains taking into account all aspects of non linearities. This element is used for the simulation of four point bending of cracked pipes. The whole experiment is simulated by the calculation taking into account very large strains at the crack tip and propagation of the crack.

Title: Pipe inside scanner for crack detection system.

Author: Hayashi,-M.; Ohtaka,-M.; Takaku,-K. (Hitachi Ltd., Ibaraki (Japan))

Corp. Author: 9. international conference on n

Source: Iida,-K.; Doherty,-J.E.; Edelmann,-X.-Proceedings of the 9th international conference on nondestructive evaluation in the nuclear industry. Metals Park, OH (USA). American Society for Metals. 1988. 686 p. p. 197-204.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Inspection methods **ID:** 325

Abstract: Surface crack configuration can be detected by the DC potential drop method. The authors report on simplified method for determining the surface crack shape developed, based on the FEM analysis for the surface cracks of various depth with different aspect ratios. The method has been applied to fatigue surface cracks introduced on the inside surface of stainless steel pipes. The crack shape could be estimated with the accuracy of +-0.3 mm. In order to verify the method, a pipe interior scanner has been devised to measurement of potential difference distribution in 12 inch piping. The scanner can pass through not only an elbow but through a vertical pipe. Crack configurations can be obtained automatically by 16 bit personal computer scanner control and data processing.

Title: Field application of integrated ultrasonic feature-based and imaging-basing analysis.

Author: Berhravesn,-M.; Avioli,-M. (Electric Power Research Inst., Palo Alto, CA (USA)); Shankar,-R.; Selby,-G. (EPRI NDE Center Charlotte, NE (USA))

Corp. Author: 9. international conference on n

Source: Iida,-K.; Doherty,-J.E.; Edelmann,-X.-Proceedings of the 9th international conference on nondestructive evaluation in the nuclear industry. Metals Park, OH (USA). American Society for Metals. 1988. 686 p. p. 489-496.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Inspection methods **ID:** 326

Abstract: Feature-based systems that combine imaging and signal analysis capabilities are shown to be useful for nondestructive evaluation (NDE) of plant components. This paper describes the field application of an integrated system for pipe weld examination to discriminate intergranular stress-corrosion cracking (IGSCC) from benign, geometrical reflectors. The integrated system consisted of a personal computer (PC)-based system capable of detailed analysis of ultrasonic signal data and an in-service inspection (ISI) imaging system used in many commercial pipe examinations for IGSCC.

Title: Investigation of the behaviour of cracks in a dissimilar weld.

Author: Benitz,-K.; Daum,-D. (Asea Brown Boveri Reaktor GmbH, Mannheim (Germany, F.R.)); Hartnagel,-W. (Asea Brown Boveri AG, Mannheim (Germany, F.R.))

Corp. Author: 15. MPA-seminar on safety and

Source: Stuttgart Univ. (Germany, F.R.). Staatliche Materialpruefungsanstalt. Safety and reliability of plant technology with special emphasis on long-term integrity of pressure components of nuclear power plants. Vol. 1 and 2. Vol. 1: Integrity of vessels and components, irradiation embrittlement, nondestructive testing. Vol. 2: Fatigue/creep processes, integrity of line-pipes, fracture mechanics. Sicherheit und Verfuegbarkeit in der Anlagentechnik mit dem Schwerpunkt 'Langzeitintegritaet der druckfuehrenden Bauteile von Kernkraftwerken'. Bd. 1 und 2. Bd. 1: Behaelter- und Komponenten-Integritaet, strahleninduzierte Versproedung, zerstoerungsfreie Pruefung. Bd. 2: Zeitstandverhalten/Kriechvorgaenge, Rohrleitungsverhalten, Bruchmechanik. 1989. 784 p. p. 5.1-5.19. Published in 2 separate volumes.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** German

Category: Test/analysis **ID:** 327

Abstract: The behaviour of postulated faults or cracks in the sense of fracture mechanics in the area of dissimilar welds on HTR pipe joints is calculated and compared with corresponding experimental results. The welded pipe sections consist of X 20 Cr Mo V 121 and X10 Ni Cr Al Ti 3220 and the weld of additional material GRINI7. The calculations were done by the finite element program system ANSYS. (DG).

Title: Ductile fracture experiment on through-wall crack piping.

Author: Tseng,-C.G.

Corp. Author: Seminar on leak-before-break:

Source: Wilkowski,-G.M. (Battelle, Columbus, OH (USA)); Chao,-K.S. (eds.) (Tawian Power Co., Taipei (Taiwan)). Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research; Battelle, Columbus, OH (USA). Leak-Before-Break: Further developments in regulatory policies and supporting research. Feb 1990. 350 p. p. 289-298.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 328

Abstract: The paper reviewed results of ductile fracture experiments on through-wall cracked pipe. The experiments were conducted on 1.5-inch diameter Schedule 20 TP304 stainless steel pipe. The pipe had a circumferential through-wall fatigue crack in the base metal that was approximately 32 percent of the pipe circumference. The pipe was loaded in four-point bending at room temperature. The J-R curve calculated from this experiment was 60-percent higher than that from the 4-inch pipe experiments conducted at Battelle for EPRI. The initiation load was found to be 97 percent of the maximum load, which is consistent with past small diameter stainless steel pipe tests. 4 refs.

Title: Stress intensity factors in axisymmetric circumferential crack in cylinder.

Author: Wu,-C.M.; Chen,-L.S. (Materials Research Lab., Hsinchu (Taiwan)) **Corp. Author:** Seminar on leak-before-break:

Source: Wilkowski,-G.M. (Battelle, Columbus, OH (USA)); Chao,-K.S. (eds.) (Tawian Power Co., Taipei (Taiwan)). Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research; Battelle, Columbus, OH (USA). Leak-Before-Break: Further developments in regulatory policies and supporting research. Feb 1990. 350 p. p. 261-276.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Research/theoretical **ID:** *329*

Abstract: In this paper an elastic stress intensity factor solution by shell analysis was compared to finite element analyses for circumferentially surface-cracked pipe in axial tension with different stress distributions through the pipe thickness. The solutions generated were for inside radius (R sub i) to pipe thickness (t) ratios of 5 to 20. The magnitude of the error was under 5 percent for flaw depths less than 60 percent of the pipe thickness, and R sub i /t of 5 to 20.

Title: Evaluation of various circumferential through-wall cracked pipe estimation schemes.

Author: Wilkowski,-G.M.; Brust,-F.W. (Battelle, Columbus, OH (USA)); Chao,-K.S. (Taiwan Power Co. (China)); Gilles,-P. (Framatome, Paris (France)) **Corp. Author:** Seminar on leak-before-break:

Source: Wilkowski,-G.M. (Battelle, Columbus, OH (USA)); Chao,-K.S. (eds.) (Tawian Power Co., Taipei (Taiwan)). Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research; Battelle, Columbus, OH (USA). Leak-Before-Break: Further developments in regulatory policies and supporting research. Feb 1990. 350 p. p. 127-160.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Methods/comparison **ID:** *330*

Abstract: The paper compared predicted moments at crack initiation and maximum load for through-wall circumferentially cracked pipe in four-point bending. Five different J-integral estimation scheme analyses were used to compare to 16 pipe experiments involving cracks in carbon steel base metal, stainless steel base metal, carbon steel SAWS, and stainless steel SAWS. The pipe diameters varied from 100 to 914 mm. The results showed that the GE/EPRI method was the most conservative, and the Paris and LBB, NRC methods were the least conservative. The LBB,ENG and LBB, NRC methods (developed in the NRC's Degraded Piping Program) were slightly conservative but were still reasonably accurate.

Title: Using acoustic emission technique to monitor fractures on the analogous pressure pipes.

Author: Zhang-Lichen (Southwest Inst. of Nuclear Reactor Engineering, SC (China)) **Corp. Author:**

Source: Jan 1989. 11 p. China Nuclear Information Centre, Beijing, BJ (China).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Inspection methods **ID:** *331*

Abstract: By using the acoustic emission technique to monitor the fractures on analogous pressure pipes of the primary circuit which has had cracks and loading with pressure was investigated. The dynamical process, from cracking to fracturing, was recorded by the acoustic emission technique. Comparing with the conventional method, this method gives more informations, such as pre-cracking, cracking growing, fast fracturing and the pressure values at different phases. During testing time a microcomputer was used for real-time data processing and locating the fracturing position. These data are useful for the mechanical analysis of the reactor components.

Title: Conditions of crack initiation in a circumferentially cracked pipe in bending; experimental determination of j on elbow

Author: Moulin,-D.; Touboul,-F.; Lebey,-J.; Acker,-D. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. d'Etudes Mecaniques et Thermiques); Foucher,-N. (Novatome, 69 - Lyon (France)) **Corp. Author:** CEA Centre d'Etudes Nucleaire

Source: 1989. 33 p. Pressure Vessel and Piping Conference. Honolulu, HI (USA). 22-26 Jul 1989.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Test/analysis **ID:** 332

Abstract: This paper describes an experimental study performed on pipes and elbows containing through circumferential cracks in bending. The study concerns the prediction of crack initiation under monotone loading conditions. A procedure for calculation of the J parameter using only the experimental results is described and applied. This procedure is compared with certain approximation methods conventionally employed. The scale function on which the experimental procedure is based is determined for pipes and elbows in opening and closing modes. The results obtained for sufficiently long cracks in both these components are similar. The tests show the influence of plasticity extending beyond the region in the vicinity of the crack. Experimental determination of J requires measurement of the potential energy transmitted to the test specimen on the part of the specimen governed by the crack.

Title: Elastic plastic finite element calculations of a cracked piping system and leak-rate evaluations.

Author: Grebner,-H.; Hoefler,-A.; Haber,-O. (Gesellschaft fuer Reaktorsicherheit mbH (GRS), Koeln (Germany, F.R.)) **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1989). v. 40(2) p. 91-105.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Research/theoretical **ID:** 333

Abstract: The failure of a large-scale piping system loaded by steady internal pressure and an increasing opening in-plane bending moment under conditions similar to those of a nuclear pressurized water reactor has been studied experimentally as part of the HDR safety program. The piping system failed with leakage through a 400 mm-long crack in the flank of a 90 sup 0 pipe elbow. In analysing the results the finite element program ADINA is used for nonlinear calculations, including fracture mechanical subroutines for the evaluation of J-integral values. A three-dimensional analysis of the complete cracked piping system would be very expensive; as a first step the complete, as yet uncracked, system was modelled using pipe elements. The resulting global displacements are used in a second step as input for three-dimensional detailed models of the cracked pipe bend with adjacent straight pipe ends. After calculations to determine the possible amount of stable crack growth, most attention is given to leakage areas with respect to the applied loading. By means of a separate computer program, leak rates were calculated using the leakage areas evaluated by finite elements. Values between 20 kg/s and 80 kg/s were obtained. (author).

Title: Acoustic emission - flaw relationships for in-service monitoring of nuclear reactor pressure boundaries.

Author: Hutton,-P.H.; Kurtz,-R.J.; Friesel,-M.A. **Corp. Author:**

Source: Nuclear Regulatory Commission, Washington, DC (USA). Div. of Engineering. Compilation of contract research for the Materials Engineering Branch, Division of Engineering. Annual report for FY 1987. Jun 1988. 423 p. p. 317-325.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Inspection methods **ID:** 334

Abstract: The acoustic emission (AE) development program addresses the objective of validating technology for continuous monitoring of reactor pressure boundaries and components to detect cracking. The work is supported by the NRC Research Office with the TVA providing supplemental funding. The FY 1987 scope includes the following items: (1) Initiate continuous, on-line AE monitoring of three sections of Watts Bar-1 (currently instrumented) during operation; (2) Complete continuous, on-line AE monitoring of cracked pipe locations at Peach Bottom-3; (3) Complete refinement of AE signal identification and AE/flaw evaluation methods; (4) Complete IGSCC/AE testing and validate AE signal identification and flaw evaluation methods applicable to IGSCC monitoring; (5) Complete investigation of the influence of low crack growth rate on detection of crack growth AE and flaw evaluation using the AE data; (6) Complete ASTM approval of a Standard Practice for Continuous Monitoring of Acoustic Emission from Metal Pressure Boundaries; (7) Submit a Code Case/Appendix to ASME Section XI for acceptance of continuous AE monitoring as a method for inspection of pipes and nozzles, and for general on-line monitoring of the entire pressure boundary; (8) Complete a review draft of a final program summary report; and (9) Prepare semi-annual progress reports and topical reports as appropriate.

Title: Analysis and development of fracture-mechanical failure concepts, with particular regard to further development and v

Author: Schmitt,-W. **Corp. Author:** Fraunhofer-Institut fuer Werkst

Source: Feb 1989. 90 p. Bundesministerium fuer Forschung und Technologie, Bonn (Germany, F.R.).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** German

Category: Research/theoretical **ID:** 335

Abstract: In the framework of this research project, the influence of the parameters a) crack and component geometry and b) crack propagation was studied. For this purpose, the behaviour of surface cracks in tensile test disks and pipes was analyzed under ductile-fracture conditions on the one hand, and on the other hand, dynamic crack resistance curves in the upper shelf viscosity were determined. Further studies which, however, did not surpass a first orientation phase were concerned with the verification of the J integral concept in the case of superposed stress (thermomechanical, tension/shear). (MM).

Title: Numeric and experimental studies on surface cracking of disks and pipes of ferritic and austenitic steels. X20 CrMoV

Author: Memhard,-D.; Klemm,-W. **Corp. Author:** Fraunhofer-Institut fuer Werkst

Source: Sep 1989. 80 p. .Bundesministerium fuer Forschung und Technologie, Bonn (Germany, F.R.).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** German

Category: Test/analysis **ID:** 336

Abstract: For safeguarding the transmission chain from the two-dimensional sample to real three-dimensional structural components, the behavior of surface cracks in disks and pipes was investigated in an experimental and a theoretic-numerical way. The objective was a quantitative failure analysis which covers essential phases resulting from operation and breakdown conditions, in fact from the first propagation of the incipient crack under fatigue loading to stabile crack growth under monotonously increasing load up to wall breakthrough and a possible subsequent instability. The investigations of the growth of fatigue cracks show that the growth of partial through-cracks in disks and pipes is generally overestimated in analytic calculations, if the constants of the crack propagation determined for CT samples are used for these calculations. The results presented here show that the J-integral concept is suited to give a good description of the behavior of cracks in constructional components under shear fracture conditions as to the quality and quantity, if the influence of the multi-axiality of the state of stress on the resistance against crack propagation is appropriately considered. The transferability of samples to constructional components seems to be possible for purely mechanical strain even with a superposed amount of bending for a crack propagation which is not too great. (orig.).

Title: Fatigue crack propagation in welded joint of austenitic steel for nuclear power engineering.

Author: Linhart,-V.; Barakova,-B. (Statni Vyzkumny Ustav Materialu, Prague (Czechoslovakia)) **Corp. Author:**

Source: Zvaranie. (Jun 1989). v. 38(6) p. 168-171.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** Russian

Category: Research/theoretical **ID:** 337

Abstract: The crack propagation characteristics were obtained for Cr-Ni type austenitic steel 08Kh18N10T under variable stress in the individual zones of a welded joint on a pipe. Measurements of the threshold deviation of the stress intensity factor, DELTA K sub p , showed that the root zone of the pipe welded joint was the weakest point as concerns crack propagation. The threshold values obtained for the filler metal on the pipe outer surface were considerably greater than those for the root zone of the welded joint and slightly greater than those for the base material and for the transition between the joint and the base material. The measured propagation response showed that the rate of fatigue crack propagation was for the base material higher by up to one order for low DELTA K than for the filler joint and the root zone of the joint. (J.B.). 5 figs., 3 tabs., 6 refs.

Title: Application of reliability techniques to prioritize BWR [boiling water reactor] recirculation loop welds for in-service i

Author: Holman,-G.S. (Lawrence Livermore National Lab., CA (USA)) **Corp. Author:** Nuclear Regulatory Commissio

Source: Dec 1989. 70 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Inspection methods **ID:** 338

Abstract: In January 1988 the U.S. NRC issued Generic Letter 88-01 together with NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," to implement NRC long-range plans for addressing the problem of SCC in BWR piping. NUREG-0313 presents guidelines for categorizing BWR pipe welds according to their SCC condition (e.g., presence of known cracks, implementation of measures for mitigating SCC) as well as recommended inspection schedules (e.g., percentage of welds inspected, inspection frequency) for each weld category. NUREG-0313 does not, however, specify individual welds to be inspected. To address this issue, the LLNL developed two recommended inspection samples for welds in a typical BWR recirculation loop. Using a PFM model, LLNL prioritized loop welds on the basis of estimated leak probabilities. The results of this evaluation indicate that riser welds and bypass welds should be given priority attention over other welds. Larger-diameter welds as a group can be considered of secondary importance compared to riser and bypass welds. A "blind" comparison between the probability-based inspection samples and data from actual field inspections indicated that the probabilistic analysis generally captured the welds which the field inspections identified as warranting repair or replacement. Discrepancies between the field data and the analytic results can likely be attributed to simplifying assumptions made in the analysis. The overall agreement between analysis and field experience suggests that reliability techniques -- when combined with historical experience -- represent a sound technical basis on which to define meaningful weld inspection programs. 13 refs., 8 figs., 5 tabs.

Title: 11. status report of the project 'HDR safety programme' of Karlsruhe Nuclear Research Center, December 9, 1987. W

Author: Katzenmeier,-G. (comp.) **Corp. Author:** Kernforschungszentrum Karlsr

Source: 1988. 411 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** German

Category: Test/analysis **ID:** 339

Abstract: The status report is subdivided as usual according to the subject-specific structure into individual projects. To the individual projects special test groups are assigned, each test group comprising of several experiments. In the past year 1987 experiments of the following test groups were the center of interest: Thermoshock on reactor pressure vessel feed pipe and on the interior wall of RPV (crack depths up to 32 mm), pipe failure test in area A, cylindrical pipe and area B; elbow, pre-test H sub 2 measurement technique, air blowdown, blowdown test in containment, fire prevention test with oil, dismantling test on pipes. In addition extensive calculations, preliminary and evaluation efforts for additionel test groups were made, in particular: blowdown of pipes and/or containment, blowdown of containment, thermal stratification on pipes, earthquake test with large shaker on building, firetests with gas, dismantling test on concrete, long-time thermoshock tests. (orig./HP).

Title: An expert system for power plant NDE.

Author: Shankar,-R.; Williams,-R. (EPRI NDE Center, Charlotte, NC (USA)); Avioli,-M. Jr. (Electric Power Research Institute, Palo Alto, CA (USA)) **Corp. Author:** 15. annual review of progress i

Source: Thompson,-D.O.; Chimenti,-D.E. (eds.). Review of progress in quantitative nondestructive evaluation. Volume 8A. New York, NY (USA). Plenum Press. 1989. 1194 p. p. 665-672.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Inspection methods **ID:** 340

Abstract: An expert system for assistance in interpretation of nondestructive evaluation (NDE) data from BWR welds has been developed on a PC system. A PC-based shell program was used to encode rules and assemble facts to discriminate IGSCC in BWR welds from benign, geometrical, weld reflectors. The system has been integrated in a PC platform capable of automatic scanning, digitally acquiring ultrasonic data, and imaging and feature-based processing. The expert system consists of approximately 200 rules and facts acquired from experts in the field. These rules include specific temporal and spatial signal behaviors that are automatically computed by feature-based imaging. The expert system combines ultrasonic and weld radiograph results to arrive at an overall decision on reflector type. The system is undergoing tests at the EPRI NDE Center on field-removed pipe weld samples with service-induced cracking.

Title: Fatigue crack growth on straight pipes under thermal shocks.

Author: Poette,-C. (CEA, Centre d'Etudes Nucleaires de Cadarache, Saint-Paul-lez-Durance (France)) **Corp. Author:** Fracture mechanics, creep and f

Source: Becht-IV,-C. (Becht Engineering Co., Inc., Liberty Corner, NJ (USA)); Bhandari,-S.K. (Framatome, Paris (France)); Tomkins,-B. (Northern Research Labs., Risley (UK)). Fracture mechanics, creep and fatigue analysis. New York, NY (USA). American Society of Mechanical Engineers. 1988. 86 p. p. 77-82.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** German

Category: LBB justification **ID:** 341

Abstract: One of the essential safety options of a future fast breeder reactor is to demonstrate the leak before break capability for the secondary pipework. For the purpose of this demonstration, accurate predictions of subcritical crack growth are necessary. The test device FORTUNA was designed to assess the particular influence of thermal stresses (which induce through wall bending and peak stresses) and of residual weld seam stresses. Finite element calculations are presented using defect linespring type shell elements and 3D brick elements to assess the thermal peak stresses effect. The analysis shows that thermal shocks should be one of the most severe case of loading for leak before break demonstration. The experimental results provide a basis for comparison.

Title: Plastic collapse analysis of pipes with arbitrarily shaped circumferential cracks.

Author: Cofie,-N.G.; Froehlich,-C.H. (NUTECH Engineers, San Jose, CA (USA)) **Corp. Author:** Fracture mechanics, creep and f

Source: Becht-IV,-C. (Becht Engineering Co., Inc., Liberty Corner, NJ (USA)); Bhandari,-S.K. (Framatome, Paris (France)); Tomkins,-B. (Northern Research Labs., Risley (UK)). Fracture mechanics, creep and fatigue analysis. New York, NY (USA). American Society of Mechanical Engineers. 1988. 86 p. p. 39-46.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Research/theoretical **ID:** 342

Abstract: In the analysis presented in this paper, a flawed pipe with an arbitrarily shaped circumferential crack is broken into various segments with corresponding crack depths. Two sets of equations are derived. The first addresses the case where the entire crack length is in tension, while the second addresses the cases where part of the crack is in compression. The equations are obtained by summing the effects of the individual segments based on equilibrium of the cracked cross section of the pipe subjected to an externally applied axial force and bending moment. Constant depth, elliptical and parabolic shaped cracks are used in this net section plastic collapse formulation to establish interaction and failure diagrams. The analyses have shown that significant differences in flaw acceptance criteria exist depending on the shape of the crack. Tables are developed for the flaw shapes considered in this paper. Application of the methodology to multiple cracks in pipes is discussed.

Title: Fracture mechanics, creep and fatigue analysis.

Author: Becht-IV,-C. (Becht Engineering Co., Inc., Liberty Corner, NJ (USA)); Bhandari,-S.K. (Framatome, Paris (France)); Tomkins,-B. (Northern Research Labs., Risley (UK)) **Corp. Author:** Fracture mechanics, creep and f

Source: New York, NY (USA). American Society of Mechanical Engineers. 1988. 86 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Experience/events **ID:** 343

Abstract: This book covers the proceedings of the 1988 ASME Pressure Vessels and Piping Converence. Topics include: Fatigue crack growth analysis of a 45 sup 0 PWR - lateral; Fatigue crack growth on straight pipes under thermal shocks; and The treatment of residual stress in defect assessment of austenitic fast reactor structures.

Title: Experimental determination of J value on circumferencially cracked stainless steel pipes under bending.

Author: Moulin,-D; Lebey,-J.; Acker,-D. **Corp. Author:** CEA Centre d'Etudes Nucleaire

Source: 1989. 9 p. 2. Conference on pipework and operation. London (UK). 21-22 Feb 1989

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Test/analysis **ID:** 344

Abstract: Development of leak before break methodology in nuclear industry requires determination of conditions of crack stability in piping products. Due to tough and strain-hardening materials involved, a promising parameter to characterize such a behaviour seems to be the driving force J. An experimental program is carrying on in CEA at Saclay in order to establish and to verify analytical and experimental methods to predict conditions of crack stability. It concerns circumferentially cracked tubes. The through wall crack center angles range from 30 sup 0 to 150 sup 0. Blunt end notches are considered, as well as fatigue precracked notches. Loading is imposed monotonically in displacement controlled conditions until maximal load is reached. The paper will present experimental procedure and first results obtained. Instrumentation and typical recordings (load, rotations at different distances from crack section, crack opening displacement, electric potential drop, ovalization) will be described. Experimental results are interpreted in terms of limit analysis and J estimation to predict crack initiation and maximal loads as a function of crack length. Results obtained permit the adjustment of experimental scaling functions usually employed for J evaluation with one single specimen. These functions are compared with analytical ones based on theoretical considerations of a simple limit state of the pipe cracked section.

Title: Application of the ACPD method in life studies on welded K joints in sea water.

Author: Lachmann,-E. (Industrieanlagen-Betriebsgesellschaft m.b.H., Ottobrunn (Germany, F.R.)) **Corp. Author:** 20. session of the DVM workin

Source: Deutscher Verband fuer Materialforschung und -pruefung e.V., Berlin (Germany, F.R.). Proceedings of the 20th meeting of the working group on fracture mechanisms. Central issue: Corrosion and fracture. 1988. 578 p. p. 393-405.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** German

Category: Test/analysis **ID:** 345

Abstract: In this contribution, crack measurements on pipe joints of steel StE 355 are carried out by means of the alternating current potential method (ACPD) and the multiposition measurement technique. The objective of these measurements was to obtain improved knowledge on the cracking and crack propagation behaviour as well as on the crack geometry. (MM).

Title: Stress corrosion and thermal fatigue, experiences and countermeasures in austenitic ss pipings of Finnish BWR-plants.

Author: Hakala,-J. (Industrial Power Co. Ltd., Olkiluoto (Finland)); Haenninen,-H.; Aaltonen,-P. (Valtion Teknillinen Tutkimuskeskus, Espoo (Finland). Metallilaboratorio) **Corp. Author:** 14. MPA-seminar on safety and

Source: Staatliche Materialpruefungsanstalt, Stuttgart (Germany, F.R.). Safety and reliability of plant technology with special emphasis on long-term integrity of pressure components of nuclear power plants. Vol. 1 and 2. Vol. 1: Plant technology, thermal shock loading, irradiation embrittlement, corrosion/wear, nondestructive testing. Vol. 2: Fatigue/creep processes, integrity of vessels and components, integrity of line-pipes. Sicherheit und Verfuegbarkeit in der Anlagentechnik mit dem Schwerpunkt 'Langzeitintegritaet der druckfuehrenden Bauteile von Kernkraftwerken'. Bd. 1 und 2. Bd. 1: Anlagentechnik, Thermoschock, strahleninduzierte Versproedung, Korrosion/Verschleiss, zerstoerungsfreie Pruefung. Bd. 2: Zeitstandverhalten/Kriechvorgaenge, Behaelter- und Komponenten-Integritaet, Rohrleitungsverhalten. 1988. 1003 p. p. 18.1-18.19. Published in 2 separate volumes.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** German

Category: Experience/events **ID:** 346

Abstract: A summary of the existence of pipe cracking in Finnish BWR plants is presented covering both thermal fatigue and IGSCC cases. Countermeasures against cracking are evaluated and the measures applied are summarized. Also the results of a research program to monitor ageing of the weld heat affected zones in a pipeline section of a shut-down cooling system are summarized. (orig.).

Title: Fracture-mechanical description of mixed-mode (presumably) hydrogen-induced circumferential cracks in a pipeline.

Author: Mattheck,-C. (Kernforschungszentrum Karlsruhe GmbH (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung 4 - Zuverlaessigkeit/Schadenskunde); Moldenhauer,-H.

Corp. Author: 20. session of the DVM workin

Source: Deutscher Verband fuer Materialforschung und -pruefung e.V. Vortraege der 20. Sitzung des Arbeitskreises Bruchvorgaenge. Schwerpunktthema: Korrosion und Bruch. 1988. 578 p. p. 225-234.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** German

Category: Test/analysis **ID:** 347

Abstract: Published in summary form only.

Title: Stress corrosion cracking of pressure vessel and pipeline steels in hot water.

Author: Magdowski,-R.M.; Speidel,-M.O. (Eidgenoessische Technische Hochschule, Zurich (Switzerland). Inst. fuer Metallforschung und Metallurgie)

Corp. Author: 20. session of the DVM workin

Source: Deutscher Verband fuer Materialforschung und -pruefung e.V., Berlin (Germany, F.R.). e. Vortraege der 20. Sitzung des Arbeitskreises Bruchvorgaenge. Schwerpunktthema: Korrosion und Bruch. 1988. 578 p. p. 119-126.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** German

Category: Test/analysis **ID:** 348

Abstract: First of all, there is a short description of the fracture-mechanical tests on stress crack corrosion. After that, crack growth curves are presented with the crack growth rate being drawn as a function of stress intensity. From this, a limit between tolerable and intolerable susceptibility to stress crack corrosion is determined. The distance to this limit with regard to the crack growth rate is defined as safe distance towards stress crack corrosion. (MM).

Title: About damage of cold headers of staem generators at NPPs with WWER-1000 reactors.

Author:

Corp. Author: Baranenko,-V.I.; Kirov,-V.S.;

Source: Atomnaya-Ehnergiya. (Nov 1993). v. 75(5). p. 391-394.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** Russian

Category: Experience/events **ID:** 349

Abstract: It is revealed that the cause of outlet ('cold') collector failures in the PGV-1000 and PGV-1000M steam generators is the mechanism of material damage in connetors between holes in collector wall, principally new in the practice of steam generator engineering. It is shown that location of the damaged connectors is determined by pipe sheet structural peculiarities damage maximum lays near the wedge area on the one hand, and by specific features of therma-hydraulic process character in steam generator volume (the second maximum lays near the edge end) on the other hand). The connector damage character may be interpreted as that of fatigue type and should be determined by thermal-hydraulic processes in staem generator volume. 7 refs., 3 figs.

Title: Conquering service water pipe corrosion.

Author:

Corp. Author: Leech,-J.N. (Public Service Ele

Source: Nuclear-Engineering-International. (Jan 1994). v. 39(474). p. 31, 33-35.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Experience/events **ID:** 350

Abstract: Damage to the components of Hope Creek's service water system from corrosion was so severe that a six-year US$37 million project to replace 2850 feet of pipe was begun in 1988. Due for completion in 1994, the bulk of the work having already been done, the project offers lessons for existing plants and for future designs. (Author).

Title: Lessons learned from fatique failures in major FWR components.

Author: Ware,-A.G.; Shah,-V.N. (Idaho National Engineering Lab., Idaho Falls (United States)) **Corp. Author:** Aging research information con

Source: Beranek,-A. (comp.). Nuclear Regulatory Commission, Washington, DC (United States). Proceedings of the Aging Research Information Conference. Volume 1. Sep 1992. 556 p. p. 275-295.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Experience/events **ID:** 351

Abstract: This paper evaluates the field fatigue failure experience and describes the lessons learned that can be employed in managing fatigue damage at the sites of these failures and at other susceptible sites. Fatigue damage has resulted in cracks on the inside surfaces of vessels and piping, and in some cases, through-wall cracks resulting in coolant leakage. All of the fatigue failures resulted from conditions or stressors that were not accounted for in the original design analyses. In some cases, it has proven difficult to discover fatigue cracks using conventional inservice inspection methods; several cracks were detected because of leakage. Supplementary monitoring and inspection techniques such as fatigue monitoring, acoustic emission monitoring, and time-of-flight-diffraction ultrasonic testing can be used to assist in identifying susceptible sites, estimating crack growth, and sizing existing fatigue cracks. It is important to identify the root cause of failures because once the stressors and degradation mechanisms are known, changes in operating procedures and designs can be implemented to mitigate future fatigue damage.

Title: Evaluation of damage in metal under high temperature creep by acoustic method.

Author: **Corp. Author:** Perevalov,-S.P.; Permikin,-V.S.

Source: Ehlektricheskie-Stantsii. (May 1992). (no.5). p. 43-47.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** Russian

Category: Inspection methods **ID:** 352

Abstract: Specimens of 12Kh1MF steel were used to investigate the effect of accumulated damage under high temperature creep on the velocity of ultrasonic waves. The specimens possessed various porosity but similar ferritic structure with coagulated carbides along grain boundaries. Ultrasound velocity was found to decrease monotoneously with an increase of damage. The change in velocity depended on the type of waves and occurred different for longuitudinal, lateral and surface waves. Based on investigation results a technique was developed for evaluating damage in pipe line bends by ultrasonic testing.

Title: 3. technical report - compilation and assessment of documentation related to specific questions in view of the further de

Author: Herter,-K.H.; Schuler,-X. **Corp. Author:** Bundesministerium fuer Umwe

Source: Dec 1990. 137 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: LBB justification **ID:** 353

Abstract: The comments made concern partial aspects of the 'leak-before-break' behaviour of peripherally damaged pipelines. In this connection it is essential to point out the prerequisites in long-term operation for ensuring the 'principle of break exclusion', in particular for peripherally damaged pipelines. (orig./DG).

Title: Methodology for addressing erosion/corrosion piping inspection program.

Author: Haramis,-V.G. (Ebasco Services Incorporated, New York, NY (United States)) **Corp. Author:** POWER-GEN '91: 4th internat

Source: Anon.-POWER-GEN '91 conference papers: Volume 11 (Fossil plant retrofit, repowering and fuel conversion) and Volume 12 (Fossil plant performance, availability and improvement). Houston, TX (United States). PennWell Conferences and Exhibitions Co. 1991. 500 p. p. 1971-1986.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Experience/events **ID:** 354

Abstract: Erosion/Corrosion or flow-induced corrosion can be described as an accelerated form of corrosion that attacks carbon steel materials in single and two-phase fluid systems. It occurs in turbulent or fast flowing fluid systems operating between certain temperature and chemistry limits. The theory behind this phenomenon is that fast flowing water or wet steam wears away the protective oxide layer that forms on the steel surface. This leads to continuous dissolution of the underlying metal, resulting in an accelerated form of corrosion. Operation under these conditions leads to continuous pipe thinning. Prolonged operation eventually reduces the wall thickness to less than the minimum required, resulting in failure of the component. At the present time the effects of E/C in carbon steel piping under single-and two-phase flow conditions in both nuclear and fossil power plants are well documented. To address this issue, generating facilities are confronted with the following tasks: (1) identifying susceptible piping systems for evaluation for potential E/C damage, (2) performing appropriate analysis to determine pipe thinning rates and selecting components for inspection, (3) implementing a piping inspection program to monitor design adequacy and safety of the system, and (4) implementing corrective actions to reduce or alleviate the problem. This paper has been prepared to provide, (1) an overview of the above tasks and (2) an approach utilizing currently available technical knowledge.

Title: Shock wave solutions for steam water mixtures in piping systems.

Author: Katze,-D.; Hamm,-J. (Bechtel Civil and Minerals, Inc., Gaithersburg, MD (United States)) **Corp. Author:** 1991 American Society of Mec

Source: Wang,-G.Y.; Shin,-Y.W. (Argonne National Lab., IL (United States)); Moody,-F.J. (GE Nuclear Engineering (United States)). Proceedings of transient thermal-hydraulics and coupled vessel and piping system responses 1991. PVP-Volume 219. New York, NY (United States). American Society of Mechanical Engineers. 1991. 112 p. p. 19-24.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Pressure ripple/water hammer **ID:** 355

Abstract: The pressure change resulting from shock phenomena caused by fluid transients can contribute to damage of equipment or piping. This paper relates the shock pressure rise to a sudden change in speed for a homogeneous steam water mixture flowing in a pipe. Equilibrium conditions are assumed and the conservation equations of mass, momentum and energy are solved across the shock discontinuity. Graphs are included allowing the determination of the maximum change in speed for a given pressure rise. The results are valid from low initial pressures to about 12 MPa.

Title: Submarine pipelines and the North Sea environment.

Author: **Corp. Author:** Haldane,-D.; Paul,-M.A.; Reub

Source: Cairns,-W.J. (International Council for Oil and the Environment, Edinburgh (United Kingdom)) (ed.). North Sea oil and the environment. Developing oil and gas resources, environmental impacts and responses. London (United Kingdom). Elsevier Applied Science. 1992. 722 p. p. 481-522.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Methods/design **ID:** 356

Abstract: The function and design of pipelines for use on the United Kingdom continental shelf are described. Environmental influences which can threaten the integrity of seabed pipelines in the North Sea include hydrodynamic forces due to residual, tidal and wave currents, the nature of seabed sediments and corrosion by seawater. Damage may be caused to pipelines by interaction with vessel anchors and with fishing gear. Special care has to be taken over the selection of the general area for the landfall of a pipeline and the engineering of the installation where the pipeline comes ashore. Trenching and other protection techniques for pipelines are discussed together with hydrostatic testing and commissioning and subsequent inspection, maintenance and repair. (UK).

Title: Contribution to safety assessment of components stressed by high temperature, assuming a crack. Final report.

Author: Roedig,-M.; Pfaffelhuber,-M.; Schubert,-F.; Nickel,-H. **Corp. Author:** Forschungszentrum Juelich Gm

Source: 29 Nov 1991. 119 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** German

Category: Test/analysis **ID:** 357

Abstract: To examine crack growth with superimposed creep and fatigue stresses, fatigue crack growth tests, with stopping times, tests with ramp loading with different duration periods and tests with successive periods of pure creep and fatigue crack growth were carried out. The investigations were done on the iron-based alloy 800 at 700degC. The crack growth in the stopping time and ramp experiments were described with the parameters of pure creep crack growth and fatigue crack growth based on a linear damage accumulation theory. This damage accumulation theory was also verified in experiments on large samples and pipe geometries with artificial faults. The second part of the report is concerned with experiments on samples of stressed or thermally stored sample material. These experiments are intended to clear up in what way the ageing of the material changes the fracture mechanics properties. To examine their transferability, experiments on creep and fatigue crack growth are carried out on pipes stressed in operation. In general, no differences in crack propagation behaviour are found for material before and after storage. (orig.).

Title: Corrosion damages and the relevant protective strategies for LWR nuclear power station.

Author: **Corp. Author:** Tadao-Ishihara (National Rese

Source: Chinese-Journal-of-Nuclear-Science-and-Engineering. (Sep 1991). v. 11(3). p. 226-238.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** Chinese

Category: Research/theoretical **ID:** 358

Abstract: The corrosion damages of metal materials in LWR environment often occur at the contact positions between materials and coolant water or steam at high temperature and high pressure. The main damages are as follows: the SCC on stainless steel pipe of BWR primary coolant system; SCC, wastage, denting and IGA on the nickel alloy tube of PWR steam generator; corrosion fatigue of carbon steel water supply pipe; nodular corrosion and PCI of zirconium alloy fuel cladding; and SCC on the split pins of control rod guide tube. Some protective strategies are presented. Finally, the paper points out that developing the long-time endurance technology, quantitatively grasping the deteriorate level of the structural materials of LWR and establishing remaining life assessment technology are the important aspects for future researches.

Title: Thermal stratification affects the operation of both nuclear and fossil power plants.

Author: Bain,-R.A.; Van-Duyne,-D.A. (Stone and Webster Engineering Corp. (United States)); Testa,-M.F. (Duquesne Light Co. (United States)) **Corp. Author:** 54. annual American power co

Source: Proceedings-of-the-American-Power-Conference. (1992). v. 54(1). p. 17-22.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Experience/events **ID:** 359

Abstract: Because of its potential impact upon plant operation, thermal stratification is a concern for both nuclear and fossil power stations. Over the past several years, most operating nuclear power plants have experienced this phenomenon in their major systems. Thermal stratification has been observed in horizontal piping runs, especially when there are low flow rates of different water temperatures in feedwater systems of both boiling water reactor (BWR) and pressurizer surge line piping, safety injection, and residual heat removal piping of PWR plants. Thermal stratification is a gravity-induced conditions, which occurs when fluid in the piping stratifies as a result of differences in thermal density of the low flowing fluid. It usually results in significant unanticipated thermal displacements of the piping, and therefore, can affect the structural integrity of the piping and its supports. Vertical piping movements of up to 4 inches have been measured on 14- to 28-inch diameter piping configurations. These unexpected displacement scan damage pipe supports if they are not considered in the design. Operating experience and data are available to help eliminate or alleviate operating conditions that cause thermal stratification or its related effects.

Title: Nucleate boiling pressure drop in an annulus: Book 6.

Author: **Corp. Author:** Westinghouse Savannah River

Source: Nov 1992. 1131 p. .FUNDING ORGANIZATION: USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Analysis of break effects **ID:** 360

Abstract: The LOCA scenario considered for SRS involves a double-ended break of a primary coolant pipe in the reactor. During a LOCA, the flow through portions of the reactor may reverse direction or be greatly reduced, depending upon the location of the break. The reduced flow rate of coolant (D sub 2 O) through the fuel assembly channels of the reactor -- downflow in this situation -- can lead to boiling and to the potential for flow instabilities which may cause some of the fuel assembly channels to overheat and melt. That situation is to be avoided. The experimental approach is to provide a test annulus which simulates geometry, materials, and flow conditions in a Mark-22 fuel assembly (Coolant Channel 3) to the extent possible. The annulus has a full-scale geometry, and in fat uses SRL dummy hardware for the inner annulus wall in the ribbed geometry. The materials aluminum. The annulus is uniformly heated in the axial direction, but the circumferential heat flux can be varied to provide "power tilt" or asymmetric heating of the inner and outer annulus walls. The test facility uses H sub 2 O rather than D sub 2 O, but it includes the effects of dissolved helium gas present in the reactor. The key analysis approaches are: To compare the minima in the measured demand curves with analytical criteria, in particular the Saha-Zuber (1974) model; and to compare the pressure and temperature as a function of length in the annulus with an integral model for flow boiling in a heated channel. This document consists of a summary of temperature measurements to include recorded minima, maxima, averages and standard deviations.

Title: Nucleate boiling pressure drop in an annulus: Book 5.

Author: **Corp. Author:** Westinghouse Savannah River

Source: Nov 1992. 67 p. .FUNDING ORGANIZATION: USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Analysis of break effects **ID:** 361

Abstract: The application of the work described in this report is the production reactors at the Savannah River Site, and the context is nuclear reactor safety. The Loss of Coolant Accident (LOCA) scenario considered involves a double-ended break of a primary coolant pipe in the reactor. During a LOCA, the flow through portions of the reactor may reverse direction or be greatly reduced, depending upon the location of the break. The reduced flow rate of coolant (D sub 2 O) through the fuel assembly channels of the reactor -- downflow in this situation -- can lead to boiling and to the potential for flow instabilities which may cause some of the fuel assembly channels to overheat and melt. That situation is to be avoided. The experimental approach is to provide a test annulus which simulates geometry, materials, and flow conditions in a Mark-22 fuel assembly (Coolant Channel 3) to the extent possible. The key analysis approaches are: To compare the minima in the measured demand curves with analytical criteria, in particular the Saha-Zuber (1974) model; and to compare the pressure and temperature as a function of length in the annulus with an integral model for flow boiling in a heated channel. Nineteen test series and a total of 178 tests were performed. Testing addressed the effects of: Heat flux; pressure; helium gas; power tilt; ribs; asymmetric heat flux. This document consists solely of the plato file index from 11/87 to 11/90.

Title: Nucleate boiling pressure drop in an annulus: Book 3.

Author: Block,-J.A.; Crowley,-C.; Dolan,-F.X.; Sam,-R.G.; Stoedefalke,-B.H. **Corp. Author:** Westinghouse Savannah River

Source: Nov 1992. 327 p. .FUNDING ORGANIZATION: USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Analysis of break effects **ID:** 362

Abstract: The LOCA scenario considered for SRS involves a DEGB of a primary coolant pipe in the reactor. During a LOCA, the flow through portions of the reactor may reverse direction or be greatly reduced, depending upon the location of the break. The reduced flow rate of coolant (D sub 2 O) through the fuel assembly channels of the reactor -- downflow in this situation -- can lead to boiling and to the potential for flow instabilities which may cause some of the fuel assembly channels to overheat and melt. That situation is to be avoided. The experimental approach is to provide a test annulus which simulates geometry, materials, and flow conditions in a Mark-22 fuel assembly (Coolant Channel 3) to the extent possible. The annulus has a full-scale geometry, and in fat uses SRL dummy hardware for the inner annulus wall in the ribbed geometry. The materials aluminum. The annulus is uniformly heated in the axial direction, but the circumferential heat flux can be varied to provide "power tilt" or asymmetric heating of the inner and outer annulus walls. The test facility uses H sub 2 O rather than D sub 2 O, but it includes the effects of dissolved helium gas present in the reactor. The key analysis approaches are: To compare the minima in the measured demand curves with analytical criteria, in particular the Saha-Zuber (1974) model; and to compare the pressure and temperature as a function of length in the annulus with an integral model for flow boiling in a heated channel. This document consists of data plots and summary files of temperature measurements.

Title: Schedule of experimental and calculation work for the assessment of steam and feedwater pipes at the Dukovany NPP,

Author: Zdarek,-J.; Ruscak,-M. **Corp. Author:** Ustav Jaderneho Vyzkumu a.s.,

Source: Jun 1993. 28 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** Czech

Category: LBB justification **ID:** 363

Abstract: Based on Decree No. 6/91 of the State Surveillance over Nuclear Safety, Czechoslovak Atomic Energy Commission, a project is being implemented, aimed at obtaining the leak-before-break statute for the main circulation pipe and the pressurizer pipe. The Decree also demands that the effect of seismic stress (at least up to level 7 on the MSK64 scale) on the integrity of safety-related pipe systems at the Dukovany NPP be assessed. Within the planned program of experiments and calculations, a complex assessment of integrity of the remaining safety-related pipe systems, the steam piping and the feedwater piping, will be accomplished at the leak-before-break methodological level. A continuation of the assessment of the primary circuit, the program will fully satisfy the requirements laid down by the State Surveillance. The results will document an extremely low probability of failure of the primary and secondary circuits and thus the optimal dimensioning of the adapted containment, emergency tanks and other innovations. The report sets forth a schedule of the experiments and calculations and their justification with respect to the safety and economic outcome. Links to other projects, qualification and competence of the persons involved in and responsible for the project, and the extent and cost of the project including its schedule are also given. (author). 11 figs., 13 refs.

Title: Analysis of the NPP-V1 primary circuit fast cooldown.

Author: Filo,-J.; Bazso,-Z.; Vranka,-L. **Corp. Author:** International workshop on WW

Source: International Atomic Energy Agency, Vienna (Austria); Nuclear Regulatory Authority, Bratislava (Slovakia). International workshop on WWER-440 reactor pressure vessel embrittlement and annealing. Working material. Scientific presentations. 1994. 366 p. p. 241-278.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Analysis of break effects **ID:** 364

Abstract: Results of thermal-hydraulic calculations of the NPP-V1 primary circuit fast cooldown during small leakage through openings of diameter 20, 32 and 50 mm as well as analyses of cooldown following the steam pipeline break at nominal and null reactor power are given in this paper. 4 refs, 24 figs, 1 tab.

Title: Leak-before-break behaviour of nuclear piping systems.

Author: Bartholome,-G.; Wellein,-R. (Siemens AG Unternehmensbereich KWU, Erlangen (Germany)) **Corp. Author:** IAEA specialist's meeting on th

Source: Gillemot,-F.; Uri,-G. (eds.). International Atomic Energy Agency, Vienna (Austria). IAEA specialist's meeting on the integrity of pressure components of reactor systems. 1992. 272 p. p. 112-117.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: LBB justification **ID:** 365

Abstract: The general concept for break preclusion of nuclear piping systems in the FRG consists of two main prerequisites: Basic safety; independent redundancies. The leak-before-break behaviour is open of these redundancies and will be verified by fracture mechanics. The following items have to be evaluated: The growth of detected and postulated defects must be negligible in one life time of the plant; the growth behaviour beyond design (i.e. multiple load collectives are taken into account) leads to a stable leak; This leakage of the piping must be detected by an adequate leak detection system long before the critical defect size is reached. The fracture mechanics calculations concerning growth and instability of the relevant defects and corresponding leakage areas are described in more detail. The leak-before-break behaviour is shown for two examples of nuclear piping systems in pressurized water reactors: main coolant line of SIEMENS-PWR 1300 MW (ferritic material, diameter 800 mm); surge line of Russian WWER 440 (austenitic material, diameter 250 mm). The main results are given taking into account the relevant leak detection possibilities. (author). 9 refs, 9 figs.

Title: Three loss-of-coolant accidents in the first wall/blanket cooling system of the SEAFP alternative plant model. SEAFP

Author: Komen,-E.M.J.; Koning,-H. **Corp. Author:** Netherlands Energy Research F

Source: Mar 1994. 104 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Analysis of break effects **ID:** 366

Abstract: This report presents the thermal-hydraulic analysis of three ex-vessel Loss-of-Coolant Accidents (LOCAs) in the first wall/blanket cooling system of the alternative SEAFP reactor design. The LOCAs are caused by a rupture of the pump suction pipe, an inlet header, and an outlet header respectively. The ex-vessel LOCAs considered result from a rupture of a cooling pipe located outside the plasma vessel. In order to determine the worst case LOCA conditions, no plasma shutdown and no other counteractions have been assumed. The analyses have been performed using the thermal-hydraulic system analysis code RELAP5/MOD3. Special attention has been paid to the transient thermal-hydraulic behaviour of the cooling system and the temperature development in the first wall and blanket. For the LOCAs considered, the temperature development in the first wall is more critical than the temperature development in the blanket. In the LOCA caused by a rupture of the pump suction pipe, melting at the midplane of the outboard first wall starts about 56 s after break initiation. In the LOCA caused by a rupture of an inlet header, melting at the midplane of the outboard first wall starts about 62 s after break initiation, whereas in the LOCA caused by a rupture of an outlet header melting at the midplane of the outboard first wall starts about 74 s after break initiation. (orig.).

Title: Metallurgical evaluation of stress corrosion cracking in large diameter piping.

Author: Wheeler,-D.A.; Rawl,-D.E. Jr.; Louthan,-M.R. Jr. (Westinghouse Savannah River Co., Aiken, SC (United States). Savannah River Lab.) **Corp. Author:**

Source: Materials-Characterization. (Jan 1994). v. 32(1). p. 25-33.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Test/analysis **ID:** 367

Abstract: Ultrasonic testing (UT) of stainless-steel piping in the primary coolant water system of Savannah River Site (SRS) reactors indicates the presence of short, partly through-wall stress corrosion cracks in the heat-affected zone of approximately 7% of the circumferential pipe welds. These cracks are thought to develop by intergranular nucleation and mixed mode propagation. Metallographic evaluations have confirmed the UT indications of crack size and provided evidence that crack growth involved the accumulation of chloride ions inside the growing crack. It is postulated that the development of an oxygen depletion cell inside the crack results in the migration of chloride ions to the crack tip to balance the accumulation of positively charged metallic ions. The results of this metallurgical evaluation, combined with structural assessments of system integrity, support the existence of leak-before-break conditions in the SRS reactor piping system.

Title: On the modelling of leak rates through cracks in pipes and tubes.

Author: Osamusali,-S.I.; Crentsil,-K.; Chu,-R.Y.; Luxat,-J.C. (Ontario Hydro, Toronto, ON (Canada)) **Corp. Author:** 32. Annual conference of the C

Source: Canadian Nuclear Society, Toronto, ON (Canada). Proceedings of the 13. annual conference of the Canadian Nuclear Society. V. 2. 1992. 762 p. [26 p.].

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Methods **ID:** 368

Abstract: A leak rate code LEAK-RATE Version 1.0 has been developed to predict two-phase critical mass fluxes, exit pressures, and pressure profiles for cracks, which form an integral part of leak-before-break analysis of pressurized reactor components such as pipes and headers. The code can also be used to calculate steam generator tube leakages. The code uses the homogeneous frozen model for determining critical mass flux, with effects of friction accounted for within the crack. The code's predictions have been compared with an extensive experimental database, and also benchmarked against similar international codes. The code predicted the leak rate and exit pressures to within +-25% of experimental data, which represents reasonably good agreement for leak rate predictions. The predicted pressure profiles within the crack agreed well with experimental data and yielded the same trend as the experimental observations. 13 refs., 20 figs., 1 tab.

Title: Calculated data as a basis for experiments on the branch pipe leading to the pressurizer, performed within application

Author: Lauerova,-D. **Corp. Author:** Ustav Jaderneho Vyzkumu CS

Source: Jul 1993. 20 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** Czech

Category: LBB justification **ID:** 369

Abstract: Calculations serving as a basis for experimental tests within the leak-before-break analysis of the primary circuit branch leading to the pressurizer are described. The crack length for the decisive measurable coolant leak was determined by the LEAKH code to be 320 mm, and the bending stress of the pipeline and the branch was calculated. The mathematical model is briefly described. (J.B.). 1 tab., 7 figs., 3 refs.

Title: A high temperature leak before break approach for pipework.

Author: Ainsworth,-R.A.; Chivers,-T.C. (Nuclear Electric plc, Berkeley (United Kingdom). Berkeley Technology Centre) **Corp. Author:** Piping engineering and operati

Source: Institution of Mechanical Engineers, London (United Kingdom); Institution of Chemical Engineers, London (United Kingdom). Piping engineering and operation. Proceedings. London (United Kingdom). Institution of Mechanical Engineers. 1993. 202 p. p. 123-134.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: LBB justification **ID:** 370

Abstract: In recent years, procedures have become available for assessing Leak-before-Break (LbB) in pipework operating at low temperatures, and also for assessing the behaviour of defects in materials operating in the creep regime. However, a major deficiency has been the absence of a methodology for the assessment of LbB in high temperature plant. This paper describes recent work which has combined the existing approaches to lay the foundation for a high temperature LbB procedure for steam pipework. The procedure developed is described as a set of steps involving: definition of the pipework stresses; calculation of critical crack sizes; characterisation of defects; calculation of creep crack growth rates; calculation of crack opening areas and associated leak rates; and selection of leak detection systems. The importance of sensitivity studies is highlighted to ensure that any safety assessment is not compromised by uncertainties in the data employed. (Author).

Title: The effect of tributary pipe breaks on the core support barrel shell responses.

Author: Jhung,-Myung-Jo (Korea Atomic Energy Research Institute, Taejon (Korea, Republic of)); Hwang,-Won-Gul (Chonnam National University, Chonnam (Korea, Republic of)) **Corp. Author:**

Source: Journal-of-the-Korean-Nuclear-Society. (Jun 1993). v.25 (2). p. 204-214.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Analysis of break effects **ID:** 371

Abstract: Work on fracture mechanics has provided a technical bases for elimination of main coolant loop double ended guillotine breaks from the structural design basis of reactor coolant system. Without main coolant loop pipe breaks, the tributary pipe breaks must be considered as design bases until further fracture mechanics work could eliminate some of these breaks from design consideration. This paper determines the core support barrel shell responses for the 3 inch pressurizer spray line nozzle break which is expected to be the only inlet break remaining in the primary side after leak-before-break evaluation is extended to smaller size pipes in the near future. The responses are compared with those due to 14 inch safety infection nozzle break and main coolant loop pipe break. The results show that, when the leak-before-break concept is applied to the primary side piping systems with a diameter of 10 inches or over, the core support barrel shell responses due to pipe breaks in the primary side are negligible for the faulted condition design. (Author).

Title: Structural integrity of whipping pipes following a postulated circumferential break - a contribution to determining strai

Author: Charalambus,-B. (Siemens AG, Bereich Energieerzeugung (KWU), Erlangen (Germany)); Labes,-M. (Siemens AG, Offenbach (Germany)) **Corp. Author:**

Source: Nuclear-Engineering-and-Design. (Oct 1993). v. 144(1). p. 91-99.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Research/theoretical **ID:** 372

Abstract: It is postulated that a break of a thin-walled pipe does not cause a subsequent break in the pipe in the vicinity of a plastic hinge even when the wall is weakened by a 60 circumferential crack of a depth of 30% of the wall thickness on the tension side. This pipe behavior is the result of plastic buckling in the compression side and applies to pipes of diameter-to-thickness ratio larger than 20. For this type of pipe, the axial strains decrease with increasing diameter-to-thickness ratio in the tension side. As the pipe is only loaded in one direction, there is no cyclic behavior that can trigger a subsequent break. (orig.).

Title: Summary report on the leak-before break program in France.

Author: Gilles,-P.; Bhandari,-S. (FRAMATOME, Paris (France)); Moulin,-D.; Petit,-M. (DEMT-CEA, Saclay (France)); Faidy,-C.; Le-Delliou,-P. (EDF-SEPTEN, Villeurbanne (France)) **Corp. Author:** 1993 pressure vessel and pipin

Source: Garud,-Y.S. (ed.). Creep, fatigue, flaw evaluation, and leak-before-break assessment. New York, NY (United States). American Society of Mechanical Engineers. 1993. 301 p. p. 195-204.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: LBB justification **ID:** 373

Abstract: The French regulations at the present stage do not require the application of leak-before-break concept on French pressurized water reactors (PWRs). However, an extensive research and development program has been carried out for a number of years in France, particularly related to the leak-before-break concept, in the frame work of the PWR piping integrity. The work is performed under three-party agreement between CEA, EDF and FRAMATOME, in strong connection with IPIRG (International Piping Integrity Research Group). The French program includes static and dynamic tests, finite element code development and validation, as well as specific work on engineering methods. The objective here is to describe briefly the main aspects of the French program along with the principle conclusions obtained through the experiments performed, Finite Element analyses conducted and the application of engineering methods on various pipe configurations. Finally the paper presents recommendations on the use of fracture assessment procedures and insists on the need of further work on improved J estimation schemes, on cyclic effects on fracture resistance and on stress classification under a seismic type of loading.

Title: Recent developments in leak-before-break technology.

Author: Quinones,-D.F.; Hardin,-T.C. (Robert L. Cloud and Associates, Inc., Berkeley, CA (United States)) **Corp. Author:** 1993 pressure vessels and pipin

Source: Bamford,-W.H. (ed.). Service experience and life management: Nuclear, fossil, and petrochemical plants. New York, NY (United States). American Society of Mechanical Engineers. 1993. 351 p. p. 79-87.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: LBB justification **ID:** 374

Abstract: Guidelines for performing leak-before-break (LBB) analyses in high energy piping were implemented by the US NRC in 1984 and modified in 1987. Since then, fracture and fluid mechanics research and LBB application experience in LWRs have identified problem areas and issues not explicitly covered by regulations. Specific examples of inadequately addressed issues are the tieing of LWR service failure experience to LBB analyses, the role of torsional loads at potential break locations for leakage and stability analyses, LBB applications to ferritic lines which may exhibit dynamic strain aging, thermal stratification/striping, the loss of fracture toughness (thermal aging or embrittlement) in cast and welded austenitic stainless steels, dynamic and cyclical effects on fracture toughness, and LBB analysis at multiple break locations. Several years of LWR application experience, using the LBB methodology provided by SRP 3.6.3 and NUREG 1061 Vol. 3, have also identified some difficulties with the regulatory methods and acceptance criteria. The purpose of this paper is to assess how these recent developments regulatory guidance.

Title: Full scale dynamic fracture testing of degraded pipe.

Author: Poole,-A.B.; Battiste,-R.L.; Clinard,-J.A. (Oak Ridge National Lab., TN (United States)) **Corp. Author:** 1993 pressure vessels and pipin

Source: Bees,-W.J. (ed.). Design analysis, robust methods, and stress classification. New York, NY (United States). American Society of Mechanical Engineers. 1993. 341 p. p. 109-128.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 375

Abstract: ORNL has completed a major task for the DOE in the demonstration that the primary piping of the New Production Reactor-Heavy Water Reactor (NPR-HWR), with its relatively moderate temperature and pressure, should not suffer an instantaneous DEGB under design basis loadings and conditions. This report provides the results of the second pipe test series in this program. In this series, a section of aged 304 stainless steel piping was loaded in fully reversed bending cycles at large loads. An initial flaw size was based on leakage flow testing at pressure and temperature for the NPR-HWR. The detectable leakage flaw was cycled in bending at large loads for 40 cycles. These tests results are discussed in this report. The results of the testing were reviewed by a special Piping Integrity Review Group (PIRG) established by DOE. Provided the caveats cited in the Applicability of Results section are in force, then PIRG agreed that the following conclusions are applicable: for DOE low pressure (<= 1.72 MPa) low temperature (approx 100C) reactors with austenitic stainless steel piping, the DEGB should not be a design basis condition; for DOE low pressure low temperature reactors subject to IGSCC the analytic results with a 360 degree deep crack indicate that instantaneous DEGB is highly improbable; PIRG believes further analysis would confirm that the instantaneous DEGB need not be a design basis condition; for Advanced Light Water Reactors (ALWRs), information is inadequate to justify relaxation of the current DEGB requirements other than that already provided by 10CFR50 GDC-4 regarding leak-before-break (LBB).

Title: Heat exchanger, head and shell acceptance criteria.

Author: Lam,-P.S.; Sindelar,-R.L. **Corp. Author:** Westinghouse Savannah River

Source: Sep 1992. 47 p. .FUNDING ORGANIZATION: USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Research/theoretical **ID:** 376

Abstract: Instability of postulated flaws in the head component of the heat exchanger could not produce a large break, equivalent to a DEGB in the PWS piping, due to the configuration of the head and restraint provided by the staybolts. Rather, leakage from throughwall flaws in the head would increase with flaw length with finite leakage areas that are bounded by a post-instability flaw configuration. Postulated flaws at instability in the shell of the heat exchanger or in the cooling water nozzles could produce a large break in the Cooling Water System (CWS) pressure boundary. An initial analysis of flaw stability for postulated flaws in the heat exchanger head was performed in January 1992. This present report updates that analysis and, additionally, provides acceptable flaw configurations to maintain defined structural or safety margins against flaw instability of the external pressure boundary components of the heat exchanger, namely the head, shell, and cooling water nozzles. Structural and flaw stability analyses of the heat exchanger tubes, the internal pressure boundary of the heat exchangers or interface boundary between the PWS and CWS, were previously completed in February 1992 as part of the heat exchanger restart evaluation and are not covered in this report.

Title: Heat exchanger staybolt acceptance criteria. Task number: 90-058-1.

Author: Lam,-P.S.; Sindelar,-R.L.; Barnes,-D.M. **Corp. Author:** Westinghouse Savannah River

Source: Feb 1992. 53 p. .FUNDING ORGANIZATION: USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Analysis of break effects **ID:** 377

Abstract: The structural integrity demonstration of the primary coolant piping system includes evaluating the structural capacity of each component against a large break or equivalent DEGBreak. A large break at the inlet or outlet heads of the H/X would occur if the restraint members of the heads become inactive. The structural integrity of the heads is demonstrated by showing the redundant capacity of the staybolts to restrain the head at design conditions and under seismic loadings. The SRS H/X head is attached to the tubesheet by 84 staybolts. Access to the staybolts is limited due to a welded seal cap over the staybolts. An UT-technique to provide an in-situ examination of the staybolts has recently been developed at SRS. Examination of the staybolts will be performed to ensure their service condition and configuration is within acceptance limits. An acceptance criteria methodology has been developed to disposition flaws reported in the staybolt inspections while ensuring adequate restraint capacity of the staybolts to maintain integrity of the heat exchanger heads against collapse. The methodology includes an approach for the baseline and periodic inspections of the staybolts. The H/X head is analyzed with a 3-D finite element model. The restraint provided by the staybolts is evaluated for several postulated cases of inactive or missing staybolts. Evaluation of specific, inactive staybolt configurations based on the UT results can be performed with the finite element model and fracture methodology in this report.

Title: Operation of Finnish nuclear power plants. Quarterly report 1 st quarter, 1993.

Author: Tossavainen,-K. (ed.) **Corp. Author:** Finnish Centre for Radiation an

Source: Sep 1993. 23 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 378

Abstract: Quarterly reports on the operation of Finnish nuclear power plants describe events and observations, relating to nuclear safety and radiation protection which the Finnish Centre for Radiation and Nuclear Safety considers safety significant. Safety-enhancing modifications at the nuclear power plants and issues relating to the use of nuclear energy which are of general interest are also reported. The reports include a summary of the radiation safety of plant personnel and the environment, as well as tabulated data on the production and load factors of the plants. In the first quarter of 1993, a primary feedwater system pipe break occurred at Loviisa 2, in a section of piping after a feedwater pump. The break was erosion-corrosion induced. Repairs and inspections interrupted power generation for seven days. On the International Nuclear Event Scale the event is classified as a level 2 incident. Other events in the first quarter of 1993 had no bearing on nuclear safety and radiation protection.

Title: Application of leak-before-break concept to design and safety of PWR piping.

Author: Kiselyov,-V.A.; Rivkin,-E.Yu.; Sudakov,-A.V. (Research and Development Inst. of Power Engineering, Moscow (Russian Federation))

Corp. Author: ANP'92: international conferen

Source: Oka,-Y.; Koshizuka,-S. (comps.) (Tokyo Univ. (Japan)). Atomic Energy Society of Japan, Tokyo (Japan). ANP'92 international conference on design and safety of advanced nuclear power plants. Tokyo (Japan). Atomic Energy Society of Japan. 1992. [2182 p.]. v. 2 p. P8.2/1-P8.2/6. Composed of four volumes.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: LBB justification **ID:** 379

Abstract: The general requirements, criteria and methodology of the 'leak-before-break' (LBB) or 'break exclusion' concept for Pressurized Water Reactors (PWRs) piping in Russia are presented. The demonstration of the applicability of LBB primary circulated loop piping in PWRs coolant system is carried out in two phases. To demonstrate that, first, a leak from a through-wall crack can be detected for crack length smaller then the critical crack size, and, second, the risk of developing a through-wall flaw is small during the planned lifetime of the plant and that any realistic end-of-life defect would be small enough not to affect the integrity of the structure. The experimental results obtained to data are very encouraging and show the applicability of the LBB concept to the actual design. (author).

Title: Accidents associated with oil and gas operations: Outer continental shelf, 1956-1990. Final report.

Author: Tracy,-L.M.

Corp. Author: Minerals Management Service,

Source: Oct 1992. 311 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Experience/events **ID:** 380

Abstract: The report is a compilation of descriptions of all blowouts, explosions and fires, pipeline breaks or leaks, significant pollution incidents, and major accidents that occurred on federally leased offshore lands from 1956 through 1990. The report identifies accidents by area, block number, lease number, platform number, well number, and operator. It describes the type of accident, corrective action taken, and the amount of pollution. It provides figures on fatalities, injuries, and property and environmental damage.

Title: Short Cracks in Piping and Piping Welds. Semiannual report, October 1991--March 1992: Volume 2, No. 2.

Author: Wilkowski,-G.M.; Brust,-F.; Francini,-R.; Ghadiali,-N.; Kilinski,-T.; Krishnaswamy,-P.; Landow,-M.; Marschall,-C.W.; Rahman,-S.; Scott,-P. (Battelle, Columbus, OH (United States))

Corp. Author: Nuclear Regulatory Commissio

Source: May 1993. 121 p. .: Nuclear Regulatory Commission, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 381

Abstract: This is the fourth semiannual report of the US Nuclear Regulatory Commission's Short Cracks in Piping and Piping Welds research program. This 4-Year program began in March 1990. The overall objective of this program is to verify and improve fracture analyses for circumferentially cracked large-diameter nuclear piping with crack sizes typically used in leak-before-break analyses or inservice flaw evaluations. Progress during this reporting period involved: (1) completing two through-wall-cracked pipe experiments and supplementary material property data, (2) an internal circumferential surface-cracked pipe experiment was completed which showed that the R/t effects on the Net-Section-Collapse Predicted loads for surface-cracked pipe to be independent of crack size, (3) the anisotropy investigation showed that pipe dimensions may be as important in determining the out-of-plane crack growth angle as the anisotropy of the toughness, (4) we initiated a probabilistic analysis of LBB to assess the potential changes in the leakage detection criteria in NRC Reg Guide 1.45, and (5) other efforts involved a sensitivity study on the effect of thermal aging of cast stainless steel on the moment-carrying capacity of the pipe as a function of time.

Title: A sensitivity study in probabilistic fracture mechanics analysis of light water reactor carbon steel pipe.

Author: Fujioka,-T.; Kashima,-K. (Central Research Inst. of Electric Power Industry, Komae, Tokyo (Japan). Komae Research Lab.) **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1992). v. 52(3). p. 403-416.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Methods **ID:** 382

Abstract: Through the efforts in leak-before-break research for light water reactor pipings based on deterministic fracture mechanics analysis, simple models which evaluate pipe fracture mechanics analysis, simple models which evaluate pipe fracture behaviour are being established. Using these models it is also becoming possible to apply probabilistic fracture mechanics analysis. This paper describes an example of such an analysis, using these proposed models. Since the authors' interests are in the range of uncertainly of the calculated failure probability, the effects of changes in the input parameters or the analytical conditions are also estimated by a sensitivity analysis. The results show that the calculated failure probability may be influenced significantly by changes in parameters concerning initial crack size distributions, and that effects due to a change in the leak detection model may appear after long operation of the plant. (author).

Title: The development of new analysis procedures for reactor internals under pipe breaks.

Author: Song,-Heuy-Gap; Jhung,-Myung-Jo; Chang,-Sang-Gyun; Lee,-Gyu-Man (Korea Atomic Energy Res. Inst., Taejon (Korea, Republic of)) **Corp. Author:** Korea Atomic Energy Research

Source: Apr 1993. 81 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Analysis of break effects **ID:** 383

Abstract: This study investigates the horizontal responses of the reactor internals due to a 14 inch safety injection nozzle break which is expected to cause the largest loads of the branch line pipe breaks defined for the YGN 3 and 4. It examines the effects of two forcing terms, RV motions and internals hydraulic loads, and suggests new procedure which can be used for the tributary pipe break analysis. The analysis result confirms the applicability of suggested procedure to a small size tributary pipe break analysis. Also, this study calculates the horizontal responses of the reactor internals due to a 3 inch pressurizer spray line nozzle break which is the only one remaining in the primary side after leak-before-break evaluation, and secondary side pipe breaks such as main steam line and economizer feedwater line. The responses are compared with those of safe shutdown earthquake(SSE) to show that SSE loads with a conservative margin may be used for the pipe break loads in the preliminary design. (Author).

Title: Comparison of effluent and inlet header breaks for an SRS reactor LOPA.

Author: Paul,-P.K.; Barbour,-K.L.; Herman,-D.T. (Westinghouse Savannah River Co., Aiken, SC (United States)) **Corp. Author:** Joint American Nuclear Societ

Source: Transactions-of-the-American-Nuclear-Society. (1992). v. 66. p. 324-325.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Analysis of break effects **ID:** 384

Abstract: The loss-of-pumping accident (LOPA) is a design-basis accident for Savannah River Site (SRS) reactors. The LOPA is defined as a double-ended guillotine break in a secondary cooling water pipe. The secondary cooling line break is termed inlet or effluent depending on break location. Upon break detection, the emergency shutdown procedure begins, the reactor scrams, the secondary cooling pump motors trip, the primary cooling pump alternating-current motors switch off, and the direct-current motor drive engages. Secondary cooling gravity flow continues flooding the building after the secondary cooling pumps are off. The emergency cooling system (ECS) activates before the dc motors flood out. Break detection time, header flooding rate, and flooding locations are different for the inlet and effluent header breaks because of different break locations. Inlet and effluent header break primary coolant temperature transients differ because primary and secondary cooling pumps continue during a break detection and reactor scram time delay for the effluent header case, whereas the pumps trip off almost immediately for the inlet header case. Design-basis accident reactor core power limits are calculated for both the inlet and effluent header breaks.

Title: Leak before break application on the primary piping of the WWER type nuclear power plants.

Author: Zdarek,-J.; Pecinka,-L. (Nuclear Research Inst., Rez (Czechoslovakia)); Palyza,-J. (Piping and Valve Research Inst., Modrany (Czechoslovakia)); Suchanek,-M. (State Research Inst. of Machine Design, Bechovice (Czechoslovakia))

Corp. Author: 7. international conference on p

Source: Vereinigung der Technischen Ueberwachungsvereine e.V., Essen (Germany). Pressure vessel technology. Proceedings. Vol. 2. Materials (2), manufacturing, quality. 1992. 613 p. p. 1355-1365.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: LBB justification **ID:** 385

Abstract: An upgrading programme is well underway for the NPP Jaslovske Bohunice and will be completed in 1992. The leak before break technology (LBB) plays an important part and a detailed project was set up for application to all the safety related piping. The first results are presented included a seismic margin assessment, detailed evaluation of stress state in critical sections, fracture and corrosion database. Full scale experiments with through-wall cracks in critical heterogeneous welds and T-type welds in elbow are well underway. The scope and initial results of these experiments are presented. The LBB Project has support from WANO, the IAEA and International Assessment Groups as well as from the Regulatory body in the CSFR and the utilities. The results could well be of interest to all V-230 type reactors as well as to the more advanced V-213 type reactors. (orig.).

Title: Dynamic analysis of the reactor core for pipe break and seismic excitations.

Author: Jhung,-M.J.; Park,-K.B.; Sohn,-G.H. (Mechanical Design Dept., Korea Atomic Energy Research Inst., Taejon (Korea, Republic of))

Corp. Author: 7. international conference on p

Source: Vereinigung der Technischen Ueberwachungsvereine e.V., Essen (Germany). Pressure vessels technology. Proceedings. Vol. 1. Design, analysis, materials (1). 1992. 857 p. p. 59-68.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Analysis of break effects **ID:** 386

Abstract: This paper investigates the lateral responses of the reactor core to main steam line and economizer feedwater line breaks in the secondary side. The tributary pipe breaks in lieu of main coolant loop breaks are considered because leak-before-break methodology has provided a technical basis for the elimination of double ended guillotine breaks of all high energy piping systems with a diameter of 10 inches or over in the primary side. This paper also calculates the lateral responses of the reactor core to the motions induced from safe shutdown earthquake and operating basis earthquake. The dynamic responses such as fuel assembly shear force, bending moment and displacement, and spacer grid impact loads are carefully investigated. Also, reported in this paper are the response characteristics of each pipe break and seismic excitation. (orig.).

Title: Scaling of the accuracy of the Relap5/ mod2 code.

Author: Bovalini,-R.; D'Auria,-F. (Dipt. di Costruzioni, Meccaniche e Nucleari, Univ. Pisa (Italy))

Corp. Author:

Source: Nuclear-Engineering-and-Design. (Feb 1993). v. 139(2). p. 187-203.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 387

Abstract: This paper presents an attempt to derive uncertainty values in the prediction of BWR and PWR transient scenarios. The small break LOCA counterpart tests performed in the BWR simulators Piper-one, First and Rosa-III, and natural circulation experiments performed in the PWR simulators Lobi, Spes and Lstf, constitute the basis of the activity. The application of Relap5/mod2 to the analyses of the above experiments, the evaluation of the comparison between predicted results and measured data, and the calculation of the BWR and PWR plants scenarios, were fundamental in achieving the proposed goal. The main result of the activity is constituted by the development of a methodology suitable for deriving uncertainty values of code calculations. The values reported for the uncertainty should be considered as the result of a demonstrative pilot application of the methodology. (orig.).

Title: Evaluation of LBB in piping considering multiple fatigue crack growth.

Author: Shibata,-Katsuyuki (Japan Atomic Energy Research Inst., Tokai, Ibaraki (Japan). Tokai Research Establishment) **Corp. Author:**

Source: Nippon-Kikai-Gakkai-Ronbunshu,-A-Hen. (Aug 1992). v. 58(552). p. 1347-1352.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** Japanese

Category: LBB justification **ID:** 388

Abstract: Through the recent development of fracture mechanics methodology in piping analysis, the LBB concept has been recognized as applicable to the safety design of LWR primary circuit piping. An important subject still remaining in the LBB (Leak-Before-Break) analysis is the consideration of multiple fatigue crack growth and development in crack geometry in piping. In the evaluation procedure currently employed, a simple crack geometry is assumed to judge whether the crack is leak-detectable or not. The paper describes a LBB evaluation procedure in which the growth of multiple fatigue cracks is considered. Two criteria, i.e., critical condition of crack coalescence preceding the penetration of wall thickness and evaluation of crack configuration at the penetration, are introduced to account for cracks that coalesce and develop to the largest possible through-wall single crack at the onset of penetration. The paper also presents the results of a case study of PLR piping of BWR using the above procedure. It is shown that LBB can be justified in PLR piping of 4 inches or greater in diameter. (author).

Title: Reactor Materials Program probability of indirectly--induced failure of L and P reactor process water piping.

Author: Daugherty,-W.L. **Corp. Author:** Du Pont de Nemours (E.I.) and

Source: 11 Mar 1988. 214 p. : USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Damage probability **ID:** 389

Abstract: The design basis accident for SRRs is the abrupt DEGB of a large process water pipe. This accident is not considered credible in light of the low applied stresses and the inherent ductility of the piping material. The Reactor Materials Program was initiated to provide the technical basis for an alternate credible design basis accident. One aspect of this work is to determine the probability of the DEGB; to show that in addition to being incredible, it is also highly improbable. The probability of a DEGB is broken into two parts: failure by direct means, and indirectly-induced failure. Failure of the piping by direct means can only be postulated to occur if an undetected crack grows to the point of instability, causing a large pipe break. While this accident is not as severe as a DEGB, it provides a conservative upper bound on the probability of a direct DEGB of the piping. The second part of this evaluation calculates the probability of piping failure by indirect causes. Indirect failure of the piping can be triggered by an earthquake which causes other reactor components or the reactor building to fall on the piping or pull it from its supports. Since indirectly-induced failure of the piping will not always produce consequences as severe as a DEGB, this gives a conservative estimate of the probability of an indirectly- induced DEGB. This second part, indirectly-induced pipe failure, is the subject of this report. Failure by seismic loads in the piping itself will be covered in a separate report on failure by direct causes. This report provides a detailed evaluation of L reactor. A walkdown of P reactor and an analysis of the P reactor building provide the basis for extending the L reactor results to P reactor.

Title: Foundation bearing capacity on soft soils.

Author: Ricciardi,-C.; Liberati,-G.; Previti,-R.; Paoli,-G. (ISMES SpA, Bergamo (Italy)) **Corp. Author:** Technical committee meeting o

Source: International Atomic Energy Agency, Vienna (Austria). Progress in development and design aspects of advanced water cooled reactors. Proceedings of a technical committee meeting held in Rome, 9-12 September 1991. Dec 1992. 311 p. p. 158-169.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** 390

Abstract: In PWRs of current design, the primary as well as the secondary volumes are minimized in order to mitigate mass and energy releases into the containment following a postulated DBA. In a typical 4-loop PWR, this event is associated with the release into the containment of the entire inventory of the primary system and of the secondary of one steam generator. This is in turn used to determine eventual containment loads. Present DBA assumptions consider a simultaneous large break LOCA within the primary pipework and within the secondary steam line. The relaxation of these assumptions which on the basis of the acquired experience have been shown to be overly conservative could have a significant impact on the conceptual development of advanced PWR concepts. Specifically, the volumes of both the primary and secondary systems could be optimized providing enhanced safety margins. Within this context, experiments conducted in the LOBI installation, an integral system test facility operated in the JRC-Ispra, have shown the potential benefit resulting from increased primary system volume with respect to the general evolution of LOCA events and related safety as well as operator requirements; generally, core thermal response was considerably mitigated when the experimental installation was configured with a larger primary volume. Although the experimental results acquired in the LOBI installation and proposed in the paper cannot be directly extrapolated to full-size plants, they are however indicative for the development of advanced reactors which, among others, are being conceived with larger and deeper pressure vessel, larger pressurizer and low core power density. (author). 14 refs, 10 figs, 2 tabs.

Title: Enhancement of advanced PWR safety margins through relaxation of PCS and containment DBA assumptions.

Author: Addabbo,-C. (Commission of the European Communities, Joint Research Centre, Ispra (Commission of the European Communities (CEC)). Safety Technology Inst.) **Corp. Author:** Technical committee meeting o

Source: International Atomic Energy Agency, Vienna (Austria). Progress in development and design aspects of advanced water cooled reactors. Proceedings of a technical committee meeting held in Rome, 9-12 September 1991. Dec 1992. 311 p. p. 152-158.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** 391

Abstract: Referring to advanced PWRs there is the tendency to conceive them with larger and deeper pressure vessel to mitigate core thermal response during anticipated accidents or abnormal events. This is however resisted by present Design Basis Accident (DBA) assumptions which would prescribe the reduction of Primary Cooling System (PCS) volume in order to minimize mass and energy release and thus pressure and temperature build up in the containment following a large break LOCA caused by a complete severance within the primary and secondary systems pipework. The relaxation of the current DBA assumptions which on the basis of the acquired experience and probabilistic risk assessment studies have been shown to be overly conservative could have a significant impact on the conceptual development of advanced PWRs. Specifically, the volumes of both the primary and secondary cooling systems could be optimized without close linkage to containment performance providing enhanced safety margins with respect to primary thermal excursions as partially confirmed by LOCA experiments conducted in the LOBI Test Facility. (author). 3 refs, 12 figs, 1 tab.

Title: Analysis of leak and break behavior in a failure assessment diagram for carbon steel pipes.

Author: Kanno,-Satoshi; Hasegawa,-Kunio; Shimizu,-Tasuku (Mechanical Engineering Research Lab., Hitachi, Ibaraki (Japan)); Saitoh,-Takashi; Gotoh,-Nobuho (Hitachi Works, Ibaraki (Japan))

Corp. Author:

Source: Nuclear-Engineering-and-Design. (Dec 1992). v. 138(3). p. 251-258.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Research/theoretical **ID:** 392

Abstract: The leak and break behavior of a cracked coolant pipe subjected to an internal pressure and a bending moment was analyzed with a failure assessment diagram using the R6 approach. This paper examines the conditions of the detectable coolant leakage without breakage. A leakage assessment curve, a locus of assessment point for detectable coolant leakage, was defined in the failure assessment diagram. The region between the leak assessment and failure assessment curves satisfies the condition of detectable leakage without breakage. In this region, a crack can be safely inspected by a coolant leak detector. (orig.).

Title: Long-term integrity of main pressure boundary components in the first generation of NPPs in Czechoslovakia.

Author: Zdarek,-J.; Pecinka,-L. (Nuclear Research Inst., Rez u Prahy (Czechoslovakia)); Brumovsky,-M. (Skoda Co., Plzen (Czechoslovakia))

Corp. Author:

Source: Nuclear-Engineering-and-Design. (Nov 1992). v. 137(3). p. 379-385.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** 393

Abstract: The determination of extremely low probability of the reactor pressure vessel and the primary main coolant pipe fracture is one of the prerequisites for further operation of the first generation nuclear power plants (NPPs) in Czechoslovakia. The project for this task includes experimental and analytical work on the reactor pressure vessel and on the main piping components. Review of the project and preliminary results are presented. The results show extremely low probability of failure. Future work is focussed on annealing studies and tests for the pressure vessel and the application of the Leak-Before-Break technology on primary piping circuit. (orig.).

Title: Structural integrity tests at the HDR pressure vessel and pipework under operating and accident conditions.

Author: Katzenmeier,-G. (Kernforschungszentrum Karlsruhe (Germany). Projektbereich Handhabungstechnik); Diem,-H. (Materialpruefungsanstalt Stuttgart (Germany))

Corp. Author:

Source: Kerntechnik-1987. (Dec 1992). v. 57(6). p. 360-367.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** 394

Abstract: Pressure vessel and pipework were subjected to static and transient loads, both thermally and mechanically, until incipient crack, crack growth, leakage or break occurred. The pressure vessel tests included thermal stratification, cyclic thermal shock tests and pressurized thermal shock tests. The pipework was subjected to alternating bending under pressure, thermal stratification, pressure surge, blowdown with valve closure, pulse loads, and simulated earthquake loadings. The experiments have shown that components made of highly ductile materials have extensive safety margins. The leak-before-break behavior of pipework was confirmed in all tests. (orig./HP).

Title: Design bases and severe accident considerations for the System 80+ trademark containment design.

Author: Schneider,-R.E.; Gerdes,-L.D. (ABB-Combustion Engineering, Inc., Windson, CT (United States)); Oswald,-J.T.; Snipes,-J.F. Jr. (Duke Engineering and Services, Inc., Charlotte, NC (United States))

Corp. Author: 5. workshop on containment int

Source: Parks,-M.B.; Hughey,-C.E. (eds.) (Sandia National Labs., Albuquerque, NM (United States)). Nuclear Regulatory Commission, Washington, DC (United States). Div. of Engineering; Sandia National Labs., Albuquerque, NM (United States). Proceedings of the fifth workshop on containment integrity. Jul 1992. 646 p. p. 163-178.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Other **ID:** 395

Abstract: Containments for Advanced Light Water Reactors (ALWRs) must not only be designed for design bases conditions but also be evaluated for postulated severe accident concerns. This paper presents the containment design description for the System 80+ ALWR, the conservative design bases specified and the System 80+ ALWR design features to prevent and mitigate the challenges considered in postulated severe accident scenarios. Included in the containment design bases are postulated primary and secondary pipe break conditions and seismic requirements for an envelope of site conditions with a control motion having much higher energy content than those used for existing reactor designs. Severe accident considerations addressed include prevention and mitigation design features incorporated into the System 80+ ALWR.

Title: Uncertainty analysis for K-reactor flow instability LOCA limits.

Author: Hardy,-B.J. (Westinghouse Savannah River Co., Aiken, SC (United States))

Corp. Author: American Nuclear Society ann

Source: Transactions-of-the-American-Nuclear-Society. (1992). v. 65. p. 230-232.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Analysis of break effects **ID:** 396

Abstract: A postulated accident scenario for the Savannah River Site (SRS) K reactor is a double-ended guillotine break loss-of-coolant accident (DEGB/LOCA) caused by a coolant pipe break at the plenum inlet. The DEBG/LOCA consists of two parts, the first of which applies to the first few seconds of the transient. The first part of the DEGB/LOCA is addressed in this paper. In the first few seconds after the pipe break, there is a rapid depressurization of the plenum, which results in a rapid reduction in the core flow rate. Safety rod insertion is not assumed to begin until 1 s after the pipe break, and the rods are assumed not to be fully inserted until approx 2 s after the break. The resulting flow-power mismatch results in coolant heating and possible flow disruption via a Ledinegg-type flow instability. It is assumed that assembly integrity will be compromised if flow disruption occurs. Because Ledinegg flow instability is the limiting phenomenon for the initial phase of the DEGB/LOCA transient, this part of the transient is called the flow instability (FI) phase.

Title: Loss of pumping accident limit calculation for Savannah River Reactor.

Author: Paul,-P.K.; Barbour,-K.L. (Westinghouse Savannah River Co., Aiken , SC (United States))

Corp. Author: American Nuclear Society ann

Source: Transactions-of-the-American-Nuclear-Society. (1992). v. 65. p. 317-318.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Analysis of break effects **ID:** 397

Abstract: For the Savannah River Site production reactors, the design basis accident reactor power limit ensures that if a double-ended guillotine break (DEGB) in a secondary cooling water pipe were to occur, the reactor will shut down safely. The primary reactor coolant is heavy water (D sub 2 O) with secondary light water (H sub 2 O) cooling. The accident scenario is a DEGB in one of two secondary coolant inlet header pipes with several assumed single failures. The recycled primary coolant loses its cooling, and the reactor core temperature begins to rise. Another possible accident is a DEGB in one of two heat exchanger secondary coolant effluent header pipes. The inlet header break is slightly more limiting than the effluent header break. Upon break detection, emergency shutdown begins and the emergency cooling system (ECS) activates. The accident scenario was constructed with regard to physical, mechanical, and human factors. The computer code TRAC simulates the accident.

Title: Characterization of material properties for assessment of integrity and application of leak-before-break technology on t

Author: Zdarek,-J.; Joch,-J.; Havel,-R.; Ruscak,-M. (Ustav Jaderneho Vyzkumu CSKAE, Rez (Czechoslovakia)) **Corp. Author:** Technical committee meeting o

Source: International Atomic Energy Agency, Vienna (Austria). Materials for advanced water cooled reactors. Proceedings of a technical committee meeting held in Plzen Czechoslovakia, 14-17 May 1991. Sep 1992. 163 p. p. 117-122.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Experience/events **ID:** *398*

Abstract: The integrity assessment and leak-before-break application on the primary and other safety important piping requires detailed fracture mechanics and corrosion data base. The experience learned from the WWER/440 type W-230 and W-231 is summarized and recommendations for the APWR are stated. The main emphasis is on the properties of the homogeneous and heterogeneous welds. (author). 4 refs, 4 figs.

Title: Determination of limits for smallest detectable and largest subcritical leakage cracks in piping systems

Author: R. Bieselt, B. Kuckartz, M. Wolf **Corp. Author:** KWU, Bayernwerk AG, RWE

Source: Nuclear Engineering and Design, Vol. 159:29-40

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** English

Category: LBB methodology **ID:** *399*

Abstract: NPP piping systems - those still in their original as-built condition as well as upgraded designs - are subject to safety analysis. In order to limit the consequences of postulated piping failures, the basic safety concept incorporating rupture preclusion criteria is applied to specific high-energy piping systems. LBB-analyses are also conducted within the framework of this concept. These analyses serve to determine the potential consequences of jet and reaction forces due to maximum subcritical leakage cracks while also establishing the minimum crack sizes that would be reliably detectable by the leakage rates resulting from these cracks. The boundary conditions for these analyses are not clearly defined. Using various examples as a basis, this paper presents and discusses how the LBB concept can be applied. 3 references

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Title: Reliability Prediction of Pipes and Valves

Author: J.E. Strutt, K. Allsopp & L. Ouchet **Corp. Author:**

Source: Journal of Quality and Reliability Engineering International, Vol. 11, No. 2 (March-April), pp 91-100

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1995 **Language:** English

Category: Reliability prediction **ID:** *400*

Abstract: ... this paper is on order [3/1/96] ...

Title: Nucleate boiling pressure drop in an annulus: Book 2.

Author: Block,-J.A.; Crowley,-C.; Dolan,-F.X.; Sam,-R.G.; Stoedefalke,-B.H. **Corp. Author:** Westinghouse Savannah River

Source: Nov 1992. 116 p. . USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Analysis of break effects **ID:** *401*

Abstract: The application of the work described in this report is the production reactors at the SRS, and the context is nuclear reactor safety. The LOCA scenario considered involves a DEGB of a primary coolant pipe in the reactor. During a LOCA, the flow through portions of the reactor may reverse direction or be greatly reduced, depending upon the location of the break. The reduced flow rate of coolant (D sub 2 O) through the fuel assembly channels of the reactor -- downflow in this situation -- can lead to boiling and to the potential for flow instabilities which may cause some of the fuel assembly channels to overheat and melt. That situation is to be avoided. The experimental approach is to provide a test annulus which simulates geometry, materials, and flow conditions in a Mark-22 fuel assembly (Coolant Channel 3) to the extent possible. The annulus has a full-scale geometry, and in fat uses SRL dummy hardware for the inner annulus wall in the ribbed geometry. The materials aluminum. The annulus is uniformly heated in the axial direction, but the circumferential heat flux can be varied to provide "power tilt" or asymmetric heating of the inner and outer annulus walls. The test facility uses H sub 2 O rather than D sub 2 O, but it includes the effects of dissolved helium gas present in the reactor. The key analysis approaches are: To compare the minima in the measured demand curves with analytical criteria, in particular the Saha-Zuber (1974) model; and to compare the pressure and temperature as a function of length in the annulus with an integral model for flow boiling in a heated channel. Nineteen test series and a total of 178 tests were performed. Testing addressed the effects of: Heat flux; pressure; helium gas; power tilt; ribs; asymmetric heat flux.

Title: Nucleate boiling pressure drop in an annulus: Book 7.

Author: **Corp. Author:** Westinghouse Savannah River

Source: Nov 1992. 1033 p. USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Analysis of break effects **ID:** *402*

Abstract: The application of the work described in this report is the production reactors at the Savannah River Site, and the context is nuclear reactor safety. The Loss of Coolant Accident (LOCA) scenario considered involves a double-ended break of a primary coolant pipe in the reactor. During a LOCA, the flow through portions of the reactor may reverse direction or be greatly reduced, depending upon the location of the break. The reduced flow rate of coolant (D sub 2 O) through the fuel assembly channels of the reactor -- downflow in this situation -- can lead to boiling and to the potential for flow instabilities which may cause some of the fuel assembly channels to overheat and melt. That situation is to be avoided. The experimental approach is to provide a test annulus which simulates geometry, materials, and flow conditions in a Mark-22 fuel assembly (Coolant Channel 3) to the extent possible. The annulus has a full-scale geometry, and in fat uses SRL dummy hardware for the inner annulus wall in the ribbed geometry. The materials aluminum. The annulus is uniformly heated in the axial direction, but the circumferential heat flux can be varied to provide "power tilt" or asymmetric heating of the inner and outer annulus walls. The test facility uses H sub 2 O rather than D sub 2 O, but it includes the effects of dissolved helium gas present in the reactor. The key analysis approaches are: To compare the minima in the measured demand curves with analytical criteria, in particular the Saha-Zuber (1974) model; and to compare the pressure and temperature as a function of length in the annulus with an integral model for flow boiling in a heated channel. This document consists solely of tables of temperature measurements; minima, maxima, averages and standard deviations being measured.

Title: Nucleate boiling pressure drop in an annulus: Book 4.

Author: Block,-J.A.; Crowley,-C.; Dolan,-F.X.; Sam,-R.G.; Stoedefalke,-B.H. **Corp. Author:** Westinghouse Savannah River

Source: Nov 1992. 379 p. .USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Analysis of break effects **ID:** 403

Abstract: The application of the work described in this report is the production reactors at the Savannah River Site, and the context is nuclear reactor safety. The Loss of Coolant Accident (LOCA) scenario considered involves a double-ended break of a primary coolant pipe in the reactor. During a LOCA, the flow through portions of the reactor may reverse direction or be greatly reduced, depending upon the location of the break. The reduced flow rate of coolant (D sub 2 O) through the fuel assembly channels of the reactor -- downflow in this situation -- can lead to boiling and to the potential for flow instabilities which may cause some of the fuel assembly channels to overheat and melt. That situation is to be avoided. The experimental approach is to provide a test annulus which simulates geometry, materials, and flow conditions in a Mark-22 fuel assembly (Coolant Channel 3) to the extent possible. The annulus has a full-scale geometry, and in fat uses SRL dummy hardware for the inner annulus wall in the ribbed geometry. The materials aluminum. The annulus is uniformly heated in the axial direction, but the circumferential heat flux can be varied to provide "power tilt" or asymmetric heating of the inner and outer annulus walls. The test facility uses H sub 2 O rather than D sub 2 O, but it includes the effects of dissolved helium gas present in the reactor. The key analysis approaches are: To compare the minima in the measured demand curves with analytical criteria, in particular the Saha-Zuber (1974) model; and to compare the pressure and temperature as a function of length in the annulus with an integral model for flow boiling in a heated channel. This document consists of data plots and summary files of temperature measurements.

Title: Status of FRJ-2 Refurbishment of tank pipes and essential results of aging analysis.

Author: Hansen,-G.; Thamm,-G.; Thome,-M. **Corp. Author:** IGORR-III: 3. meeting of the i

Source: Japan Atomic Energy Research Inst., Tokyo (Japan). Proceedings of the third meeting of the international group on research reactors (IGORR-III). 1994. 359 p. p. 87-114.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Other **ID:** 404

Abstract: An aging evaluation program for FRJ-2 (DIDO) of the Forschungszentrum Juelich GmbH has been developed and is currently executed in cooperation with the licensing and regulatory and TUV experts to determine the overall life expectancy of the facility and to identify critical systems and components that need to be upgraded or refurbished for future safe reactor operation. In Phase A (completed) a master list of the FRJ-2 mechanical, electrical and structural components was compiled on a system-by system basis and the operational documentation with respect to regular inspections, maintenance, repair and unusual occurrences was carefully examined. Critical components were selected and their ageing respectively life limiting mechanisms identified. In Phase B (currently under way) special inspections, examinations and tests for critical systems/components are being elaborated, executed and evaluated. Current work is being concentrated on non replaceable components (e.g. reactor aluminium tank (RAT) and the connecting pipes to the primary cooling circuit, the reactor steel tank and pipe work inside the concrete reactor block). As a consequence of first results of the aging evaluation program and due to leaks in the weir and drain pipes of the RAT a repair/refurbishment program was set up for the Al-RAT pipes (risers, downcomers, weir and drain pipes) and the steel guide tubes. Details of the r/r program which is in far progress and first essential results of the aging evaluation will be presented. The results achieved until today are encouraging with respect to safe reactor operation on short and medium term. (J.P.N.).

Title: Caluclation of the dynamic opening behaviour for two through cracks on a pipe.

Author: Grebner,-H. (Gesellschaft fuer Reaktorsicherheit (GRS) mbH, Koeln (Germany)); Fischer,-F. (BEB Erdgas und Erdoel GmbH, Hannover (Germany)); Hoefler,-A. (Gesellschaft fuer Reaktorsicherheit (GRS) mbH, Koeln (Germany))

Corp. Author: 24. DVM fracture workshop m

Source: Deutscher Verband fuer Materialforschung und -pruefung e.V., Berlin (Germany). Fracture characteristics under high stress velocities. Proceedings. Bruchvorgaenge unter hohen Beanspruchungsgeschwindigkeiten. Vortraege. 1992. 517 p. p. 279-289.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** German

Category: Analysis of break effects **ID:** 405

Abstract: Two cases of cracks on a DN 400 pipe (inside radius 184.7 mm, wall thickness 18.5 mm) were examined. One was an axial through crack with 2a = 180 mm, the other was a circumferential through crack extending over 2 alpha = 90 . The pipe consists of StE 290.7 steel. Finite element calculations were carried out to describe the dynamic leak opening behaviour of the longitudinal and circumferential crack. The process of penetration through the wall is modelled in a simplified way, as the existing through crack is first kept closed by additional stable elements. Penetration through the wall is simulated by 'switching off' the stable elements. The results obtained can be summarized as follows: in the case of the longitudinal crack, a maximum leakage area of about 56 mm sup 2 (on the inside of the pipe) and a maximum J value of approximately 25 Nmm sup - sup 1 are reached about 0.3 millisecs after the crack starts to open. On the pipe with the circumferential crack there is a maximum leakage area of about 60 mm sup 2 and a J value of about 15 Nmm sup - sup 1 about 0.6 millisecs after the crack starts to open. (orig.).

Title: Analysis of a large break LOCA in the cold leg of the WWER-440/W-213 plant Griefswald, Unit 5.

Author: Horche,-W. (Gesellschaft fur Anlagen-und Reaktorsicherheit, Garching (Germany))

Corp. Author: 2. Japan Society of Mechanical

Source: Peterson,-P.F. (ed.) (Univ. of California, Berkeley, CA (United States)). 2nd ASME-JSME international conference on nuclear engineering -- 1993. Volume 1. New York, NY (United States). American Society of Mechanical Engineers. 1993. 770 p. p. 613.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Analysis of break effects **ID:** 406

Abstract: The GRS has performed a safety evaluation of Greifswald-5 in cooperation with the French IPSN and other partners. Within this project an independent accident analysis is performed by GRS to assess the results of existing analysis and to supplement them. In this paper the analysis of DEGB of one cold leg of the main circulation pipe is described. The major objective of the calculation was investigation of the accident sequence with reduced availability of the ECCS (single failure criterion). In addition, the simultaneous LOSP and failure of scram were assumed. The thermal-hydraulic system code ATHLET/FLUT was applied. The pressure in the confinement, the back pressure for the discharge model, was calculated as a function of time for this accident separately with GRS-Code RALOC. Furthermore, it was necessary to model the local concentration of direct accumulator injection into the reactor vessel with the help of a special two-channel model of the core and upper plenum. For this model, results were considered obtained from the 1:1 scaled test facility UPTF. It was assumed that only 25% of the upper plenum and core volume is directly penetrated by the injected water. The DEGB was defined in that loop, which is connected with one of three low-pressure injection subsystems. This means that this injected water flows towards the leak without passing the core. As single failure the failure of one of three diesel generators was assumed. The full paper will contain nodalization schemes, which are generated by the ATHLET-Input-Grafic.

Title: Study of thermal fluid leaking between piping and insulator: Basic experiment by air.

Author: Toda,-Saburo; Hsu,-Wensheng; Hashizume,-Hidetoshi; Hori,-Yutaka (Tohoku Univ., Sendai (Japan). Dept. of Nuclear Engineering)

Corp. Author: 2. Japan Society of Mechanical

Source: Peterson,-P.F. (ed.) (Univ. of California, Berkeley, CA (United States)). 2nd ASME-JSME international conference on nuclear engineering -- 1993. Volume 1. New York, NY (United States). American Society of Mechanical Engineers. 1993. 770 p. p. 131-134.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Analysis of break effects **ID:** *407*

Abstract: An experimental study was performed to evaluate effect of steam leakage on temperature distributions of pipes in nuclear power plants. Heated air was used as thermal fluid and defect of the pipe was simulated by a pin hole. Experimental results indicate that the surface temperature distributions of covers surrounding the pipe are categorized into two patterns due to location of the pin hole. A new method to predict the location of the defect based on these temperature distributions is proposed through this study.

Title: Free-blowing of pipe elbows in original geometry UPTF-TRAM integral experiment A5 for testing small leak incident

Author: Sonnenburg,-H.G. (Gesellschaft fuer Anlagen- und Reaktorsicherheit (GRS) mbH, Garching (Germany))

Corp. Author: Annual meeting on nuclear tec

Source: Bauer,-K.G. (ed.). Deutsches Atomforum e.V., Bonn (Germany); Kerntechnische Gesellschaft e.V., Bonn (Germany). Annual meeting on nuclear technology '94. Proceedings. Jahrestagung Kerntechnik '94. Tagungsbericht. Bonn (Germany). Inforum Verl. 1994. 598 p. p. 49-52.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** German

Category: Test/analysis **ID:** *408*

Abstract: Short communication.

Title: Summary of important results and SCDAP/RELAP5 analysis for OECD LOFT experiment LP-FP-2.

Author: Coryell,-E.W. (EG and G Idaho, Inc., Idaho Falls, ID (United States))

Corp. Author: Nuclear Regulatory Commissio

Source: Apr 1994. 161 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Analysis of break effects **ID:** *409*

Abstract: This report summarizes technical findings from the LP-FP-2 Experiment sponsored by the OECD and conducted in the LOFT facility at INEL. The overall technical objective of the test was to contribute to the understanding of fuel rod behavior, hydrogen generation, and fission product release, transport, and deposition during a V-sequence accident scenario that resulted in severe core damage. An 11x11 test bundle, comprised of 100 pre-pressurized fuel rods, 11 control rods, and 10 instrumented guide tubes, was surrounded by an insulating shroud and contained in a specially designed central fuel module, that was inserted into the LOFT reactor. The simulated transient was an ISLOCA scenario featuring a pipe break in the LPIS line attached to the hot leg of the LOFT broken loop piping. The transient was terminated by reflood of the reactor vessel when the outer wall shroud temperature reached 1517 K. With sustained fission power and heat from oxidation and metal-water reactions, elevated temperatures resulted in zircaloy melting, fuel liquefaction, material relocation, and the release of hydrogen, aerosols, and fission products. A description and evaluation of the major phenomena, based upon the response of on line instrumentation, analysis of fission product data, postirradiation examination of the fuel bundle, and calculations using the SCDAP/RELAP5 computer code, are presented.

Title: Experiments and calculation on crack opening and leak rate of a pipe branch within the HDR-program.

Author: Grebner,-H. (Gesellschaft fuer Reaktorsicherheit (GRS) mbH, Koeln (Germany)); Hunger,-H. (Kernforschungszentrum Karlsruhe GmbH (Germany)); Hoefler,-A. (Gesellschaft fuer Reaktorsicherheit (GRS) mbH, Koeln (Germany))

Corp. Author:

Source: Nuclear-Engineering-and-Design. (Jan 1994). v. 147(1). p. 79-84.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Test/analysis **ID:** 410

Abstract: In this paper experiments and calculations on crack opening and leak rates of a pipe branch are presented. The pipe branch has an artificial through-crack located in the weldment between nozzle and the larger pipe. A superposition of internal pressure and bending load is considered. The experiment was part of a series of experiments on straight pipes, branches and elbows, which were performed at the HDR-facility at Karlstein/Germany. For the pipe branch under consideration experimental and numerical results and comparisons between both are presented. (orig.).

Title: Application of the leak-before-break concept to steam generator tubes.

Author: Keim,-E.; Kastner,-W. (Siemens. Erlangen

Corp. Author: IAEA Specialist's meeting on st

Source: IAEA Specialist's meeting on steam generator problems and replacement. Madrid (Spain). CIEMAT. 1994. 653 p. p. 515-527.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: LBB justification **ID:** 411

Abstract: The Leak-Before-Break (LBB) behaviour of a piping component means that the length of a crack resulting in a leak is smaller than the critical crack length and that the leak is safety detectable by a suitable monitoring system. The LBB-concept of Siemens/KWU is based on computer codes for the evaluation of critical crack lengths, crack openings, leakage areas and leakage rates, developed by Siemens/KWU. The fracture mechanics analysis supplies the input for the thermal-hydraulic analysis. The resulting leakage rate related to the crack length of a longitudinal or circumferential crack and the minimum detectable values of leakage rate and crack length lead to two criteria, which allow for the LBB-behaviour of the pipe: - the critical crack length must be larger than the crack length being safety detected by leakage monitoring systems (LMS) - the critical crack length must be larger than the crack length being safety detected by non-destructive examination (NDE). This LBB-concept is applied to steam generator (SG) tubes. Two examples, which will be presented, show that this concept is a very useful and effective tool which allows the prediction of LBB-behaviour of SG tubes. (Author).

Title: Application of intrinsic germanium spectral gamma-ray logging for characterization of high-level nuclear waste tank l

Author: Brodeur,-J.R.; Kiesler,-J.P.; Kos,-S.E.; Koizumi,-C.J.; Nicaise,-W.F.; Price,-R.K.

Corp. Author: Westinghouse Hanford Co., Ri

Source: Nov 1993. 14 p. USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Inspection methods **ID:** 412

Abstract: Spectral gamma-ray logging with a high-resolution, intrinsic germanium logging system was completed in boreholes surrounding two high-level nuclear waste tanks at the US Department of Energy's Hanford Site. The purpose was to characterize the concentrations of man-made radionuclides in the unsaturated zone sediments and identify any new leaks from the tanks. An intrinsic germanium detection system was used for this work because it was important to positively identify the specific radionuclides and to precisely assay those radionuclides. The spectral gamma log data were processed and displayed as log plots for each individual borehole and as three-dimensional plots of sup 1 sup 3 sup 7 Cs radionuclide concentrations. These data were reviewed to identify the sources of the contamination. The investigation did not uncover a new or active leak from either of the tanks. Most of the contamination found could be related to known pipeline leaks, to surface contamination from aboveground liquid spills, or to leaks from other tanks. The current spectral gamma ray data now provide a new baseline from which to compare future log data and identify any changes in the radioelement concentration.

Title: LBB technology application to the primary piping system of the NPP V1 Jaslovske Bohunice.

Author: Zdarek,-J. (Nuclear Research Inst. plc, Integrity and Materials Div., Prague (Czech Republic)); Pecinka,-L. (Nuclear Research Inst. plc, Integrity and Materials Div., Prague (Czech Republic)); Joch,-J. (Nuclear Research Inst. plc, Integrity and Materials Div., Prague (Czech Republic))

Corp. Author:

Source: Nuclear-Engineering-and-Design. (Oct 1993). v. 144(1). p. 69-76.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: LBB justification **ID:** 413

Abstract: Due to several deficiences of the WER Model 230 type reactor a leak before break demonstration of this reactor is of primary importance. The complex project for NPP V1 Jaslovske Bohunice includes a static and dynamic stress analysis of the primary piping, a fatique damage analysis, leak rate assessments and an analysis of the stability of the heavy components supports. The material database includes data on fracture mechanics, on assessment of corrosion properties, and on the influence of 100 000 hr service exposure on base metal and welds including disimilar welds. The program was supported by large scale experiments on RPV safe-end, pressurizer safe-end, elbow welds with through-wall cracks and leak rate measurements. The results and applications are discussed. (orig.).

Title: Calculation of leakage areas and leak rates for wall penetrating cracks in pipes loaded with internal pressure and bendi

Author: Grebner,-H. (Gesellschaft fuer Reaktorsicherheit (GRS) mbH, Koeln (Germany)); Hoefler,-A. (Gesellschaft fuer Reaktorsicherheit (GRS) mbH, Koeln (Germany)); Hunger,-H. (Kernforschungszentrum Karlsruhe, Projekt HDR-Sicherheitsprogramm (PHDR) (Germany))

Corp. Author:

Source: Nuclear-Engineering-and-Design. (Oct 1993). v. 144(1). p. 101-109.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 414

Abstract: Calculations on the leak opening and leak rates of piping components with through cracks are presented. Mostly the analyses are post-calculations to experiments in order to verify the models used in the calculations. The experiments under consideration were performed at MPA-Stuttgart, Siemens-KWU and at the HDR-facility. Straight pipes, pipe bends and branches with different crack locations were considered. As far as possible numerical and experimental results are compared. (orig.).

Title: Gentilly-2 secondary-side break study.

Author: Lafreniere,-P. (Hydro-Quebec, Gentilly (Canada). Gentilly Generating Station); Shill,-R. (Atomic Energy of Canada Ltd., Montreal, PQ (Canada). CANDU Operations)

Corp. Author: Canadian Nuclear Society 11.

Source: Rouben,-B. (ed.). Canadian Nuclear Society, Toronto, ON (Canada). Proceedings of the 11th Annual Conference of the Canadian Nuclear Society. 1990. 440 p. p. 4.23-4.31.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** French

Category: Damage probability **ID:** 415

Abstract: The AECB asked Hydro-Quebec in July 1986 to study the consequences for station safety of a significant rupture of a secondary side pipe break in the powerhouse. This study was to deal with the guillotine rupture of the main steam line, of the feedwater supply and of the steam balance header. A preliminary examination carried out by Hydro-Quebec showed that it would be necessary to ensure the availability of Group 2 systems during and after a secondary side rupture. It was also decided that the study would cover ruptures outside the powerhouse, and particularly ruptures in room S2-246 in the service building. In June 1988 Hydro-Quebec asked AECL to carry out a study of the safety of the Gentilly-2 station during and after a secondary side rupture, covering the area outside the reactor building. The study included breaks of every size, guillotine and non-guillotine ruptures. The study also covered the response of Group 1 systems under the postulated conditions in order to gain a better understanding of station response. To evaluate the safety of the Gentilly-2 station, four main safety criteria were examined: reactor shutdown, heat sink, containment, and monitoring. The probabilistic techniques used by the study of secondary side ruptures have shown the eventual advantage of several modifications designed to prevent the spread of steam within the station and to ensure safe access to the secondary control area. (Author).

Title: Overview of reliability test program on primary coolant piping of light water reactors.

Author: Shibata,-Katsuyuki; Isozaki,-Toshikuni; Ueda,-Syuzo; Kurihara,-Ryoichi; Onizawa,-Kunio; Kosaka,-Atsuo (Japan Atomic Energy Research Inst., Tokai, Ibaraki (Japan). Tokai Research Establishment)

Corp. Author:

Source: Nippon-Genshiryoku-Gakkai-Shi. (Oct 1993). v. 35(10). p. 923-939.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** Japanese

Category: LBB verification **ID:** 416

Abstract: Upon request by the Science and Technology Agency of Japanese Government, the JAERI has conducted Piping Reliability Test Program to demonstrate the safety and reliability of light water reactor primary pipings. In this report, the results of the program are summarized. In the test program, pipe fatigue tests, Leak-Before-Break (LBB) verification tests and pipe rupture tests were carried out to examine the integrity of pipings, to verify the LBB concept and to demonstrate the effectiveness of the protective measures against jet impingement and pipe whip under pipe rupture event, respectively. In the pipe fatigue tests, a procedure to predict the fatigue crack growth was developed and the integrity of piping during plant service life was demonstrated. In the LBB verification tests, pipe fracture tests and leak rate tests were performed using cracked pipes. Based on the test results, LBB in the primary pipings was demonstrated. In the pipe rupture tests, the influence of jet impingement on the target plate and the interaction between whipping pipe and restraint were investigated. Using the test results, the effect of jet impingement and the effectiveness of pipe whip restraints were demonstrated. (author).

Title: Comparison of leak opening and leak rate calculations to HDR experimental results.

Author: Grebner,-H.; Hoefler,-A. (Gesellschaft fur Anlagen- und Reaktorsicherheit mbH, Cologne (Germany)); Hunger,-H. (Kernforschungszentrum, Karlsruhe (Germany))

Corp. Author: 1993 pressure vessel and pipin

Source: Garud,-Y.S. (ed.). Creep, fatigue, flaw evaluation, and leak-before-break assessment. New York, NY (United States). American Society of Mechanical Engineers. 1993. 301 p. p. 217-229.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 417

Abstract: During the last years a number of calculations of leak opening and leak rate for through cracks in piping components have been performed. Analyses are pre- or mostly post-calculations to experiments performed at the HDR facility under PWR operating conditions. Piping components under consideration were small diameter straight pipes with circumferential cracks, pipe bends with longitudinal or circumferential cracks and pipe branches with weldment cracks. The components were loaded by internal pressure and opening as well as closing bending moment. The finite element method and two-phase flow leak rate programs were used for the calculations. Results of the analyses are presented as J-integral values, crack opening displacements and areas and leak rates as well as comparisons to the experimental results.

Title: Pipe fracture evaluations for leak-rate detection: Applications to BWR and PWR piping.

Author: Rahman,-S.; Wilkowski,-G.; Ghadiali,-N. (Battelle Memorial Inst., Columbus, OH (United States). Dept. of Engineering Mechanics)

Corp. Author: 1993 pressure vessel and pipin

Source: Garud,-Y.S. (ed.). Creep, fatigue, flaw evaluation, and leak-before-break assessment. New York, NY (United States). American Society of Mechanical Engineers. 1993. 301 p. p. 269-285.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Damage probability **ID:** 418

Abstract: This is the last in series of three papers generated from studies on pipe fracture evaluations for leak-rate detection. This paper focuses on the application of proposed deterministic and probabilistic models described in References 1 and 2 for stochastic pipe fracture evaluations of nuclear piping in BWR and PWR for leak-rate detection. The computer codes developed in the previous phases of the study were used to determine the conditional probability of failure of nuclear piping in BWR and PWR plants. Several pipe sizes, such as small, intermediate, and large and several pipe materials, such as stainless steel, carbon steel, and cast stainless steel were considered. The computational effort involved calculation of conditional failure probability of 10 BWR pipes and 6 PWR pipes and evaluation of adequacy for the current safety margin of 10 used for leak-rate by explicitly considering the statistical variability of crack morphology variables. As an end-product from this study, various plots of conditional failure probability versus leak rate were generated. A comparison of the above conditional failure probabilities will provide a technical basis for any changes in the maximum allowable unidentified leak rates allowed by Regulatory Guide 1.45 with reference to the leak-before-break procedures of NRC.

Title: Pipe fracture evaluations for leak-rate detection: Probabilistic models.

Author: Rahman,-S.; Wilkowski,-G.; Ghadiali,-N. (Battelle Memorial Inst., Columbus, OH (United States). Dept. of Engineering Mechanics) **Corp. Author:** 1993 pressure vessel and pipin

Source: Garud,-Y.S. (ed.). Creep, fatigue, flaw evaluation, and leak-before-break assessment. New York, NY (United States). American Society of Mechanical Engineers. 1993. 301 p. p. 255-267.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Damage probability **ID:** 419

Abstract: This paper focuses on the development of novel probabilistic models for stochastic performance evaluation of degraded nuclear piping systems. It was accomplished here in three distinct stages. First, a statistical analysis was conducted to characterize various input variables for thermo-hydraulic analysis and elastic-plastic fracture mechanics, such as material properties of pipe, crack morphology variables, and location of cracks found in nuclear piping. Second, a new stochastic model was developed to evaluate performance of degraded piping systems. It is based on accurate deterministic models for thermo-hydraulic and fracture mechanics analyses described in the first paper, statistical characterization of various input variables, and state-of-the-art methods of modem structural reliability theory. From this model. the conditional probability of failure as a function of leak-rate detection capability of the piping systems can be predicted. Third, a numerical example was presented to illustrate the proposed model for piping reliability analyses. Results clearly showed that the model provides satisfactory estimates of conditional failure probability with much less computational effort when compared with those obtained from Monte Carlo simulation. The probabilistic model developed in this paper will be applied to various piping in boiling water reactor and pressurized water reactor plants for leak-rate detection applications.

Title: Pipe fracture evaluations for leak-rate detection: Deterministic models.

Author: Wilkowski,-G.; Rahman,-S.; Paul,-D.; Ghadiali,-N. (Battelle Memorial Inst., Columbus, OH (United States). Dept. of Engineering Mechanics) **Corp. Author:** 1993 pressure vessel and pipin

Source: Garud,-Y.S. (ed.). Creep, fatigue, flaw evaluation, and leak-before-break assessment. New York, NY (United States). American Society of Mechanical Engineers. 1993. 301 p. p. 243-254.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Damage probability **ID:** 420

Abstract: Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems was published by NRC in May 1973, and its update is being considered. Updating this procedure can involve accounting for the current leak-detection instrumentation capabilities, experience from the accuracy of leak-detection systems in the past, and current analysis methods to assess the significance of the detectable leakage relative to the structural integrity of the plant. In this study, a three-phase effort was undertaken to conduct circumferentially cracked pipe fracture evaluations for applications to leak-rate detection requirement. Results from these probabilistic analyses can be used as a technical basis for future changes to leak-rate detection criterion. In this paper, a state-of-the-art review was conducted to evaluate the adequacy of current deterministic models for thermal-hydraulic analysis for estimation of leak rates, crack-opening area analysis for determination of crack geometry, and elastic-plastic fracture mechanics for prediction of maximum load-carrying capacity of circumferentially cracked piping systems (Phase 1). The results predicted from the above deterministic models were compared with experimental data obtained from the past NRC research programs. Based on the comparisons, it was concluded that the models considered in this study provide reasonably accurate estimates of leak rates, area of crack opening, and maximum load-carrying capacity of circumferentially cracked pipes. These validated deterministic models will be used for subsequent development of novel probabilistic models to evaluate structural reliability of degraded piping systems (Phase 2). Using these models, stochastic pipe fracture evaluation will be conducted for applications to leak-rate detection of piping in boiling water reactor and pressurized water reactor plants (Phase 3).

Title: Short cracks in piping and piping welds. Semiannual report, April 1992--September 1992: Volume 3, No. 1.

Author: Wilkowski,-G.M.; Brust,-F.; Francini,-R.; Ghadiali,-N.; Kilinski,-T.; Krishnaswamy,-P.; Landow,-M.; Marschall,-C.W.; Rahman,-S.; Scott,-P. (Battelle, Columbus, OH (United States))

Corp. Author: Nuclear Regulatory Commissio

Source: Oct 1993. 164 p. FNuclear Regulatory Commission, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Research/theoretical **ID:** 421

Abstract: This is the fifth semiannual report of the USNRC research program entitled "Short Cracks in Piping and Piping Welds." This 4-year program began in March 1990. The program objective is to verify and improve fracture analyses for circumferentially cracked large-diameter nuclear piping with crack sizes typically used in leak-before-break analyses or in-service flaw evaluations. During this reporting period, the overall program as well as the results to date were reviewed very critically. It was found that several changes to the current program were needed to meet the final objectives at the end of the 4 years. Hence, the program was restructured. As a result, several activities were put on hold during this reporting period until restructuring was finalized. The changes to the existing program as well as the deliverables from the additional activities are detailed in this report. In the surface- cracked pipe evaluations, work progress involved: (1) evaluating the tensile and Charpy V-notch data for a carbon-manganese submerged arc weld metal (Plate DP2-F49W), and (2) conducting 3D finite-element (FE) analyses of uncracked stainless steel pipe experiments conducted in Japan to resolve the discrepancies between experimental data and FE predictions. Significant efforts during this period involved quantifying the leak rate from cracked pipe using advanced probabilistic analysis. A new PC version of the code to evaluate circumferential surface-cracked pipe, NRCPIPES Version 1.0, was completed and sent to the NRC for testing along with a user's manual. Most of the analysis of the influence of the residual stress field on cracks in welds, being conducted under a subcontract to the University of Michigan, was completed during this reporting period and is included here.

Title: Thermal stratification of feedwater piping in a BWR plant.

Author: Wang,-W.Y.; Cokonis,-A.J.; Casella,-R.C.; Fox,-J.O. (Stone and Webster Engineering Corp., Cherry Hill, NJ (United States)); Prunotto,-L.P. (Niagara Mohawk Power Corp., Syracuse, NY (United States))

Corp. Author: 1993 pressure vessel and pipin

Source: Dermenjian,-A.A. (ed.). Piping, supports, and structural dynamics. New York, NY (United States). American Society of Mechanical Engineers. 1993. 181 p. p. 7-19.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 422

Abstract: Thermal stratification was suspected in the Feedwater (FW) piping of a BWR plant during its initial power ascension and later confirmed during a turbine trip from full power. This experience and the operating conditions that cause thermal stratification are discussed in this paper so that plants with similar potential can benefit from the lessons learned. Since thermal stratification was not considered in the original plant design, an assessment of thermal stratification effects on piping structural integrity was performed. Based on the analysis, field measurements and the actual observation of pipe coupling leakage, support modifications were implemented and the piping has since performed well. The methodology used in this assessment is also discussed here.

Title: Experimental and estimated crack mouth opening displacements on carbon and stainless steel pipes under monotonic a

Author: Maricchiolo,-C.; Milella,-P.P.; Pini,-A. (ENEA, Rome (Italy)) **Corp. Author:** ANP'92: international conferen

Source: Oka,-Y.; Koshizuka,-S. (comps.) (Tokyo Univ. (Japan)). Atomic Energy Society of Japan, Tokyo (Japan). ANP'92 international conference on design and safety of advanced nuclear power plants. Tokyo (Japan). Atomic Energy Society of Japan. 1992. [2182 p.]. v. 2 p. 20.2/1-20.2/5. Composed of four volumes.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** 423

Abstract: During last ten years ENEA funded several research programs with the objective of studying the fracture behavior of cracked pipes. In the framework of national programs almost 100 carbon and stainless steel pipes were tested under quasi-static bending moment. In 1988 ENEA joined the International Piping Integrity Research Group (IPIRG), a consortium of organizations from nine nations: Canada, France, Italy, Japan, Sweden, Switzerland, Taiwan, the United Kingdom and the United States. This 5-year program was conducted at Battelle Laboratories and was completed in June 1991. The scope of this paper is the comparison between Crack Mouth Opening Displacement (CMOD) estimation scheme calculations and IPIRG data relative to displacement controlled tests. Since CMODs are necessary to evaluate the leak area associated to through-wall cracks postulated in Leak Before Break analysis of nuclear power plant pipeline, the accuracy in the prediction of CMODs becomes fundamental. (author).

Title: Nuclear fluid handling equipment: Are sparks still in the ashes?.

Author: O'Keefe,-W. **Corp. Author:**

Source: Power. (Jul 1993). v. 137(7). p. 29-37.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Methods/design **ID:** 424

Abstract: This article addresses the innovation and improvement to design for components of nuclear power plants such as pumps, valves and piping components. The topics of the article include reactor coolant pumps, feed pumps, gate valves, packing materials, swing-check valves, valve actuators, piping and leak monitoring and detection, and loose parts monitoring.

Title: Complex evaluation of properties for some thermal insulating materials of NPP.

Author: Yurchenko,-V.G.; Nazarova,-G.A.; Yakunichev,-V.N.; Potulov,-V.V.; Kazakova,-K.A. **Corp. Author:**

Source: Ehnergeticheskoe-Stroitel'-stvo. (Dec 1991). (no.12). p. 42-43.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** Russian

Category: Other **ID:** 425

Abstract: The effects of the main operational factors (temperature, ionizing radiation, increased humidity) on some most widely applied fibrous materials are investigated. The samples were irradiated by sup 6 sup 0 Co gamma photons at the PKhM-gamma-20 device in air at temperature of 40+-1 deg C in order to analyze the radiation resistance of thermal insulating materials. The analysis and generalization of the results of laboratory tests give an opportunity to make the following conclusions. The thermal insulation articles and constructions made of superfine basalt fiber may be used in the zones of rigorous regime. The superfine glass fibers (GF) are recommended to be used for equipment and pipeline shielding in the zones of rigorous control only as a part of multilayer insulation as the second or next layers and only in places where leaks are impossible.

Title: Detailed leak detection test plan and schedule for the Oak Ridge National Laboratory LLLW active pipelines. Environ

Author: Douglas,-D.G.; Starr,-J.W.; Juliano,-T.M.; Maresca,-J.W. Jr. (Vista Research, Inc., Mountain View, CA (United States)) **Corp. Author:** Oak Ridge National Lab., TN (

Source: Sep 1993. 71 p. USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Inspection methods **ID:** 426

Abstract: This document provides a detailed leak detection test plan and schedule for leak testing many of the pipelines that comprise the active portion of the liquid low-level waste (LLLW) system at the Oak Ridge National Laboratory (ORNL). This plan was prepared in response to the requirements of the Federal Facility Agreement (FFA) between the US Department of Energy (DOE) and two other agencies, the US Environmental Protection Agency (EPA) and the Tennessee Department of Environment and Conservation (TDEC). The LLLW system is an interconnected complex of tanks and pipelines. The FFA distinguishes four categories of tank and pipeline systems within this complex: new systems (Category A), doubly contained systems (Category B), singly contained systems (Category C), and inactive systems (Category D). The FFA specifically requires leak testing of the Category C systems. This plan and schedule addresses leak testing of the Category C pipelines and those doubly contained pipelines that do not fully meet the requirements for secondary containment as listed in the FFA.

Title: Evaluation of pipeline leak detection systems.

Author: Glauz,-W.D.; Flora,-J.D.; Hennon,-G.J. (Midwest Research Inst., Kansas City, MO (United States)) **Corp. Author:** Symposium on leak detection f

Source: Durgin,-P.B. (ed.) (Veeder-Root Co., Simsbury, CT (United States)); Young,-T.M. (ed.). Leak detection for underground storage tanks. Philadelphia, PA (United States). American Society for Testing and Materials. 1993. 241 p. p. 151-161.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Inspection methods **ID:** 427

Abstract: Leaking underground storage tank system presents an environmental concern and a potential health hazard. It is well known that leaks in the piping associated with these systems account for a sizeable fraction of the leaks. EPA has established performance standards for pipeline leak detection systems, and published a document presenting test protocols for evaluating these systems against the standards. This paper discusses a number of facets and important features of evaluating such systems, and presents results from tests of several systems. The importance of temperature differences between the ground and the product in the line is shown both in theory and with test data. The impact of the amount of soil moisture present is addressed, along with the effect of frozen soil. These features are addressed both for line tightness test systems, which must detect leaks of 0.10 gal/h (0.38 L/h) at 150% of normal line pressure, or 0.20 gal/h (0.76 L/h) at normal line pressure, and for automatic line leak detectors that must detect leaks of 3 gal/h (11 L/h) at 10 psi (69 kPa) within an hour of the occurrence of the leak. This paper also addresses some statistical aspects of the evaluation of these systems. Reasons for keeping the evaluation process "blind" to the evaluated company are given, along with methods for assuring that the tests are blind. Most importantly, a test procedure is presented for evaluating systems that report a flow rate (not just a pass/fail decision) that is much more efficient than the procedure presented in the EPA protocol, and is just as stringent.

Title: Pipeline leak detection using volatile tracers.

Author: Thompson,-G.M. (Tracer Research Corp., Tucson, AZ (United States)); Golding,-R.D. (Special Projects, Tucson, AZ (United States))

Corp. Author: Symposium on leak detection f

Source: Durgin,-P.B. (ed.) (Veeder-Root Co., Simsbury, CT (United States)); Young,-T.M. (ed.). Leak detection for underground storage tanks. Philadelphia, PA (United States). American Society for Testing and Materials. 1993. 241 p. p. 131-136.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Inspection methods **ID:** 428

Abstract: A method of leak detection for underground storage tanks and pipelines adds volatile tracers to the products in the tanks and analyzes the surrounding shallow soil gases for tracer vapors. This method has several advantages: the success of the test is not limited by the size and structural design of the vessels, tanks can be tested at any fill level without taking the tank out of service, the location of a leak along a pipeline is clearly marked by the location of the tracer, and liquid leaks as small as 0.2 liters per hour (lph) can be detected. A limitation is: the backfill material must have some degree of air permeability in the zone above the water table. Several field tests document the success achieved using this method. A tracer leak detection system was installed at Homestead AFB after several other testing methods failed to locate a leak at a valve pit location along approximately 4 kilometers of fuel transfer piping. The leak was detected to the side of the valve pit at a depth of approximately 2.5 meters below the ground surface. Another installation of Edwards AFB involved the collection of 415 soil gas samples along approximately 3,050 meters of 15.25-centimeter fiberglass pipeline. Fourteen separate leaks were detected.

Title: Increased protection required against secondary pipe failures.

Author:

Corp. Author:

Source: Canadian-Energy-News. (15 Dec 1992). v. 7(24). p. 191.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Inspection methods **ID:** 429

Abstract: The AECB (Atomic Energy Control Board) has concluded that Point Lepreau and Gentilly 2 operators must ensure that the main control room and other critical equipment is adequately protected against all steam and feedwater pipe failures. This can be ensured by means short of actually relocating pipes, including (but not limited to) highly reliable in-service inspection and leak detection.

Title: Technical report on the Piping Reliability Proving Tests at the Japan Atomic Energy Research Institute.

Author:

Corp. Author: Japan Atomic Energy Research

Source: May 1993. 468 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** Japanese

Category: Test/analysis **ID:** 430

Abstract: JAERI conducts Piping Reliability Proving Tests from 1975 to 1992 based on the contracts between JAERI and Science and Technology Agency of Japan (STA) under auspices of the special account law for electric power development promotion. The purpose of these tests are to prove the structural reliability of primary cooling piping constituting a part of the pressure boundary in the light water reactor power plants. The tests with large experimental facilities had ended already in 1990. Presently piping reliability analysis by the PFM method is being done. Until now annual reports concerning the proving tests were produced and submitted to STA, whereas this report summarizes the test results done during these 16 years. Objectives of the piping reliability proving tests are to prove that the primary piping of the light water reactor (1) be reliable throughout the service period, (2) have no possibility of rupture, (3) bring no detrimental influence on the surrounding instrumentations or equipments near the break location even if it ruptured suddenly. To attain these objectives (i) pipe fatigue tests, (ii) unstable pipe fracture tests, (iii) pipe rupture tests and also the analyses by computer codes were done. After carrying out these tests, it is verified that the piping is reliable throughout the service period.

Title: Pipe break testing of primary loop piping similar to Department of Energy's New Production Reactor-Heavy Water Re

Author: Poole,-A.B.; Clinard,-J.A.; Battiste,-R.L.; Hendrich,-W.R. **Corp. Author:** Oak Ridge National Lab., TN (

Source: [1993]. 7 p. .USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 431

Abstract: The subject of this paper is to review the recent failure testing of the Savannah River C-reactor piping weldment, which will be referred to as the C-pipe in the remainder of the paper. The intent of this paper is to further familiarize the technical community with Oak Ridge National Laboratory's (ORNL) pipe test program and associated activities surrounding the C-pipe test as conducted on behalf of the Department of Energy New Production Reactor (DOE-NPR) Program.

Title: Abnormality diagnosis device of reactor.

Author: Hirayama,-Tatsuya; Honma,-Hitoshi **Corp. Author:** Toshiba Corp., Kawasaki, Kan

Source: 24 Nov 1992; 10 May 1991. 4 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992; 199 **Language:** Japanese

Category: Inspection methods **ID:** 432

Abstract: The device of the present invention can rapidly detect a small amount of leakage of primary coolants from a heat transfer pipe in a steam generator of a PWR type reactor. That is, an off gas monitor comprising a radiation detector is disposed for detecting radiation leakage of the primary coolants to a secondary system. Further, a radiation detector for sup 1 sup 6 N, as an object of measurement, is disposed to the upstream of a secondary main steam pipeline. A calculation and processing system is disposed, to which signals detected by both of the radiation detectors are inputted. The calculation and processing system applies time sequential processing to the signals detected by both of the radiation detectors and judges as to whether the processing signals are meaningful signals due to leakage, or they are fluctuation of natural radiation or noises in the instrumentation system. The measured data from both of the radiation detectors are calculated and processed on real time. Accordingly, if a fluctuation of a radiation dose is measured at a time based on the consideration for the time of arrival between both of the radiation detectors at upstream and downstream, it is diagnosed as a fluctuation due to passage of same radioactive materials, that is, a leakage. (I.S.).

Title: Radiation detector.

Author: Noda,-Masanori **Corp. Author:** Nuclear Fuel Industries Ltd., T

Source: 18 Nov 1992; 30 Apr 1991. 6 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992; 199 **Language:** Japanese

Category: Inspection methods **ID:** 433

Abstract: The device of the present invention detects leakage of primary coolants to a pipeline of a secondary system in a PWR type plant and estimates a portion of the leakage. That is, a detector capable of discriminatively detecting nuclides, which release high energy gamma rays and have a short half life, is disposed to a secondary coolant pipe or a branch thereof. Alternatively, another detector is disposed in addition to the detector described above. Since the target nuclides concerned with the leakage are sup 1 sup 6 N, they release the gamma ray at a high energy of 4.5 to 7 MeV and have a short half life of about 7 sec. None of nuclides present in natural field has characteristics identical with both of them. Accordingly, sup 1 sup 6 N is discriminated based on the energy or the half life to detect a slight leakage of primary coolants to secondary coolants at an early stage of the leakage. (I.S.).

Title: Generation of deposits and self ignited fires in H sub 2 S-H sub 2 O services (Paper No. 4.6).

Author: Agarwal,-A.K.; Hiremath,-S.C. (Heavy Water Plant, Kota (India)) **Corp. Author:** SCOPEX-92 : national sympos

Source: Department of Atomic Energy, Bombay (India). Heavy Water Board. National symposium on commissioning and operating experiences in heavy water plants and associated chemical industries [Preprint volume]. Bombay (India). Bhabha Atomic Research Centre. Feb 1992. [501 p.]. p. 4.6.1-4.6.6.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Other **ID:** *434*

Abstract: The Heavy Water Plant (Kota) uses a large inventory of H sub 2 S gas at a nominal pressure and temperature. The plant has used mild steels/carbon steels as the material of construction of vessels, piping, flanges and fasteners. The entire construction is with flanged joints with raised face and spiral wound gaskets. Any leakages from any of the pipe line, flanged joints, heat exchanger covers, valve bonnets, valve glands etc causes H sub 2 S and H sub 2 O to leak out which generate deposits around the leakage paths after reaction with mild steel/carbon steels. The deposits grow into hard material, cause corrosion and thinning of stud bolts and gasket outer rings, weaken the confidence in the joint, and also cause ignited fires as they provide a source of ignition under certain conditions. (author). 2 refs.

Title: A technique of including the effect of aging of passive components in probabilistic risk assessments.

Author: Phillips,-J.H. (Idaho National Engineering Lab., Idaho Falls (United States)); Weidenhamer,-G.H. (Nuclear Regulatory Commission, Washington, DC (United States)) **Corp. Author:** Aging research information con

Source: Beranek,-A. (comp.). Nuclear Regulatory Commission, Washington, DC (United States). Proceedings of the Aging Research Information Conference. Volume 1. Sep 1992. 556 p. p. 114-138.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Inspection methods **ID:** *435*

Abstract: PRAs generally focus on active component failures. Potential failures of passive components is given little consideration. We are developing methods for selecting risk-significant passive components and including them in PRAs. These methods provide ways to prioritize passive components for inspection, and where inspection reveals aging damage, mitigation or repair can be employed to reduce the likelihood of component failure. We demonstrated a method by selecting a weld in the AFWS, basing our selection on expert judgement of the likelihood of failure and on an estimate of the consequence of component failure to plant safety. We then modified and used the PRAISE computer code to perform a probabilistic structural analysis to calculate the probability that crack growth due to aging would cause the weld to fail. The PRAISE code was modified to include the effects of changing design material properties with age and changing stress cycles. The calculation included the effects of mechanical loads and thermal transients typical of the service loads for this piping design and the effects of thermal cycling caused by a leaking check valve. However, this particular calculation showed little change in low component failure probability and plant risk for 48 years of service. However, sensitivity studies showed that if the probability of component failure is high, the effect on plant risk is significant. The success of this demonstration shows that this method could be applied to nuclear power plants. The demonstration showed the method is too involved (PRAISE takes a long time to perform the calculation and the input information is extensive) for handling a large number of passive components and therefore simpler methods are needed.

Title: Crack opening and leak rate evaluation for piping components with through cracks.

Author: Grebner,-H.; Hoefler,-A. (Gesellschaft fuer Reaktorsicherheit (GRS) mbH, Koeln (Germany)) **Corp. Author:**

Source: Nuclear-Engineering-and-Design. (Jun 1992). v. 135(2). p. 161-170.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Methods/comparison **ID:** *436*

Abstract: For components of piping it may be necessary in many cases to evaluate the crack opening and leakage area of anticipated through cracks in order to estimate maximum or minimum leak rates for the transported gaseous or liquid medium. In this paper several analytical solutions for the crack opening and for the leakage area found in literature are described and comparisons are made between results gained with these methods and values from elastic-plastic finite element calculations. The piping components under consideration are straight pipes with longitudinal and circumferential cracks and pipe bends with longitudinal cracks. The loading cases studied are either internal pressure or bending moment or a combination of both. Crack opening and leak rate values obtained for straight pipes are compared to results of experiments carried out in the frame of the German HDR (overheated steam reactor) - Safety Program. (orig.).

Title: Analyses of fatigue crack growth and fracture in carbon steel piping. Technical note.

Author: Kashima,-K.; Matsubara,-M.; Miura,-N. (Central Research Inst. of Electric Power Industry, Tokyo (Japan)); Ando,-Y. (Tokyo Univ./NUPEC (Japan)); Takumi,-K. (NUPEC (Japan))

Corp. Author:

Source: Nuclear-Engineering-and-Design. (Jun 1992). v. 135(2). p. 179-186.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: LBB justification **ID:** 437

Abstract: The objective of the present study is to evaluate the fatigue crack growth behavior and fracture conditions of Japanese carbon steel piping, which are relevant for the Leak-Before-Break behavior. A fatigue crack growth analysis was conducted for a circumferential inner-surface crack in the main feedwater line, the main stream line and the ECCS line under the design loading conditions of BWR and PWR plants. The fatigue analysis showed that a crack does not penetrate the pipe wall under the design loading conditions if the initial crack depth is equal to or smaller than 10-20% of the pipe wall thickness. Conditions for stable crack growth and pipe failure were analyzed by non-linear fracture mechanics. Good agreement between experiment and analysis was shown in the load-deformation relationship during stable crack growth. Fracture conditions were described with a good accuracy by the net-section collapse criterion for a STS42 6-inch diameter pipe. (orig.).

Title: Computation of leak areas of circumferential cracks in piping for application in demonstrating leak-before-break beha

Author: Bhandari,-S. (Framatome, 92 - Paris-la-Defense (France)); Faidy,-C. (EDF-Septen, 69 - Villeurbanne (France)); Acker,-D. (CEA-DEMT, CEN-Saclay, 91 - Gif-sur-Yvette (France))

Corp. Author:

Source: Nuclear-Engineering-and-Design. (Jun 1992). v. 135(2). p. 141-149.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Methods **ID:** 438

Abstract: This paper presents a simplified engineering method to evaluate lower bounds of leak areas for circumferential through-wall cracks in view of their application in demonstrating the Leak-Before-Break behaviour of pipes. Starting from the simple elastic solution in a flat plate, bulging and plasticity correction factors are applied to determine leak rates in pipes. An amplification factor due to bulging is based in Sanders' shell solutions while, for plasticity, the Dugdale-Barenblatt model is used, introducing the reference-stress concept. The method is thus applicable for a large range of diameter-to-thickness ratios and for non-uniform applied loading situations. The method is tested against a number of results obtained either numerically (finite-elements analysis) or experimentally. The results are also compared with other known simplified methods used in the USA and FRG. The simplified approach presented here, to evaluate the crack leak areas in circumferential through-cracks, has been validated on a large range of diameter-to-thickness ratios and for non-uniform applied loads. The method seems promising for demonstration of Leak-Before-Break behaviour of pipes. (orig.).

Title: Basis UST leak detection systems.

Author: Silveria,-V. (Arizona Instrument Corp., Goldsboro, NC (United States))

Corp. Author:

Source: Plant-Engineering. (13 Aug 1992). v. 46(13). p. 74-77.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Inspection methods **ID:** 439

Abstract: This paper reports that gasoline and other petroleum products are leaking from underground storage tanks (USTs) at an alarming rate, seeping into soil and groundwater. Buried pipes are an even greater culprit, accounting for most suspected and detected leaks according to Environmental Protection Agency (EPA) estimates. In response to this problem, the EPA issued regulations setting standards for preventing, detecting, reporting, and cleaning up leaks, as well as fiscal responsibility. However, federal regulations are only a minimum; some states have cracked down even harder Plant managers and engineers have a big job ahead of them. The EPA estimates that there are more than 75,000 fuel USTs at US industrial facilities. When considering leak detection systems, the person responsible for making the decision has five primary choices: inventory reconciliation combined with regular precision tightness tests; automatic tank gauging; groundwater monitoring; interstitial monitoring of double containment systems; and vapor monitoring.

Title: International piping integrity research group (IPIRG) program final report.

Author: Schmidt,-R.; Wilkowski,-G.; Scott,-P.; Olsen,-R.; Marschall,-C.; Vieth,-P.; Paul,-D. (Battelle, Columbus, OH (United States))

Corp. Author: Atomic Energy Control Board,

Source: Apr 1992. 385 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** 440

Abstract: This is the final report of the International Piping Integrity Research Group (IPIRG) Programme. The IPIRG Programme was an international group programme managed by the USNRC and funded by a consortium of organizations from nine nations: Canada, France, Italy, Japan, Sweden, Switzerland, Taiwan, the United Kingdom, and the United states. The objective of the programme was to develop data needed to verify engineering methods for assessing the integrity of nuclear power plant piping that contains circumferential defects. The primary focus was an experimental task that investigated the behaviour of circumferentially flawed piping and piping systems to high-rate loading typical of seismic events. To accomplish these objectives a unique pipe loop test facility was designed and constructed. The pipe system was an expansion loop with over 30 m of 406-mm diameter pipe and five long radius elbows. Five experiments on flawed piping were conducted to failure in this facility with dynamic excitation. The report: provides background information on leak-before-break and flaw evaluation procedures in piping; summarizes the technical results of the programme; gives a relatively detailed assessment of the results from the various pipe fracture experiments and complementary analyses; and, summarizes the advances in the state-of-the-art of pipe fracture technology resulting from the IPIRG Program.

Title: Experimental study on a simulated primary-pipe rupture accident of HTGR. Experimental results of air ingress behavi

Author: Takenaka,-Satsuki; Takeda,-Tetsuaki; Hishida,-Makoto; Agake,-Takashi; Emori,-Koichi (Japan Atomic Energy Research Inst., Tokai, Ibaraki (Japan). Tokai Research Establishment)

Corp. Author: Japan Atomic Energy Research

Source: Mar 1994. 65 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** Japanese

Category: Test/analysis **ID:** 441

Abstract: A primary-pipe rupture accident is a critical design base accident of a HTGR. At the accident it is expected that air enters into the reactor core from the breach and the complicated natural convection of multi-component gas mixtures with graphite oxidation takes place. In order to investigate these phenomenon, therefore, we constructed an experimental apparatus simulating the pipe rupture and performed air ingress experiments. As a result, it was known that in the isothermal cases the period from the pipe rupture to the onset of the natural circulation of air (the first stage) lasts for a few days. In this report, in order to simulate the similar condition of the real plant, we report the non-uniform temperature cases and uniform-and-non-constant cases of the air ingress experiments. The results are as follows. (1) The period of the first stage of the non-uniform temperature cases is shorter than that of isothermal cases. (2) It owes the cooling speed of the reactor core, whether the natural circulation of air takes place or not. (author).

Title: Sub-project 'Development of moisture gauges' of the project 'Improvement of HTR safety by means of further develop

Author: Winkenbach

Corp. Author: Asea Brown Boveri AG, Mann

Source: 1991. 56 p.Bundesministerium fuer Forschung und Technologie, Bonn (Germany).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** German

Category: Inspection methods **ID:** 442

Abstract: The necessity of safely and quickly detecting high moistures as an indicator of heat exchanger pipe ruptures, in particular in the primary gas of the HTR 500, requires the development of a high-moisture measuring system. The demands on, and set objectives for, such high-moisture measuring system are presented. (DG).

Title: Vibrations of steam generator heat exchange tubes in accident regimes.

Author: Krupa,-V.; Pecinka,-L. **Corp. Author:** Ustav Jaderneho Vyzkumu a.s.,

Source: Apr 1993. 17 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** Czech

Category: Other **ID:** 443

Abstract: The draft regulations concerning the integrity condition and blinding of steam generator tubes at PWR nuclear power plants require a proof of their stability against hydrodynamic excitation in the normal operation and in accident regimes. Analysis for accidents involving rapid detachment of the collector lid, steam pipe rupture and feedwater pipe rupture gave evidence that: 1. no static instability or flutter due to longitudinal flow will occur during normal operation or after lid detachment; 2. hazardous self-excited vibrations at the water-steam interface can only occur after steam pipe rupture; and 3. stimulated vibrations can also take place only after steam pipe rupture. Hence, the tension values can be classed as safe. Only steam pipe rupture is thus found potentially hazardous, so that proof of integrity of the steam pipe system is sufficient. (Z.S.). 4 figs., 9 refs.

Title: Consequences of expansion joint bellows rupture.

Author: Daugherty,-W.L.; Miller,-R.F.; Cramer,-D.S. (Westinghouse Savannah River Co., Aiken, SC (United States). Savannah River Technology Center) **Corp. Author:** 2. Japan Society of Mechanical

Source: Peterson,-P.F. (ed.) (Univ. of California, Berkeley, CA (United States)). 2nd ASME-JSME international conference on nuclear engineering -- 1993. Volume 2. New York, NY (United States). American Society of Mechanical Engineers (ASME). 1993. 914 p. p. 855-858.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Damage probability **ID:** 444

Abstract: Expansion joints are used in piping systems to accommodate pipe deflections during service and to facilitate fitup. Typically, the expansion joint bellows is the thinnest part of the pressure boundary, a fact that is reflected in estimates of bellows rupture frequencies that are typically several orders of magnitude higher than pipe rupture frequencies. This paper reviews an effort to estimate the flow rates associated with bellows rupture. The Level 1 PRA for the SRS production reactors made the bounding assumption that bellows rupture would produce the maximum possible leakage--that of a DEGB. This assumption resulted in predictions of flooding of the reactor building with a high conditional probability that a Loss of Pumping Accident and core melting would follow. This paper describes analyses that were performed to develop a realistic break area and leak rate resulting from bellows rupture and therefore reduce the impact that bellows rupture can have on the estimated total core melt frequency. In the event of a 360 degree circumferential break of the bellows (conservative assumption), the resulting two sections will separate to the point where the force from the internal pressure acting to push the bellows open is just balanced by the spring force of the bellows itself. For the bellows addressed in this analysis, the equilibrium separation distance is 0.7 inches with normal pump lineup, providing a leak rate much less than would result from an assumed DEGB. This paper also discusses several related issues to place this result in perspective with regard to use in the PRA.

Title: Complex testing of 12Kh1MF steel on technical diagnosis of metal used in power machinery.

Author: Bugaj,-N.V.; Lebedev,-A.A.; Sharko,-A.V. (Novosibirskij Inst. Inzhenerov Zheleznodorozhnogo Transporta, Novosibirsk (Russian Federation)) **Corp. Author:**

Source: Defektoskopiya. (1992). (no.5). p. 47-53.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** Russian

Category: Inspection methods **ID:** 445

Abstract: The technique for complex estimation of metal strenght properties using three methods of nondestructive examination, when the strength characteristics being under testing are determined analytically according to the multiple regression equation connecting the results of nondestructive tests, is suggested. Taking as an example the results of testing steam pipelines of steel 12Kh1MF using the acoustic emission, electromagnetic and sampleless methods it is shown that introduction of complex tests decreases the volume of sampling rupture tests by 70-80%.

Title: Branch line pipewhip.

Author: Baum,-M.R. (Nuclear Electric plc, Berkeley (United Kingdom). Berkeley Technology Centre) **Corp. Author:** Piping engineering and operati

Source: Institution of Mechanical Engineers, London (United Kingdom); Institution of Chemical Engineers, London (United Kingdom). Piping engineering and operation. Proceedings. London (United Kingdom). Institution of Mechanical Engineers. 1993. 202 p. p. 179-196.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Analysis of break effects **ID:** 446

Abstract: There are many cases on power plant where large diameter pipes containing high pressure gas or steam have branch connections of a much smaller diameter. Frequently branch lines contain a closed valve. Here, pipe rupture adjacent to the main pipe means that the section of pipe on the valve side of the break is subject to a transient thrust, as the limited quantity of fluid between the closed valve and the open end is expelled. This section of pipe may also experience a force exerted by the impingement of the jet emerging from the main pipe. This paper considers the resulting pipewhip motion. The relative significance of the thrust resulting from the expulsion of the limited volume of fluid within the pipe and the subsequent jet impingement force, are explored. In addition the extent of the hazard zone and the peak kinetic energy of the pipe are determined. (Author).

Title: Benchmarking of MELCOR against RELAP5/MOD2 and plant data during the blowdown phase.

Author: Syrtmadzhiev,-A.; Ivanova,-A.; Balabanov,-E. (Energoproekt, Sofia (Bulgaria)) **Corp. Author:** Seminar on mathematical mode

Source: Committee on the Use of Atomic Energy for Peaceful Purposes, Sofia (Bulgaria). Mathematical models in nuclear safety and radiation protection. Collection of papers. 993. 248 p. p. 24-37.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** Bulgarian

Category: Analysis of break effects **ID:** 447

Abstract: Blowdown phase calculations with MELCOR and RELAP5/Mod2 for WWER-440/213 are made. The adequate simulation of the blowdown phase is of major importance to determine the starting time of core uncovering and heating-up, hence the fuel and cladding temperature gradients, the time for radioactive releases, the loss of integrity of the defensive barriers, etc. The benchmarking is performed for three cases of a small primary LOCA (a) Cold leg LOCA with an equivalent diameter 100 mm with a blackout and only one hydroaccumulator injecting to the upper plenum; (b) LOCA from the steam part of the pressurizer - a rupture of the pipe to the safety valves, with a complete loss of ECCS; (c) Faulty opening of the pressurizer safety valves for the Rovno-2. It is concluded that as a whole MELCOR well predicts the main thermohydraulic parameters in the leakage phase at suitable modelling of the breakage. 20 figs. (author).

Title: Rupture of pressurised tubes by multiple cracking and fragmentation.

Author: Ford,-I.J. (AEA Industrial Technology, Harwell (United Kingdom)) **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1994). v. 57(1). p. 21-29.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Methods **ID:** 448

Abstract: The likelihood of stable propagation of an axial crack away from a rupture site in a pressurised tube is a problem of concern in a number of areas, including the gas and nuclear industries. A model of crack propagation is developed which provides the crack velocity and deformation geometry and predicts a minimum driving pressure. Emphasis is placed upon the stability of propagation against small perturbations. The model also offers a criterion for the appearance of multiple cracks and subsequent fragmentation of the tube wall due to excessive bending strains. Calculations of interest in gas pipeline rupture and fast reactor fuel pin failure are presented. (author).

Title: The experience of RELAP4/MOD6 adaptation to analysis of RBMK accidents resulting from postulated coolant loop

Author: Dostov,-A.I.; Moskalev,-A.M.; Nikonov,-A.P. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Moscow (Russian Federation). Inst. Atomnoj Ehnergii)

Corp. Author:

Source: Gagarinskij,-A.Yu. (ed.). Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Moscow (Russian Federation). Inst. Atomnoj Ehnergii. Problems of nuclear science and technology. Scientific-technical collection. 1992. 96 p. p. 44-50.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** Russian

Category: Analysis of break effects **ID:** *449*

Abstract: RELAP4-MOD6 program widely used for PWR and BWR reactors is applied to the study of channel-type reactors RBMK-1000 and RBMK-1500 coolant loops thermohydraulics processes by accidents resulting from postulated ruptures of a pressure circuit pipelines. At the first stage of calculations is solved the general circuit problem. Then the service channel model is used for elaborate investigations of different power channels. The detailed description of the nodalization diagrams and used models is given. It is shown that fuel element can temperature variation during the first 20-30 s of the process has the character of a short-time splash. The temperature is maximal at ruptures with area of 25-30% of the pressure circuit pipeline cross section area. 2 refs.; 7 figs.; 6 tabs.

Title: Pipeline calculation for emergency rupture.

Author: Kostovetskij,-D.L.

Corp. Author:

Source: Teploehnergetika. (Jan 1992). (no.1). p. 49-52.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** Russian

Category: Research/theoretical **ID:** *450*

Abstract: The problem of calculation of NPP pipelines for emergency rupture is considered. The problem is being solved in simplified statement, namely, the plane unbranched pipeline is studied. It is supposed that the vector of the disturbing force (the reaction of outflowing jet) lays in the pipeline axis plane and its projections are the known functions of time. As a results the pipeline rigidity matrix is determined.

Title: Analysis of processes in the WWER-440/B-230 unit compartments during loss of coolant accident.

Author: Marinov,-M.; Popov,-E.; Dimitrov,-B.; Khinovski,-I.

Corp. Author:

Source: Ehlektricheskie-Stantsii. (Aug 1992). (no.8). p. 8-10.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** Russian

Category: Analysis of break effects **ID:** *451*

Abstract: Problem on preserving steam generator unit integrity at NPPs with WWER-440/B-230 reactors is considered. Results on studying processes in hermetically sealed rooms by joining steam generator compartments at the Kozloduj NPP four units at the place of explosive valves through coupling valve with cross section of 15 m sup 2 are presented. Loss-of-coolant accident in the primary circuit of one of the steam generator compartment by the pipeline rupture with equivalent diameter of 200 mm is studied. The calculations were performed by XEPMO program. The conclusion is made that the accident under consideration may be localized within the frames of sealed rooms.

Title: Safety criteria and safety evaluation results of HTTR.

Author: Tanaka,-T.; Iyoku,-T.; Kunitomi,-K.; Sawa,-K.; Nakagawa,-S.; Sudo,-Y. (Japan Atomic Energy Research Inst., Oarai, Ibaraki (Japan). Oarai Research Establishment); Okamoto,-F.

Corp. Author: ANP'92: international conferen

Source: Oka,-Y.; Koshizuka,-S. (comps.) (Tokyo Univ. (Japan)). Atomic Energy Society of Japan, Tokyo (Japan). ANP'92 international conference on design and safety of advanced nuclear power plants. Tokyo (Japan). Atomic Energy Society of Japan. 1992. [2182 p.]. v. 2 p. 18.1/1-18.1/7. Composed of four volumes.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Criteria **ID:** 452

Abstract: The High-Temperature Engineering Test Reactor (HTTR) developed by the Japan Atomic Energy Research Institute (JAERI) is a test reactor with thermal output of 30MW and outlet temperature of 950degC to establish basic technologies for advanced High Temperature Gas-cooled Reactors (HTGRs). At the JAERI, various safety evaluation has been performed to confirm the validity of safety design of the HTTR facility. This paper describes a brief description of the acceptance criteria of the HTTR and analytical results of depressurization accidents caused by a rupture of co-axial double pipes of primary cooling system and also a rupture of stand pipe of control rod drive mechanism housing, which are the major core heat up events with graphite oxidation and radiation exposure. (author).

Title: Leakage detection system in nuclear reactor container.

Author: Kurosawa,-Masahiko

Corp. Author: Toshiba Corp., Kawasaki, Kan

Source: 12 Mar 1993; 5 Sep 1991. 5 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993; 199 **Language:** Japanese

Category: Inspection methods **ID:** 453

Abstract: The present invention comprises an injection means for adding radioactive materials to coolants in a container cooler, a gamma ray amplitude analyzer connected to coolant pipelines and a means for recording/transferring the data of the result of the measurement, a gamma ray amplitude analyzer connected to a drain water sump and a means for recording/transferring the data of the result of the measurement, a gamma ray amplitude analyzer connected to various kinds of pipelines and a means for recording/transferring the data of the result of the measurement, and a data processing means for comparing and analyzing the measured data of each of the gamma ray amplitude analyzers inputted from each of date recording/transferring means. The gamma ray amplitude analysis for each of the pipelines and drain water sump are conducted at an appropriate frequency, and the measured data are compared and analyzed, to improve the detection accuracy for a trace amount of leakage from each of the pressure pipelines and the container cooler coolant pipelines, thereby enabling to specify the pipeline having leakage. Maintenance efficiency is improved, and severe rupture accident in each of pressure pipelines is prevented previously. (N.H.).

Title: The model experiments on the stationary outflow in the retaining system.

Author: Kucak,-L. (Slovenska Vysoka Skola Technicka, Bratislava (Czechoslovakia). Strojnicka Fakulta)

Corp. Author: ENS Topform '92: ENS East-

Source: European Nuclear Society (ENS), Bern (Switzerland); Czech Nuclear Society, Prague (Czech Republic); Slovak Nuclear Society, Bratislava (Slovakia). Topform '92: the safe and reliable operation of LWR NPPs. Vol. II. Poster papers. [Jan 1993]. 245 p. p. 132-134.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Analysis of break effects **ID:** 454

Abstract: The existing protection system for WWER-type nuclear power plants was designed to cope with a rupture of the pipeline with diameter less than 200 mm. The new requirements on the protection system, ie., to retain accident product outflow from the 500 mm main circulation pipe represent a great technical problem. In order to obtain data for designing a new primary pipe accident retaining system for the Bohunice V-1 power plant with overpressure not exceeding 100 kPa, model experiments were carried out. Basic features of the experiments are given. (Z.S.) 1 fig.

Title: Passive siphon break in a submerged pipe.

Author: Cole,-R.F.; Schindler,-C.R.; Sink,-A.M.; Morgan,-C.D. (Virginia Military Inst., Lexington (United States)) **Corp. Author:** American Nuclear Society ann

Source: Transactions-of-the-American-Nuclear-Society. (1992). v. 65. p. 484-486.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Other **ID:** 455

Abstract: A typical nuclear power generating facility includes an auxiliary spent-fuel storage tank to provide a safe storage location for spent-fuel assemblies. The assemblies must be completely submerged in water. In the event of an emergency, the suction side of the cooling system pipe could rupture creating a siphon. If the siphon remained unbroken, the water level in the tank would drop below the top of the fuel assemblies. The US Nuclear Regulatory Commission requires the use of a passive shut-off system to ensure termination of the siphon. To create an automatic siphon-terminating device, a 1.27-cm-diam hole was placed in the horizontal section of the suction pipe. A drop in the water level to that of the level of the 1.27-cm hole would result in air flow into the siphon. Sufficient air flow would terminate the siphon. There is no documented evidence that a 1.27-cm hole is sufficient. The purpose of this work is to develop a method to size the hole.

Title: the

Author: Futagawa,-Kiyoshi **Corp. Author:** Ishikawajima-Harima Heavy I

Source: 22 Mar 1994; 4 Sep 1992. 4 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994; 199 **Language:** Japanese

Category: Test/analysis **ID:** 456

Abstract: In a manufacturing step for IGSCC-induced for stainless steel pipelines, pipe are abutted against with each other and welded, and a heat affected portion is applied with a sensitizing heat treatment. Further, a crevice jig is attached near the heat affected portion at the inner surface of the pipe and kept in a chlorine ion added water under high temperature and high pressure at a predetermined period of time. If tap water is used instead of purified water for C.P.T. test in a step of forming sample of IGSCC, since the chlorine ion concentration in the tap water is relatively high, TGSCC (intragranular stress corrosion crackings caused in all of the samples. A heat input and an interlayer temperature are determined for the material of stainless pipe having a carbon content of more than 0.05% so that the welding residual stress on the inner surface is applied as tension. The condition for the heat treatment is determined as, for example, 500degC x 24hr, and the samples are kept under water at high temperature and high pressure applied with chlorine ions for 500 to 200hours. As a result, since samples of TGSCC can be formed by utilizing the manufacturing step for IGSCC, there is no requirement for providing devices for applying environmental factors separately. (N.H.).

Title: Stress intensity factor calculation for surface and subsurface semi-elliptical cracks in the outlet branch pipe zone of the

Author: Korinets,-A.R.; Chernysh,-T.A. (Moskovskij Inzhenerno-Fizicheskij Inst., Moscow (Russian Federation)); Maksimov,-Yu.M.; Semishkin,-V.P. **Corp. Author:**

Source: Atomnaya-Ehnergiya. (Aug 1992). v. 73(2). p. 87-91.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** Russian

Category: Research/theoretical **ID:** 457

Abstract: Calculation of stress intensity factor for subfused and surface semi-elliptical cracks in the zone of branch pipe joining with the WWER-440 vessel under conditions of maximum credible accident is discussed. The coefficient obtained by the method of crack virtual growth and asymptotic displacement method is compared with that determined on the bases of the influence functions for subfused and surface cracks in a plate.

Title: Detection and sizing of intergranular stress corrosion cracks in austenitic stainless steel piping of BWR.

Author: Bandyopadhyay,-M.; Mangsulikar,-M.D.; Nanekar,-P.P.; Shah,-B.K.; Kulkarni,-P.G. (Bhabha Atomic Research Centre, Bombay (India). Atomic Fuels Division) **Corp. Author:** AMNF-94: 1. national symposi

Source: Soman-Pillai,-M.D.; Sinha,-A.K.; Srinivasan,-V.S.; Srinivasan,-G.R. (comps.) (Nuclear Power Corporation of India Ltd., Bombay (India)). Department of Atomic Energy, Bombay (India). Board of Research in Nuclear Sciences. Ageing management of nuclear facilities (AMNF-94): proceedings. Bombay (India). Nuclear Power Corporation of India Ltd. 1994. [647 p.]. p. S7-25-S7-32.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Inspection methods **ID:** 458

Abstract: In boiling water reactors (BWR), austenitic stainless steels (Grades AISI 304 and 316) have been used for piping system. The welding of these pipes gives rise to sensitized microstructure and residual stress. In addition to this, presence of high temperature oxygenated water due to radiolysis provides highly corrosive environment. The conjoint action of the above three factors cause cracking along the grain boundaries which is referred to as intergranular stress corrosion cracking (IGSCC). Extensive investigations have been carried out in the last two decades to develop a technique to detect, locate and monitor the extent of IGSCC. Ultrasonic testing is found to be the most preferred NDT tool for this purpose. We have been carrying out periodic in-service inspection of BWR piping at Tarapur Atomic Power Station by ultrasonic testing to monitor the initiation and growth of IGSCC in weld heat affected zone of austenitic stainless steel piping. Of late, it has been realised that the existing technique has several limitations. Some new techniques based on ultrasonics like flaw tip diffraction method creeping longitudinal wave method, shear longitudinal inspection characteristics (SLIC) method, etc. are being developed to improve over the existing techniques. (author). 5 refs., 3 figs.

Title: SCC-induced failure of a 304 stainless steel pipe.

Author: Tapping,-R.L.; Disney,-D.J.; Szostak,-F.J. (Chalk River Laboratories, Ontario (Canada)) **Corp. Author:** 6. international symposium on

Source: Gold,-R.E.; Simonen,-E.P. (eds.). Proceedings of the sixth international symposium on environmental degradation of materials in nuclear power systems - water reactors. Warrendale, PA (United States). Minerals, Metals ampersand Materials Society. 1993. 963 p. p. 351-359.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 459

Abstract: On 1991 January 12, a 304 Stainless Steel (SS) suction line in the AECL-Research NRU reactor failed, shutting down the reactor for approximately 12 months. The pipe, a 32 mm schedule 40 304 stainless steel line exposed to D sub 2 O at temperatures <=35 degrees C had been in service for approximately 20 years, although no manufacturing data or composition specifications were available. The failure and resultant leak resulted in a small loss of D sub 2 O moderator from the reactor vessel. The pipe cracked approximately 180 degrees C around the circumference of a weld. This failure was unexpected and hense a thorough metallographic examination was carried out on the failed section, on the rest of the line (Line 1212), and on representative samples from the rest of the reactor in order to assess the integrity of the remaining piping.

Title: Metallurgical evaluation of weld overlaid pipe sections from Brunswick unit 2 Nuclear Power Station.

Author: Czajkowski,-C. (Brookhaven National Laboratory, Upton, NY (United States)) **Corp. Author:** 6. international symposium on

Source: Gold,-R.E.; Simonen,-E.P. (eds.). Proceedings of the sixth international symposium on environmental degradation of materials in nuclear power systems - water reactors. Warrendale, PA (United States). Minerals, Metals ampersand Materials Society. 1993. 963 p. p. 419-425.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 460

Abstract: A metallurgical assessment of four sections of weld overlaid pipe was performed. The investigation consisted of strain gage measurements, metallographic sectioning and mounting, scanning electron microscopy, hardness and ferrite measurements, radiography and dye penetrant examinations. A review of the fabrication history and original preservice and inserviceexaminations was performed and comparison was made to the actual cracks revealed after sectioning. In general, the report concludes that the weld quality of the overlays was consistent with ASME quality code class welds with adequate average 'as deposited' ferrite readings of FN>7. THe chemical analysis of the welds were normal for the alloys used (type 304 stainless steel, type 308 weld metal). The study also concludes that the ultrasonic inspection techniques used for inservice inspection of the overlays may not accurately depict the top 25% of the pipe in all cases and that crack growth is possible after weld overlay under certain conditions.

Title: Improving pitting corrosion resistance of type 304 austenitic stainless steel pipe weldments using purging gases with lo

Author: Huang,-W.; Paciej,-R.; Link,-L. (BOC, Murray Hill, NJ (United States)); Mckeown,-M. (BOC, Morden, London (United Kingdom))

Corp. Author: 6. international symposium on

Source: Gold,-R.E.; Simonen,-E.P. (eds.). Proceedings of the sixth international symposium on environmental degradation of materials in nuclear power systems - water reactors. Warrendale, PA (United States). Minerals, Metals ampersand Materials Society. 1993. 963 p. p. 387-390.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Research/theoretical **ID:** 461

Abstract: Pitting corrosion on the welded joints of water pipe lines may cause the shut down of nuclear power plants. The susceptibility to pitting corrosion is usually increased by high temperature oxidation during welding. In this study, an attempt was made to improve the pitting corrosion resistance of type 304 austenitic stainless steel pipe weldments by reducing the oxygen level in the purging gases. The pitting corrosion resistance of the as-received weldments was determined using stepwise cyclic polarization in aqueous solution containing chloride ions. It was found that relatively high oxygen levels (1-2%) in the purging gases caused severe high temperature oxidation, resulting in a non-uniform, porous, cracked, and thick surface oxide layer on the fusion and heat affected zones on the root pass of the pipe weldments. This high temperature oxidation also created a non-uniform distribution of chromium in the surface oxide layer, which, in turn, caused preferential pitting in the chromium depleted areas.

Title: Stress corrosion cracking of cold worked austenitic stainless steel pipes in BWR reactor water.

Author: Taehtinen,-S.; Haenninen,-H. (Technical Research Centre of Finland, Espoo (Finland)); Trolle,-M. (Swedish Nuclear Power Inspectorate (Sweden))

Corp. Author: 6. international symposium on

Source: Gold,-R.E.; Simonen,-E.P. (eds.). Proceedings of the sixth international symposium on environmental degradation of materials in nuclear power systems - water reactors. Warrendale, PA (United States). Minerals, Metals ampersand Materials Society. 1993. 963 p. p. 265-274.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 462

Abstract: The results of the failure analysis, material characterization and stress corrosion cracking (SCC) tests of cold bent, seamless austenitic stainless steel pipes removed from scram system 354 of Ringhals 1 BWR plant after 11 years operating time at 240 degrees C are presented. The material characterization and also the stress corrosion cracking susceptibility study of the cold bent pipes were performed for materials taken from different depths of the pipe wall thickness. Stress corrosion tests were carried out in simulated BWR water environments at 288 degrees C with varying oxygen contents. The results indicated that highly cold worked material is susceptible to IGSCC without any marked sensitization under constant loading, while dynamic loading results in TGSCC.

Title: Review of environmental effects on fatigue crack growth of austenitic stainless steels.

Author: Shack,-W.J.; Kassner,-T.F. (Argonne National Lab., IL (United States))

Corp. Author: Nuclear Regulatory Commissio

Source: May 1994. 28 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Test/analysis **ID:** 463

Abstract: Fatigue and environmentally assisted cracking of piping, pressure vessel cladding, and core components in light water reactors are potential concerns to the nuclear industry and regulatory agencies. The degradation processes include intergranular stress corrosion cracking of austenitic stainless steel (SS) piping in boiling water reactors (BWRs), and propagation of fatigue or stress corrosion cracks (which initiate in sensitized SS cladding) into low-alloy ferritic steels in BWR pressure vessels. Crack growth data for wrought and cast austenitic SSs in simulated BWR water, developed at Argonne National Laboratory under USNRC sponsorship over the past 10 years, have been compiled into a data base along with similar data obtained from the open literature. The data were analyzed to develop corrosion-fatigue curves for austenitic SSs in aqueous environments corresponding to normal BWR water chemistries, for BWRs that add hydrogen to the feedwater, and for pressurized water reactor primary-system-coolant chemistry.

Title: Real time pipe crack monitoring with OD surface probes of ID cracks.

Author: Solomon,-H.D.; Catlin,-W.R. (GE R ampersand D Center, Schenectady NY (United States)); Weinstein,-D. (GE Nuclear Energy, San Jose (Canada)); Pathania,-R. (Electric Power Research Institute, Palo Alto, CA (United States))

Corp. Author: 6. international symposium on

Source: Gold,-R.E.; Simonen,-E.P. (eds.). Proceedings of the sixth international symposium on environmental degradation of materials in nuclear power systems - water reactors. Warrendale, PA (United States). Minerals, Metals ampersand Materials Society. 1993. 963 p. p. 255-262.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** *464*

Abstract: Electrical potential measurements were made on the OD of a type 304 SS pipe, to measure stress corrosion crack growth emanating from the ID surface. The pipe was weld-sensitized and loaded in tension in 288 degrees C oxygenated water. No starter defects were made. This study monitored naturally initiated cracks and followed their growth to ultimate failure of the pipe. Since it was not known where cracks would initiate, probe wires were placed in the HAZ around the entire circumference of the pipe. Interpretation of the electrical potential changes was made with the aid of a computer program that analyzed these potentials in terms of the perturbation in the electric field caused by the crack.

Title: Experiments and calculations in support of the safety philosophy for the reconstruction of the V-1 NPP.

Author: Zdarek,-J.; Pecinka,-L.

Corp. Author: Ustav Jaderneho Vyzkumu a.s.,

Source: May 1993. 14 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** Czech

Category: LBB justification **ID:** *465*

Abstract: The IAEA document "Basic Safety Principles for NPP" (INSAG-3), which is currently recognized as the standard in the nuclear safety of steam-generating nuclear facilities, requires installation of reactor core emergency cooling systems as part of reconstruction of the Jaslovske Bohunice V-1 NPP, as well as the existence of a final barrier, i.e., the containment. While the first requirement can be satisfied, it is virtually impossible to implement the other. Reference to the assigned LBB (leak-before-break) statute is of no value in this context. An approach is suggested which has its logic with respect to current safety trends. In analogy to the definition of a "severe accident", it is possible to define a "severe LBB", or "0.1 F" (F is the primary piping cross section). The project proposes an approach which is extended to cases where cracks corresponding to a flow rate of 40 l/min occur but whose through-flow area is 10% with respect to the circulation piping cross section (0.1 F). The area of 0.1 F corresponds to a full cut of the inner diameter of 165 mm according to the newly proposed ECCS system. (Z.S.). 5 refs.

Title: Metallurgical evaluation of stress corrosion cracking in large diameter piping.

Author: Wheeler,-D.A.; Rawl,-D.E. Jr.; Louthan,-M.R. Jr. (Westinghouse Savannah River Co., Aiken, SC (United States). Savannah River Lab.)

Corp. Author:

Source: Materials-Characterization. (Jan 1994). v. 32(1). p. 25-33.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Test/analysis **ID:** *466*

Abstract: Ultrasonic testing (UT) of stainless-steel piping in the primary coolant water system of Savannah River Site (SRS) reactors indicates the presence of short, partly through-wall stress corrosion cracks in the heat-affected zone of approximately 7% of the circumferential pipe welds. These cracks are thought to develop by intergranular nucleation and mixed mode propagation. Metallographic evaluations have confirmed the UT indications of crack size and provided evidence that crack growth involved the accumulation of chloride ions inside the growing crack. It is postulated that the development of an oxygen depletion cell inside the crack results in the migration of chloride ions to the crack tip to balance the accumulation of positively charged metallic ions. The results of this metallurgical evaluation, combined with structural assessments of system integrity, support the existence of leak-before-break conditions in the SRS reactor piping system.

Title: Finite-element computation of large circumferentially cracked pipes.

Author: Faidy,-C.; Coustillas,-F.; Setz,-W.; Bhandari,-S.; Debaene,-J.P. **Corp. Author:** ASME-PVP Conference. New

Source: Bhandari,-S.; Milella,-P.P.; Pennell,-W.E. (eds.). Pressure vessel fracture, fatigue, and life management: PVP-Volume 233. New York, NY (United States). American Society of Mechanical Engineers. 1992. 312 pp. 187-192.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** *467*

Abstract: Results of analytical studies of a cooperative joint fracture mechanics program are presented. The program is concerned with bending of original DN 700 straight pipes with circumferential through-wall cracks. Material is the austenitic stainless steel 316L SPH. This paper is complementary to previous publications on the experimental part (Nucl. Engrg. Des. 108 (1988) 447-456) and the analytical part (Nucl. Engrg. Des. 119 (1990) 337-354). Details are given on studies using the finite element technique. Results are obtained only for crack initiation phase without modelizing the stable crack growth and comparison is made with those obtained in the experiments.

Title: Damage evaluation of a stratified feedwater line.

Author: Barthez,-M.; L'Huby,-Y.; Laclau,-J.N.; Faidy,-C. (EDF-SEPTEN, Villeurbanne (France)) **Corp. Author:** American Society of Mechanic

Source:

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Experience/events **ID:** *468*

Abstract: Following different cracked piping systems in different countries related to stratification, EDF decided on different Research and Development studies and some detailed analysis of practical situations. Analysis of the feedwater line is presented as an example of the methodology used in different locations in PWR plants to evaluate the consequences of these stratified situations with respect to fatigue damage.

Title: A benchmark on computational simulation of a CT fracture experiment. Compact Tension specimens.

Author: Franco,-C. (Framatome, Paris-la Defense (France)); Brochard,-J. (CEA-Saclay, Gif-sur-Yvette (France)); Ignaccolo,-S. (EDF-SEPTEN, Villeurbanne (France)); Eripret,-C. (EDF-DER, Moret-sur-Loing (France)) **Corp. Author:** American Society of Mechanic

Source: Bhandari,-S.; Milella,-P.P.; Pennell,-W.E. (eds.). Pressure vessel fracture, fatigue, and life management: PVP-Volume 233. New York, NY (United States). American Society of Mechanical Engineers. 1992. 312 p. p. 89-97.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Research/theoretical **ID:** *469*

Abstract: For a better understanding of the fracture behavior of cracked welds in piping, FRAMATOME, EDF and CEA have launched an analytical research program. This program is mainly based on the analysis of the effects of the geometrical parameters (the crack size and the welded joint dimensions) and the yield strength ratio on the fracture behavior of several cracked configurations. Two approaches have been selected for the fracture analyses: on one hand, the global approach based on the concept of crack driving force J and on the other hand, a local approach of ductile fracture. In this approach the crack initiation and growth are modelized by the nucleation, growth and coalescence of cavities in front of the crack tip. The model selected estimates only the growth of the cavities using the RICE and TRACEY relationship. The present study deals with a benchmark on computational simulation of CT fracture experiments using three computer codes : ALIBABA developed by EDF the CEA's code CASTEM 2000 and the FRAMATOME's code SYSTUS. The paper present the experimental procedure for high temperature toughness testing of two CT specimens taken from a welded pipe, characteristic of pressurized water reactor primary piping. Secondly, considerations are outlined about the FEM-analysis and the application procedure. A detailed description is given on boundary and loading conditions, on the mesh characteristics, on the numerical scheme involved and on the void growth computation. Finally, the comparisons between numerical and experimental results are presented up to the crack initiation, the tearing process being not taken into account in the present study. The variations of J and of the local variables used to estimate the damage around the crack tip (triaxiality and hydrostatic stresses, plastic deformations, void growth ...) are computed as a function of the increasing load.

Title: Recent progress in structural integrity assessment techniques for components subject to service-induced degradation.

Author: Mehta,-H.S. (General Electric Nuclear Energy, San Jose, CA (United States)) **Corp. Author:** 2. Japan Society of Mechanical

Source: Peterson,-P.F. (ed.) (Univ. of California, Berkeley, CA (United States)). 2nd ASME-JSME international conference on nuclear engineering -- 1993. Volume 2. New York, NY (United States). American Society of Mechanical Engineers (ASME). 1993. 914 p. p. 123-132.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Methods **ID:** 470

Abstract: NPP components are exposed to a wide range of environmental and loading conditions which can cause degradation over time. Aging embrittlement, erosion-corrosion, irradiation embrittlement, stress corrosion cracking, and corrosion fatigue are examples of aging mechanisms which could reduce structural margins in reactor components. The degradation effects from these mechanisms have been seen more frequently with the aging of the early nuclear plants. Since there is a strong incentive for keeping these older plants running for longer periods of time without compromising safety, proper plant management to minimize damage from degradation mechanisms is extremely important. Structural margin assessment, monitoring, and maintenance are important elements of such a management plan. Significant progress has been recently made in the understanding, evaluation and monitoring of these degradation mechanisms. This has led also to new requirements in the ASME Code design basis for nuclear plants. Current state of understanding and new developments in the ASME Code to address some of these degradation mechanisms are covered in this paper. Cast stainless steels used in pump casings and valve bodies have been known to experience thermal aging embrittlement at reactor operating temperatures. Recent predictive models of thermal aging effects on material toughness, developed at Argonne National Lab are reviewed and applied to assess ASME Code structural margins of a reactor pump casing. A recent ASME Code Case provides methods for the evaluation and acceptance criteria for reactor pressure vessels having ductile fracture toughness values reduced below the requirements of 10CFR50 due to irradiation embrittlement. Background and application of this code case to an older BWR vessel is described. The occurrence of stress corrosion cracking in austenitic stainless steel piping highlighted the need for evaluation methods for structural margin assessment in piping.

Title: Monitoring and prediction of environmentally assisted crack growth in stainless steel piping.

Author: Ranganath,-S. (GE Nuclear Energy, San Jose, CA (United States)) **Corp. Author:** 12. international corrosion con

Source: Anon.-Corrosion control for low-cost reliability: Preceedings. Electric power industry: Volume 6. Houston, TX (United States). NACE International. 1993. 258 p. p. 4185-4199.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Inspection methods **ID:** 471

Abstract: Stainless steel piping components used in nuclear power plants are exposed to the high temperature water environment and subjected to cyclic stresses as well as sustained stresses due to pressure, thermal and weld residual stresses. Crack initiation and subsequent growth due to corrosion fatigue and IGSCC are the potential environmentally assisted cracking (EAC) mechanisms that should be considered in the piping design. One way of monitoring and protecting against EAC is to perform periodic inspections of piping to provide assurance of piping integrity. If crack indications are discovered as a result of the inspections, an immediate question that arises is what the expected crack growth rate is and whether continued operation can be justified on a short term basis. Determination of the crack growth rate requires some form of monitoring and analytical prediction. This paper describes several monitoring techniques for predicting crack growth in austenitic stainless steel piping in BWR. The three types of monitoring systems -- the crack arrest verification system, the in-pipe electrochemical potential (ECP) monitoring and the in-core stress corrosion monitor -- provide plant specific environmental data. Prediction of plant component crack growth rate still requires extrapolation of the results of the monitoring system with crack growth predictive models. A major benefit of plant monitoring is that is enables measurement of the actual water chemistry parameters instead of relying on bounding values. This allows realistic crack growth predictions that can be used in planning and prioritizing inspections and in making operate as is vs. repair decisions. The environmental monitoring systems provide valuable water chemistry information which can be used to take corrective actions when operational problems arise and are also important when mitigation measures such as hydrogen water chemistry are implemented.

Title: J-integral of circumferential crack in large diameter pipes.

Author: Ji,-Wei; Chao,-Y.J.; Sutton,-M.A. (Univ. of South Carolina, Columbia, SC (United States). Dept. of Mechanical Engineering); Lam,-P.S.; Mertz,-G.E. (Westinghouse Savannah River Co., Aiken, SC (United States). Savannah River Technology Center)

Corp. Author: 2. Japan Society of Mechanical

Source: Peterson,-P.F. (ed.) (Univ. of California, Berkeley, CA (United States)). 2nd ASME-JSME international conference on nuclear engineering -- 1993. Volume 2. New York, NY (United States). American Society of Mechanical Engineers (ASME). 1993. 914 p. p. 139-149.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 472

Abstract: Large diameter thin-walled pipes are encountered in low pressure nuclear power piping system. Fracture parameters, such as K and J, associated with postulated cracks are needed to assess the safety of the structure, for example, prediction of the onset of the crack growth and the stability of the crack. The Electric Power Research Institute (EPRI) has completed a comprehensive study of cracks in pipes and handbook-type data is available. However, for some large diameter, thin-walled pipes the needed information is not included in the handbook. This paper reports the authors' study of circumferential cracks in large diameter, thin-walled pipes (R/t= 30 to 40) under remote bending or tension loads. Elastic-plastic analyses using finite element method were performed to determine the elastic and fully plastic J values for various pipe/crack geometries. A non-linear Ramberg-Osgood material model is used, with strain hardening exponent, n, ranging from 3 to 10. A number of circumferential, through thickness cracks were studied with half crack angles ranging from 0.063 pi to 0.5 pi. Results are tabulated for use with the EPRI estimation scheme.

Title: Application of a nonlinear spring element to analysis of circumferentially cracked pipe under dynamic loading.

Author: Olson,-R.; Scott,-P.; Wilkowski,-G.M. (Battelle, Columbus, OH (United States))

Corp. Author: American Society of Mechanic

Source: Bhandari,-S.; Milella,-P.P.; Pennell,-W.E. (eds.). Pressure vessel fracture, fatigue, and life management: PVP-Volume 233. New York, NY (United States). American Society of Mechanical Engineers. 1992. 312 p. p. 279-292.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Research/theoretical **ID:** 473

Abstract: As part of the US NRC's Degraded Piping Program, the concept of using a nonlinear spring element to simulate the response of cracked pipe in dynamic finite element pipe evaluations was initially proposed. The nonlinear spring element is used to represent the moment versus rotation response of the cracked pipe section. The moment-rotation relationship for the crack size and material of interest is determined from either J-estimation scheme analyses or experimental data. In this paper, a number of possible approaches for modeling the nonlinear stiffness of the cracked pipe section are introduced. One approach, modeling the cracked section moment rotation response with a series of spring-slider elements, is discussed in detail. As part of this discussion, results from a series of finite element predictions using the spring-slider nonlinear spring element are compared with the results from a series of dynamic cracked pipe system experiments from the International Piping Integrity Research Group (IPIRG) program.

Title: SM -- A new and unique method for monitoring of corrosion and cracking internally in piping systems and vessels. Fie

Author: Strommen,-R.D.; Horn,-H.; Wold,-K.R. (CorrOcean, Trondheim (Norway)) **Corp. Author:** 12. international corrosion con

Source: Anon.-Corrosion control for low-cost reliability: Preceedings. Electric power industry: Volume 6. Houston, TX (United States). NACE International. 1993. 258 p. p. 4141-4153.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Inspection methods **ID:** 474

Abstract: Over the last couple of decades there has been a substantial growth worldwide in the number of plants for generation of power, mainly electricity. As the plants grow older, the need for inspection and monitoring becomes ever more important to ensure a safe and non-interrupted operation of these plants. At the same time it is a challenge to optimize inspection and monitoring programs to reduce the expenditures for such programs. This paper describes a new technique known as the FSM, the Field Signature Method, that offers a means of continuous monitoring of the condition of pipes, pressurized vessels etc. of such plants, and of any corrosion, pitting or cracking that might take place, and of the remaining wall thickness at any time of a pipe or a vessel. It is claimed that this new FSM technique combines the advantages of corrosion probes and NDT/inspection: It offers high sensitivity and responds to changes in corrosion of the actual pipe wall in real time. This combines with an ability to cover relatively large areas of the actual structure. FSM removes the need for access fittings, for replacement of probes and for retrieval operations, reducing costs and improving safety. The System requires virtually no maintenance nor replacement of consumables. The service life of an FSM System equals that of the pipe work itself. FSM is therefore an excellent technique for monitoring any piping system, pressure vessels etc, and in particular inaccessible areas like buried and subsea pipelines and hazardous areas in nuclear power stations.

Title: Design and testing of equipment for nondestructive detection and identification of the location and dimensions of mate

Author: Wuensch,-W. **Corp. Author:**

Source: Feb 1992. 196 p. Bundesministerium fuer Umwelt, Naturschutz und Reaktorsicherheit, Bonn (Germany).Stuttgart Univ. (Germany). Staatliche Materialpruefungsanstalt.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** German

Category: Inspection methods **ID:** 475

Abstract: The prototype of a testing device for the nondestructive detection and identification of defect location and dimensions in a piping, especially of cracks in welded joints, has been evaluated on a laboratory scale. For a variety of reasons, it was not possible yet to perform trials in an industrial-scale system, as eg. in a power plant pipe system or the like. (orig./BBR).

Title: Short cracks in piping and piping wells. Volume 3, No. 2: Semiannual report, October 1992--March 1993.

Author: Wilkowski,-G.M.; Brust,-F.; Francini,-R. (Battelle, Columbus, OH (United States)) (and others) **Corp. Author:** Nuclear Regulatory Commissio

Source: Mar 1994. 103 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Research/theoretical **ID:** 476

Abstract: This is the sixth semiannual report of the USNRC's 4-year research program "Short Cracks in Piping and Piping Welds" which began in March 1990. The objective is to verify and improve fracture analyses for circumferentially cracked nuclear piping with cracks sizes typically found during in-service flaw evaluations. Progress is the through-wall-cracked pipe efforts involved (1) verification of deformation plasticity under nonproportional loading, (2) evaluation of the effect of weld metal strength on various J-estimation schemes, and (3) development of new GE/EPRI functions. Surface-cracked pipe evaluations involved (1) material characterization of B ampersand W C-Mn-Mo submerged arc weld metal, and (2) 3D finite-element mesh refinement study. The toughness of the bimetallic weld fusion line was evaluated and showed unusual fracture behavior based on the results of the Charpy tests. The dynamic strain aging J-R tests confirmed the screening criterion developed earlier in the program. The results from this program to date necessitated several additional efforts. These were initiated and have been reported here. Presentation of the results from this program to the ASME Section XI Pipe Flaw Evaluation Working Group is also summarized here.

Title: Insights for aging management of light water reactor components: Metal containments. Volume 5.

Author: Shah,-V.N.; Sinha,-U.P. (EG and G Idaho, Inc., Idaho Falls, ID (United States)); Smith,-S.K. (Ogden Environmental and Energy Services, Southfield, MI (United States)) **Corp. Author:** Nuclear Regulatory Commissio

Source: Mar 1994. 100 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Experience/events **ID:** 477

Abstract: This report evaluates the available technical information and field experience related to management of aging damage to light water reactor metal containments. A generic aging management approach is suggested for the effective and comprehensive aging management of metal containments to ensure their safe operation. The major concern is corrosion of the embedded portion of the containment vessel and detection of this damage. The electromagnetic acoustic transducer and half-cell potential measurement are potential techniques to detect corrosion damage in the embedded portion of the containment vessel. Other corrosion-related concerns include inspection of corrosion damage on the inaccessible side of BWR Mark I and Mark II containment vessels and corrosion of the BWR Mark I torus and emergency core cooling system piping that penetrates the torus, and transgranular stress corrosion cracking of the penetration bellows. Fatigue-related concerns include reduction in the fatigue life (a) of a vessel caused by roughness of the corroded vessel surface and (b) of bellows because of any physical damage. Maintenance of surface coatings and sealant at the metal-concrete interface is the best protection against corrosion of the vessel.

Title: Corrosion damage examples and brittleness affecting containing carbon steel materials of the PWR type reactors.

Author: Millet,-L.; Dordonat,-M.; Guttmann,-D.; Calle,-P. (Electricite de France (EDF), 93 - Saint-Denis (France)) **Corp. Author:** Autumn Days of the Societe Fr

Source: Revue-de-Metallurgie-Paris. (Sep 1993). v. 90(9). p. 1173.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** French

Category: Experience/events **ID:** 478

Abstract: Intercrystalline corrosion has been observed in carbon steel heat exchanger tubes. Waterproof turbine boxes composed of graphite rings and carbon steel, some coated with a KANIGEN chemical nickel, may develop a galvanic coupling corrosion between the graphite rings and the carbon steel body. In a steam impulsion pipe circuit, fatigue corrosion and stress corrosion cracking may appear. Brittleness of carbon steel is linked to an anomalous composition with an excess content of phosphorus and nitrogen. Lamellar wrenchings are observed on steam pipes connection. 5 refs., 3 figs.

Title: Modeling of residual stress mitigation in austenitic stainless steel pipe girth weldment.

Author: Li,-M.; Atteridge,-D.G.; Anderson,-W.E. (Oregon Graduate Inst., Portland, OR (United States)); West,-S.L. (Westinghouse Savannah River Co., Aiken, SC (United States)) **Corp. Author:** Westinghouse Savannah River

Source: [1994]. 10 p. USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Research/theoretical **ID:** 479

Abstract: This study provides numerical procedures to model 40-cm-diameter, schedule 40, Type 304L stainless steel pipe girth welding and a newly proposed post-weld treatment. The treatment can be used to accomplish the goal of imparting compressive residual stresses at the inner surface of a pipe girth weldment to prevent/retard the intergranular stress corrosion cracking (IGSCC) of the piping system in nuclear reactors. This new post-weld treatment for mitigating residual stresses is cooling stress improvement (CSI). The concept of CSI is to establish and maintain a certain temperature gradient across the pipe wall thickness to change the final stress state. Thus, this process involves sub-zero low temperature cooling of the inner pipe surface of a completed girth weldment, while simultaneously keeping the outer pipe surface at a slightly elevated temperature with the help of a certain heating method. Analyses to obtain quantitative results on pipe girth welding and CSI by using a thermo-elastic-plastic finite element model are described in this paper. Results demonstrate the potential effectiveness of CSI for introducing compressive residual stresses to prevent/retard IGSCC. Because of the symmetric nature of CSI, it shows great potential for industrial application.

Title: Numerical evaluation of stress intensity factor for vessel and pipe subjected to thermal shock.

Author: Kim,-Y.W.; Lee,-H.Y.; Yoo,-B. (Korea Atomic Energy Research Inst., Daeduk (Korea, Republic of). Mechanical Structure Development Div.) **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1994). v. 58(2). p. 215-222.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Research/theoretical **ID:** 480

Abstract: The thermal weight function method and the finite element method were employed in the numerical computation of the stress intensity factor for a cracked vessel and the cracked pipe subjected to thermal shock. A wall subjected to thermal shock was analyzed, and it has been shown that the effect of thermal shock on the stress intensity factor is dominant for the crack with small crack length to thickness ratio. Convection at the crack face had an influence on the stress intensity factor in the early stage of thermal shock. (Author).

Title: Safety analysis of its reactor system and its components. Calculation of stress intensity factors in A 32"-20" cracked pi

Author: Balestreri,-F. (Socotec Industrie (France)); Churier-Bossennec,-H. (EDF/SEPTEN, Villeurbanne (France)) **Corp. Author:** 20. annual meeting on nuclear t

Source: Deutsches Atomforum e.V., Bonn (Germany); Kerntechnische Gesellschaft e.V., Bonn (Germany). Jahrestagung Kerntechnik '93. Tagungsbericht. Bonn (Germany). INFORUM Verl. May 1993. 494 p. p. 191-194.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Damage probability **ID:** 481

Abstract: Short communication.

Title: Pre-cracked piping members made of tough materials and their behaviour under water hammer loads.

Author: Kussmaul,-K. (MPA, Stuttgart (Germany)); Kobes,-E. (MPA, Stuttgart (Germany)); Diem,-H. (MPA, Stuttgart (Germany)); Schrammel,-D. (KFK/PHDR, Karlsruhe (Germany)) **Corp. Author:** Annual meeting on nuclear tec

Source: Deutsches Atomforum e.V., Bonn (Germany); Kerntechnische Gesellschaft e.V., Bonn (Germany). Annual meeting on nuclear technoloy '92. Proceedings. Jahrestagung Kerntechnik '92. Tagungsbericht. Bonn (Germany). INFORUM Verl. May 1992. 532 p. p. 381-384.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** German

Category: Pressure ripple/water hammer **ID:** 482

Abstract: Short communication.

Title: Cyclic crack growth evaluation of a 20MnMoNi55 piping steel in high-oxygen reactor water.

Author: Aaltonen,-P. (Technical Research Centre of Finland (VTT), Metals Lab., Espoo (Finland)); Rintamaa,-R. (Technical Research Centre of Finland (VTT), Metals Lab., Espoo (Finland)); Haenninen,-H. (Technical Research Centre of Finland (VTT), Metals Lab., Espoo (Finland)); Ehrnsten,-U. (Technical Research Centre of Finland (VTT), Metals Lab., Espoo (Finland)); Arilahti,-E. (Technical Research Centre of Finland (VTT), Metals Lab., Espoo (Finland))

Corp. Author:

Source: Nuclear-Engineering-and-Design. (Oct 1993). v. 144(1). p. 111-122.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** 483

Abstract: Samples of a low alloy steel piping material taken from the full scale corrosion fatigue test loop of the Heissdampfreaktor (HDR) plant have been tested at 240 C in high oxygen reactor water. The small-scale specimens (CT25) were exposed to a similar loading spectrum to that which has been used in the full-scale corrosion fatigue tests at the HDR-plant. During the autoclave tests cyclic crack growth rates were determined. Fracture surface investigations were performed not only for the laboratory test specimens but also for the fracture surface of a sample taken from the HDR test loop piping after the full scale test. In this paper the autoclave testing results and fracture surface observations are presented and compared to those obtained in the HDR piping tests. (orig.).

Title: Numerical study of cracked pipe's behavior in the frame of ductile fracture.

Author: Meister,-E.

Corp. Author: Electricite de France (EDF), 92

Source: Aug 1992. 18 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** 484

Abstract: In order to characterize crack initiation, the energy release rate is calculated with the THETA method which consists, in a virtual kinematic of the crack, to solve the elastic problems (linear or non-linear) in the Lagrangian configuration. The considered models are two circumferentially surface-cracked pipes tested under four point bending loads. This kind of internal defect needs a special mesh in the thickness of the pipe and, associated with the non-linearity of materials (plasticity), leads to a large finite element model that is computed with the code PERMEAS on a CRAY YMP. Global values of load and crack opening displacement are calculated and compared to experimental values, with a good agreement. Local and global values of the energy release rate are also calculated and the stability of the THETA method is discussed. Calculating the fracture mechanics parameters under the hypothesis of 'proportional loading' is also discussed.

Title: Welding residual stresses at the intersection of a small diameter pipe penetrating a thick plate.

Author: Mochizuki,-Masahito (Mechanical Engineering Research Lab., Hitachi Ltd., Ibaraki (Japan)); Enomoto,-Kunio (Mechanical Engineering Research Lab., Hitachi Ltd., Ibaraki (Japan)); Okamoto,-Noriaki (Mechanical Engineering Research Lab., Hitachi Ltd., Ibaraki (Japan)); Saito,-Hideyo (Hitachi Works, Hitachi Ltd., Ibaraki (Japan)); Hayashi,-Eisaku (Hitachi Works, Hitachi Ltd., Ibaraki (Japan))

Corp. Author:

Source: Nuclear-Engineering-and-Design. (Nov 1993). v. 144(3). p. 439-447.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Research/theoretical **ID:** 485

Abstract: The pipe is welded to the plate, and TIG cladding is melted on the inner surface of the pipe to protect it from stress corrosion cracking due to environmentally-induced changes in nuclear power plant components. Welding residual stresses are calculated by heat conduction and thermal elastoplastic analyses using an assumption of 'simplified pass' to save the computing time, and are also measured by the strain-relief technique. Welding residual stresses after TIG cladding are shown to have no corrosive influence on the inner pipe surface, and the residual stresses are compressive enough to protect the pipe against stress corrosion cracking on the outer surface. (orig.).

Title: 3D computation of cracked piping components.

Author: Ignaccolo,-S.; Proix,-J.M.; Churier-Bossennec,-H.; Faidy,-C. (EDF-SEPTEN, Villeurbanne (France). Engineering and Construction Div.) **Corp. Author:** 1993 pressure vessel and pipin

Source: Garud,-Y.S. (ed.). Creep, fatigue, flaw evaluation, and leak-before-break assessment. New York, NY (United States). American Society of Mechanical Engineers. 1993. 301 p. p. 137-146.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Methods/comparison **ID:** 486

Abstract: The major rules for piping flaw evaluation are derived for simple geometries like plates and cylinders. During the past three years, many flaw evaluations considered more complex situations like elbows, tees or nozzles. The objective here is to present the main features of the actual french methodology for flaw evaluation and to discuss the transposition of these methods to complex piping components. Different examples of three dimensional cracked models are presented, with direct computation of elastic J and plastic J. The main objective of these computations is to develop a modified engineering method to analyze elbows, tees or nozzles.

Title: The conservatism of the net-section stress procedure for predicting the failure of cracked piping systems: The effect of

Author: Smith,-E. (Manchester Univ. (United Kingdom). UMIST Materials Science Centre) **Corp. Author:** 1993 pressure vessel and pipin

Source: Garud,-Y.S. (ed.). Creep, fatigue, flaw evaluation, and leak-before-break assessment. New York, NY (United States). American Society of Mechanical Engineers. 1993. 301 p. p. 161-170.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Damage probability **ID:** 487

Abstract: Interest in the integrity of cracked piping systems fabricated from ductile materials has been motivated, in large part, by the technological problem of intergranular stress corrosion cracking of Type 304 stainless steel piping in boiling water nuclear reactor piping systems. The failure of cracked steel piping is often predicted by assuming that failure conforms to a net-section stress criterion using as input an appropriate value for the critical net-section stress together with a knowledge of the anticipated loadings. The stresses at the cracked section are usually calculated via a purely elastic analysis based on the piping being uncracked. However because the piping is built-in at the ends into a larger component, and since the onset of crack extension requires some plastic deformation, use of the net-section stress approach can give overly conservative failure predictions. An earlier paper has quantified the extent of this conservatism, and has shown how it depends on the material ductility and the elastic flexibility of a piping system. Using the results of analyses for simple model systems, the present paper shows that, for the same cracked section geometry, the degree of conservatism is markedly influenced by the location of the cracked section within the system.

Title: Low cycle fatigue crack growth and ductile fracture under dynamic/cyclic loadings for Japanese carbon steel piping: P

Author: Miura,-Naoki; Fujioka,-Terutaka; Kashima,-Koichi (Central Research Inst. of Electric Power Industry, Tokyo (Japan). Fast Breeder Reactor Dept.); Kanno,-Satoshi; Hayashi,-Makoto (Hitachi Ltd., Ibaraki (Japan). Mechanical Engineering Research Lab.); Ishiwata,-Masayuki; Gotoh,-Nobho (Hitachi Ltd., Ibaraki (Japan). Dept. of Nuclear Engineering) **Corp. Author:** 1993 pressure vessel and pipin

Source: Garud,-Y.S. (ed.). Creep, fatigue, flaw evaluation, and leak-before-break assessment. New York, NY (United States). American Society of Mechanical Engineers. 1993. 301 p. p. 175-182.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 488

Abstract: Dynamic fracture behavior of circumferentially cracked pipe is important to evaluate the structural integrity of nuclear piping from the view point of leak-before-break concept under seismic conditions. Fracture tests were conducted for Japanese carbon steel (STS42) pipes which were subjected to cyclic or dynamic monotonic loading. This paper describes the analytical studies for these pipe tests. J-integral approach was applied to evaluate the cyclic crack growth. A new equation for calculating DELTA J for a circumferentially throughwall cracked pipe subjected to bending was proposed. The effects of dynamic or cyclic loading on pipe fracture were also investigated.

Title: Low cycle fatigue crack growth and ductile fracture under dynamic/cyclic loadings for Japanese carbon steel piping: P

Author: Kanno,-Satoshi; Kimoto,-Hiroshi; Hayashi,-Makoto (Hitachi Ltd., Ibaraki (Japan). Mechanical Engineering Research Lab.); Ishiwata,-Masayuki; Gotoh,-Nobho (Hitachi Ltd., Ibaraki (Japan). Dept. of Nuclear Engineering); Miura,-Noaki; Fujioka,-Terutaka; Kashima,-Koichi (Central Research Inst. of Electric Power Industry, Tokyo (Japan). Komae Research Lab.)

Corp. Author: 1993 pressure vessel and pipin

Source: Garud,-Y.S. (ed.). Creep, fatigue, flaw evaluation, and leak-before-break assessment. New York, NY (United States). American Society of Mechanical Engineers. 1993. 301 p. p. 171-174.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 489

Abstract: To evaluate the structural integrity of power plant piping during earthquakes, dynamic fracture strength of 4-inch carbon steel pipes with circumferential defect's were examined. Pipes were subjected to monotonic and alternate bending loads at room temperature. In monotonic loading tests, the maximum load to failure increased slightly with loading rate. The number of cycles to failure can be expressed by the ratio of load amplitude to plastic collapse load. Since the load ratio is independent of length and configuration but does not depend on whether a defect is partly or entirely through the pipe wall, it is useful for estimating the strength of piping subjected to seismic loads.

Title: A database to evaluate stress intensity factors of elbows with throughwall flaws under combined internal pressure and

Author: Chattopadhyay,-J.; Dutta,-B.K.; Kushwaha,-H.S.; Mahajan,-S.C.; Kakodkar,-A. (Bhabha Atomic Research Centre, Bombay (India). Reactor Design and Development Group)

Corp. Author: Bhabha Atomic Research Centr

Source: 1993. 24 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: LBB justification **ID:** 490

Abstract: The advent of LBB-concept has replaced the traditional design basis event of DEGB in the design of primary heat transport (PHT) piping. The use of LBB concept requires postulation of largest credible cracks at highly stressed locations and demonstration of its stability under the maximum credible loading conditions. Stress analysis of PHT piping in nuclear power plants shows that the highly stressed piping components are normally elbows and branch tees. This necessitates detailed fracture mechanics evaluation of piping connections by computing Stress Intensity Factor (SIF) and/or J-integral. Simple analytical solutions for evaluation of SIF and J-integral for cracks in straight pipes are readily available in literature. However, the same type of solutions for elbows and tees are limited in open literature. In the present work, a database is generated to evaluate SIF for throughwall circumferential and longitudinal cracks under combined internal pressure and bending moment. Different parameters to characterise a cracked elbow are pipe factor (h), pipe bore radius to thickness ratio (r/t) and crack length. Another parameter (sigma) is used to consider the relative magnitude of stresses due to internal pressure and remote bending moment. The database has been used to derive closed form expressions to evaluate SIF for elbow with cracks in terms of the aforementioned parameters. (author). 8 refs., 12 figs., 3 tabs.

Title: Promising materials for steam generator pipelines at NPP.

Author: Gerasimov,-V.I.; Zvezdin,-Yu.I.; Kuznetsov,-E.V.; Nosov,-G.F.; Sandler,-N.G.; Kharbina,-I.L.

Corp. Author:

Source: Teploehnergetika. (Oct 1992). (no.10). p. 44-48.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** Russian

Category: Other **ID:** 491

Abstract: The conditions of steam generator (SG) operation at NPP are considered. The conclusion is made that the main cause of SG piping system failure is the chloride cracking. Estimation of corrosion cracking resistance for several home-made steels is made. The promising steels for SG pipe manufacturing are the next ones: 08Kh14MF, 015Kh18M25, 04Kh15N6M3V, 08Kh18N10T.

Title: Fracture mechanics investigations on a pipe with a circumferential flaw supported by FEM.

Author: Brocks,-W. (Bundesanstalt fuer Materialforschung und -pruefung, Berlin (Germany)); Mueller,-W. (Bundesanstalt fuer Materialforschung und -pruefung, Berlin (Germany)); Noack,-H.D. (Bundesanstalt fuer Materialforschung und -pruefung, Berlin (Germany)); Veith,-H. (Bundesanstalt fuer Materialforschung und -pruefung, Berlin (Germany))

Corp. Author:

Source: Nuclear-Engineering-and-Design. (Sep 1993). v. 143(2-3). p. 171-185.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 492

Abstract: The transferability of crack resistance properties obtained from fracture mechanics specimens to analyse stable crack growth of a 120 surface flaw in a pipe of large diameter under pure bending is discussed supported by results of an elastic-plastic FEM calculation. The ratio of triaxiality, hydrostatic stress and the von Mises effective stress, is about 2.6 and does neither depend on the location phi at the crackfront (if phi<+-45) nor on the bending stress (if sigma sub b >100 MPa). Thus, stable crack growth of a circumferential surface flaw in a pipe under bending may conservatively be estimated using the J sub R -curve of a large or side-grooved CT specimen. An elastic FEM analysis of the pipe under four-point bending according to a test shows that the distribution of the bending stress across the cross-section is getting asymmetrical due to the ovalization of the pipe. The evaluation of the J-integral was performed using the bending moment versus crack mouth opening displacement curves measured at various locations along a 120 circumferential notch under four-point bending (notch radius 0.25 mm). The result reveals a much higher J-integral level caused by the same bending moment when the ovalization of the pipe is not taken into account. Thus, the question may be raised whether the four-point bending test on large diameter pipes with flaws will meet the worst case because in the vicinity of the connection between pipe and pressure vessel the high local stiffness of the system will prevent ovalization of the pipe. An estimate of the crack resistance under four-point bending with ovalization indicates that the J sub R -curve for the circumferential notch corresponds better to that of a CT-25 specimen with a fatigue precrack than to the J sub R -curve of a CT-25 specimen with a notch of 0.1 mm resp. 0.25 mm notch radius. (orig.).

Title: Evaluation of fatigue induced crack growth in primary coolant circuit piping of Bohunice V-1 nuclear power plant.

Author: Samohyl,-P.

Corp. Author: Ustav Jaderneho Vyzkumu a.s.,

Source: Feb 1993. 23 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** Czech

Category: Experience/events **ID:** 493

Abstract: The fatigue-induced crack growth in the primary coolant circuit piping of the Bohunice V-1 nuclear power plant was analyzed. Sections of the piping with the least favorable combination of the stress and material properties for the base and weld material were found. The maximum permissible crack depth according to ASME IVB-3500 Section XI was postulated in each section. Fatigue-induced crack growth analysis was carried out following ASME A-3000 and C-3000 Non-mandatory Appendices to Section XI. The loading blocks were determined from actual operating conditions. The technological crack growth rates were compared with experimental data, and a conservative procedure for determining the growth rate was developed. A computer program was set up to calculate the crack increment in a preselected time period. Evidence was gained that the probability of fatigue-induced piping failure is low. (Z.S.). 4 tabs., 8 figs., 8 refs.

Title: Environmentally assisted cracking in light water reactors. Semiannual report April--September 1992.

Author: Ruther,-W.E.; Chung,-H.M.; Chopra,-O.K.; Kassner,-T.F.; Majumdar,-S.; Park,-J.Y.; Sanecki,-J.E.; Hins,-A.G.; Shack,-W.J. (Argonne National Lab., IL (United States))

Corp. Author: Nuclear Regulatory Commissio

Source: Jun 1993. 63 p.: USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 494

Abstract: This report summarizes work performed by ANL on fatigue and environmentally assisted cracking (EAC) in LWRs during April-September '92. Topics include (1) fatigue and SCC of low-alloy steel used in piping, steam generators. and reactor pressure vessels. (2) EAC of cast stainless steels (SSs), and (3) radiation-induced segregation and irradiation-assisted SCC of Type 304 SS after accumulation of relatively high fluence. Data on fatigue of low-alloy steel in LWR environments have been reviewed. Based on fracture-mechanics models and engineering judgement, interim fatigue design curves were developed that are consistent with available fatigue-life data. Crack growth data were obtained on fracture-mechanics specimens of A533-Gr B and A106-Gr B ferritic steels and on cast austenitic SSs in the as-received and thermally aged conditions in simulated BWR water at 289 degrees C. The data were compared with predictions based on crack growth correlations for ferritic steels in oxygenated water and correlations for wrought austenitic SS in oxygenated water developed at ANL and rates in air from Section M of the ASME Code. Microchemical and microstructural changes in high- and commercial-purity Type 304 SS specimens from control-blade absorber tubes and a control-blade sheath from operating BWRs were studied by Auger electron spectroscopy and scanning electron microscopy. Slow-strain-rate-tensile tests were conducted on irradiated specimens in air and simulated BWR water.

Title: Boiling water reactor pipe cracking : prediction, detection and life extension.

Author: Rastogi,-P.K.; Shah,-B.K.; Kulkarni,-P.G. (Bhabha Atomic Research Centre, Bombay (India). Atomic Fuels Div.)

Corp. Author:

Source: Indian-Journal-of-Technology. (Jul 1993). v. 31(7). p. 530-534.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Methods/comparison **ID:** 495

Abstract: In boiling water reactors (BWRs), austenitic stainless steels (grades AISI 304 and 316) have been used for various piping systems. The welding of these pipes provides the sensitized microstructure as well as residual stress whereas high temperature oxygenated water due to radiolysis provides specific environment for intergranular stress corrosion cracking (IGSCC). After initial IGSCC incidents in BWR pipes in 1974, extensive investigations have been carried out in the two decades to develop various remedial measures. This paper reviews the various methods developed for prediction, detection and life extension of IGSCC affected BWR pipes. (author). 11 refs., 3 tabs., 8 figs.

Title: A limit load criterion to predict crack growth in stainless steel pipes.

Author: Kassir,-M.K.; Hofmayer,-C.H.; Bandyopadhyay,-K.K. (Brookhaven National Lab., Upton, NY (United States). Nuclear Energy Dept.)

Corp. Author:

Source: Engineering-Fracture-Mechanics. (1992). v. 43(5). p. 807-813.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** 496

Abstract: In a recent test program, specimens of circumferentially cracked type 304 stainless steel pipes were subjected to dynamic cyclic loading. The experimental data indicated a linear correlation between the limit load of the pipe's cross-section, assuming elastic-plastic material behavior, and the logarithm of the number of loading cycles which are required to drive the crack through the pipe's thickness. A similar criterion is postulated to investigate the crack growth behavior observed in a High Level Vibration Test (HLVT) Program performed on a large scale modified model of a pressurized water reactor primary coolant system made of an equivalent stainless steel material. The input motion in the HLVT Program induced inelastic stresses which were responsible for the crack propagation. Reasonable results are obtained in terms of the number of loading cycles required to propagate a part-through circumferential crack through the pipe's thickness. (author).

Title: Stress intensity factor solution for thin-walled straight pipes DN 700 under bending.

Author: Grueter,-L.; Setz,-W. (Siemens/KWU, Bergisch Gladbach (Germany)); Bhandari,-S.; Deschanels,-H. (Novatome, Lyon (France)); Faidy,-C. (Electricite de France (EDF), 69 - Villeurbanne (France)) **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1992). v. 52(3). p. 379-390.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** *497*

Abstract: Within a cooperative fracture mechanics programme between Electricite de France, Novatome and Siemens-Interatom, bending tests on circumferentially cracked straight stainless steel 316L pipes of typical 'liquid metal fast breeder reactor' (LMFBR) main piping dimensions were performed. In this report, the fracture properties for elastic conditions are summarized; experimental data are compared with finite element calculations. Additional data published in the literature are considered. Based on experiments and accompanying finite element calculations, the stress intensity factor, i.e. the solution for the elastic case, is derived. A recommended procedure for technical application is outlined. (author).

Title: Thermalhydraulic study of a stratified flow in a piping elbow (Application to the model Coufast).

Author: Peniguel,-C.; Stephan,-J.M. **Corp. Author:** Electricite de France (EDF), 92

Source: Nov 1992. 11 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** French

Category: Test/analysis **ID:** *498*

Abstract: In PWR's, mechanical damages (cracks) have been detected at the internal faces of steam generator feedwater piping and also in dead legs, when thermal stratification occurs. To gain some understanding on these issues, experimental and numerical programs have been set up at EDF. This paper reports a thermalhydraulic study of an elbow geometry under operating conditions leading to the establishment of a stable stratified flow. Results obtained with ESTET (a three dimensional finite differences-finite volume code solving the averaged Navier-Stokes equations) and comparisons with experimental data obtained on COUFAST (an analytical mock up, scale 1 of a French 900-MW PWR steam generator pipe elbow) are shown.

Title: Effect of aging on the predicted maximum moment-carrying capacity of circumferentially cracked cast stainless steel p

Author: Krishneswany,-P.; Scott,-P.; Wilkowski,-G. (Battelle, Columbus, OH (United States)) **Corp. Author:** Aging research information con

Source: Beranek,-A. (comp.). Nuclear Regulatory Commission, Washington, DC (United States). Office of Nuclear Regulatory Research. Aging research information conference: Proceedings. Volume 2. Sep 1992. 461 p. p. 341-368.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Research/theoretical **ID:** *499*

Abstract: Cast stainless steel used in LWR primary system components such as valve bodies, pump castings, pipe fittings, and piping is susceptible to thermal embrittlement at reactor operating temperatures, 280-320 C (536-608 F). This process of thermal aging causes an increase in the hardness and ultimate tensile strength of the steel, and at the same time a decrease in toughness. Work at ANL has shown that such thermal embrittlement due to changes in the microstructure can occur during the reactor lifetime of 40 years. The effect of this thermal degradation on the load-carrying capacity of circumferentially cracked piping is the subject of this work. In this study, both lower-bound and typical values of the J-R curves and the tensile properties for CF8M and CF8A cast stainless steels, which have been artificially aged to simulate 4, 8, 16, 32, and 48 years of service at 300 C (572 F), were used to predict the maximum load-carrying capacity of circumferentially cracked pipes. The effect of aging, that is, reduced toughness and increased strength, for different pipe diameters, crack geometries [i.e., through-wall cracks (TWC) and surface cracks (SC)], and crack sizes has been investigated. Since complete stress-strain curve fits as a function of aging were not available at this time, only three analyses methods could be used. The three analyses methods used to estimate the maximum load-carrying capacity of cracked pipes were: (1) a J-estimation scheme for TWC pipes developed by Paris, (2) a Plastic-Zone-Screening Criteria (DPZP) developed at Battelle which is applicable to both TWC and SC pipe, and (3) the R6 Option 1 method developed by CEGB which is also applicable for both TWC and SC pipe.

Title: Acceptance size of erosion thinning in carbon steel pipes subjected to internal pressure and tensile load.

Author: Hasegawa,-Kunio (Hitachi Europe Ltd., Maidenhead (United Kingdom)); Kanno,-Satoshi; Hirano,-Akihiko; Gotoh,-Nobuho; Saito,-Takashi

Corp. Author:

Source: Journal-of-Nuclear-Science-and-Technology-Tokyo. (Nov 1992). v. 29(11). p. 1080-1085.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Criteria **ID:** 500

Abstract: Structural integrity evaluation of local wall thinning caused by erosion is important for maintaining the integrity of piping systems in power generating plants. The pipe of interest is a STS 42 carbon steel pipe loaded by an axial load and subjected to an internal pressure. Acceptable thinning length and width were determined from the allowable size of circumferential and axial cracks in pipes, and the wall thickness is determined from the local membrane stress rule. Thus the acceptable extent and depth of wall thinning were proposed. Double-ended fracture can then be prevented if the local wall thinning is kept within this acceptable size. (author).

Title: Study of the sulphide stress corrosion cracking (SSCC) resistance of API SL GR B and X60 pipeline steels.

Author: Bao-Iturbe,-C. (Babcock and Wilcox Espanola. S.A. Bilbao (Spain)); Gutierrez-de-Saiz-Solabarria (Univ. Pais Vasco. Departamento Ingenieria Metalurgica y Control de Materiales. Bilbao (Spain))

Corp. Author:

Source: Revista-de-Metalurgia. (1993). v. 29(1). p. 3-12.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** Spanish

Category: Experience/events **ID:** 501

Abstract: A study of the sulphide stress corrosion cracking resistance at room temperature of API 5L Cr B and X60 pipeline steels has been carried out. The theoretical mechanisms in order to explain these phenomena and several operational failures of pipeline steel due to SSCC have been reviewed and the National Association of Corrosion Engineers (NACE) standard concerning SSCC has been described. The main factors of influence of the SSCC have been analysed, results are presented and conclusions are elaborated. (Author) 32 ref.

Title: Fatique and environmentally assisted cracking in light water reactors.

Author: Kassner,-T.F.; Ruther,-W.E.; Chung,-H.M.; Hicks,-P.D.; Hins,-A.G.; Park,-J.Y.; Shack,-W.J. (Argonne National Lab., IL (United States))

Corp. Author: Aging research information con

Source: Beranek,-A. (comp.). Nuclear Regulatory Commission, Washington, DC (United States). Proceedings of the Aging Research Information Conference. Volume 1. Sep 1992. 556 p. p. 189-210.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Experience/events **ID:** 502

Abstract:

Title: The first twenty years with section XI: Responding to service experience.

Author: Bamford,-W.H. (Westinghouse Energy systems, Pittsburgh, PA (United States)) **Corp. Author:** 7. international conference on p

Source: Vereinigung der Technischen Ueberwachungsvereine e.V., Essen (Germany). Pressure vessel technology. Proceedings. Vol. 2. Materials (2), manufacturing, quality. 1992. 613 p. p. 1266-1277.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Experience/events **ID:** 503

Abstract: This paper will provide an overall review of some of the key incidents of service-induced degradation in the safety related systems of light water reactors since the inception of Section XI. Included in the review will be the detection of each degradation incident, its subsequent resolution, and changes in Section XI which resulted from the issue. Examples of issues to be reviewed are BWR reactor vessel nozzle cracking, steam generator girth weld cracking (PWR), stainless steel pipe cracking (BWR) and ferritic feedwater pipe cracking (PWR). The history of service-induced degradation will then be used as a basis for review of the responsiveness of Section XI. The mechanisms of damage which have been observed will be discussed, along with those which might occur in the future as power plants age further. (orig.).

Title: Fracture mechanics calculations of pressure vessels and piping components using shell elements.

Author: Grebner,-H.; Hoefler,-A. (Gesellschaft fuer Reaktorsicherheit (GRS) mbH, Koeln (Germany)) **Corp. Author:** 7. international conference on p

Source: Vereinigung der Technischen Ueberwachungsvereine e.V., Essen (Germany). Pressure vessels technology. Proceedings. Vol. 1. Design, analysis, materials (1). 1992. 857 p. p. 477-490.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** 504

Abstract: Finite element analyses of pressure vessels and piping components with through the wall cracks have been performed using shell elements. For the fracture mechanics assessment of the components a postprocessor program for ADINA shell elements has been developed to evaluate J-integral values as crack driving forces for ductile material behaviour and wall penetrating cracks of regular simple shape. The performance and accuracy of this procedure have been proved on several test cases with pressure, bending and thermal loads. The method seems to be well suited for the assessment of such cracks against stable crack initiation. As a rule the results obtained with shell elements are calculated with approximately the same accuracy but with less overall amount than equivalent 3d-continuum element models would require. (orig.).

Title: Creep-fatigue crack propagation tests and the development of an analytical evaluation method for surface cracked pipe

Author: Shimakawa,-T. (Nuclear Systems Div., Kawasaki Heavy Ind. Ltd., Tokyo (Japan)); Takahashi,-H. (Research and Development Center, Toshiba Corp., Kawasaki (Japan)); Doi,-H. (Mechanical Engineering Research Lab., Hitachi Ltd., Ibaraki (Japan)); Watashi,-K. (Materials Development Section, Power Reactor and Nuclear Fuel Development Corp., Ibaraki (Japan)); Asada,-Y. (Dept. of Mechanical Engineering, Univ. Tokyo (Japan)) **Corp. Author:**

Source: Nuclear-Engineering-and-Design. (Mar 1993). v. 139(3). p. 283-292.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 505

Abstract: This paper shows test results and 3D/FEM estimations of the surface crack growth in a straight pipe and elbow under creep-fatigue conditions. Simplified estimation schemes such as CEGB/R6, CEA and GE/EPRI were also applied to straight pipe tests. The electrical potential method was successfully applied to measure the surface crack geometry; so crack propagation rates both for surface and thickness direction were measured. Predicted growth rates by 3D inelastic FEM analyses were compared with test data and the coincidence between test results and predictions was confirmed. Crack growth rates evaluated by the simplified method were also compared with test results and FEM results. The applicability of the simplified estimation scheme is discussed. (orig.).

Title: Assessment of the effects of surface preparation and coatings on the susceptibility of line pipe to stress-corrosion cracki

Author: Beavers,-J.A. (Cortest Columbus Technologies, Inc., OH (United States)) **Corp. Author:** American Gas Association, Inc.

Source: 24 Feb 1992. 196 p.American Gas Association, Inc., Arlington, VA (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** 506

Abstract: Objectives were to evaluate susceptibility of pipeline steel to SCC when coated with coal-tar enamel, fusion-bonded epoxy (FBE), and polyethylene tape coatings. The tests included standard cathodic disbondment tests, potential gradients beneath disbonded coatings, electrochemical measurements, and SCC tests. It was concluded that factors affecting relative SCC susceptibility of pipelines with different coatings are the disbonding resistance of the coating and the ability of the coating to pass cathodic protection (CP) current. FBE coated pipelines would be expected to exhibit good SCC resistance, since the FBE coating had high cathodic disbonding resistance and could pass CP current. Grit blasting at levels used at coating mills may be beneficial or detrimental to SCC susceptibility. Excellent correlation was found between th Almen strip deflection and change in SCC threshold stress. It appears to be beneficial to remove as much mill scale as possible, and a white surface finish probably should also be specified. 50 figs, 10 tabs.

Title: Assessment of susceptibility of Type 304 stainless steel to intergranular stress corrosion cracking in simulated Savann

Author: Ondrejcin,-R.S.; Caskey,-C.R. Jr. **Corp. Author:** Westinghouse Savannah River

Source: 1 Dec 1989. 206 p.FUNDING ORGANIZATION: USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Test/analysis **ID:** 507

Abstract: Intergranular stress corrosion cracking (IGSCC) of Type 304 stainless steel rate tests (CERT) of specimens machined was evaluated by constant extension from Savannah River Plant (SRP) decontaminated process water piping. Results from 12 preliminary CERT tests verified that IGSCC occurred over a wide range of simulated SRP envirorments. 73 specimens were tested in two statistical experimental designs of the central composite class. In one design, testing was done in environments containing hydrogen peroxide; in the other design, hydrogen peroxide was omitted but oxygen was added to the environment. Prediction equations relating IGSCC to temperature and environmental variables were formulated. Temperature was the most important independent variable. IGSCC was severe at 100 to 120C and a threshold temperature between 40C and 55C was identified below which IGSCC did not occur. In environments containing hydrogen peroxide, as in SRP operation, a reduction in chloride concentration from 30 to 2 ppB also significantly reduced IGSCC. Reduction in sulfate concentration from 50 to 7 ppB was effective in reducing IGSCC provided the chloride concentration was 30 ppB or less and temperature was 95C or higher. Presence of hydrogen peroxide in the environment increased IGSCC except when chloride concentration was 11 ppB or less. Actual concentrations of hydrogen peroxide, oxygen and carbon dioxide did not affect IGSCC. Large positive ECP values (+450 to +750 mV Standard Hydrogen Electrode (SHE)) in simulated SRP environments containing hydrogen peroxide and were good agreement with ECP measurements made in SRP reactors, indicating that the simulated environments are representative of SRP reactor environments. Overall CERT results suggest that the most effective method to reduce IGSCC is to reduce chloride and sulfate concentrations.

Title: Stress corrosion cracking of steam generator tube and primary pipe in PWR type nuclear power plants.

Author: Zhang-Weiguo; Gao-Fengqin; Zhou-Hongyi (Academia Sinica, Beijing, BJ (China). Inst. of Atomic Energy) **Corp. Author:** China Nuclear Information Ce

Source: Mar 1992. 20 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** Chinese

Category: Test/analysis **ID:** 508

Abstract: The behavior of stress corrosion cracking (SCC) was studied by slow strain rate test (SSRT), constant load test (CLT) and low frequency cyclic loading test (LFCLT). The purpose of these tests is to get the test data for evaluating the integrity of pressurized boundary of pipes in Qinshan and Guangdong nuclear power plants (NPPs). Tested materials are 316 nuclear grade stainless steel (SS) for primary pipes in welded heat affected zone (WHAZ) and tubes of heat transfer, such as Incoloy-800, Inconel-600 and 321 SS which are used for steam generator in PWR NPPs. The effects of material metallurgy, shot peening treatment, tensile load, strain rate, cyclic load and water chemistry on the behavior of SCC were considered.

Title: Short cracks in piping and piping welds.

Author: Wilkowski,-G.; Ahmad,-J.; Brust,-F.; Francini,-R.; Krishnaswamy,-P.; Landow,-M.; Marschall,-C.; Rahman,-S.; Scott,-P.; Vieth,-P.

Corp. Author: 18. water reactor safety inform

Source: Weiss,-A.J. (comp.) (Brookhaven National Lab., Upton, NY (United States)). Nuclear Regulatory Commission, Washington, DC (United States). Office of Nuclear Regulatory Research; Brookhaven National Lab., Upton, NY (United States). Eighteenth water reactor safety information meeting. Volume 3, Pressure vessel integrity; Piping and NDE; Aging and components: Proceedings. Apr 1991. 574 p. p. 235-250.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Research/theoretical **ID:** 509

Abstract: The overall objective of the Short Cracks in Piping and Piping Welds Program is to verify and improve engineering analyses to predict the fracture behavior of circumferentially cracked pipe under quasi-static loading. Specific efforts focus on clarification of technical issues that were unresolved in the Degraded Piping Program - Phase II. In fiscal year 1990, the program was started in several different areas. The program consists of 7 technical tasks. The tasks are as follows: (1) short through-wall cracked (TWC) pipe evaluations; (2) short surface-cracked (SC) pipe evaluations; (3) bi-metallic weld crack evaluations; (4) dynamic strain aging and crack instabilities; (5) fracture evaluations of anisotropic pipe; (6) crack-opening-area evaluations; and (7) Nuclear Regulatory Commission's (NRCPIPE) Code improvements. Summary of task progress is provided for each active task.

Title: Statement of incidents at nuclear installations: third quarter 1992.

Author:

Corp. Author: Health and Safety Executive, L

Source: Quarterly-Statement-on-Nuclear-Incidents. (4 Jan 1993). (4 Jan 1993 issue). [3 p.].

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 510

Abstract: Three incidents are reported for the third quarter of 1992. During a radiological survey of British Nuclear Fuel's site at Sellafield in June, contamination of the ground under a cracked pipebridge was found. Contamination of two workers was removed by washing; the contaminated soil was removed and contained in drums. In September on the same site, a pipe failure occurred and plutonium nitrate leaked into the secondary containment cell leading to a shutdown of the reprocessing plant. However, no discharge of radioactivity to the environment and no additional radiation exposure to workers occurred. This was subsequently classified as a level 3 incident. 25 spots of radioactive contamination of a service road at the United Kingdom Atomic Energy Authority's Winfrith site were removed and disposed of without injury or contamination. Recommendations to improve the site roads and car parks were made. (UK).

Title: Stress corrosion cracking of 316 SS and Incoloy-800 in high temperature aqueous containing sulfate and chloride.

Author: Zhang-Weiguo; Lin-Fangliang; Gao-Fengqin; Zhou-Hongyi; Cao-Xiaoning (Academia Sinica, Beijing, BJ (China). Inst. of Atomic Energy)

Corp. Author: China Nuclear Information Ce

Source: Mar 1992. 10 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** 511

Abstract: The stress corrosion cracking (SCC) susceptibility of 316 stainless steel (SS) which was welded for primary pipe and Incoloy-800 (shot peening) for steam generator (SG) tube have been investigated by a slow strain rate test (SSRT) at a strain rate of 4.2 x 10 sup - sup 6 /s. Tests were conducted at 315 C degree for 316 SS and 270 C degree for In-800 in the oxygenated simulated resin intrusion environment (acidic sulfate). Tests of the effect of combination of SO sub 4 sup 2 sup - and Cl sup - on SCC of Incoloy-800 were also carried out. The results indicate that Incoloy-800 is unsusceptible to SCC either in the environment with SO sub 4 sup 2 sup - (from a few ppm to 1000 ppm, pH 3 approx 4) or in the environment of combination of SO sub 4 sup 2 sup - (1000 ppm) and Cl sup - (from 2 to 1000 ppm). The 316 NG SS is susceptible to transgranular stress corrosion cracking (TGSCC) in the resin intrusion environment with SO sub 4 sup 2 sup - in high temperature water.

Title: Surface crack testing - state of technique and trends in development. Proceedings.

Author: **Corp. Author:** Deutsche Gesellschaft fuer Zers

Source: 1991. 87 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** German

Category: Inspection methods **ID:** 512

Abstract: This Seminar contains 12 lectures on the following subjects: State of technique in magnetic powder testing (K. Goebbels); Recognisability of faults and probability of faults in surface crack testing (W. Morgner); Requirements for picture processing systems for proving and assessing crack indications (M. Stadthaus); Possibilities and limits of automatic crack recognition in magnetic powder testing (V. Deutsch); Development of equipment for eddy current testing (M. Junger); Signal processing - a way of improving the recognisability of faults in eddy current testing (R. Becker); Methods of testing steel products for surface faults and their practical limits of fault recognisability (D. Thiery); Surface crack testing in pipe manufacture (R. Pawelletz); Surface crack testing in powerstation construction (L. v. Bernus); Trends in automation in surface crack testing (G. Maier); Eddy current testing in engine construction (E. Dickhaut); Eddy current testing in aircraft repair (F. Schur). (orig.).

Title: Procedure of crack shape determination by Reversing DC Potential Method.

Author: Hashimoto,-Yukihiro; Urabe,-Yoshio (Mitsubishi Heavy Industries Ltd., Takasago Research and Development Center, Hyogo (Japan)); Masamori,-Shigero; Kamiwaki,-Yoshiharu (Mitsubishi Heavy Industries Ltd., Kobe Shipyard and Machinery Works, Hyogo (Japan)); Baba,-Kinji (Mitsubishi Heavy Industries Ltd., System Engineering Dept., Kobe, Hyogo (Japan)) **Corp. Author:**

Source: Nuclear-Engineering-and-Design. (Dec 1992). v. 138(3). p. 259-268.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Inspection methods **ID:** 513

Abstract: On-line monitoring of a crack and evaluation of component integrity are needed for maintaining the safety of a plant. The authors have been developing this kind of system using the Reversing DC Potential Method (RDCPM). The crack shape estimation based on RDCPM is performed by comparing the measured potential difference with the analytical potential difference. In this paper the simplified method for determing the crack shape is discussed and its application to a pipe is shown. (orig.).

Title: Stable crack growth of axial flaws in pressure vessels. StE 460, 20 MnMoNi 5 5.

Author: Brocks,-W.; Krafka,-H.; Kuenecke,-G.; Wobst,-K. (Bundesanstalt fuer Materialforschung und -pruefung (BAM), Berlin (Germany)) **Corp. Author:**

Source: Nuclear-Engineering-and-Design. (Jun 1992). v. 135(2). p. 151-160.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** 514

Abstract: The ductile crack growth of axial through and part-through cracks in a vessel under internal pressure has been studied experimentally to contribute to the fundamental problem whether or not and under which conditions resistance curves obtained from specimens can be transferred to large scale components. The experiments and numerical analyses are part of a research program on fracture mechanics failure concepts for the safety assessment of nuclear components. Whereas only an averaged crack extension is determined in specimen tests, the local propagation of cracks may be of main importance for surface cracks in vessels and pipes. In the present experiments, the surface cracks revealed the well known canoe shape, i.e. a larger crack extension has occured in the axial direction than in the wall thickness direction. Two of these tests have been analysed by finite element calculations to obtain the variation of the J-integral along the crack front and the stress and strain state in the vicinity of the crack. The local resistance appeared to depend on the local stress state. To predict ductile crack extension correctly, J sub R -curves have to account for the varying triaxiality of the stress state along the crack front. (orig.).

Title: Bringing longer life to LWR pipe: update on MSIP. Success of mechanical stress improvement process in Boiling Wat

Author: Porowski,-J.S.; Badlani,-M.L. (AEA O'Donnell, Pittsburgh, PA (United States)) **Corp. Author:**

Source: Nuclear-Engineering-International. (Jul 1992). v. 37(456). p. 40-42.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Methods/design **ID:** 515

Abstract: Intergranular stress corrosion cracking (IGSCC) is a recognized problem that has damaged weldments in BWR plants, and many piping systems in older plants have had to be replaced. The mechanical stress improvement process (MSIP) was developed to eliminate tensile stresses from weldments where sensitization in the presence of reactor water makes the material susceptible to stress corrosion attack. Tensile stress is a major contributor to this problem, and its elimination permanently protects the weldment. MSIP arrests cracks that have already developed and prevents further cracks initiating, extending the life of the weldments for as long as remaining plant life. For replaced piping, MSIP provides ultimate protection of weldments for the expected life of the plant. Use of MSIP is especially beneficial for nozzle weldments which are not immune to stress corrosion cracking, even in replaced piping systems. MSIP has had a 100% successful track record of performance in actual plants since its first application in 1986. (author).

Title: Fatigue and environmentally assisted cracking in light water reactors.

Author: Kassner,-T.F.; Ruther,-W.E.; Chung,-H.M.; Hicks,-P.D.; Hins,-A.G.; Park,-J.Y.; Shack,-W.J. (Argonne National Lab., IL (United States)) **Corp. Author:** 19. Nuclear Regulatory Comm

Source: Weiss,-A.J. (comp.) (Brookhaven National Lab., Upton, NY (United States)). Nuclear Regulatory Commission, Washington, DC (United States). Office of Nuclear Regulatory Research; Brookhaven National Lab., Upton, NY (United States). Proceedings of the US Nuclear Regulatory Commission nineteenth water reactor safety information meeting. Volume 1, Plenary session; Pressure vessel and piping integrity; Metallurgy and NDE; Aging and components; Probabilistic risk assessment topics. Apr 1992. 523 p. p. 127-150.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Experience/events **ID:** 516

Abstract: Fatigue and environmentally assisted cracking of piping, pressure vessels, and core components in light water reactors (LWRs) are important concerns as extended reactor lifetimes are envisaged. Topics that have been investigated during this year include fatigue and stress corrosion cracking (SCC) of low-alloy steel used in piping and in steam generator and reactor pressure vessels, role of chromate and sulfate in simulated boiling water reactor (BWR) water on SCC of sensitized Type 304 SS, and radiation-induced segregation (RIS) and irradiation-assisted SCC of Type 304 SS after accumulation of relatively high fluence. Fatigue data obtained on medium-sulfur-content A533-Gr B and A106-Gr B pressure-vessel and piping steels in high-purity (HP) deoxygenated water, in simulated pressurized water reactor (PWR) water, and in air all lie above the ASME design curve. Crack-growth-rate (CGR) measurements on composite specimens of A533-Gr B/Inconel-182/Inconel-600 and on homogeneous specimens of A533-Gr B material indicate that CGRs increased markedly during small-amplitude cyclic loading in HP water with approx 300 ppb dissolved oxygen. Under cyclic loading, crack growth was observed at K sub m sub a sub x values that produced no crack growth under constant loading. The CGR dependence on dissolved-oxygen concentration was also investigated under different loading conditions. Possible synergistic reactions involving chromate and sulfate in SCC of sensitized Type 304 SS have been investigated by fracture-mechanics CGR tests. Microchemical and microstructural changes in HP and commercial-purity Type 304 SS specimens from control-blade absorber tubes used in two operating BWRs were studied by Auger electron spectroscopy and scanning electron microscopy, and slow-strain-rate-tensile tests were conducted on tubular specimens in air and in simulated BWR water at 289C.

Title: Characteristics of fatigue crack development of carbon steel for piping STS 42 in high temperature atmosphere.

Author: Nakamura,-Haruo (Tokyo Inst. of Tech. (Japan). Faculty of Engineering) **Corp. Author:**

Source: Haikan-Gijutsu. (Nov 1992). v. 34(13). p. 55-60.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** Japanese

Category: Test/analysis **ID:** 517

Abstract: The high temperature range in which fatigue-creep interaction does not occur is called medium-high temperature range. For the piping of light water reactors, it is the present state to use mostly carbon steel instead of stainless steel, accordingly, it is required to accumulate the data on fatigue crack development in relation to the evaluation of structural soundness. In this report, on the carbon steel for LWR piping STS 42, on its characteristics of fatigue crack development in the medium-high temperature range around 288degC in the simulated environment of LWRs (supposing internal cracks), the effect of accelerating and retarding crack development and the effect that high temperature oxidation films exert to the lower limit characteristics are discussed in comparison with those in the room temperature atmosphere. Besides, based on the results of the statistical analysis of the rate of fatigue crack development in the low alloy steel for LWR pressure vessels, the formula for the rate of crack development in carbon steel is presented, and its application to the evaluation of structural soundness is explained. (K.I.).

Title: Causes of failure of WWER-440 primary coolant circuit materials.

Author: Kupca,-L.; Beno,-P.; Brezina,-M. (Vyskumny Ustav Jadrovych Elektrarni, Trnava (Czechoslovakia)) **Corp. Author:**

Source: Spravodajca-Vyskumny-Ustav-Jadrovych-Elektrarni. (1992). v. 9(3). p. 9-18.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** Slovak

Category: Experience/events **ID:** 518

Abstract: Failures of the various technological assemblies of the primary coolant circuits of WWER-440 type reactor units are discussed. Particular examples illustrating the causes of failures are presented. Stress corrosion cracking (of the hot steam generator collector or the collector tube plate, etc.) and point corrosion (e.g. defects in heat exchangers of the emergency cooling system) are frequent causes of failure. Inappropriate technology of manufacture of some components and inappropriate technological assembling procedures at the nuclear power plant also contribute. This is illustrated with the following examples: unsuitable material for primary piping elbows, crack in a blind plug of a steam generator heat exchanger tube, and cracks in the austenitic overlay of the pressure vessel of unit 1 at the Bohunice NPP in the radius reducer of the necks and at places where lining has been repaired. (Z.S.). 2 tabs., 9 figs., 15 refs.

Title: Environmentally assisted cracking in light water reactors. Semiannual report, October 1991--March 1992: Volume 14

Author: Chung,-H.M.; Kassner,-T.F.; Majumdar,-S.; Park,-J.Y.; Purohit,-A.; Ruther,-W.E.; Sanecki,-J.E.; Shack,-W.J. (Argonne National Lab., IL (United States)) **Corp. Author:** Nuclear Regulatory Commissio

Source: Aug 1992. 55 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Research/theoretical **ID:** 519

Abstract: Fatigue and environmentally assisted cracking of piping, pressure vessels, and core components in light water reactors are important concerns as extended reactor lifetimes are envisaged. Topics investigated during this year include (1) fatigue and stress corrosion cracking (SCC) of low-alloy steel used in piping and in steam generator and reactor pressure vessels, (2) radiation-induced segregation (RIS) and irradiation-assisted SCC of Type 304 SS after accumulation of relatively high fluence, and (3) update of a crack growth data base for austenitic and ferritic steels in high-temperature water. Existing data on fatigue of low-alloy steel in LWR environments have been reviewed. Based on fracture-mechanics models and engineering judgement, interim fatigue design curves are being developed that are consistent with available fatigue-life data. Microchemical and microstructural changes in high- and commercial-purity type 304 SS specimens from control-blade absorber tubes and a control-blade sheath from operating BWRs were studied by Auger electron spectroscopy and scanning electron microscopy. Slow-strain-rate-tensile tests were conducted on irradiated specimens in air and in simulated BWR water at 289 degree C. Crack growth data on fracture-mechanics specimens of austenitic and ferritic steels in simulated BWR water, developed in this program over the past eight years, were compiled into a data base along with references that contain details of test methods, material compositions, metallographic information, and comparisons of data with predictions based on the new crack growth curves proposed for inclusion in Section 9 of the ASME Code.

Title: Intergranular stress corrosion cracking: A rationalization of apparent differences among stress corrosion cracking tend

Author: Louthan,-M.R. **Corp. Author:** Westinghouse Savannah River

Source: 28 Sep 1990. 19 p. : USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Experience/events **ID:** 520

Abstract: The frequency of stress corrosion cracking in the near weld regions of the SRS reactor tank walls is apparently lower than the cracking frequency near the pipe-to-pipe welds in the primary cooling water system. The difference in cracking tendency can be attributed to differences in the welding processes, fabrication schedules, near weld residual stresses, exposure conditions and other system variables. This memorandum discusses the technical issues that may account the differences in cracking tendencies based on a review of the fabrication and operating histories of the reactor systems and the accepted understanding of factors that control stress corrosion cracking in austenitic stainless steels.

Title: Pressure surge analyses for conventional and nuclear power plants.

Author: Buehl,-G. (Mannesmann Anlagenbau AG, Duesseldorf (Germany)); Grams,-J. (Mannesmann Anlagenbau AG, Duesseldorf (Germany)); Reiners,-U. (Mannesmann Anlagenbau AG, Duesseldorf (Germany)) **Corp. Author:**

Source: 3-R.-Rohre,-Rohrleitungsbau,-Rohrleitungstransport. (Sep 1993). v. 32(9). p. 508-516.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** German

Category: Pressure ripple/water hammer **ID:** 521

Abstract: There is need of pressure surge analyses when valves or pumps are activated or piping systems fail (pipe rupture). Based on actual problems the influences of boundary conditions upon fluid simulation results are discussed. Hints concerning realistic dynamic analyses of piping systems are presented. Some of the simulations results are compared with measurements. (orig.).

Title: Low temperature sensitization of austenitic stainless steel: an ageing effect during BWR service.

Author: Shah,-B.K.; Sinha,-A.K.; Rastogi,-P.K.; Kulkarni,-P.G. (Bhabha Atomic Research Centre, Bombay (India). Atomic Fuels Division) **Corp. Author:** AMNF-94: 1. national symposi

Source: Soman-Pillai,-M.D.; Sinha,-A.K.; Srinivasan,-V.S.; Srinivasan,-G.R. (comps.) (Nuclear Power Corporation of India Ltd., Bombay (India)). Department of Atomic Energy, Bombay (India). Board of Research in Nuclear Sciences. Ageing management of nuclear facilities (AMNF-94): proceedings. Bombay (India). Nuclear Power Corporation of India Ltd. 1994. [647 p.]. p. S7-20-S7-24.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Research/theoretical **ID:** 522

Abstract: Sensitization in austenitic stainless steel refers to chromium carbide precipitation at the grain boundaries with concomitant depletion of chromium below 12% near grain boundaries. This makes the material susceptible to either intergranular corrosion (IGC) or intergranular stress corrosion cracking (IGSCC). This effect is predominant whenever austenitic stainless steel is subjected to thermal exposure in the temperature range 723-1073K either during welding or during heat treatment. Low temperature sensitization (LTS) refers to sensitization at temperature below the typical range of sensitization i.e. 723-1073K. A prerequisite for LTS phenomenon is reported to be the presence of chromium carbide nuclei at the grain boundaries which can grow during boiling water reactor service even at a relatively lower temperature of around 560K. LTS can lead to failure of BWR pipe due to IGSCC. The paper reviews the phenomenological and mechanistic aspects of LTS. Studies carried out regarding effect of prior cold work on LTS are reported. Summary of the studies reported in literature to examine the occurrence of LTS during BWR service has also been included. (author). 10 refs., 3 figs.

Title: Corrosion survelliance for reactor materials in the calandria vault of Pickering NGS a unit 1.

Author: Quirk,-G.P. (Capcis March Ltd., Manchester (United Kingdom)); H-Mirzai,-M.; Bek,-W.W. (Ontario Hydro, Toronto (Canada)); Doherty,-P.E.

Corp. Author: 6. international symposium on

Source: Gold,-R.E.; Simonen,-E.P. (eds.). Proceedings of the sixth international symposium on environmental degradation of materials in nuclear power systems - water reactors. Warrendale, PA (United States). Minerals, Metals ampersand Materials Society. 1993. 963 p. p. 335-341.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** *523*

Abstract: This paper presents selected results from an 18 month period of corrosion surveillance for structural materials inside the calandria vault of Pickering NGS Unit 'A'. Cooling pipe leaks have resulted in humidity build up inside the air-filled vault, and subsequent radiolysis of the water and air by the high gamma radiation field from the operating reactor results in the formation of nitric acid. Condensation of the nitric acid on cooler components (pipes, support brackets) promotes general corrosion and possibly localized corrosion of carbon steel structures. The corrosion surveillance system was installed to directly monitor changes in corrosion rates of selected materials. One period has been chosen in which, due to a bioshield cooling water leak, the corrosion rates increased.

Title: Review of elastic stress and fatigue-to-failure data for branch connections and tees in relation to ASME design criteria

Author: Rodabaugh,-E.C.; Moore,-S.E.; Gwaltney,-R.C.

Corp. Author: ORNL / U.S. NRC-NRR

Source: May 1994. 120 p. Nuclear Regulatory Commission, Washington (DC)

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Experience/events **ID:** *524*

Abstract: This is the third in a series of reports on the state-of-the art design guidance for piping system branch connections and tees provided by Section III of the ASME Boiler and Pressure Vessel Code. The other reports covered primary or limit-loads and nozzle flexibility. The principal objective of this report, as with the others, was to identify and collect the pertinent literature on the the subject and to identify needed improvements in the design methods and criteria of the Code based on the evaluation of the available information. This report does not propose changes in the design procedure of the Code. This report discusses the evaluation of stresses in branch connections and tees, correlation of these stresses with fatigue failures, and the Code rules for protection against fatigue failure in design applications. Because of the extensive amount of available information, the report was divided into two parts. Part I discusses cyclic internal pressure loading and Part II discusses moment loadings for the branch and run. The cyclic pressure loading fatigue parameters are mostly based on leakage, whereas, if the parameters were based on crack initiation, different and possibly higher valves would be developed. The fatigue evaluation procedure, which attempts to relate fatigue strength of piping components to strain-controlled, polished bar, and fatigue data appears to be inaccurate on the conservative side for high amplitude cycles and inaccurate on the unconservative side for low amplitude cycles. The report proposes additional analytical and experimental work.

Title: Non-linear dynamic analysis of pipe whip.

Author: Attab,-M.; Ajam,-W. (Atomic Energy of Canada Ltd., Montreal, PQ (Canada). CANDU Operations); Baset,-S. (Atomic Energy of Canada Ltd., Chalk River, ON (Canada). Chalk River Nuclear Labs.)

Corp. Author: 32. Annual conference of the C

Source: Canadian Nuclear Society, Toronto, ON (Canada). Proceedings of the 13. annual conference of the Canadian Nuclear Society. V. 1. 1992. 740 p. [11 p.].

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Research/theoretical **ID:** *525*

Abstract: The dynamic non-linear behavior of pipe whip of a steam line piping system is studied using the general purpose finite element program MARC. The study showed that plastic deformation of the steam line will occur at strains higher than the maximum permissible material strain of 35%. It also indicates that the magnitude of the blowdown force following a guillotine break may be reduced by the deformation of the pipe cross section. This study can be used as a guideline for assessing the behavior of other steam lines following a pipe break. 4 refs., 13 figs., 1 tab.

Title: Evaluation of fatigue crack growth in the primary circuit pipeline of a WWER 440/213c type nuclear power plant.

Author: Samohyl,-P. **Corp. Author:** Ustav Jaderneho Vyzkumu CS

Source: Jul 1993. 69 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** Czech

Category: Research/theoretical **ID:** 526

Abstract: The fatigue damage of the primary circuit of WWER-440/213c reactors was evaluated proceeding from actual and design operating data of units 3 and 4 of the Bohunice V-2 nuclear power plant. A complex computation model was set up, encompassing the main circulation pipeline, pressurizer pipeline, emergency core aftercooling system pipeline, steam pipeline, and feedwater pipeline. The standardized STATIC code was applied to the stress analysis, and the FATLBB code was used to determine the crack increment for all operating states and primary circuit sections. The probability of fatigue failure of the pipelines was found to be low. (J.B.). 55 tabs., 3 figs., 9 refs.

Title: Computation of the mechanical behaviour of nuclear reactor components.

Author: Brosi,-S.; Niffenegger,-M.; Roesel,-R.; Reichlin,-K.; Duijvestijn,-A. (Paul Scherrer Inst. (PSI), Villigen (Switzerland)) **Corp. Author:**

Source: Neall,-F.B. (ed.) (Paul Scherrer Inst. (PSI), Villigen (Switzerland)). Paul Scherrer Institut annual report 1993. Annex IV: PSI nuclear energy research progress report 1993. Villigen PSI (Switzerland). Paul Scherrer Institut. 1994. 96 p. p. 75-82.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 527

Abstract: A possible limiting factor of the service life of a reactor is the mechanical load carrying margin, i.e. the excess of the load carrying capacity over the actual loading, of the central, heavy section components. This margin decreases during service but, for safety reasons, may not fall below a critical value. Therefore, it is essential to check and to control continuously the factors which cause the decrease. The reasons for the decrease are shown at length and in detail in an example relating to the test which almost achieved failure of a pipe emanating from a reactor pressure vessel, weakened by an artificial crack and undergoing a water-hammer loading. The latter was caused by a sudden valve closure supposed to follow upon a break far downstream. The computational and experimental difficulties associated with the simultaneous occurrence of an extreme weakening and an extreme loading in an already rather complicated geometry are explained. It is concluded that available computational tools and present know-how are sufficient to simulate the behaviour under such conditions as would prevail in normal service, and even to analyse departures from them, as long as not all difficulties arise simultaneously. (author) figs., tabs., refs.

Title: Model for heat-up of structures in VICTORIA.

Author: Bixler,-N.E. **Corp. Author:** Sandia National Labs., Albuqu

Source: Dec 1993. 38 p. N: USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Analysis of break effects **ID:** 528

Abstract: VICTORIA is a mechanistic computer code that treats fission product behavior in the reactor coolant system during a severe accident. During an accident, fission products that deposit on structural surfaces produce heat loads that can cause fission products to revaporize and possibly cause structures, such as a pipe, to fail. This mechanism had been lacking from the VICTORIA model. This report describes the structural heat-up model that has recently been implemented in the code. A sample problem shows that revaporization of fission products can occur as structures heat up due to radioactive decay. In the sample problem, the mass of deposited fission products reaches a maximum, then diminishes. Similarly, temperatures of the deposited film and adjoining structure reach a maximum, then diminish.

Title: NRC Information No. 90-40: Results of NRC-sponsored testing of motor-operated valves.

Author: Rossi,-C.E. (Nuclear Regulatory Commission, Washington, DC (United States). Office of Nuclear Reactor Regulation) **Corp. Author:**

Source: Anon.-Nuclear EQ sourcebook: A compilation of documents for nuclear equipment qualification. Piscataway, NJ (United States). IEEE Standards Press. 1992. 1360 p. p. 6, Paper 127.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Other **ID:** 529

Abstract: The NRC Office of Nuclear Regulatory Research (RES) has been sponsoring an MOV testing program in support of the resolution of Generic Safety Issue 87 (GI-87), "Failure of HPCI Steam Line Without Isolation." The initial scope of GI-87 involved the evaluation of the capability of certain motor-operated flexible wedge gate containment isolation valves to mitigate the loss of reactor coolant inventory in the event of a pipe break outside of the containment building at boiling-water-reactor (BWR) plants. The particular MOVs involved in the GI-87 program were those in the turbine steam supply lines for the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems, and in the supply line to the reactor water cleanup (RWCU) system. This information notice is intended addressees with specific information regarding the results of recent NRC-sponsored testing of motor-operated valves (MOVs) which was discussed at a public meeting on April 18, 1990.

Title: Guillotine failure of fixed-end pipes, pressurized with hot water.

Author: Shewfelt,-R.S.W.; Leitch,-B.W.; Godin,-D.P. (Atomic Energy of Canada Ltd., Pinawa, MB (Canada). Whiteshell Labs.) **Corp. Author:**

Source: International-Journal-of-Pressure-Vessels-and-Piping. (1994). v. 57(2). p. 211-221.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Test/analysis **ID:** 530

Abstract: It is extremely unlikely that a pressure tube and its calandria tube would rupture during normal operation in a CANDU (CANada Deuterium Uranium) reactor. However, if this unlikely scenario did happen, it would result in the pressure tube containing an axial, through-wall crack while still filled with water at 250-300 sup o C. This crack could run in the axial direction until it stopped, or it could turn and run in the circumferential direction, possibly causing a guillotine failure. As the path of this crack controls the resulting damage to the reactor, instrumented small-scale burst tests were done to determine the parameters controlling guillotine failure. These tests were analysed using the dynamic finite element code, VEC/DYNA3D. There was reasonable agreement between the measured and predicted deformation and the time and location of the guillotine failure. (Author).

Title: High-temperature service and time dependent failure.

Author: Swindeman,-R.W.; Asada,-Y.; Chang,-S.J.; Todd,-J.A. (eds.) **Corp. Author:** 1993 pressure vessel and pipin

Source: New York, NY (United States). American Society of Mechanical Engineers. 1993. 231 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Other **ID:** 531

Abstract: Separate abstracts were prepared for the technical papers presented at the American Society of Mechanical Engineers 1993 Pressure Vessels and Piping Conference on July 25--29 in Denver, Colorado. This volume contains twelve papers related to materials and design methods for high temperatures, eight papers related to time dependent failure evaluation and prevention in pressure vessels and piping, and five papers related to constitutive equations in high temperature design.

Title: Failure rate of piping in hydrogen sulphide systems.

Author: Hare,-M.G. **Corp. Author:** Atomic Energy Control Board,

Source: Aug 1993. 68 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Failure probability **ID:** 532

Abstract: The objective of this study is to provide information about piping failures in hydrogen sulphide service that could be used to establish failures rates for piping in 'sour service'. Information obtained from the open literature, various petrochemical industries and the Bruce Heavy Water Plant (BHWP) was used to quantify the failure analysis data. On the basis of this background information, conclusions from the study and recommendations for measures that could reduce the frequency of failures for piping systems at heavy water plants are presented. In general, BHWP staff should continue carrying out their present integrity and leak detection programmes. The failure rate used in the safety studies for the BHWP appears to be based on the rupture statistics for pipelines carrying sweet natural gas. The failure rate should be based on the rupture rate for sour gas lines, adjusted for the unique conditions at Bruce.

Title: Probabilistic based design rules for intersystem LOCAS in ABWR piping.

Author: Ware,-A.G. (EGandG Idaho, Inc., Idaho Falls, (United States)); Wesley,-D.A. (EQE Engineering Consultants, Irvine, CA (United States)) **Corp. Author:** 1993 pressure vessel and pipin

Source: Dermenjian,-A.A. (ed.). Piping, supports, and structural dynamics. New York, NY (United States). American Society of Mechanical Engineers. 1993. 181 p. p. 105-120.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Methods **ID:** 533

Abstract: A methodology has been developed for probability based standards for low-pressure piping systems that are attached to the reactor coolant loops of advanced light water reactors (ALWRs) which could experience reactor coolant loop temperatures and pressures because of multiple isolation valve failures. This accident condition is called an intersystem loss-of-coolant accident (ISLOCA). The methodology was applied to various sizes of carbon and stainless steel piping designed to advanced boiling water reactor (ABWR) temperatures and pressures.

Title: Rational design of piping systems.

Author: Esselman,-T.C. (Altran Corp., Boston, MA (United States)); Thailer,-H.J. (Pacific Gas and Electric Co., San Francisco, CA (United States)) **Corp. Author:** 1993 pressure vessel and pipin

Source: Dermenjian,-A.A. (ed.). Piping, supports, and structural dynamics. New York, NY (United States). American Society of Mechanical Engineers. 1993. 181 p. p. 121-123.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 534

Abstract: Recent industry failures in piping systems have been attributed to normal and abnormal operating conditions. However, the emphasis on design and analysis of piping systems in nuclear power plants is heavily weighted towards seismic capability, with increasing emphasis on the use of more sophisticated analytic methodologies. The priorities appear to be out-of-tune with industry concerns and needs. Emphasis on seismic design has detracted from normal operating design considerations. This shift in emphasis is unfortunate, since limited resources are available to address all piping system structural issues. The combination of limited resources and less flexible piping systems have both contributed to less optimal designs. The methodology proposed in this paper represents a rational approach to the design of piping systems by reducing stresses during normal plant operation yet accommodating seismic response. The paper emphasizes the importance of designing for normal operating conditions and proposes a simplified methodology for designing for seismic events.

Title: On the reasons of damages in NPP pipelines and expertizing their design decisions.

Author: Kaliberda,-I.V.; Dolitsaj,-E.V.; Morina,-M.V.; Teslitskij,-A.L. **Corp. Author:**

Source: Ehnergeticheskoe-Stroitel'-stvo. (Nov 1991). (no.11). p. 27-30.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** Russian

Category: Experience/events **ID:** 535

Abstract: The main causes of NPP pipeline damage are enumerated. It is noted that it is necessary to form and manage the database dealing with failures of pilepine elements, to develop a set of mathematical models and software, as well as to make studies into NPP pipeline safety levels in order to realize the expert activity.

Title: Nuclear piping criteria for Advanced Light-Water Reactors, Volume 1--Failure mechanisms and corrective actions.

Author: **Corp. Author:**

Source: Welding-Research-Council-Bulletin. (Jan 1993). (no.382). p. 1-45.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 536

Abstract: This WRC Bulletin concentrates on the major failure mechanisms observed in nuclear power plant piping during the past three decades and on corrective actions taken to minimize or eliminate such failures. These corrective actions are applicable to both replacement piping and the next generation of light-water reactors. This WRC Bulletin was written with the objective of meeting a need for piping criteria in Advanced Light-Water Reactors, but there is application well beyond the LWR industry. This Volume, in particular, is equally applicable to current nuclear power plants, fossil-fueled power plants, and chemical plants including petrochemical. Implementation of the recommendations for mitigation of specific problems should minimize severe failures or cracking and provide substantial economic benefit. This volume uses a case history approach to high-light various failure mechanisms and the corrective actions used to resolve such failures. Particular attention is given to those mechanisms leading to severe piping failures, where severe denotes complete severance, large "fishmouth" failures, or long throughwall cracks releasing a minimum of 50 gpm. The major failure mechanisms causing severe failure are erosion-corrosion and vibrational fatigue. Stress corrosion cracking also has been a common problem in nuclear piping systems. In addition thermal fatigue due to mixing-tee and to thermal stratification also is discussed as is microbiologically-induced corrosion. Finally, water hammer, which represents the ultimate in internally-generated dynamic high-energy loads, is discussed.

Title: Prioritizing aged piping for inspection using a simple probabilistic structural analysis model.

Author: Bishop,-B.A. (Westinghouse Electric Corp., Pittsburgh, PA (United States). Nuclear and Advanced Technology Div.); Phillips,-J.H. (Tenera L.P., Idaho Falls, (United States). Safety Services and Risk Assessment) **Corp. Author:** 1993 pressure vessels and pipin

Source: Phillips,-J.H. (ed.). Reliability and risk in pressure vessels and piping. New York, NY (United States). American Society of Mechanical Engineers. 1993. 168 p. p. 141-152.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Inspection methods **ID:** 537

Abstract: Aging causes increased failure rates for some piping segments and welds in mechanical systems. The identification of these specific locations of concern is complicated by the large number of welds and segments and by the time necessary to apply probabilistic models allow these calculations to be done very quickly. Sensitivity studies can identify groups of welds and piping segments that could have transients causing failure rates to increase. These models can also be used to optimize the inspection strategy necessary to assure that the structural failure rates remain low. This paper discusses the development of simplified probabilistic models of piping structural reliability and provides a demonstration of their use.

Title: Statement of nuclear incidents at nuclear installations. Second quarter 1993.

Author: **Corp. Author:** Health and Safety Executive, L

Source: Oct 1993. 5 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 538

Abstract: Three incidents were reported in April-June 1993. The first was on the British Nuclear Fuel plc (BNFL) site at Sellafield and concerned leakage of 0.5 TBq of alpha activity from plutonium contaminated waste stored in a steel drum. This was subsequently double contained and moved so it could be inspected regularly. No contamination of personnel occurred. The second concerned the leakage of thorium liquor from a pipe at the UKAEA's Thorium reprocessing plant at Dounreay. Two temporary repairs were made and no personnel were contaminated. The third was at the Sellafield site where a small quantity (5 mls) of plutonium containing liquor had leaked from a package and released alpha activity. The bags were temporary containment of engineering debris which may have had sharp edges. The bags had been piled up and one of the bags had torn. Recommendations were made following inquiries into each of the incidents to improve procedures and prevent similar incidents occurring. (UK).

Title: Calculation tools for testing a model of heterogeneous welded joint of the safe-end of reactor pressure vessel within the

Author: Lauerova,-D. **Corp. Author:** Ustav Jaderneho Vyzkumu a.s.,

Source: Jan 1993. 16 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** Czech

Category: LBB justification **ID:** 539

Abstract: Calculations necessary to perform experimental tests on a model of the safe-end DN 250 (for emergency reactor core cooling) are given. The tests are to be performed within the application of the leak-before-break (LBB) concept to V-213c type nuclear power plants. The methodology of the LBB concept is outlined briefly, an arrangement of the experiment is proposed, and the procedures for the calculation of the crack length recommended for the experiment and for the calculation of the limiting bending load are described. The LBB approach is designed to test the integrity of the various parts of the NPP primary coolant circuit. Predictions of the limiting bending load are given for the first and second stages of the experiment. The predictions were obtained by two different methods. For the first stage of the experiment, in which the experimental model will be stressed by bending and overpressure, the force from the load body corresponding to the limiting bending load lies within the region of 240-280 kN, whereas for the second stage, in which the model will be stressed by bending solely, the predicted limiting force from the load body lies within the range of 288-298 kN. (Z.S.). 1 tab., 7 figs., 3 refs.

Title: Failure and factors of safety in piping system design.

Author: Antaki,-G.A. **Corp. Author:** Westinghouse Savannah River

Source: [1993]. 8 p. : USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 540

Abstract: An important body of test and performance data on the behavior of piping systems has led to an ongoing reassessment of the code stress allowables and their safety margin. The codes stress allowables, and their factors of safety, are developed from limits on the incipient yield (for ductile materials), or incipient rupture (for brittle materials), of a test specimen loaded in simple tension. In this paper, we examine the failure theories introduced in the B31 and ASME III codes for piping and their inherent approximations compared to textbook failure theories. We summarize the evolution of factors of safety in ASME and B31 and point out that, for piping systems, it is appropriate to reconsider the concept and definition of factors of safety.

Title: Analysis of failed nuclear plant components.

Author: Diercks,-D.R. **Corp. Author:** Argonne National Lab., IL (Un

Source: Jul 1992. 9 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Experience/events **ID:** 541

Abstract: Argonne National Laboratory has conducted analyses of failed components from nuclear power generating stations since 1974. The considerations involved in working with and analyzing radioactive components are reviewed here, and the decontamination of these components is discussed. Analyses of four failed components from nuclear plants are then described to illustrate the kinds of failures seen in service. The failures discussed are (a) intergranular stress corrosion cracking of core spray injection piping in a boiling water reactor, (b) failure of canopy seal welds in adapter tube assemblies in the control rod drive head of a pressure water reactor, (c) thermal fatigue of a recirculation pump shaft in a boiling water reactor, and (d) failure of pump seal wear rings by nickel leaching in a boiling water reactor.

Title: The application of radiotracers in the leak detection of underground pipes.

Author: Tong,-Yungchien; Chung,-Showen (Institute of Nuclear Energy Research, Lung-Tan (Taiwan, Province of China)) **Corp. Author:** 2. topical meeting on industrial

Source: Transactions-of-the-American-Nuclear-Society. (1992). v. 65(1). p. 44-45.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Inspection methods **ID:** 542

Abstract: Leaks in chemical processing plants can be both expensive and dangerous. Leakage into the subsoil from underground pipes may cause environmental problems, but it is very difficult to locate the leak area in underground pipes. The development of radioactive techniques has greatly facilitated the detection of underground pipe leakages. The use of radioactive tracers affords an extremely sensitive means of measurement and permits the detection of tracers in low concentrations. The radiotracer method discussed in this paper was applied to five 6- to 12-in.-i.d., 70-m-long underground pipes that collect the water and/or oil from a large petrochemical processing plant in Taiwan. Bromine-82 was chosen as the tracer for this experiment because it emits gamma rays, can be prepared easily, and has a convenient half-life. The underground pipe was filled with an ammonium sup 8 sup 2 Br aqueous solution, and the system was kept closed for 2 to 3 h to ensure the free flow of the radiotracer through leak areas into the surrounding soil.

Title: Fluid-structure interaction model to check up discharging pipe system.

Author: Sainz-Mejia,-E. (Instituto Nacional de Investigaciones Nucleares, Mexico City (Mexico)) **Corp. Author:** 6. Seminar of the IIE-ININ-IM

Source: Instituto de Investigaciones Electricas, Cuernavaca (Mexico); Instituto Nacional de Investigaciones Nucleares, Mexico City (Mexico); Instituto Mexicano de Petroleo, Mexico City (Mexico). 6. Seminar of the IIE-ININ-IMP on technological specialties. Topic 3: thermal fluids.1992. 171 p. [8 p.].

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** Spanish

Category: Research/theoretical **ID:** 543

Abstract: Within phenomena group that occur in a pipelines system that lead some fluid in stationary state, the loss of lateral stability is which one of the more common and important of them since it is showed in periodic vibrations or aleatories, way against whose effects it will have to be designed the piping to avoid catastrophic failures. The present work is a part of the realized effort for incorporating to the programs of digital computers used for the structural analysis of piping systems based in the finite element method. It is a model that includes the lateral effect that induces the fluid on the pipes. For this effect was planted and obtained a model or element for straight pipes segments. It was through the use of analytical variational methods and polynomial approximations (typical techniques using in finite elements). When were effected the calculations of characteristic frequencies in straight pipe sections configurations. It was obtained concordance with the analytical predictions. There fore it was demonstrated that the model is correct. A continuation of this work will be to obtain the models for curved segments of piping. (Author).

Title: Location of leaks in pressurized underground pipelines.

Author: Eckert,-E.G.; Maresca,-J.W. Jr. (Vista Research, Inc., Mountain View, CA (United States)) **Corp. Author:** 13. biennial international confe

Source: Anon.-1993 International oil spill conference: Prevention, preparedness, response. Washington, DC (United States). American Petroleum Institute. 1993. 931 p. p. 806-809.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Inspection methods **ID:** 544

Abstract: Millions of underground storage tanks (UST) are used to store petroleum and other chemicals. The pressurized underground pipelines associated with USTs containing petroleum motor fuels are typically 2 in. in diameter and 50 to 200 ft in length. These pipelines typically operate at pressures of 20 to 30 psi. Longer lines, with diameters up to 4 in., are found in some high-volume facilities. There are many systems that can be used to detect leaks in pressurized underground pipelines. When a leak is detected, the first step in the remediation process is to find its location. Passive-acoustic measurements, combined with advanced signal-processing techniques, provide a nondestructive method of leak location that is accurate and relatively simple, and that can be applied to a wide variety of pipelines and pipeline products.

Title: 4th technical report, evaluation of bedded pipes under loads similar to operational load.

Author: Diem,-H. **Corp. Author:** Bundesministerium fuer Umwe

Source: 1992. 191 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** German

Category: Test/analysis **ID:** 545

Abstract: The study aims to identify locations of possible failures and the resulting component failure depending on the type of strain on bended pipes (out-of-plane strain, in-plane bending, inner pressure). Results on 1. deformation behaviour of bended tubes, elastic and plasto-elastic 2. deformation of straight pipe sections downstream with and without additional reinforcement 3. distribution of elongation across wall thickness 4. location and orientation of cracks in the area of the bend are evaluated and discussed. (orig./MM).

Title: Predictions of failure for several of the international pipe tests using the R6 method.

Author: Darlaston,-B.J. (Nuclear Electric, Berkeley Nuclear Labs. (United Kingdom)); Bhandari,-S.; Franco,-C. (FRAMATOME, 92 - Paris la Defense (France)) **Corp. Author:** 7. international conference on p

Source: Vereinigung der Technischen Ueberwachungsvereine e.V., Essen (Germany). Pressure vessels technology. Proceedings. Vol. 1. Design, analysis, materials (1). 1992. 857 p. p. 327-345.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Test/analysis **ID:** 546

Abstract: Experimental results on pipes with circumferential cracks have been analysed using the CEGB R6 Revision 3 Defect Assessment Procedure. The experimental data covers pipes with surface as well as through thickness defects under bending and/or pressure loading. Carbon steel and stainless steel base materials as well as welds were considered. The analytical results were compared with the experimental pipe data to demonstrate the need to limit the range of application of the procedure. The R6 method is based on the demonstration of fracture avoidance but for leak-before-break application a predictive approach is desirable. By imposing limits on the application in terms of geometry of pipe and crack and using best estimate data, the analytical predictions are within 10% of the experimental data. R6 is a well founded engineering assessment method initially developed to demonstrate avoidance of failure. With sound engineering judgement the method can be used in a predictive mode. This provides the necessary confidence in using R6 on plant components for leak-before-break assessments and in general for defect assessment. (orig.).

Title: Observations on seismic design of piping systems.

Author: Habip,-L.M. (Siemens AG, Power Generation (KWU), Offenbach (Germany)); Schrammel,-D. (Project HDR Safety Program, Karlsruhe Nuclear Research Center (Germany))

Corp. Author: 7. international conference on p

Source: Vereinigung der Technischen Ueberwachungsvereine e.V., Essen (Germany). Pressure vessels technology. Proceedings. Vol. 1. Design, analysis, materials (1). 1992. 857 p. p. 46-58.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Methods/design **ID:** 547

Abstract: Practical aspects of piping system design for seismic loads are considered. Main topics are structural effects of natural earth-quakes, full-scale dynamic tests - with emphasis on work performed at the HDR plant - and implications for the design and qualification of industrial systems and equipment. Experimental evidence and past experience indicate that design-by-rule or qualification-by-inspection can be used at this time to achieve dependable seismic performance, pending the development of piping failure criteria for cyclic overloads of short duration. (orig.).

Title: Safety of existing installations under dynamic loads: observations on nonlinear response of piping systems - experimen

Author: Habip,-L.M.; Jedlicka,-J. (Siemens AG Unternehmensbereich KWU, Offenbach am Main (Germany)); Kerkhof,-K. (Stuttgart Univ. (Germany). Inst. fuer Materialpruefung, Werkstoffkunde und Festigkeitslehre); Schrammel,-D. (Kernforschungszentrum Karlsruhe GmbH (Germany). Projektbereich Heissdampfreaktor - Sicherheitsprogramm/ Handhabungstechnik)

Corp. Author: ENS Topform '92: ENS East-

Source: European Nuclear Society (ENS), Bern (Switzerland); Czech Nuclear Society, Prague (Czech Republic); Slovak Nuclear Society, Bratislava (Slovakia). Topform '92: the safe and reliable operation of LWR NPPs. Vol. II. Poster papers. [Jan 1993]. 245 p. p. 123-126.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 548

Abstract: The nonlinear response of piping systems under base excitation or due to pressure waves caused by simulated breaks and valve closure has been investigated experimentally at the HDR reactor. Structural analysis of ruptured piping and the related design of pipe whips restraints are usually performed on the basis of nonlinear material behavior, with powerful computational techniques being used increasingly. Some aspects of these developments (high-level earthquake tests, high-level pressure wave tests, pipe rupture nonlinear analyses) are summarized with implications for qualification and optimal backfitting of operating nuclear power plants. (Z.S.) 7 refs.

Title: Reactor Materials Program process water piping indirect failure frequency.

Author: Daugherty,-W.L.

Corp. Author: Westinghouse Savannah River

Source: 30 Oct 1989. 216 p. : USDOE, Washington, DC (United States).

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Damage probability **ID:** 549

Abstract: Following completion of the probabilistic analyses, the LOCA Definition Project has been subject to various external reviews, and as a result the need for several revisions has arisen. This report updates and summarizes the indirect failure frequency analysis for the process water piping. In this report, a conservatism of the earlier analysis is removed, supporting lower failure frequency estimates. The analysis results are also reinterpreted in light of subsequent review comments.

Title: Aging risk of passive components.

Author: Phillips,-J.H.; Roesener,-W.S.; Magleby,-M.L. (Idaho National Engineering Lab., Idaho Falls (United States)); Geidl,-V.

Corp. Author: 18. water reactor safety inform

Source: Weiss,-A.J. (comp.) (Brookhaven National Lab., Upton, NY (United States)). Nuclear Regulatory Commission, Washington, DC (United States). Office of Nuclear Regulatory Research; Brookhaven National Lab., Upton, NY (United States). Eighteenth water reactor safety information meeting. Volume 3, Pressure vessel integrity; Piping and NDE; Aging and components: Proceedings. Apr 1991. 574 p. p. 337-354.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Damage probability **ID:** 550

Abstract: This paper presents an approach for determining the increasing failure probability of an aging passive component and for calculating its resulting effect on plant risk by modifying an existing commercial nuclear reactor probabilistic risk assessment (PRA). A technique was developed for introducing aging into failure probability calculations using probabilistic structural analysis (PSA) techniques. Various probabilistic structural analysis methods were reviewed, and the PRAISE computer code was selected to perform the PSA. A component was selected that could fail and have a significant effect on the risk of core damage frequency. This component is a weld in the auxiliary feedwater system (AFW) of a pressurized water reactor (PWR). The stress on the AFW weld, for input in PRAISE, was determined for piping design loads, plant transient loads, and a thermal cyclic load that could cause crack growth and ultimate pipe failure. One PRAISE calculation might be made with the possibility of water hammer introduced to determine the effect on core damage frequency. An existing PRA (for a NUREG 1150 plant) was modified to include the failure of the AFW weld. Because this work is not complete, only preliminary conclusions and recommendations are presented.

Title: The incorporation of seismic loadings within the failure criteria for cracked piping systems.

Author: Smith,-E. (Manchester Univ., UMIST Materials Science Center, Manchester (United Kingdom))

Corp. Author: 1991 American Society of Mec

Source: Ware,-A.G. (Idaho National Engineering Laboratory, ID (United States)). Proceedings of seismic engineering 1991. PVP-Volume 220. New York, NY (United States). American Society of Mechanical Engineers. 1991. 337 p. p. 215-220.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Methods/comparison **ID:** 551

Abstract: The technological problem of intergranular stress corrosion cracking of stainless steel piping in Boiling Water Nuclear Reactor piping systems has been responsible for considerable attention being given to the question of the integrity of cracked piping systems that are fabricated from ductile materials. This paper reports that in performing a structural integrity assessment, the usual procedure is to calculate the stress in the region of a crack, assuming the piping system to be uncracked. Most of the theoretical underpinning of the failure criteria for cracked piping has been with regard to the case where the system loadings are essentially static. By analyzing specific simulation models, this paper shows that the safety margin, introduced by basing the integrity assessment on the stress calculated on the assumption of the system being uncracked, is essentially unaffected by the fact that the loadings might be seismically induced, whether these by inertial loadings or displacement loadings. Particular consideration is given to the net-section stress criterion for the onset of crack extension, and the criterion for instability at the onset of crack extension or at some later stage in the crack extension process.

Title: Significance of high level test data in piping design.

Author: McLean,-J.L. (Altran Corp., Boston, MA (United States)); Bitner,-J.L. (Robert L. Cloud and Associates, Inc., Bethel Park, PA (United States))

Corp. Author: 1991 American Society of Mec

Source: Ware,-A.G. (Idaho National Engineering Laboratory, ID (United States)). Proceedings of seismic engineering 1991. PVP-Volume 220. New York, NY (United States). American Society of Mechanical Engineers. 1991. 337 p. p. 41-48.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Methods/design **ID:** 552

Abstract: During the 1980's the piping technical community in the U.S. initiated a series of research activities aimed at reducing the conservatism inherent in nuclear piping design. One of these activities was directed at the application of the ASME Code rules to the design of piping subjected to dynamic loads. This paper surveys the test data obtained from three groups in the U.S. and none in the U.K., and correlates the findings as they relate to the failure modes of piping subjected to seismic loads. The failure modes experienced as the result of testing at dynamic loads significantly in excess of anticipated loads specified for any of the ASME Code service levels are discussed. A recommendation is presented for modifying the Code piping rules to reduce the conservatism inherent in seismic design.

Title: Pretest analysis of a pipe system for high-level vibration response and failure.

Author: Severud,-L.K.; Weiner,-E.O. (Westinghouse Hanford Co., Richland, WA (United States)) **Corp. Author:** 1991 American Society of Mec

Source: Ware,-A.G. (Idaho National Engineering Laboratory, ID (United States)). Proceedings of seismic engineering 1991. PVP-Volume 220. New York, NY (United States). American Society of Mechanical Engineers. 1991. 337 p. p. 117-122.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Test/analysis **ID:** *553*

Abstract: This paper reports on simplified elastic and inelastic analyses for high level vibration response and cyclic failure capacity of a prototypic light-water reactor pipe system which were carried out in a pretest environment. The system consists of a steam generator and a circulating pump with associated piping that has been tested on a shake table. Five analyses, ranging from standard linear elastic to detailed inelastic transient analysis, are compared in terms of response. With the inelastic analysis, subsequent failure analysis indicated that strain in the 3% to 4% range can be expected if the planned inputs are realized. Possible cyclic failure was predicted by through-wall cracking and leaking in 20 to 40 cycles of maximum strain range, caused by ratchet-fatigue in the pressurized system.

Title: A review of fatigue failures in LWR plants in Japan.

Author: Iida,-Kunihiro (Inst. of Tech., Tokyo (Japan)) **Corp. Author:**

Source: Nuclear-Engineering-and-Design. (Dec 1992). v. 138(3). p. 297-312.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Experience/events **ID:** *554*

Abstract: A review was made of fatigue failures of nuclear power plant components in Japan, which were experienced in service and during periodical inspection. No case has been recently reported of a service fatigue failure of a reactor pressure vessel itself, excluding nozzle corner cracks, that occurred many years ago. But, service fatigue failures have been occasionally experienced in piping systems, pumps, and valves, on which fatigue design seems to have been inadequately applied. The causes of fatigue failures can be divided into two categories: Mechanical-vibration-induced fatigue and thermal-fluctuation-induced fatigue. Vibration-induced fatigue failure occurs more frequently than is generally thought. The lesson gleaned from the present survey is a recognition that a service fatigue failure may occur due to any one or a combination of the folowing factors: (1) Lack of communication between designers and fabrication engineers, (2) lack of knowledge about a possibility of fatigue failure and poor consideration about the effects of residual stresses, (3) lack of consideration on possible vibration in the design and fabrication stages, and (4) lack of fusion or poor penetration in a welded joint. (orig.).

Title: Analysis of the LaSalle Unit 2 Nuclear Power Plant, Risk Methods Integration and Evaluation Program (RMIEP). Vol

Author: Ferrell,-W.L. (Science Applications International Corp., Albuquerque, NM (United States)); Payne,-A.C. Jr.; Daniel,-S.L. (Sandia National Labs., Albuquerque, NM (United States)) **Corp. Author:** Nuclear Regulatory Commissio

Source: Oct 1992. 273 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Other **ID:** *555*

Abstract: This report is a description of the internal flood analysis performed on the LaSalle County Nuclear Generating Station, Unit 2. A more detailed integration with the internal events analysis than in prior flood risk assessments was accomplished. The same system fault trees used for the internal events analysis were also used for the flood analysis, which included modeling of components down to the contact pair level. Subsidiary equations were created to map the effects of pipe failures. All component locations were traced and mapped into the fault trees. The effects of floods were then mapped directly onto the internal plant model and their relative importance was evaluated. A detailed screening analysis was performed which showed that most plant areas had a negligible contribution to the flood-induced core damage frequency. This was influenced strongly by the fact that the LaSalle plant was designed with a high level of concern about the effects of external events such as fire and flood and significant separation was maintained between systems in the original design. Detailed analysis of the remaining flood scenarios identified only two that contributed significantly to risk. The flood analysis resulted in a total (mean) core damage frequency of 3.23E-6 per year.

Title: Assessment of high confidence of low probability of failure of NPP V1 Jaslovske Bohunice safety significant pipings (

Author: Pecinka,-L.; Zdarek,-J. **Corp. Author:** Ustav Jaderneho Vyzkumu CS

Source: Feb 1992. 83 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Research/theoretical **ID:** 556

Abstract: An improved assessment was made of the high confidence of low probability of failure of the primary piping, pressurizer surge lines, steam and feed-water piping for the non-seismic design. In accordance with the preliminary safety report for the V-1 nuclear power plant, the design earthquake acceleration was chosen as 0.1 g or 0.125 g; this corresponds to the intensity of 7 on the 64-degree macroseismic intensity scale. The high confidence of low probability of failure of primary piping and all three pressurizer surge lines was shown to be greater than the upper limit of 0.125 g, the high confidence of low probability of failure of the steam piping is less than 0.1 g. Based on the calculated stress state of all weldments, diagrams of high-stressed cross-sections before and after earthquake are shown as a guide for non-destructive examinations. (author) 7 figs., 30 tabs., 15 refs.

Title: 22. technical report, failure analysis of pipes and containers with longitudinal faults.

Author: Stoppler,-W.; Shen,-S.M.; Boer,-A.-de **Corp. Author:** Bundesministerium fuer Umwe

Source: Jan 1992. 53 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** German

Category: Experience/events **ID:** 557

Abstract: The pressures at failure of 134 pipes and containers with longitudinal faults were calculated with different semi-empirical calculation expression, toughness-, flow stress-, plastic instability and ligament stress criteria, and compared with the experimentally determined pressures at failure. It was found in all calculation processes that the calculated pressure at failure differs more or less greatly from the experimentally determined pressure at failure. (orig./MM).

Title: Pipe failures in US commercial nuclear power plants. Interim report.

Author: Jamali,-K. **Corp. Author:** Electric Power Research Inst.,

Source: Jul 1992. 157 p.

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Experience/events **ID:** 558

Abstract: Recent NRC mandates require utilities to perform probabilistic risk assessments as part of their individual plant examinations (IPEs). To date, a significant number of IPEs have identified small-break loss-of-coolant accidents (LOCAs) as a major contributor to nuclear power plant risk. Most existing databases that address pipe failure rates have been based on judgmental estimates from industry experts. EPRI has developed a methodology and database that uses actual experiences to support failure rate calculations on a plant-or system-specific basis. This document discusses this methodology.

Title: Short cracks in piping and piping welds. Semiannual report, April--September 1991: Volume 2, No. 1.

Author: Wilkowski,-G.M.; Brust,-F.; Francini,-R.; Ghadiali,-N.; Kilinski,-T.; Krishnaswamy,-P.; Landow,-M.; Marschall,-C.W.; Rahman,-S.; Scott,-P. (Battelle, Columbus, OH (United States)) **Corp. Author:** Nuclear Regulatory Commissio

Source: Sep 1992. 207 p.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Research/theoretical **ID:** 559

Abstract: This is the third semiannual report of the US Nuclear Regulatory Commission's Short Cracks in Piping and Piping Welds research program. This 4-year program began in March 1990. The overall objective of this program is to verify and improve fracture analyses for circumferentially cracked large-diameter nuclear piping with crack sizes typically used in leak-before-break analyses or inservice flaw evaluations.

Title: Microstructural evolution of pipelines for thermal electric power plants after a prolongated operation.

Author: Twentyman,-M.; Rosetti,-R.; Porta,-G. (Instituto Nacional de Tecnologia Industrial (INTI), Buenos Aires (Argentina)) **Corp. Author:** Metallurgical sessions; 2. ALA

Source: Comision Nacional de Energia Atomica, Buenos Aires (Argentina). Gerencia de Desarrollo. Metallurgical sessions. Second ALAMET congress. Jornadas metalurgicas. Segundo congreso ALAMET (Asociacion Latinoamericana de Metalurgia). Buenos Aires (Argentina). CNEA. 1991. 309 p. p. 207-210.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** Spanish

Category: Experience/events **ID:** 560

Abstract: The study of failures originated in pipelines for thermal electric power plants allows an evaluation of the limit microstructural conditions that turn the system to critical conditions. A set of pipe samples with different microsctructural evolution which had been affected by direct flame were prepared. The samples were taken close to failures, away from them, from out of use pipes, etc. Metallographic studies were carried out using optical microscopy and scanning electron microscopy. Phase distribution, morphology and their relation with the different stages of aging were observed. (Author).

Title: Comparison of fuel spill fate models in soil and groundwater.

Author: Leinberry,-B.E. (Naval Facilities Engineering Command, Washington, DC (United States). Chesapeake Div.); Regan,-R.W. Sr. (Pennsylvania State Univ., University Park, PA (United States). Environmental Resources Research Inst.) **Corp. Author:** 23. mid-Atlantic industrial was

Source: Neufeld,-R.D.; Casson,-L.W. (Univ. of Pittsburgh, PA (United States)). Proceedings of the twenty-third Mid-Atlantic industrial waste conference. Hazardous and industrial wastes. Lancaster, PA (United States). Technomic Publishing Co., Inc. 1991. 405 p. p. 106-110.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Damage probability **ID:** 561

Abstract: It has been estimated that 96 percent of the 1.4 million underground storage tanks in the U.S. contain petroleum products such as gasoline and fuel oil. The Environmental Protection Agency (EPA) further estimates that 84 percent of the existing underground storage tanks (UST's) are bare-steel, single-wall tanks with no corrosion protection, leak-prevention or leak-detention devices. As many as 40 percent of these tanks could be leaking now or in the near future due to corrosion, installation mistakes, or piping failures. This paper discusses the potential sources of fuel oil leaks or spills, describes the physical and chemical fate of the hydrocarbon contaminants, and reviews current literature reports which model the fate of petroleum contaminants in the subsurface environment.

Title: What went wrong? Case histories of process plant disasters

Author: Kletz-TA **Corp. Author:**

Source: Gulf Publishing Company, Book Division, P.O. Box 2608, Houston, Texas 77252-2608, USA, 2nd ed. 1988. xvii, 238p. Illus. Bibl.ref. Index.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Experience/events **ID:** 562

Abstract: Reports of process plant accidents are presented to illustrate what went wrong in the past and to suggest how similar incidents might be prevented in the future. Incidents are described under the following headings: preparation for maintenance; modifications; accidents caused by human error; labelling; storage tanks; stacks; leaks; liquefied flammable gases; pipe and vessel failures; other equipment; entry to vessels; hazards of common materials; tank trucks and cars; testing of trip controls and other protective systems; static electricity; materials of construction; operating methods; reverse flow and other unforeseen deviations; problems with computer control.

Title: Zinc embrittlement of stainless steel - The problem in perspective.

Author: Elliott-D **Corp. Author:**

Source: Process Engineering, London, United Kingdom, July 1976, p.67-71. 19 ref.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1976 **Language:** English

Category: Experience/events **ID:** 563

Abstract: This literature survey begins with a discussion of one of the theories of contributing causes of the explosion at a chemical plant in Flixborough (United Kingdom) - zinc-induced cracks in the stainless steel pipe. The published data on zinc-stainless steel interaction (type of steel, temperature, time, form of attack) are tabulated. Experiments to reproduce hazardous conditions and an earlier accident are described.

Title: The Flixborough disaster - Report of the Court of Inquiry.

Author: Department of Employment, London. **Corp. Author:**

Source: H.M. Stationery Office, P.O. Box 569, London S.E.1, United Kingdom, 1975. 56p. Illus. Price: #2.50.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1975 **Language:** English

Category: Experience/events **ID:** 564

Abstract: Report of the formal investigation into the explosion on 1 June 1974 at the Nypro factory at Flixborough, United Kingdom, which killed 28 people and injured 36, causing extensive damage. Essentially, a vapour cloud explosion occurred as a result of leakage of cyclohexane, some weeks after a faulty cyclohexane reactor had been removed from the plant and replaced by a dog-leg assembly consisting of 20-inch pipe. The report describes the Flixborough site, the company, and particularly Section 25A of the plant, where the explosion occurred. Events leading to the explosion are recapitulated and possible explanations stated. 2-stage rupture of the by-pass assembly was rejected on account of lack of evidence. The main possibilities investigated are the "8-inch hypothesis" and the "20-inch hypothesis", i.e. rupture of a nearby 8-inch pipe and of the 20-inch assembly respectively. Conclusion: one-stage failure of the 20-inch assembly. A section is devoted to lessons to be learned from the disaster, and matters to be referred to other bodies are listed. Appendices: technical investigations, photographs taken before, during and after the fire, list of witnesses, explanatory drawings, etc.).

Title: Flixborough - The implications for management.

Author: Taylor-HD **Corp. Author:**

Source: Keith Shipton Developments, Adelaide House, London Bridge, London EC4R 9DS and Woodhead-Faulkner, 7 Rose Crescent, Cambridge CB2 3LL, United Kingdom, 1975. 30p. Illus. Price: #1.50.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1975 **Language:** English

Category: Experience/events **ID:** 565

Abstract: This booklet describes the explosion which occurred in June 1974 at a chemical plant at Flixborough, United Kingdom, resulting in the death of 28 people. The explosion was due to the leakage of hot flammable liquid (cyclohexane) from a pipe connecting 2 vertical reactors. The report of the court of inquiry into the causes of the disaster is summarised, and the events preceding the disaster and management structure at the plant are analysed. Lessons for management are drawn (plant layout, construction and siting; planning procedures; storage arrangements; essential records; disaster planning; proper communication of instructions; safety training; risk management; loss prevention and insurance; plant integrity; inspection mandatory after any modification or repair of pressure systems; nitrogen supplies; nitrate stress corrosion; crack propagation in clad mild steel; zinc embrittlement and creep cavitation of stainless steel; vapour cloud explosions, etc.).

Title: Continuous digesters.

Author: **Corp. Author:**

Source: Data Sheet 645, National Safety Council, 425 North Michigan Avenue, Chicago, Illinois 60611, USA, 1974. 6p. Illus. 9 ref.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1974 **Language:** English

Category: Other **ID:** 566

Abstract: This data sheet is concerned with the prevention of accidents on continuous digesters and defibrators for breaking down wood chips for pulp. Hazards can range from gland or gasket leaks, through pipe and valve failures, to ruptures of major elements, and hazardous gases. Sections are devoted to: guarding; walking surfaces; valves; lighting; control panel; protective equipment; inspection; cleaning; entering tanks and enclosed spaces (chip bins, steaming vessels); maintenance.

Title: Hydraulic fluids.

Author: **Corp. Author:**

Source: Data Sheet 1-471-78, Revised 1978, National Safety Council, 444 North Michigan Avenue, Chicago, Illinois 60611, USA, 1978. 4p. Illus. 5 ref.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1978 **Language:** English

Category: Other **ID:** 567

Abstract: This data sheet gives information on types of hydraulic fluids, their industrial uses and hazards (fire hazards; failure of pipes, gaskets, valves and fittings; skin and eye irritants) and contains sections on design of equipment, replacement of flexible tubing, deterioration due to vibration, changing to fire-resistant fluids, and preventive maintenance, with safety rules for maintenance work.

Title: Gas pipeline rupture - Holocaust.

Author: **Corp. Author:**

Source: ACC Report, Wellington, New Zealand, Sep. 1978, Vol.3, No.4, p.10-11. Illus.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1978 **Language:** English

Category: Experience/events **ID:** 568

Abstract: Earthmoving machinery drivers are often unaware of the hazards of excavating near flammable gas or liquid pipelines. This article gives examples of accidents which show the catastrophic consequences of a pipeline breaking, and advises contractors to notify the gas distribution authority or pipeline inspector before commencing operations, and especially to use hand tools instead of earthmoving machinery for this kind of work.

Title: Pressure and leakage testing of pressure vessels and piping

Author: **Corp. Author:**

Source: Meddelanden 1978:21, National Board of Occupational Safety and Health (Arbetarskyddsstyrelsen), Fack, 100 26 Stockholm, Sweden, 13 June 1978. 6p. Gratis.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1978 **Language:** Swedish

Category: Other **ID:** 569

Abstract: This notification (effective 1 Jan. 1979) prescribes safety rules for hydraulic and pneumatic pressure and leakage testing: definitions, units of measurement; general rules (supervision by qualified staff, protection against flying particles during tests on vessels made of brittle material, checking of pressure gauges, ventng of air pockets, etc.); rules for compressed air testing (>3bar, <3bar); accessory equipment (flanges for pipe connections, covers, plugs, gaskets, etc.).

Title: Guidance notes on the use of acoustic emission testing in process plants

Author: **Corp. Author:**

Source: (The Institution of Chemical Engineers, George E. Davis Bldg, 165-171 Railway Terrace, Rugby, Warwickshire CV21 3HQ, United Kingdom, 1985. 74p. 133 ref. Price: #7.50.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: Inspection methods **ID:** 570

Abstract: These guidance notes have been prepared by a working party set up by the International Study Group on Hydrocarbon Oxidation and summarise the collective experience of a number of major companies in the application of acoustic emission testing for the inspection of vessels and pipelines in process plants. This type of testing is used to detect defects and stress or corrosion cracks in vessels and pipes made of various materials such as reinforced plastics and steels. The advantage of the method is that a whole system can be monitored under service conditions. However, it is an expensive technique which requires highly skilled personnel.

Title: Technical rules on flammable liquids

Author: German Federal Ministry of Labour and Social Affairs (Bundesministerium fur Arbeit und Sozialordnung) **Corp. Author:**

Source: Bundesarbeitsblatt; Dec. 1982, No.12, p.34-81. Illus.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1982 **Language:** German

Category: Other **ID:** 571

Abstract: Notification of amendments to and new versions (published Dec. 1982) of the technical regulations issued under the Ordinance of 27 Feb. 1980 on flammable liquids: TRbF001-General requirements, structure and application of the regulations; TRbF111 - Storage and emptying depots, fuelling stations on airfields; TRbF211 - Storage and emptying depots (for transport tanks); TRbF231 - Piping in factories, including feed pipes for oil burners; TRbF501 - Directive and design and testing principles for tank leakage indicators; TRbF502 - Directive and design and testing principles for leak indicators for double wall piping.

Title: Leak analysis in compliance with the major accident hazard control ordinance

Author: Strohmeier-K **Corp. Author:**

Source: Chemie-Ingenieur-Technik; Dec. 1990, Vol.62, No.12, p.1003-1007. Illus.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** German

Category: Other **ID:** 572

Abstract: The Major Accident Hazard Control Ordinance (Germany, see CIS 81-293) requires identification of hazards which might lead to failure of pressure vessels and systems. Leaks pose one such hazard. For pipes, containers, tanks, fittings and flanges the methods of detecting cracks and of predicting crack propagation as required by the major accident hazard control ordinance are outlined.

Title: LPG pipeline and Trans Siberian Railway explosion and fire

Author: Lewis-DJ **Corp. Author:**

Source: Loss Prevention Bulletin; Dec. 1989, No.090, p.11-12.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Experience/events **ID:** 573

Abstract: Report on an explosion and fire on the Trans Siberian Railway in June 1989 caused by leakage from an LPG pipeline (462 dead, 706 injured, a great many with burns up to 70-80%). The pipeline was reported to have been leaking for several days, with a gas cloud drifting for several miles and gas pockets forming in low-lying areas along the railway line. Turbulence caused by 2 passing trains mixed LPG mist and vapour with overlying air to form a flammable cloud which was sparked off by one of the trains. Other explosions and a fire followed. The blast was reported to be equivalent to 10,000 tons TNT, making it the largest known aerial explosion. A full investigation has been ordered.

Title: (1) A theoretical study of NH3 concentrations in moist air arising from accidental releases of liquefied NH3, using the

Author: Wheatley-CJ; Safety and Reliability Directorate **Corp. Author:**

Source: United Kingdom Atomic Energy Authority, Wigshaw Lane, Culcheth, Warrington WA3 4NE, United Kingdom, Feb. 1987. 52p. + 22p. Illus. Bibl. Price: GBP 5.00. + GBP 4.00.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Other **ID:** *574*

Abstract: The computer code TRAUMA calculates the consequences of accidentally releasing liquefied ammonia into moist atmospheres through a pipe or tank wall rupture. In this report, TRAUMA is used to study 4 typical accidental releases: release of pressurised ammonia through a pipe and through a tank wall rupture; release of refrigerated ammonia and of semi-refrigerated ammonia through a pipe. Results for discharge rates and speeds, flashing at the outlet, drop sizes, settling speeds and entrainment of air are presented and discussed. TRAUMA can provide information concerning the safe storage of ammonia in a range of circumstances.

Title: A strategy for plant management to prevent loss - 7 ways for managers to cut incidents by up to 44%

Author: Dunford-N **Corp. Author:**

Source: Loss Prevention Bulletin; June 1990, No.93, p.25-31. Illus.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Experience/events **ID:** *575*

Abstract: In a recent study on the human contribution to pipework and in-line equipment failure frequencies, a 3-way classification scheme was used to define failures in terms of direct cause, origin of failure (underlying cause) and recovery (preventive) mechanism. Underlying causes of analysed incidents were examined in combination with preventive actions, such as hazard study, human factors, task checking and routine checking, so as to provide a framework for use by managers in prioritising a detailed action plan for prevention. Effective action by management would theoretically have prevented 44% of the analysed incidents.

Title: Sabotage causes propane release

Author: **Corp. Author:**

Source: Loss Prevention Bulletin 077; Oct. 1987, No.077, p.17-25. Illus. 1 ref.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Experience/events **ID:** *576*

Abstract: The propane pipeline explosion and fire in 1981 in Sweden are described. Covered are: description of the damaged pipeline; development of the incident; ignition of the vapour cloud; fire fighting; damage; cause of the release; the vapour cloud; source of ignition.

Title: Use of Risk Assessment as an Offshore Design Tool

Author: Shaw-SJ **Corp. Author:**

Source: Journal of Loss Prevention in the Process Industries, Vol. 5, No. 1, pages 10-17, 2 references

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Other **ID:** 577

Abstract: Applying risk assessment techniques to the United Kingdom (UK) offshore petroleum industry was discussed. The methodology for performing an offshore risk assessment was considered. The basic components of an offshore risk analysis were similar to those of an onshore assessment. They consisted of hazard identification, frequency estimation, consequence prediction, risk summation, and evaluation. The main difference between an onshore and an offshore risk assessment was that onshore assessments were concerned primarily with evaluating the effects of an accident upon the environment and the general population whereas offshore assessments were concerned with the effects on the platform crew because of their close proximity to the hazardous material, the high pressures created in most offshore processes, and the fact that the platform is usually in the middle of the sea. Major hazards encountered at offshore sites included blowouts, process related events such as gas leaks or well fluid leaks, riser and pipeline failures, collisions with passing ships or supply boats, helicopter crashes, structural failures caused by earthquakes and wind and wave action, design failures, and nonprocess related fires. Preliminary results from the public investigation of the explosion and fire on the Piper Alpha offshore platform that destroyed the platform and killed 167 workers were summarized. Recommendations resulting from the investigation were discussed.

Title: Probabilistic Safety Analysis in Chemical Installations

Author: Papazoglou-IA; Nivolianitou-Z; Aneziris-O; Christou-M **Corp. Author:**

Source: Journal of Loss Prevention in the Process Industries, Vol. 5, No. 3, pages 181-191, 33 references

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Other **ID:** 578

Abstract: Applying probabilistic safety analysis (PSA) techniques to chemical facilities was discussed. The basic features of a PSA were summarized. PSA is a scheme for performing a systematic analysis of hazards and quantification of risks which can then be used to support safety related decision making. A PSA methodology suitable for a chemical installation was presented. The procedures consisted of hazard identification, accident sequence modeling, data acquisition, and parameter estimation, accident sequence quantification, hazardous substance release categories assessment, assessment of the probability and consequences of a release, and integration of the results of the accident sequence quantification, hazardous substance release categorization, and the consequences assessment. The methodology was illustrated by applying it to a refrigerated ammonia storage facility. The PSA indicated 21 initiating events that could cause one of five events leading to accidental ammonia releases. The events that could result in an accidental ammonia release were a ship to tank piping failure, seismic failure of a storage tank, an overpressure failure of the storage tank, a tank to facility piping failure, and an underpressure failure of the storage tank. The probabilities of the events occurring were estimated to be 3.8x10(-3), 1.1x10(-3), 1.3x10(-3), 5.9x10(-4), and 1.0x10(-5) per year, respectively. The authors conclude that the PSA methodology is suitable for chemical installations where toxic materials are handled or stored.

Title: A Classification Scheme for Pipework Failures to Include Human and Sociotechnical Errors and Their Contribution to

Author: Hurst-NW; Bellamy-LJ; Geyer-TAW; Astley-JA **Corp. Author:**

Source: Journal of Hazardous Materials, Vol. 26, No. 2, pages 159-186, 21 references

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Experience/events **ID:** 579

Abstract: The results were presented of a study which analyzed over 900 reported incidents involving failures of fixed pipework on chemical and major hazard facilities. In about 500 cases the data were sufficient to fully classify the incident using the scheme developed here. An important part of the scheme involved the development of a failures classification scheme, used to analyze recorded incident accounts. A three dimensional scheme was developed which consisted of a number of layers of immediate causes. Each immediate cause was overlaid with a two way matrix of underlying causes of failure and preventive mechanism. Thus each incident was classified in three ways. Operating error was the largest known immediate contributor to incidents. Overpressure and corrosion were the next largest categories of known immediate causes. The other major areas of human contribution to immediate causes were human initiated impact and incorrect installation of equipment. For the underlying causes of failure maintenance and design were the largest contributors. The largest potential preventive mechanisms were human factors review, hazard study and checking and testing of completed tasks. A hierarchical scale of accident causation from the most immediate direct causes to increasingly remote causes was also constructed. The levels of the hierarchy were engineering reliability, operator reliability, communication information and feedback control, organization and management, and system climate.

Title: Risk Assessment for Installations Where Liquefied Petroleum Gas (LPG) Is Stored in Bulk Vessels above Ground

Author: Clay-GA; Fitzpatrick-RD; Hurst-NW; Carter-DA; Crossthwaite-PJ **Corp. Author:**

Source: Journal of Hazardous Materials, Vol. 20, pages 357-374, 15 references

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Other **ID:** 580

Abstract: The methodology and models used in efforts to assess the risks involved at liquid petroleum gas (LPG) storage areas were reviewed. The methods and models described were used in some examples of the outputs available. The purpose of having a qualified risk assessment method for LPG within the Health and Safety Executive's office (HSE) was to improve the technical basis and, therefore, the quality of HSE's advice by a more precise consideration of the events which can occur and their likelihood, thereby giving a refined impression of the risks. The sensitivity of the results obtained to the various assumptions made, and to the precise nature of the various submodels included in the method were considered. The main inputs for the whole vessel failure calculation were the vessel size and fuel type. The environment within which the installation was located was considered in terms of the distribution of population and potential ignition sources. Pipework sizes and process conditions were used as inputs for the part of the assessment which dealt with events other than whole vessel failure. The model calculated the probabilities that certain levels of thermal radiation dose and blast overpressure would be experienced at the center of each grid point for a hypothetical individual either indoors and outdoors. These data were used to calculate radiation and overpressure contours.

Title: Walk-Through Survey Report No. CT-101-22a, Control Technology For Chemical Batch Unit Operations, Mobil Ch

Author: Wang-CCK **Corp. Author:**

Source: Division of Physical Sciences and Engineering, NIOSH, U.S. Department of Health and Human Services, Cincinnati, Ohio, Report No. CT-101-22a, 8 pages

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: Other **ID:** 581

Abstract: A walk through survey was conducted to assess a phosphorus products manufacturing facility at Mobile Chemical Company (SIC-2819), Charleston, South Carolina in December 1983. The facility produced phosphorus-trichloride (7719122) (PT). The primary engineering control was a water scrubber to remove PT vapors from gas streams moving at 1000 to 5000 cubic feet per minute. The acidic solution resulting from the scrubber operation was neutralized by sodium-hydroxide and then sent for waste water treatment. Double mechanical seals were used to prevent PT leaks from pumps, valves, and piping. All storage tanks were constructed of nickel coated steel. Sampling ports for product quality control were equipped with ventilated sample hoods. Ambient air was continuously monitored for hydrogen-sulfide (7783064) and organophosphorus compounds. Five percent of the finished product was stored in 50 gallon plastic drums in a partially enclosed shed, and was then shipped. Workers were required to wear respirators, protective outer garments, and gloves. Eye washers and safety showers were available. Monthly safety meetings were held. The author concludes that the facility has adequate, though not outstanding, control technology. The drumming operation poses a potential exposure hazard due to its being labor intensive.

Title: Preliminary Site Visit Report, Cherry Point Refinery, Control Technology Assessment of Petroleum Refinery Operatio

Author: Anonymous **Corp. Author:**

Source: Occupational Safety and Health Division, Radian Corporation, Salt Lake City, Utah, Report No. CT-102-12a, Contract No. 210-81-7102, 36 pages

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1982 **Language:** English

Category: Other **ID:** 582

Abstract: An on site visit was made to the Cherry Point Refinery, near Ferndale, Washington, to evaluate control technology strategies used by the refinery to control worker exposure to potentially toxic chemical agents and harmful physical agents. Specific attention was focused on naphtha reforming and oil/water separation. This facility could process up to 120,000 barrels per stream day of Alaskan North Slope crude. Efforts to control exposures included the identification of all hazardous chemical and physical agents by workplace, employee exposure assessment, control of unacceptable exposures, training and information offered, medical surveillance, and a documentation and record keeping system. Area hydrogen-sulfide (7783064) monitors were located at the sulfur recovery unit and an API separator was used for oil and water separations. Several interesting control techniques noted during the visit included a closed oil water sewer with sample points directly piped to the sewer, use of computer programs to calculate correct bolt tightening stress, pump inspections for seal leaks, valve inspections for steam and hydrocarbon leaks, use of temperature sensitive paints on vessels to indicate higher temperatures, and the use of acoustical testing for locating stress/corrosion cracking while the vessel was in service.

Title: Failure of High Pressure Synthesis Pipe

Author: Prescott-GR; Blommaert-P; Grisolia-L **Corp. Author:**

Source: Ammonia Plant Safety (and Related Facilities), Vol. 26, American Institute of Chemical Engineers, New York, pages 228-233

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Experience/events **ID:** 583

Abstract: An explosion that occurred in The Netherlands in 1984 due to failure of a high pressure synthesis pipe associated with an ammonia (7664417) converter was described and corrective measures suggested. Just before the accident, the cold ammonia pump flow to storage showed some irregularities; the explosion occurred as the operator was investigating the problem, and he was fatally injured. The failure of the feed line was characterized by a sudden rupture with an instantaneous release of energy; severe hydrogen attack on the carbon/steel was found to be the cause of failure. The selection of steels to prevent hydrogen attacks was based on the Nelson Curves; revisions in the curves over the years, based on experience and accumulated data, involved changes in material specified in the lines as a function of temperature and hydrogen pressure. Older facilities were likely to be operating in regions of the curves currently considered unsafe by American Petroleum Institute (API) standards. Also, the lack of portable alloy analyzers at the time older facilities were built prevented detection of improper substitutions of materials; components with insufficient alloy content could undergo a slow process of deterioration by hydrogen attack. The authors conclude that all operators of ammonia facilities should check the alloy content of components and weldments for safe operation based on the current API curves; portable analyzers are available to do the job quickly and efficiently.

Title: Risk Management of a Petroleum Refining Facility under Design

Author: Leach-DS; Maher-ST; Sharp-DR; Sherbine-CA **Corp. Author:**

Source: Automation for Safety in Shipping and Offshore Petroleum Operations, C. Kuo, A. J. Thunem and N. P. Sundby, Editors; Elsevier Science Publishers, B.V., pages 219-224

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Other **ID:** 584

Abstract: A quantitative hazards analysis was performed for a small refinery which produces diesel fuel from a crude stream, in a facility composed of several subsystems. The hazards analysis was performed using fault tree analysis (FTA) techniques. Some of the general hazards identified were: overpressure at various sites, high temperature piping failure, vacuum formation causing collapse of the crude tower and spillage of hydrocarbon fluids. Each subsystem was analyzed using FTA, with the identified hazards serving as second level events. The control and protection systems were developed to the component level. This analysis resulted in several low cost, large benefit design changes. Examples of a fault tree and a crude tower were presented, and a model problem report was included. In general, the study verified the design of the facility, provided suggestions for design modifications to enhance plant availability and personnel safety and provided the facility operators with an increased awareness of potential facility problems. The facility owners used the hazard analysis as a decision making aid for risk management of the facility.

Title: Norwegian Petroleum Directorates Requirements Relating to Safety Evaluation of Platform Conceptual Design

Author: Berg-O; Thuestad-O **Corp. Author:**

Source: Automation for Safety in Shipping and Offshore Petroleum Operations, C. Kuo, A. J. Thunem and N. P. Sundby, Editors; Elsevier Science Publishers, B.V., pages 207-218

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Other **ID:** 585

Abstract: Functions of the Norwegian Petroleum Directorate (NPD) with regard to offshore oil platform safety design were discussed. Risk analyses, or safety evaluations, as a tool for enhancing and verifying the safety of offshore production drilling, production, and quarters platforms, were implemented in the mid 1970's. At that stage the regulations were too specialized, approaching safety from set angles or by individual component or system. The problem was reduced by implementing a total safety review by a group of engineering and risk analysis experts. The NPD guidelines stipulated that escape ways, shelter areas and main support structures remain at least partly functional during any of the several Design Accidental Events (DAEs). The DAEs were particular scenarios in which a failure (such as pipe rupture) was considered under particular conditions (such as wind direction). The possibility of accidental events which would make escape impossible should not exceed 1E-4/year. After some initial resistance, most operators actively applied safety evaluations; the NPD has been revising the guidelines to emphasize the importance of a continuous evaluation process. According to the NPD, the application of the guidelines contributes significantly to offshore safety.

Title: The Detection and Monitoring of Cracks in Structures, Process Vessels and Pipework by Acoustic Emission

Author: Rogers-LM **Corp. Author:**

Source: Hazards in the Process Industries: Hazards IX, The Institution of Chemical Engineers Symposium Series No. 97, The Institution of Chemical Engineers, pages 201-214, 6 references

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Inspection methods **ID:** 586

Abstract: A new method of acoustic emission (AE) analysis is described and results of onshore and offshore performance evaluation trials are presented. The Vulcan-8 AE system comprises an array of 8 transducers which sense the AE in the structure. The underwater header is fastened to the structure near the node being monitored. Tension cables carry signals to a computer on or near the installation. Validation is performed by pattern recognition. The transducers perform sensing and calibration. Rebooting is carried out automatically. An online computer displays location and density of AE, giving a cumulative picture and a relative depth of cracking. The history of crack growth is represented by histogram plots which monitor buildup of cracks at 6 day intervals. Laboratory tests on several large scale tubular joints showed that there was a good correlation between AE results and those obtained for the same joints by NDE. Noise resulting from compliance from a non-propagating crack or from corrosion products was found to be filtered out under normal operating conditions. The AE system was installed on a production platform jacket node joint and the transducers continued to operate to performance specification after severe hurricane conditions. The remote results obtained on crack location, increase in crack length and depth measurement were consistent with subsequent subsea nondestructive testing. AE monitoring of a section of a single point mooring with known defects showed the presence of cracks in areas thought to be free of defects. The AE system was also successfully used to detect cracks and crack growth on semisubmersible structures. The author concludes that AE is capable of detecting and locating crack propagation in welds sufficiently early to allow low cost repairs.

Title: Parametric Cost-Benefit Analysis Applied To Underwater Pipeline Safety

Author: Glickman-TS **Corp. Author:**

Source: Journal of Safety Research, Vol. 15, No. 3, pages 91-96, 2 references

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: Other **ID:** 587

Abstract: A parametric approach was used for evaluation of policy pertaining to under water pipeline safety. The effectiveness of signs used to mark places where gas and liquid pipelines cross under water was evaluated in the absence of experimentation. A cost/benefit analysis was executed treating the ability of these signs to reduce accidental pipeline damage as the unknown variable, with cost/benefit measures as the functions of the parameter. Pertinent regulations and accident reports are reviewed. Professional opinions were sought. Marine activity near the pipelines was analyzed. Equations were derived to indicate the percent reduction in accidents and the average annual benefit of having markers in place considering marker and upkeep costs. The safety record for gas pipelines indicated under water pipeline damage was not a serious problem. For liquid pipelines safety records were better, though consequences were more serious in human terms. The consensus of professional opinion was that signs have some usefulness, impossible to quantitate. Marine activity ranged up to 265,000 crossings annually in the Atlantic Coast area. Even at the lowest computed values of effectiveness, the traffic was high enough at pipeline crossings to justify signs even if only slightly effective. The author concludes that total deregulation of signs would be inadvisable. The method gives policy makers as much information as possible in the absence of experimentation. The degree of deregulation or increased sign density or visibility can be planned on a cost/benefit basis.

Title: Selected Safety-Related Events Reported In September And October 1983

Author: Casto-WR **Corp. Author:**

Source: Nuclear Safety, Vol. 25, No. 1, pages 115-117, 7 references

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: Experience/events **ID:** *588*

Abstract: Three safety related incidents involving nuclear power facilities reported during September and October 1983 were examined. At the first facility in Illinois, alarms on two control panels failed. The alarm recorder for the computer remained in operation and the facility was able to maintain full power. The circuit malfunction was traced and repaired. At a second facility, a reactor trip caused a water hammer in feed water lines resulting in a feed water pipe rupture. Cool down was begun. Subsequent inspection revealed cracking due to water hammer and thermal shock due to extreme temperature differences between normal and auxiliary feed water sources. Cracked piping was replaced and design changes were implemented to minimize water hammer potential and reduce thermal cycling. In October 1982 all three positive displacement charging pumps stopped when a third facility in Florida was in hot stand by. The volume control tank was found to be dry. Pumps were started again at reduced flow in under 30 minutes. Although the volume control tank was empty, its liquid sensors indicated all was well because the shared reference leg was empty. Pressure transducers were replaced and a separate instrument line for each instrument was installed. Personnel were advised to check reference legs routinely.

Title: Study on Current Practices, Technologies, Problems and Recommendations Relating to the Overall Safety of Gas Pipe

Author: Bartol-JA; Nichols-RO **Corp. Author:**

Source: Office of Pipeline Safety Operations, Materials Transportation Bureau, Department of Transportation, Washington, D.C., Report no. DOT/MTB/OPSO-76/01, 113 pages, 34 references.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1975 **Language:** English

Category: Experience/events **ID:** *589*

Abstract: The major safety problem of the million mile gas piping network in the US is the unintentional leakage of gas in a manner or location where it becomes a potential hazard. The discovery, prevention, and handling of leaks has been part of the business of the Office of Pipeline Safety Operations. Eight topics directly bear on these safety problems: assessment of pipelines, corrosion, outside forces, odorization, plastic pipe, emergency plans, valving and rapid shutdown, and master metering. The physical condition of most pipelines is only known after a leak has been located. Through the use of more acoustic emission tests to locate and assess flaws, and ultrasonic tests to note sizes and depths of flaws, deteriorating pipe can be identified before failure occurs. Corrosion accounts for the largest number of repaired leaks each year. To combat corrosion, more pipeline should be subject to cathodic protection and more electrical measurements should be taken around pipelines to determine corrosion susceptible pipes. Outside forces, such as excavators, also cause many leaks. Better pipeline markings is one means of protection. Better communications between evacuators and operators of underground facilities through the Utility Location and Coordination Committees and One Call systems can also reduce this problem. Odorization serves as a warning of the presence of natural gas. Odorant fading within the pipelines can be reduced by greater use of tertiary butyl mercaptan blends. Fading due to soil contact requires further research and development. Plastic pipe, while corrosion free, is more susceptible to outside force damage, and needs to be improved in terms of dimensional tolerances, brittle fractures, joining failures, pressure failures, stress cracking, and heat resistance. Better training of personnel in the implementation of emergency plans is needed. Automatic shut-off valves should be installed only after cost-benefit analyses have determined their usefullness. Master metering by distributors has unfortunately made the assessment and regulation of gas piping difficult. A survey of the piping used by customers on master meters is therefore important.

Title: Directives Concerning Oil Pipelines -- Report of the Committee on the Storage of Dangerous Substances

Author: Anonymous **Corp. Author:**

Source: Arbeidsinspectie, Directoraat-Generaal van de Arbeid, Voorburg, The Netherlands, 49 pages

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1973 **Language:** German

Category: Methods/design **ID:** *590*

Abstract: Dutch safety directives on design, laying, testing, and maintenance of oil pipelines are given. Topics covered include: authorization procedure; design and calculations; materials; execution of layout; inspection; testing before putting into operation; operating plant, including pumping stations; shut-off valves; electrical installations; fire precautions; control room and safety devices; and maintenance, including emergency procedure. Rules for pipe socket welding reinforcement of tank openings, protection against corrosion removal of polluting oil from water are also covered, and a model of a printed form for notification of leaks is shown. (Dutch)

Title: Safety Procedures in the Gas Industry

Author: Chinnock-JHJ **Corp. Author:**

Source: Occupational Safety and Health, Vol. 10, No. 1, pages 12-27

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1980 **Language:** English

Category: Other **ID:** 591

Abstract: Safety factors and procedures in the British gas industry are discussed. Potential hazards, such as pipeline leaks, explosions, and ground movements are reviewed. The distribution system of the British gas pipeline network is described. Safeguard and repair methods, dealing primarily with planned leakage survey programs are discussed, along with emergency procedures. The authors conclude that better communications between utility industries will increase safety and efficiency.

Title: What Really Happened at Flixborough?

Author: Kinnersly-P **Corp. Author:**

Source: New Scientist, pages 520-522

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1975 **Language:** English

Category: Experience/events **ID:** 592

Abstract: The Flixborough, England, disaster was an explosion of cyclohexane vapor which killed 28 workers, and caused many injuries and extensive damage. The evidence presented at the public inquiry is reviewed. Evidence was produced to suggest that a split in a small pipeline allow cyclohexane to leak, a small explosion than ruptured a much larger pipe, causing the massive explosion, but this evidence was disputed. Much information in metallurgy has been obtained, and a number of recommendations made by the Department of Employment. Some of the recommendations are embodied in a new Health and Safety Act which came into force in April 1975. Since the disaster many chemical plants have been examining their handling of cyclohexane to improve safety.

Title: The AEC and the Loss of Coolant Accident

Author: Wilson-R **Corp. Author:**

Source: Nature, Vol. 241, No. 5388, pages 317-320, 18 references

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1973 **Language:** English

Category: Failure probability **ID:** 593

Abstract: Review of the risks associated with the possible loss of coolant from a nuclear reactor. Topics include loss of reactor coolant accident, operation of nuclear power stations as steam generating power stations, failure in the reactor vessel or steam piping that can lead to a catastrophe, the need for core cooling, prediction of the failure of the steam system, the role of the Atomic Energy Commission in safety research, cost of nuclear power stations, and fossil fuel prices.

Title: Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors

Author: **Corp. Author:** U.S. Nuclear Regulatory Com

Source: NUREG-0691

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1980 **Language:** English

Category: Operating experience **ID:** 594

Abstract: This report summarizes an investigation of known cracking incidents in PWRs. Several instances of cracking in FW-piping in 1979, together with reported cases of SCC at Three Mile Island-1 led to the establishment of the third Pipe Crack Study Group (PCSG). Major differences between the scope of the 3rd PCSG and the previous two are: (1) the emphasis given to system safety implications of cracking, and (2) the consideration given all cracking mechanisms known to affect PWR piping, including the failure of small lines in secondary safety systems. The present PCSG reviewed existing information on cracking of PWR pipe systems, either contained in written records or collected from meetings in the US, and made recommendations in response to the PCSG charter questions and to other major items that may be considered to either reduce the potential for cracking or to improve licensing bases.

Title: Safety Spacing Between Parallel Pipelines for Combustible Liquids and Gases

Author: Helwig-N; Nabert-K **Corp. Author:**

Source: Arbeitsschutz, No. 5, pages 105-108, 12 references

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1969 **Language:** German

Category: Experience/events **ID:** 595

Abstract: When pipelines are laid parallel one to another, there is a danger that in the event of an accident to one, the next one may be damaged. The authors assess the consequences of fires and explosions and the mechanical effects of escaping liquids and gases on the basis of a study of the few accident reports available. When pipes are laid underground for greater safety, serious damage to parallel pipes is most likely to occur in the case of gas pipelines. In the light of available information regarding the size of explosion trench to be expected, the authors propose provisional guide distances for the safe spacing of gas and liquid pipelines, in relation to pipe diameter. (German)

Title: Steam Boilers

Author: Anonymous **Corp. Author:**

Source: Accidents, pages 35-40

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1971 **Language:** English

Category: Experience/events **ID:** 596

Abstract: Review of reported accidents involving steam system components. Topics include damage by water hammer caused by water traveling along the pipes at a very high velocity and striking a fitting to produce a hammer-like blow; and bow water alarm where an explosion is caused by the firebox plate becoming over-heated due to lack of water in the boiler. Illustrations are given for both types of accidents. Details of the causes of accidents and the steps to be taken for notifying the proper authorities are discussed.

Title: COLD WEATHER EFFECTS ALTER SAFSTOR EMPHASIS.

Author: Anagnostopoulos-H **Corp. Author:**

Source: Nuclear Engineering International. Aug.1994, vol.39, no.481, 18-19.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Experience/events **ID:** 597

Abstract: Reports that severe cold weather in January 1994 caused several piping systems to freeze and break in Commonwealth Edison's retired Dresden 1 reactor in the United States of America (USA). This potentially caused a drain down of the spent fuel pool. A brief review of the incident and corrective action taken to avoid the risk is presented. The unit is now in the final stages of a decommissioning effort aimed at preparing the unit for SAFSTOR status.

Title: A DATABASE TO EVALUATE STRESS INTENSITY FACTORS OF ELBOWS WITH THROUGHWALL FLA

Author: Chattopadhyay-J; Dutta-BK; and-others **Corp. Author:**

Source: International Journal of Pressure Vessels and Piping. 1994, vol.60, no.1, 71-83.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Test/analysis **ID:** 598

Abstract: The leak-before-break (LBB) concept has widely replaced the traditional design basis event of a double-ended guillotine break (DEGB) in the design of primary heat transport (PHT) piping. The LBB concept requires postulation of the largest credible cracks in highly stressed locations and demonstration of their stability under the maximum credible loading conditions. Stress analysis of PHT piping in nuclear power plants shows that the highly stressed piping components are normally elbows and branch trees. A database is described to evaluate the stress intensity factors (SIF) for throughwall circumferential and longitudinal cracks under combined internal pressure and bending moment.

Title: PREDICTION OF DISCHARGE RATE FROM PRESSURIZED VESSEL BLOWDOWN THROUGH SHEARED

Author: Khajehnajafi-S; Shinde-A **Corp. Author:**

Source: Process Safety Progress. Apr.1994, vol.13, no.2, 75-82.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Test/analysis **ID:** 599

Abstract: Describes a general purpose, quasi-steady model which can calculate the discharge rate from a tank-pipe system (rupture in the tank or shearing of an attached pipe). The model output includes history of the tank variables (pressure, temperature, mass content, liquid level) as well as the transient outflow of chemicals as a function of momentary tank thermodynamic conditions. Results of the model for two phase flow through a pipe are validated against experimental data. 18 refs.

Title: ENHANCED ULTRASONIC EXAMINATION OF FEEDWATER PIPE-TO-NOZZLE WELDS.

Author: Bisbee-LH; Burns-ST **Corp. Author:**

Source: Nuclear Plant Journal. Mar./Apr.1994, vol.12, no.2, 42, 44, 46, 53.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Inspection methods **ID:** 600

Abstract: Enhanced volumetric examination methods for steam generator feedwater pipe-to-nozzle welds and associated counterbore regions are necessary to improve reliability in the detection, sizing, and evaluation of thermal fatigue cracking and to enhance confidence in repair decisions. The Focused Array Transducer System (FATS) and TestPro data acquisition and analysis system offer utilities the most advanced nondestructive examination (NDE) system available for this application. Details are given of a demonstration and the documented results which confirm the stated advantages of using the TestPro/FATS system for the reliable detection and characterisation of relevant indications in the subject welds.

Title: REACHING THE PARTS NO-ONE HAS REACHED BEFORE.

Author: Kristensen-WD; Jeppesen-L **Corp. Author:**

Source: Nuclear Engineering International. Jul.1994, vol.39, no.480, 22-23.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Experience/events **ID:** 601

Abstract: Reports on work being carried out at Oskarshamn-1 in Sweden. In the course of doing the work to address problems which were found in its emergency core cooling systems, extensive pipework inspection revealed cracks in four of the six feedwater pipes inside the reactor vessel. To change the pipes, all the internals of the reactor vessel had to be removed, providing a unique opportunity to carry out a complete verification of the reactor vessel itself.

Title: CREEP-FATIGUE CRACK PROPAGATION TESTS AND THE DEVELOPMENT OF AN ANALYTICAL EVA

Author: Shimakawa-T; Takahashi-H; and-others **Corp. Author:**

Source: Nuclear Engineering and Design. Mar.1993, vol.139, no.3, 283-292.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 602

Abstract: Shows test results and estimations of the surface crack growth in a straight pipe and elbow under creep-fatigue conditions. The electrical potential method was successfully applied to measure the surface crack geometry; so crack propagation rates both for surface and thickness direction were measured.

Title: HYDROGEN PEROXIDE DEFLAGRATION IN WASTE WATER TREATMENT TANK.

Author: Anonymous **Corp. Author:**

Source: Loss Prevention Bulletin. Apr.1994, no.116, 17-20.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Experience/events **ID:** 603

Abstract: Describes an incident which occurred at 12.20 hours on 3 August 1991 inside a waste-water treatment tank in an unspecified plant making a metal extractant, MOC-45. The tank sustained damage to its manway and associated pipework. There was one minor injury. A description of the process is given along with details of the investigation and conclusions drawn.

Title: THE CONSERVATISM OF THE R6 PROCEDURE WHEN APPLIED TO THE ASSESSMENT OF THE INTEG

Author: Smith-E **Corp. Author:**

Source: International Journal of Pressure Vessels and Piping. 1994, vol.57, no.2, 163-168.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Test/analysis **ID:** 604

Abstract: Demonstrates, by analysing a specific simulation model, the extent to which the R6 procedure is unduly conservative when it is applied to the assessment of the integrity of a cracked piping system which displays limited elastic follow-up.

Title: EXPERIMENTAL AND ANALYTICAL FRACTURE ASSESSMENT OF 165-MM DIAMETER AISI 304 SEAM

Author: Shin-CS; Tseng-RP **Corp. Author:**

Source: International Journal of Pressure Vessels and Piping. 1994, vol.57, no.2, 169-185.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Methods/comparison **ID:** 605

Abstract: Based on data produced from fracture experiments on 165-mm diameter piping with through-wall cracks, the predictive capabilities of the three engineering assessment methods are compared: the J-integral method, the R6 revision 3 method, and the J-estimation method.

Title: GUILLOTINE FAILURE OF FIXED-END PIPES, PRESSURIZED WITH HOT WATER.

Author: Shewfelt-RSW; Leitch-BW; and-others **Corp. Author:**

Source: International Journal of Pressure Vessels and Piping. 1994, vol.57, no.2, 211-221.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Test/analysis **ID:** 606

Abstract: Describes instrumented small-scale burst tests carried out in order to determine the parameters controlling guillotine failure in the event of a pressure tube and its calandria tube rupturing during normal operation in a CANDU (Canada deuterium uranium) reactor.

Title: SASKATCHEWAN, CANADA (PIPELINE ACCIDENT).

Author: Anonymous **Corp. Author:**

Source: Lloyds Casualty Week. 25 Feb.1994, vol.295, no.7, 137-138.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Experience/events **ID:** 607

Abstract: Gives details of a gas pipeline which ruptured on 17 February 1994 causing a huge fireball. No-one was injured.

Title: NUMERICAL EVALUATION OF STRESS INTENSITY FACTOR FOR VESSEL AND PIPE SUBJECTED TO T

Author: Kim-YW; Lee-HY; and-others **Corp. Author:**

Source: International Journal of Pressure Vessels and Piping. 1994, vol.58, no.2, 215-222.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Methods/comparison **ID:** 608

Abstract: Presents the results of a study in which the thermal weight function method and the finite element method were employed in the numerical computation of the stress intensity factor for a cracked vessel and a cracked pipe subjected to thermal shock. 17 refs.

Title: A COMPREHENSIVE PROGRAM FOR PREVENTING CYCLOHEXANE OXIDATION PROCESS PIPING FAI

Author: Sadler-DL; Matusz-BT **Corp. Author:**

Source: Process Safety Progress. Jan.1994, vol.13, no.1, 45-49.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Experience/events **ID:** 609

Abstract: In 1971 the cyclohexane oxidation unit at the Monsanto plant at Pensacola, Florida suffered a pipe rupture which resulted in the formation of a cyclohexane vapour cloud. There was no ignition of the cloud. The incident and past and present piping system remediation programmes are described, involving a survey of critical piping systems utilizing flexibility analysis, acoustic emission testing, radiographic and ultrasonic inspection, safety studies, and distributed process controls.

Title: RUPTURE OF PRESSURISED TUBES BY MULTIPLE CRACKING AND FRAGMENTATION.

Author: Ford-IJ **Corp. Author:**

Source: International Journal of Pressure Vessels and Piping. 1994, vol.57, no.1, 21-29.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Methods **ID:** 610

Abstract: The likelihood of stable propagation of an axial crack away from a rupture site in a pressurised tube is a problem of concern in a number of areas, including the gas and nuclear industries. A model of crack propagation is developed which provides the crack velocity and deformation geometry and predicts a minimum driving pressure. The model also offers a criterion for the appearance of multiple cracks and subsequent fragmentation of the tube wall due to excessive bending strains. Calculations of interest in gas pipeline rupture and fast reactor fuel pin failure are presented. 15 refs.

Title: CONQUERING SERVICE WATER PIPE CORROSION.

Author: Leech-JN; Miller-DJ; and-others **Corp. Author:**

Source: Nuclear Engineering International. Jan.1994, vol.39, no.474, 31, 33-35.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Experience/events **ID:** 611

Abstract: The Public Service Electric and Gas Company (USA) and a number of other utilities in the United States of America have experienced piping leaks in their power plant service water cooling system. Investigation of the causes of these leaks have shown that microbiologically influenced corrosion (MIC), galvanic corrosion and other corrosion effects were prime contributors to the piping's detioration. It is stressed that companies must develop comprehensive plans for minimising the potential for such damage.

Title: POWER STATION, VOHBURG, BAVARIA. (EXPLOSION CAUSED BY NATURAL GAS, 15 FEBRUARY 199

Author: Anonymous **Corp. Author:**

Source: Lloyd's Weekly Casualty Reports. 13 Mar.1992, vol.287, no.10, 215.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Experience/events **ID:** 612

Abstract: The no.3 generator, boiler and connecting pipework at the electrical power station owned by Isar Amper Werke in Vohburg, Bavaria, sustained severe damage on 15 February 1992 due to an explosion in the boiler as natural gas fed into the burner. The repairs will take 3-4 months to carry out. Loss is estimated at more than 3 million Deutschmarks.

Title: A FRACTURE MECHANICS EVALUATION OF THE FAILURE OF FLAKE GRAPHITE CAST IRON PIPE.

Author: Norton-G; Dutton-J; and-others; Health and Safety Executive **Corp. Author:**

Source: 1993. (IR/L/MM.ME/93/01) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 613

Abstract:

Title: PIPE FAILURES IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.

Author: Jamali, K. **Corp. Author:**

Source: EPRI-TR-100380 Interim Report)

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Experience/events **ID:** 614

Abstract:

Title: PIPELINE LEAK DETECTION BASED ON MASS BALANCE: IMPORTANCE OF THE PACKING TERM.

Author: Stouffs-P; Giot-M **Corp. Author:**

Source: Journal of Loss Prevention in the Process Industries. 1993, vol.6, no.5, 307-312.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Inspection methods **ID:** 615

Abstract: Presents some mass balance systems, after a brief survey of pipeline leak detection systems. Such systems use a pipeline flow model in order to compute the change in pipeline inventory during a transient flow. The packing term is a function of the speed of sound in the pipeline. Describes the consequences of leak detection thresholds.

Title: COMPARISONS BETWEEN FINITE-ELEMENT ANALYSIS PREDICTIONS AND PIPE FRACTURE EXPERI

Author: Brust-FW; Ahmao-J; and-others **Corp. Author:**

Source: Nuclear Engineering and Design. Sep.1993, vol.143, nos.2/3, 201-215.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 616

Abstract: Presents the results of ten finite-element analyses of cracked pipe subjected to bending loads compared to the corresponding experimental results produced from full-scale tests. Detailed results from two international round-robin problems are also presented. In all, nine through-wall cracked pipe and one surface cracked pipe is considered. The cracked pipe includes stainless, carbon, and welded pipe.

Title: FRACTURE MECHANICS INVESTIGATIONS ON A PIPE WITH A CIRCUMFERENTIAL FLAW SUPPORTE

Author: Brocks-W; Mueller-W; and-others **Corp. Author:**

Source: Nuclear Engineering and Design. Sep.1993, vol.143, nos.2/3, 171-185.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 617

Abstract: The transferability of crack resistance properties obtained from fracture mechanics specimens to analyse stable crack growth of a 120 degree surface flaw in a pipe of large diameter under pure bending is discussed supported by results of an elastic-plastic FEM (finite element method) calculation. 20 refs.

Title: CYCLIC CRACK GROWTH EVALUATION OF 20 MNMON155 PIPING STEEL IN HIGH-OXYGEN REACT

Author: Aaltonen-P; Rintamaa-R; and-others **Corp. Author:**

Source: Nuclear Engineering and Design. Oct.1993, vol.144, no.1, 111-122.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 618

Abstract: Samples of a low alloy steel piping material taken from the full scale corrosion fatigue test loop of the Heissdampfreaktor (HDR) plant have been tested at 240 degrees celsius in high oxygen reactor water. Autoclave testing results and fracture surface observations are preserved. 15 refs.

Title: RESEARCHES ON AIR INGRESS ACCIDENTS OF THE HTTR.

Author: Hishida-M; Fumizawa-M; and-others **Corp. Author:**

Source: Nuclear Engineering and Design. Oct.1993, vol.144, no.2, 317-325.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 619

Abstract: Deals with experimental and analytical studies which have been performed to understand air ingress processes during primary-pipe and stand-pipe rupture accidents of the HTTR (High Temperature Gas-Cooled Reactor). Air ingress processes are summarised during the first stage of the primary-pipe rupture accident and during the stand-pipe rupture accident.

Title: CREEP, FATIGUE, FLAW EVALUATION, AND LEAK-BEFORE-BREAK ASSESSMENT: TECHNOLOGY IN

Author: Graud-YS; American Society of Mechanical Engineers **Corp. Author:**

Source: New York, 1993. (PVP - vol.266) 295pp.1993 PRESSURE VESSELS AND PIPING CONFERENCE, DENVER, COLORADO, JULY 25-29, 1993.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Research/theoretical **ID:** 620

Abstract:

Title: CONTROLLING STRESS CORROSION CRACKING IN BOILING WATER REACTORS.

Author: Jones-RL; Gilman-JD; and-others **Corp. Author:**

Source: Nuclear Engineering and Design. Aug.1993, vol.143, no.1, 111-123.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Methods **ID:** 621

Abstract: Presents a description of the pipe cracking remedies that were developed during the major research and development programme on boiling water reactor (BWR) pipe cracking co-funded by the Electric Power Research Institute (EPRI), the General Electric Company (GE), and the BWR Owners Group for IGSCC (Intergranular Stress Corrosion Cracking) research between 1979 and 1988. The prospects of adapting these remedies for the protection of internals and attachments are discussed.

Title: INTERFACING SYSTEMS LOCA (ISLOCA) COMPONENT PRESSURE CAPACITY METHODOLOGY AND

Author: Wesley-DA **Corp. Author:**

Source: Nuclear Engineering and Design. Aug.1993, vol.142, nos.2 and 3, 209-224.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Damage probability **ID:** 622

Abstract: A proposed methodology and sample results from several plant investigations are presented for evaluating the expected pressure capacity of nuclear power plant components which could potentially be subjected to Interfacing Systems. Loss of coolant accident conditions. The probabilities of failure, as a function of internal pressure, are evaluated as well as the variabilities associated with them. Leak rates or leak areas are estimated for the controlling modes of failure. Pressure capacities for the pipes and vessels are evaluated using limit-state analyses for the various failure modes considered. 13 refs.

Title: EVALUATION OF FLAWED PIPING UNDER DYNAMIC LOADING.

Author: Nickell-RE; Quinones-DF; and-others **Corp. Author:**

Source: Nuclear Engineering and Design. Jul.1993, vol.142, no.1, 77-87.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Research/theoretical **ID:** 623

Abstract: Describes analytical studies of several of the large-scale flawed pipe experiments conducted for the International Piping Integrity Research Group (IPIRG), including detailed discussion of the test with the longest loading duration. Dynamic excitation with increasing load amplitude leads to failure of the piping at a predesignated test section containing a large manufactured flaw. Here, elastic analysis is shown to describe the system response reasonably well, provided that an appropriate value of structural damping can be selected. A simplified two degree-of-freedom model displays sensitivity to damping and is used to help select the optimal damping value for use in subsequent finite element calculations. The discussion includes comparisons of the calculated IPIRG results with ASME Code-suggested analysis damping values. 14 refs.

Title: VALIDATION OF ROOM-TEMPERATURE PRIMARY CREEP CRACK-GROWTH ANALYSIS FOR SURFAC

Author: Leis-BN; Brust-FW **Corp. Author:**

Source: Nuclear Engineering and Design. Jul.1993, vol.142, no.1, 69-75.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Test/analysis **ID:** 624

Abstract: Reviews the theoretical considerations that underlie the development of an engineering analysis of ductile, time dependent flaw growth as adapted to axial surface flaws in cylindrical containers such as pipes and tanks. Thereafter the validation of this analysis by comparison of the predicted and observed behaviour of an extensive database for part-through-wall defects in pipes is presented. 18 refs.

Title: EVALUATION OF CRACK OPENING TIMES AND LEAKAGE AREAS FOR LONGITUDINAL CRACKS IN

Author: Bhandari-S; Leroux-JC **Corp. Author:**

Source: Nuclear Engineering and Design. Jul.1993, vol.142, no.1, 15-19.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Research/theoretical **ID:** 625

Abstract: The present study (Part I and Part II) deals with a method of evaluating the average time to crack opening as well as the maximum leakage areas in the case of longitudinal through-wall cracks in a cylinder with internal pressure. Part I deals with the leakage areas. Starting from the linear elastic theory as applied to the case of a central crack in a plate, leak areas are evaluated in a cylinder under elastoplastic conditions by using an amplification factor and a plasticity correction factor. A reasonable upper bound is proposed which takes into account the interaction between plasticity and curvature effects as a first approximation and considers the crack opening uniform all over the crack surface. The method is validated using the available experimental and/or computational results. 10 refs.

Title: AUTOMATING HYDROSTATIC TESTS FOR PIPE LEAKS.

Author: Baker-B; Musilli-M **Corp. Author:**

Source: Chemical Engineering. Dec.1992, vol.99, no.12, 153-154.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Inspection methods **ID:** 626

Abstract: Describes a fast, computer-based technique that can locate small leaks along 100 mile lengths of pipeline.

Title: A PROBABILISTIC FRACTURE MECHANICS ANALYSIS FOR CRACKED PIPE USING 3-D MODEL.

Author: Yagawa-G; Ye-GW **Corp. Author:**

Source: Reliability Engineering and System Safety. 1993, vol.41, no.2, 189-196.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Research/theoretical **ID:** 627

Abstract: For the purpose of a probabilistic fracture mechanics (PFM) estimation based on elastic-plastic fracture mechanics (EPFM) in the field of reliability analysis of pressure vessesl and piping, a 3-D EPFM database of fully plastic solutions for surface cracks and a PFM code for the integrity evaluation of nuclear structural components based on the above database are given. As an example, a comparison study of the PFM analysis is performed between the 2-D and the 3-D solutions to demonstrate the 3-D effects on the solutions. 23 refs.

Title: STRATAS: DEVELOPMENT OF AN HSE AUDIT SCHEME FOR LOSS OF CONTAINMENT INCIDENTS. PA

Author: Ratcliffe-KB **Corp. Author:**

Source: Loss Prevention Bulletin. Aug.1993, no.112, 1-6.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Failure probability **ID:** 628

Abstract: Looks at the development by the Health and Safety Executive of a system called SRTATAS (Structured Audit Technique for the Assessment of Safety Management Systems). This is the first in a series of 3 articles which describes the empirical basis for the system using a 3 dimensional classification scheme for vessel and pipework failures. The 3 components are: 1) the apparent direct cause of the failure; 2) the origin in the plant's lifecycle of the failure; and 3) the prevention or recovery mechanisms which were available but failed to identify the fault. The author is a member of staff of the Health and Safety Executive. 12 refs.

Title: CONSTRUCTION AND TESTING OF A TEST FACILITY FOR THE NON-DESTRUCTIVE DETECTION AN

Author: Wuensch-W; Germany (Federal Republic). Bundesministerium fuer Umwelt, Naturschutz und Reaktorsicherheit **Corp. Author:**

Source: Bonn, 1992. (Schriftenreihe, Reaktorsicherheit und Strahlenschutz) (BMU-1993-368) 150pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** German

Category: Test/analysis **ID:** 629

Abstract:

Title: STANDARD TEST METHOD. EVALUATION OF PIPELINE STEELS FOR RESISTANCE TO STEPWIRE CR

Author: National Association of Corrosion Engineers **Corp. Author:**

Source: Houston, Tex., 1987. (NACE standard TM0284-87) 6pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Test/analysis **ID:** 630

Abstract:

Title: COMPUTATION OF NATURAL GAS PIPELINE RUPTURE PROBLEMS USING THE METHOD OF CHARA

Author: Olorunmaiye-JA; Imide-NE **Corp. Author:**

Source: Journal of Hazardous Materials. Apr.1993, vol.34, no.1, 81-98.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Analysis of break effects **ID:** 631

Abstract: The flow in a long, high pressure, natural gas pipeline following sudden rupture was modelled as unsteady one-dimensional isothermal flow. The set of hyperbolic partial differential equations were solved with a numerical method of characteristics. To assess the hazard of a natural gas pipeline rupture, it is necessary to know the rate of outflow of the gas at the breakpoint as a function of time. The predicted mass flow rate of gas out of the broken end was 18 per cent lower than that predicted using adiabatic flow theory, whereas, there was good agreement with the results of earlier workers who also used isothermal flow theory whose computation method was based on weighted residuals.

Title: PREDICTION OF VESSEL AND PIPING FAILURE RATES IN CHEMICAL PROCESS PLANTS USING THE

Author: Medhekar-SR; Bley-DC; and-others **Corp. Author:**

Source: Process Safety Progress. Apr.1993, vol.12, no.2, 123-125.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Damage probability **ID:** 632

Abstract: Describes the use of the Thomas model which was developed using a large database for predicting vessel and pipe failures for the nuclear industry but can be used in the chemical industry. The Thomas model here was used to predict failure rates of process vessels sharing corrosive and hazardous chemicals.

Title: FIVE HURT AT NUCLEAR PLANT.

Author: Anonymous **Corp. Author:**

Source: Chemical Engineer. 24 Jun.1993, no.545, 7.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 633

Abstract: A pipe in an emergency cooling system burst during a test at Quad. Cities nuclear power plant in Cordova, Illinois, United States on 9 June 1993. Five workers were burned, one seriously, after the pipe burst due to the failure of a pump which could send cooling water into the reactor in an emergency. The seriously injured woman worker suffered 30 percent burns. The incident is being investigated by the company and the Nuclear Regulatory Commission.

Title: COMBINING AI AND NDE TO AID PIPEWORK REPAIR AND INSPECTION DECISIONS.

Author: Miyoshi-S **Corp. Author:**

Source: Nuclear Engineering International. Jul.1993, vol.38, no.468, 31-32, 34.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Inspection methods **ID:** 634

Abstract: NDE (Non Destructive Examination) techniques such as computer tomography, ultrasonic holography and electromagnetic acoustic transducer technology have been combined with software using AI (Artificial Intelligence) principles to create a system that focuses inspection work at nuclear power plants on the parts of plant pipework most vulnerable to cracking. Ultimately the system could enable staff without specialist knowledge to take decisions about repairs.

Title: CHERNOBYL-1 FORCED DOWN BY SMALL LEAK IN PUMP.

Author: Anonymous **Corp. Author:**

Source: Nuclear News. Apr.1993, vol.36, no.5, 64.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 635

Abstract: The Chernobyl-1 reactor was off-line 1-4 March 1993 for the repair of a small leak in the pipework of one of the eight coolant circulation pumps.

Title: LOVIISA FEEDWATER PIPE BREAK LIKE ONE IN 1990.

Author: Anonymous **Corp. Author:**

Source: Nuclear News. Apr.1993, vol.36, no.5, 63.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 636

Abstract: The Utility Imatran Voima Oy (IVO), , has announced that a feedwater pipe break occurred on 25 February 1993. The incident was initially rated at level 1 on the international nuclear event scale, but there was speculation it might be uprated to level 2 because of its similarity to a previous event in May 1990.

Title: SERVICE WATER PIPE BREAK.

Author: Anonymous **Corp. Author:**

Source: Nuclear News. May 1993, vol.36, no.7, 26.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 637

Abstract: Describes how a broken water pipe made of fibre glass forced the Cleveland Electric Illuminating Company Perry-1 boiling water reactor off-line on 26 March 1993. Also gives details of another incident on 22 December 1991 involving a fibreglass pipe carrying secondary coolant.

Title: A CLASSIFICATION SCHEME FOR PIPEWORK FAILURES TO INCLUDE HUMAN AND SOCIOTECHNIC

Author: Hurst, N. W. **Corp. Author:**

Source: Journal of Hazardous Materials. Mar.1991, vol.26, no.2, 159-186.

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Experience/events **ID:** 638

Abstract: Analyses the contribution of human error and sociotechnical failures to pipework failure frequencies and introduces a failure classification scheme used to investigate incidents involving pipework failures at chemical and major hazard plant. Results show that 90 per cent of the analysed incidents could have potentially been prevented by preventive mechanisms within the scope of management control. The implications of the classification schemes are considered with respect to the contributions which compose the generic failure rates which are used in the calculation of risk for major hazard plant. Mr N. Hurst is a member of staff of the Health and Safety Executive. 23 refs.

Title: A STATISTICAL THEORY OF CORROSION FAILURE IN PIPELINES.

Author: Davies-JKW **Corp. Author:** Research/theoretical

Source:

SKI Project File: Nej **Transfer:** Nej **Publ year:** **Language:** English

Category: Research/theoretical **ID:** 639

Abstract: This paper attempts to answer the question, 'what may be inferred about the rate of occurrence of leaks in the pipeline if no leak has been observed in a given period of time?' A simple statistical model of corrosion failure of a pipeline is described, fr

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Title: THE REGRESSION ANALYSIS OF POISSON RARE-EVENT DATA.

Author: Davies-JKW **Corp. Author:**

Source: United Kingdom Atomic Energy Authority National Centre of Systems Reliability, 1982.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1982 **Language:** English

Category: Research/theoretical **ID:** 640

Abstract: The author is a member of staff of the Health and Safety Executive. In a paper read at the Second National Reliability Conference 1979, Davies showed by means of a simple a priori argument that corrosion failures in pipelines could well follow the Poisson rare-event distribution, thus underlining the fact that much failure data is Poisson in character. In practice one would like to be able to relate such rare-event data to a set of relevant physical factors. The usual assumptions of linear regression analysis - normally distributed errors with constant variance - do not apply in the case, in which case one is obliged either to transform the observations so that these assumptions are approximately satisfied or else to formulate a parallel theory in which the observations are Poisson-distributed from the outset. Such a theory is presented in this paper and illustrated by an example.

Title: ELASTIC-PLASTIC FINITE ELEMENT ANALYSIS OF CRACK GROWTH IN LARGE COMPACT TENSION

Author: Ahmad-J; Nakagaki-M; and-others; Battelle. Columbus Division United States. Nuclear Regulatory Commission **Corp. Author:**

Source: USGPO, 1986. (NUREG/CR-4573) (BMI-2135) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Research/theoretical **ID:** 641

Abstract:

Title: A REVIEW OF FATIGUE CRACK GROWTH OF PRESSURE VESSEL AND PIPING STEELS IN HIGH-TEMP

Author: Cullen-WH; Torronen-K; United States. Naval Research Laboratory United States. Nuclear Regulatory Commission **Corp. Author:**

Source: Nuclear Regulatory Commission, 1980. (NUREG/CR-1576) (NRL-MR-4298) (AD A089 697) 126pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1980 **Language:** English

Category: Research/theoretical **ID:** 642

Abstract:

Title: AMMONIA IN SHIPHOLD KILLS 7, INJURES 7.

Author: Anonymous **Corp. Author:**

Source: Safety Engineering News. Mar.1983, no.8, 8.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: Experience/events **ID:** 643

Abstract: Seven Japanese workers were killed and seven suffered various degrees of toxic poisoning when ammonia fumes leaked from a broken pipe in a hold while they were unloading a South Korean fish carrier in Kesennuma, Miyagi prefecture, on April 11, 1982.

Title: ONE KILLED, 15 HURT AS STEAM LINE RUPTURES AT POWER PLANT IN NEVADA.

Author: McMillan-P; Thackrey-T **Corp. Author:**

Source: Los Angeles Times. 10 Jun.1985, vol.104, section 1, 3, 18.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: Experience/events **ID:** 644

Abstract: One person was killed and 15 injured on 9 June 1985 at the Southern Californian Edison's coal-fired Mohave Generating Plant in Laughlin, Nevada. The control room was destroyed by the blast, and the generating units were closed down. The cause was not immediately known, but occurred when a 30 inch steel reheat pipe carrying steam at 1000 degrees Celsius at 600 pounds/square inch ruptured.

Title: BULGARIA BLAMES NEGLECT FOR CHEMICAL DISASTER. (EXPLOSION AT A CHEMICAL PLANT IN

Author: Anonymous **Corp. Author:**

Source: New York Times. 4 Nov.1986, vol.136, 7.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Experience/events **ID:** 645

Abstract: Briefly describes an incident in which 17 people were killed in an explosion at a chemical complex in Devnya, Bulgaria on 1st November 1986. Experts believe that a rupture in the pipe which connects the workshop handling vinyl chloride with polyvinyl chloride caused the explosion, and said that if the piping had been regularly checked by x-rays, the fault would have been detected.

Title: PIPELINE ACCIDENT REPORT, SUN PIPELINE CO., RUPTURE OF 8 INCH PIPELINE, ROMULUS, MICHI

Author: National Transportation Safety Board, **Corp. Author:**

Source: 1976. (PB 257 671) 22 pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1976 **Language:** English

Category: Experience/events **ID:** 646

Abstract:

Title: PIPELINE ACCIDENT REPORT : PACIFIC GAS AND ELECTRIC COMPANY, NATURAL GAS LEAK, SAN

Author: National Transportation Safety Board **Corp. Author:**

Source: NTIS, 1982. (PB 82 91650) (NTSB-PAR-82-1) 29pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1982 **Language:** English

Category: Experience/events **ID:** *647*

Abstract:

Title: NEW ENGINEERING RULES FOR FLAMMABLE LIQUIDS.

Author: Germany (Federal Republic). Bundesministerium fuer Arbeit **Corp. Author:**

Source: Bundesarbeitsblatt. Dec.1982, no.12, 34-81.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1982 **Language:** German

Category: Other **ID:** *648*

Abstract: Lists changes, supplements and revisions approved by the BMA (Bundesministerium fuer Arbeit) in June 1982, dealing with refuelling and draining facilities, airfield refuelling facilities and pipelines. Also lists new BMA guidelines on leakage indicators in containers and pipelines.

Title: LESSON LEARNT FROM BLASTS 8 YEARS AGO. (GAS EXPLOSIONS).

Author: Copps-A **Corp. Author:**

Source: Daily Telegraph. 11 Jan.1985.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: Experience/events **ID:** *649*

Abstract: Discusses the spate of gas explosions which occurred in 1977 and which led to the instigation of the King Inquiry into gas explosions. The explosions, which cost ten lives and caused eight hundred thousand pounds worth of damage, was caused by shrinkage of iron mains pipe during extremely cold weather, followed by a hot dry summer which resulted in soil movement. Several other gas explosions which have occurred in recent times are outlined.

Title: PIPELINE ACCIDENT REPORT : WILLIAMS PIPE LINE COMPANY LIQUID PIPELINE RUPTURE AND FI

Author: National Transportation Safety Board **Corp. Author:**

Source: Washington, D.C., 1987. (PB87-916502) (NTSB/PAR-87/02) 58pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Experience/events **ID:** *650*

Abstract:

Title: EVALUATE LNG'S STORAGE HAZARD.

Author: Rigard-J; Vadot-L **Corp. Author:**

Source: Hydrocarbon Processing. Jul.1979, vol.58, no.7, 267-268.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1979 **Language:** English

Category: Other **ID:** 651

Abstract: An investigation of LNG hazard protection is described. Vapour concentrations around the storage tank for three LNG escape flows were charted, and the flammability limits were determined. The escape flows represented a valve leak, a pipe failure and the destruction of the entire tank. The simulation was based on the water analogue technique. It attempts to solve the LNG storage hazard problem, and can be used to establish protection required at given storage points.

Title: AN EXPERIMENTAL STUDY OF THE IGNITION OF NATURAL GAS IN A SIMULATED PIPELINE RUPTU

Author: Hoff-ABM **Corp. Author:**

Source: Combustion and Flame. Jan.1983, vol.49, no.1/3, 51-58.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: Test/analysis **ID:** 652

Abstract: Experiments have been made with the ignition of natural gas in a simulated pipeline rupture, to study the pressure wave generated. Various conditions were used to produce the gas-air mixture, which was ignited by firing an incandescent bullet into it. The pressure waves that resulted from the ignition of the mixtures were recorded on a storage oscilloscope and/or an ultraviolet recorder via microphones placed at various points. The shape of the pressure wave was not always the same, due to the ignition starting at different points in the mixture. Some of the experiments were recorded on film by a high-speed camera. 11 refs.

Title: TRANSPORTATION OF LIQUIDS BY PIPELINE : PROCEDURES FOR OPERATION, MAINTENANCE AND

Author: United States. Department of Transportation **Corp. Author:**

Source: Federal Register. 10 Aug. 1978, vol. 43, no. 155, 35513-35517. .

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1978 **Language:** English

Category: Other **ID:** 653

Abstract: Failure analyses, accident investigations and recommendations of the NTSB indicate that, in many cases, pipeline carriers of hazardous liquids have not followed proper procedures for handling normal operations and maintenance, abnormal operations and emergencies. This notice proposes to establish the essentials that the procedures must cover as well as requirements for communications, training of personnel and educating the public about the hazards involved, with emphasis on highly volatile commodities like LPG (liquefied petroleum gas). (Fuel and Energy Abstracts. May 1979, vol.20, no. 3, 197.)

Title: TWENTY-SIX-INCH PIPE NDE INSTRUMENT SURVEILLANCE TEST.

Author: Bickford-RL; Clark-RA; Electric Power Research Institute Battelle Pacific Northwest Laboratories **Corp. Author:**

Source: Palo Alto, Electric Power Research Institute, 1983. (EPRI-NP-2869) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: Inspection methods **ID:** 654

Abstract:

Title: PREVENTION OF CATASTROPHIC FAILURE OF PRESSURE VESSELS AND PIPING. RESULTS OF PRESS

Author: Rintamaa-R; Torronen-K; and-others; Technical Research Centre of Finland **Corp. Author:**

Source: Espoo, 1988. (Technical Research Centre of Finland Research Report 515) 52pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Test/analysis **ID:** 655

Abstract:

Title: PIPEWORK FAILURES : A REVIEW OF HISTORICAL INCIDENTS.

Author: Blything-KW; Parry-ST **Corp. Author:** UKAEA

Source: SRD R441) 34pp.

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Experience/events **ID:** 656

Abstract: Historical incident data has been fathered from different sources and classified into the four plant categories - Chemical, refinery, nuclear and steam. However, the available world-wide data was found to be surprisingly limited and it should be regarded as indicative of typical problems rather than statistically significant. The incident data has been analysed to determine failure cause and the underlying reasons for failure defined as root causes. Data concerning leak severity has been gathered from some sources and this has been classified as leaks or ruptures with the number of incidents in each category. Brief descriptions are given for a selection of incidents to illustrate the types of failure and their consequences.

Title: PREVENTION OF CATASTROPHIC FAILURE IN PRESSURE VESSELS AND PIPINGS. FINAL REPORT OF

Author: Rintamaa-R; Wallin-K; and-others; Nordic Liaison Committee for Atomic Energy **Corp. Author:**

Source: Randers, Grafisk Center Kronjylland, 1989. 49pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Damage probability **ID:** 657

Abstract:

Title: Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants

Author: **Corp. Author:** U.S. Nuclear Regulatory Com

Source: NUREG-0531; 98 pages

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1979 **Language:** English

Category: IGSCC **ID:** 658

Abstract: This report covers the investigation of the possible IGSCC of large diameter piping. During 1978, IGSCC was reported for the first time in large diameter piping (> 20") in a BWR in Germany. This discovery, together with the reported questions concerning the interpretation of ultrasonic inspections, led to the activation of a new Pipe Crack Study Group (PCSG). The charter of the new PCSG was expanded to: (1) review the potential for IGSCC in PWRs and BWRs, (2) examine operating experiences in foreign reactors relevant to IGSCC, and (3) specifically address five PCSG charter questions. The specific areas considered by the PCSG and summarized in this report are PWR and BWR cracking experience, metallurgy associated with pipe cracking, reactor coolant chemistry, pipe configuration and stress levels, Duane Arnold safe-end cracking, methods of detecting significance of cracks, and recent developments relevant to controland detection of IGSCC. In the report, conclusions and recommendations by the PCSG are presented.

Title: STRESS-INTENSITY FACTORS FOR SURFACE CRACKS IN PIPES : A COMPUTER CODE FOR EVALUATI

Author: Science Applications, Inc.Electric Power Research Institute **Corp. Author:**

Source: Palo Alto, Electric Power Research Institute, 1982. (EPRI NP 2425)(Research project 1757-8) 72pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1982 **Language:** English

Category: Methods **ID:** 659

Abstract:

Title: A METHOD OF COUNTERACTING STRESS CORROSION CRACKING OF PIPING COMPONENTS BY ME

Author: Tanaka-Y; Umemoto-T **Corp. Author:**

Source: Ishikawajima-Harima Engineering Review. 1988, vol.28, no.3, 151-155.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** Japanese

Category: Methods **ID:** 660

Abstract:

Title: TOPICAL REPORT ON ENVIRONMENT SENSITIVE CRACKING (LOW PH STRESS CORROSION CRACKI

Author: Parkins-RN; American Gas Association Battelle **Corp. Author:**

Source: Columbus, Ohio, Battelle, 1990. (AGA L51623) (NG-18 report no.191) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Experience/events **ID:** 661

Abstract:

Title: TECHNICAL REPORT INVESTIGATION AND EVALUATION OF CRACKING IN AUSTENITIC STAINLES

Author: United States. Nuclear Regulatory Commission **Corp. Author:**

Source: 1975. (NUREG/75/067) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1975 **Language:** English

Category: Experience/events **ID:** 662

Abstract:

Title: FATIGUE STUDIES IN THE LEAK BEFORE BREAK ASSESSMENT OF PRESSURISED PIPES.

Author: Hellen-RAJ; Connors-DC; Berkeley Nuclear Laboratories **Corp. Author:**

Source: 1980. (RD/B/N47.39) 7pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1980 **Language:** English

Category: LBB justification **ID:** 663

Abstract:

Title: ESTIMATION OF CRACK EXTENSION IN A PIPING ELBOW USING FRACTURE MECHANICS TECHNIQ

Author: James-LA; American Society of Mechanical Engineers **Corp. Author:**

Source: 1974. (ASME paper no.74-PVP-14) 6pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1974 **Language:** English

Category: Research/theoretical **ID:** 664

Abstract:

Title: EVALUATION OF CRACKING IN FEEDWATER PIPING ADJACENT TO THE STEAM GENERATORS IN M

Author: Goldberg-A; Streit-RD; United States. Nuclear Regulatory Commission Lawrence Livermore Laboratory **Corp. Author:**

Source: Lawrence Livermore Laboratory, 1980. (NUREG/CR-1603) (UCRL-53000) 190pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1980 **Language:** English

Category: Experience/events **ID:** 665

Abstract:

Title: SAFETY CONCERNS ASSOCIATED WITH PIPE BREAKS IN THE BWR SCRAM SYSTEM.

Author: Rubin-SD; United States. Nuclear Regulatory Commission **Corp. Author:**

Source: 1981. (NUREG-0785 draft) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1981 **Language:** English

Category: Research/theoretical **ID:** 666

Abstract:

Title: Review of Erosion Corrosion in Single Phase Flows

Author: G. Cragnolino, C. Czajkowski and W.J. Stack **Corp. Author:** Argonne National Laboratory,

Source: ANL-88-25 (NUREG/CR-5156); 91 pages

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Erosion-corrosion **ID:** 667

Abstract: This report contains two literature reviews (prepared by Brookhaven National Laboratory and Argonne National Laboratory, respectively) on the available data and current mechanistic understanding of erosion-corrosion, and a failure analysis (prepared by Brookhaven National Laboratory) of a tee-elbow joint from the Surry-2 plant that failed in December 1986. It also includes suggestions for additional research that should be performed by the U.S.NRC to increase the capability to rank plants and/or location within plants in terms of susceptibility to erosion-corrosion and to ensure that proposed inspection and mitigation programs are soundly based.

Title: DESIGN BASIS FOR PROTECTION OF LIGHT WATER NUCLEAR POWER PLANTS AGAINST EFFECTS

Author: American Nuclear Society **Corp. Author:**

Source: La Grange Park, Illinois, 1980. (ANSI/ANS 58.2-1980) 89pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1980 **Language:** English

Category: Analysis of break effects **ID:** 668

Abstract:

Title: TROJAN NUCLEAR PLANT : ANALYSES OF PIPE SYSTEM BREAKS OUTSIDE THE CONTAINMENT.

Author: Bechtel Power Corp.Portland General Electric Co. **Corp. Author:**

Source: 3rd ed., Portland General Electric Co., 1975. (PGE-1004) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1975 **Language:** English

Category: Experience/events **ID:** 669

Abstract:

Title: DETERMINATION OF DESIGN PIPE BREAKS FOR THE WESTINGHOUSE REACTOR COOLANT SYSTE

Author: Szy-Slow-Ski-JJ; Salvatori-R; Westinghouse Electric Corp. **Corp. Author:**

Source: rev.ed., Pittsburgh, 1972. (WCAP-7503) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1972 **Language:** English

Category: Methods **ID:** 670

Abstract:

Title: SIZEWELL B PWR PRE-CONSTRUCTION SAFETY REPORT REFERENCE REPORT: EVENT TREE ANAL

Author: National Nuclear Corp. Ltd. **Corp. Author:**

Source: Whetstone, 1982. (PWR/RX 540) 123pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1982 **Language:** English

Category: Failure probability **ID:** 671

Abstract:

Title: SIZEWELL B PWR PRE-CONSTRUCTION SAFETY REPORT REFERENCE REPORT: POSTULATED PIPE

Author: Austin-RW; National Nuclear Corp. Ltd. **Corp. Author:**

Source: Whetstone, 1982. (PWR/RX 286) 15pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1982 **Language:** English

Category: Damage probability **ID:** 672

Abstract:

Title: PIPE BREAKS FOR ANALYSIS OF THE LOCA ANALYSIS OF THE WESTINGHOUSE PRIMARY LOOP.

Author: Westinghouse Electric Corp. **Corp. Author:**

Source: Pittsburgh, 1975. (WCAP-8172-A) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1975 **Language:** English

Category: Damage probability **ID:** 673

Abstract:

Title: STATIC STRESS INTENSITY FACTORS AND DYNAMIC CRACK PROPAGATION IN PIPES. ANNUAL RE

Author: Emery-AF; Kobayashi-AS; and-others; Washington University Electric Power Research Institute **Corp. Author:**

Source: Palo Alto, Electric Power Research Institute, 1981. (EPRI NP-2024) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1981 **Language:** English

Category: Methods **ID:** 674

Abstract:

Title: THE GROWTH AND STABILITY OF STRESS CORROSION CRACKS IN LARGE-DIAMETER BWR PIPING.

Author: Electric Power Research InstituteGeneral Electric Co. **Corp. Author:**

Source: Palo Alto, Electric Power Research Institute, 1982. (EPRI NP2472 SY) 2 vols.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1982 **Language:** English

Category: Methods **ID:** 675

Abstract:

Title: A PWR SECONDARY SYSTEM BEHAVIOUR FOR POSTULATED PIPE RUPTURE.

Author: Chu-AW; Ramchandani-M; and-others; American Society of Mechanical Engineers Burns and Roe Inc. **Corp. Author:**

Source: New York, American Society of Mechanical Engineers, 1980. (ASME paper 80-WA/NE-1) 11pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1980 **Language:** English

Category: Methods **ID:** 676

Abstract:

Title: CRACK GROWTH EVALUATION FOR SMALL CRACKS IN REACTOR COOLANT PIPING.

Author: Simonen-FA; Mayfield-ME; Battelle Pacific Northwest Laboratories United States. Nuclear Regulatory Commission **Corp. Author:**

Source: Nuclear Regulatory Commission, 1983. (NUREG/CR-3176) (PNL-4642) 61pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: Research/theoretical **ID:** 677

Abstract:

Title: PARAMETRIC CALCULATIONS OF FATIGUE CRACK GROWTH IN PIPING.

Author: Simonen-FA; Goodrich-CW; United States. Nuclear Regulatory Commission Battelle Pacific Northwest Laboratories **Corp. Author:**

Source: Nuclear Regulatory Commission, 1983. (NUREG/CR-3059) 33pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: Methods **ID:** 678

Abstract:

Title: CRACKS AND LEAKS IN SMALL DIAMETER PIPING.

Author: United States. Nuclear Regulatory Commission **Corp. Author:**

Source: 1983. (Engineering evaluation report : report no.: AEOD/E308) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: Experience/events **ID:** 679

Abstract:

Title: THE APPLICATION OF FRACTURE PROOF DESIGN METHODS USING TEARING INSTABILITY THEOR

Author: Paris-PC; Tada-H; Del Research Corp.United States. Nuclear Regulatory Commission **Corp. Author:**

Source: Nuclear Regulatory Commission, 1983. (NUREG/CR-3464) 190pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: Methods **ID:** 680

Abstract:

Title: FINITE ELEMENT MOMENT-ROTATION TEARING CURVE GENERATION AND USE OF SIMPLIFIED EL

Author: Macek-RW; Sadik-S; E G and G Idaho, Inc.United States. Nuclear Regulatory Commission **Corp. Author:**

Source: Nuclear Regulatory Commission, 1983. (NUREG/CR-3465) (EGG-EA-6244) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: Methods **ID:** 681

Abstract:

Title: A STUDY OF THE REGULATORY POSITION ON POSTULATED PIPE RUPTURE LOCATION CRITERIA :

Author: Woo-HH; Laurence Livermore National Laboratory United States. Nuclear Regulatory Commission **Corp. Author:**

Source: Lawrence Livermore National Laboratory, 1984. (NUREG/CR 3483) (UCRL 53490) 35pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: Damage probability **ID:** 682

Abstract:

Title: RELIABILITY OF REACTOR PRESSURE COMPONENTS : PROCEEDINGS OF AN INTERNATIONAL SYM

Author: International Atomic Energy Agency **Corp. Author:**

Source: Vienna, 1983. (STI/PUB/645) 415pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: Other **ID:** 683

Abstract:

Title: GERMAN STANDARD PROBLEM NO.3 (SECONDARY CONTAINMENT STANDARD PROBLEM) : "WAT

Author: Nguyen-DL; Winkler-W. **Corp. Author:**

Source: 1983. (Schriftenreihe, Reaktorsicherheit und Strahlenschutz) (BMI-1983-019) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1983 **Language:** German

Category: Test/analysis **ID:** 684

Abstract:

Title: Proceedings of the CSNI Specialist Meeting on Leak-Before-Break in Nuclear Reactor Piping

Author: **Corp. Author:** U.S. NRC

Source: 1984. (NUREG/CP-0051) (CSNI report no.82) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: LBB justification **ID:** 685

Abstract:

Title: Report of the U.S. NRC Piping Review Committee: Investigation and Evaluation of Stress Corrosion Cracking in Pipi

Author: **Corp. Author:** U.S. NRC

Source: 1984. (NUREG-1061, vol.1) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: Experience/events **ID:** 686

Abstract:

Title: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERIN

Author: Lo-TY; Mensing-RW; and-others; Lawrence Livermore National Laboratory United States. Nuclear Regulatory Commission **Corp. Author:**

Source: Nuclear Regulatory Commission, 1984. (NUREG/CR-3663) (UCRL-53500 vol.2) 81pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: Damage probability **ID:** 687

Abstract:

Title: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLAN

Author: Woo-HH; Mensing-RW; and-others; United States. Nuclear Regulatory Commission Lawrence Livermore National Laboratory **Corp. Author:**

Source: Washington, Nuclear Regulatory Commission, 1984. (NUREG/CR-3660, vol.2) (UCID-19988, vol.2) 62pp.

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: Damage probability **ID:** 688

Abstract:

Title: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERIN

Author: Ravindra-MK; Campbell-RD; and-others; Lawrence Livermore National Laboratory United States. Nuclear Regulatory Commission **Corp. Author:**

Source: Lawrence Livermore National Laboratory, 1984. (NUREG/CR-3663) (UCRL-53500, vol.3) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: Failure probability **ID:** 689

Abstract:

Title: FATIGUE CRACK GROWTH RATES OF LOW-CARBON AND STAINLESS PIPING STEELS IN PWR ENVI

Author: Cullen-WH; Materials Engineering Associates, Inc.United States. Nuclear Regulatory Commission **Corp. Author:**

Source: Nuclear Regulatory Commission, 1985. (NUREG/CR-3945) (MEA-2055) 56pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: Methods **ID:** 690

Abstract:

Title: AN EVALUATION OF STRESS CORROSION CRACK GROWTH IN BWR PIPING SYSTEMS.

Author: Kassir-M; Sharma-S; and-others; Brookhaven National Laboratory United States. Nuclear Regulatory Commission **Corp. Author:**

Source: Nuclear Regulatory Commission, 1985. (NUREG/CR-4221) (BNL-NUREG-51874) 66pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: Methods **ID:** 691

Abstract:

Title: REPORT TO THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE : SUMMA

Author: United States. Nuclear Regulatory Commission **Corp. Author:**

Source: 1985. (NUREG-1061 volume 2, addendum) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: Experience/events **ID:** 692

Abstract:

Title: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLAN

Author: Chou, C.K., Holman, G. S. **Corp. Author:** Lawrence Livermore Natl. Lab.

Source: UCID-19988 (NUREG/CR-3660-VI)

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: Failure probability **ID:** 693

Abstract:

Title: A REVIEW OF THE MODELS AND MECHANISMS FOR ENVIRONMENTALLY-ASSISTED CRACK GROW

Author: Cullen-W; Gabetta-G; and-others; Materials Engineering Associates, Inc.United States. Nuclear Regulatory Commission **Corp. Author:**

Source: Nuclear Regulatory Commission, 1985. (NUREG/CR-4422) (MEA-2078) 106pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: Methods/comparison **ID:** 694

Abstract:

Title: DESIGN SAFETY ARRANGEMENTS FOR THE RBMK-1000 REACTOR.

Author: Cherkashov-YM; International Atomic Energy Agency **Corp. Author:**

Source: (IAEA-SM-268/84) 8pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1984 **Language:** Russian

Category: Other **ID:** 695

Abstract: The paper discusses some technical and economic factors involved in the operation of nuclear power plants with RBMK-1000 reactors. It indicates the local factors and the operational availability factors that have been obtained for the equipment. It analyses the variations in the main parameters of nuclear power plants during transient processes associated with the switching off of equipment and with ruptures in pipelines. It presents a scheme for emergency shut-down cooling of the reactor core and algorithms for the triggering of this system.

Title: THE RUSSIAN APPROACH TO NUCLEAR REACTOR SAFETY.

Author: Lewin-J **Corp. Author:**

Source: Nuclear Safety, Vol. 18:438-450.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1977 **Language:** English

Category: Other **ID:** 696

Abstract: Soviet reactor design initially proceeded from a safety philosophy that did not acknowledge a loss-of-coolant accident caused by a double-ended pipe break nor a massive core meltdown as credible eventualities to be considered in the design of systems and details. Generally, engineered safeguards and conservatism in design have been regarded as adequate insurance against accidents that could escalate to a point where there is significant radiation damage to either plant personnel or the public.

Title: NRC LEAK-BEFORE-BREAK (LBB.NRC) ANALYSIS METHOD FOR CIRCUMFERENTIALLY THROUGH-

Author: Klecker-R; Brust-F; and-others; Battelle. Columbus Division United States. Nuclear Regulatory Commission **Corp. Author:**

Source: Washington, D.C., Nuclear Regulatory Commission, 1986. (NUREG/CR-4572) (BMI-2134) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: LBB justification **ID:** 697

Abstract:

Title: Probability of Pipe Failure in the Reactor Coolant Loops of Babcock & Wilcox PWR Plants. Vol. 1: Summary Report

Author: Holman-GS; Chou-CK; Lawrence Livermore National Laboratory United States. Nuclear Regulatory Commission **Corp. Author:**

Source: Washington D.C., Nuclear Regulatory Commission, 1986. (NUREG/CR-4290, vol.1) (UCRL-53644, vol.1) 56pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Damage probability **ID:** 698

Abstract:

Title: REACTOR COOLANT AND ASSOCIATED SYSTEMS IN NUCLEAR POWER PLANTS : A SAFETY GUIDE.

Author: International Atomic Energy Agency **Corp. Author:**

Source: Vienna, 1986. (Safety series no.50-SG-D13)(STI/PUB/731) 70pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Other **ID:** 699

Abstract:

Title: AN ASSESSMENT OF CIRCUMFERENTIALLY COMPLEX-CRACKED PIPE SUBJECTED TO BENDING.

Author: Kramer-G; Papaspvropoulos-V; Battelle. Columbus Division United States. Nuclear Regulatory Commission **Corp. Author:**

Source: USGPO, 1986. (NUREG/CR-4687) (BMI-2142) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Test/analysis **ID:** 700

Abstract:

Title: AN EXPERIMENTAL AND ANALYTICAL ASSESSMENT OF CIRCUMFERENTIAL THROUGH-WALL CR

Author: Scott-P; Brust-F; Battelle. Columbus Division United States. Nuclear Regulatory Commission **Corp. Author:**

Source: USGPO, 1986. (NUREG/CR-4574) (BMI-2136) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Test/analysis **ID:** 701

Abstract:

Title: SURVEY OF PIPING FAILURES FOR THE REACTOR PRIMARY COOLANT PIPE RUPTURE STUDY.

Author: Gibbons-WS and Hackney-BD **Corp. Author:** General Electric

Source: GEAP-4574

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1964 **Language:** English

Category: Operating experience **ID:** 702

Abstract: An industrial piping failure survey covering 701 contacts in electric utilities, petroleum refineries, chemical processing, marine applications, architect-engineers, component manufacturers, piping fabricators and erectors, insurance companies, and others was conducted as part of the Reactor Primary Coolant Rupture Study. The total of 315 replies received, when combined with published failure cases, provided 399 failure case histories of interest to the study. The term "failure" was defined as any defective condition encountered during start-up, testing, and/or service that required repair or replacement corrective action.

Title: INSTABILITY PREDICTIONS FOR CIRCUMFERENTIALLY CRACKED TYPE 304 STAINLESS STEEL PIP

Author: Zahoor-A; Wilkowski-G; and-others; Battelle Columbus Laboratories **Corp. Author:**

Source: Palo Alto, Calif., Electric Power Research Institute, 1982. (EPRI NP-2347) 2 vols.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1982 **Language:** Engl;ish

Category: Methods **ID:** 703

Abstract:

Title: FRACTURE MECHANICS ANALYSES ON DAMAGED PIPELINES : GRS PARTICIPATION IN USNRC DEG

Author: Azodi-D; Sievers-J; Germany (Federal Republic). Bundesministerium fuer Umwelt, Naturschutz und Reaktorsicherheit **Corp. Author:**

Source: Bonn, 1986. (Schriftenreihe, Reaktorsicherhiet und Strahlenschutz) (BMU-1986-123) 94pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** German

Category: Methods **ID:** 704

Abstract:

Title: FATIGUE CRACK GROWTH RATES IN PRESSURE VESSEL AND PIPING STEELS IN LWR ENVIRONME

Author: Cullen-WH; Materials Engineering Associates, Inc. United States. Nuclear Regulatory Commission **Corp. Author:**

Source: USGPO, 1987. (NUREG/CR-4724) (MEA-2175) 54pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Methods **ID:** 705

Abstract:

Title: APPLICATION OF THE LEAK-BEFORE-BREAK APPROACH TO WESTINGHOUSE PWR PIPING.

Author: Westinghouse Electric Corp.Electric Power Research Institute **Corp. Author:**

Source: Palo Alto, Calif., Electric Power Research Institute, 1986. (EPRI NP-4971) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: LBB justification **ID:** 706

Abstract:

Title: PROBABILITY OF FAILURE IN BWR REACTOR COOLANT PIPING : GUILLOTINE BREAK INDIRECTLY

Author: Hardy-GS; Campbell-RD; and-others; NTS/Structural Mechanics Associates United States. Nuclear Regulatory Commission **Corp. Author:**

Source: USGPO, 1986. (NUREG/CR-4792) (UCID-20914 vol.4) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Damage probability **ID:** 707

Abstract:

Title: LEAK DETECTION IN NUCLEAR PIPING OUTSIDE CONTAINMENT.

Author: Bausch-HP; Wyle Laboratories Nuclear Safety Analysis Center **Corp. Author:**

Source: Palo Alto, Calif., Nuclear Safety Analysis Center, 1987. (NSAC 110) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Methods **ID:** 708

Abstract:

Title: EXPERIMENTAL AND ANALYTICAL ASSESSMENT OF CIRCUMFERENTIALLY SURFACE-CRACKED PI

Author: Scott-PM; Ahmad-J; Battelle. Columbus Division United States. Nuclear Regulatory Commission **Corp. Author:**

Source: USGPO, 1987. (NUREG/CR-4872) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Test/analysis **ID:** 709

Abstract:

Title: Pipe Break Frequency Estimation for Nuclear Power Plants.

Author: R.E. Wright, J.A. Steverson & W.F. Zuroff **Corp. Author:** EG&G Idaho, Inc., Idaho Falls

Source: EGG-2421 (NUREG/CR-4407)

SKI Project File: Ja **Transfer:** Ja **Publ year:** 1987 **Language:** English

Category: Pipe break frequency **ID:** 710

Abstract: This study empirically develops frequencies of safety-significant pipe failures in commercial nuclear power plants (NPPs). Its primary purpose is to update the pipe break frequencies reported in the Reactor Safety Study, WASH-1400, which are used in many risk analyses. The study involved reviewing various data sources for actual piping failure events of significant magnitude. When extant in the documentation reviewed, information was extracted concerning conditional factors such as the system in which the failure occurred, operational mode of the plant, and size of the pipe involved to estimate conditional pipe break frequencies useful to risk analysts. Because of the high quality piping used in NPPs, there have been few significant pipe failures. An attempt was made to augment the analysis with synthetic data from a deplhi approach, but the wide uncertainty bounds on the resulting estimates rendered the results unsuitable for combining data.

Title: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERIN

Author: Holman-GS; Lo-T; Lawrence Livermore National Laboratory United States. Nuclear Regulatory Commission **Corp. Author:**

Source: USGPO, 1985. (NUREG/CR-3663) (UCRL-53500) 64pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: Damage probability **ID:** 711

Abstract:

Title: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOP OF WESTINGHOUSE PWR PLANT

Author: Ravindra-MK; Campbell-RD; and-others; United States. Nuclear Regulatory Commission Lawrence Livermore National Laboratory **Corp. Author:**

Source: Washington, Nuclear Regulatory Commission, 1984. (NUREG/CR 3660) (UCID-19988, vol.3) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: Damage probability **ID:** 712

Abstract:

Title: APPLYING LEAK-BEFORE-BREAK TO HIGH-ENERGY PIPING.

Author: Cloud (R.L.) Associates, Inc. Nuclear Safety Analysis Center **Corp. Author:**

Source: Palo Alto, Nuclear Safety Analysis Center, 1987. (NSAC-114) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: LBB justification **ID:** 713

Abstract:

Title: Fracture Mechanical Analysis of Piping Damaged During Operations.

Author: Azodi-D **Corp. Author:**

Source: Bonn, 1987. (Schriftenreihe, Reaktorsicherheit und Strahlenschutz) (BMU-1987-167) 34pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** German

Category: Methods **ID:** 714

Abstract: Contribution of GRS (SR 271/3) to USNRC's "Degraded Piping Program" : Numerical Simulation of Crack Arrest Tests to Verify the Dynamic J-Integrals.

Title: HIGH-LEVEL SEISMIC RESPONSE AND FAILURE PREDICTION METHODS FOR PIPING.

Author: Severud-LK; Anderson-MJ; and-others; **Corp. Author:**

Source: USGPO, 1988. (NUREG/CR - 5023) (WHC-EP-0081) 197pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Damage probability **ID:** 715

Abstract:

Title: Lead Plant Application of Leak-Before-Break to High Energy Piping.

Author: Server-WL; Beaudoin-BF; and-others; Nuclear Safety Analysis Center Cloud (R.L.) Associates, Inc. **Corp. Author:**

Source: Palo Alto, Calif., Nuclear Safety Analysis Center, 1989. (NSAC-141) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: LBB justification **ID:** 716

Abstract:

Title: FATIGUE CRACK GROWTH OF PART-THROUGH CRACKS IN PRESSURE VESSEL AND PIPING STEELS

Author: Cullen-WH; Jolles-MR; Materials Engineering Associates, Inc.United States. Nuclear Regulatory Commission **Corp. Author:**

Source: USGPO, 1988. (NUREG/CR-4828) (MEA-2198) 41pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Test/analysis **ID:** 717

Abstract:

Title: AMERICAN NATIONAL STANDARD. DESIGN BASIS FOR PROTECTION OF LIGHT WATER NUCLEAR P

Author: American Nuclear Society **Corp. Author:**

Source: La Grange Park, Ill., 1988. (ANSI/ANS-58.2-1988) 71pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Other **ID:** 718

Abstract:

Title: PROBABILITY OF FAILURE IN BWR REACTOR COOLANT PIPING : VOL.1, SUMMARY REPORT.

Author: Holman-GS; Chou-CK; Lawrence Livermore National Laboratory United States. Nuclear Regulatory Commission **Corp. Author:**

Source: USGPO, 1989. (NUREG/CR-4792) (UCD-20914 VOL.1) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Damage probability **ID:** 719

Abstract:

Title: PROBABILITY OF FAILURE IN BWR REACTOR COOLANT PIPING: VOL.2, PIPE FAILURE INDUCED BY

Author: Lo-T; Bumpus-SE; and-others; United States. Nuclear Regulatory Commission **Corp. Author:**

Source: USGPO, 1989, (NUREG/CR-4792) (UCID-20914 vol.2) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Damage probability **ID:** 720

Abstract:

Title: TECHNICAL MEMORANDUM FOR IN-SERVICE INSPECTION OF CLASS 2 PIPING DESIGNATED 'NO-BR

Author: Central Electricity Generating Board. Sizewell B Project Management Board **Corp. Author:**

Source: Knutsford, 1989. (SXB-IM-090520) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Inspection methods **ID:** 721

Abstract:

Title: Inspection and Assessment of Documents With Regard to Safety Related Problems and Their Consideration in the Fur

Author: Herter-KH; Germany (Federal Republic). Bundesministerium fuer Umwelt, Naturschutz und Reaktorsicherheit **Corp. Author:**

Source: Bonn, 1989. (BMU-1989-215) (Schriftenreihe, Reaktorsicherheit und Strahlenschutz) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** German

Category: Other **ID:** 722

Abstract:

Title: APPROXIMATE METHODS FOR FRACTURE ANALYSES OF THROUGH-WALL CRACKED PIPES.

Author: Brust-FW; Battelles's Columbus Division United States. Nuclear Regulatory Commission **Corp. Author:**

Source: Washington D.C., USGPO, 1987. (NUREG/CR-4853) (BMI-2145) (RF,R5) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Methods **ID:** 723

Abstract:

Title: Comments on the Leak-Before-Break Concept for Nuclear Power Plant Piping Systems.

Author: E.C. Rodabaugh **Corp. Author:** ORNL

Source: ORNL/Sub/82-22252/3 (NUREG/CR-4305); 58 pages

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: LBB **ID:** 724

Abstract: Leak-before-break (LBB) entails the concept that, with a high degree of probability, failure of the pressure boundary of piping systems will be signaled by a detectable leak which will provide ample time to shut down and repair that leak. The status of the LBB concept is dicussed in this report, including a review of industrial and nuclear power plant experience with respect to LBB, fracture mechanics and potential elimination of postulated pipe breaks in NPP piping design

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Title: MECHANICAL FRACTURE PREDICTIONS FOR SENSITISED STAINLESS STEEL PIPING WITH CIRCUM

Author: Kanninen-MF; Broek-D; and-others **Corp. Author:**

Source: Palo Alto, Calif., Electric Power Research Institute, 1976. (EPRI NP-192) (Research project 585-1) (Final report) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1976 **Language:** English

Category: Damage probability **ID:** 725

Abstract:

Title: EVALUATION LEAK AND FAILURE PROBABILITY OF NUCLEAR PLANT EQUIPMENT ELEMENT AND

Author: Tkachev-VV; Vasilyev-VG; and-others; Scientific and Engineering Centre for Safety in Industry and Nuclear Power Kurchatov Institute of Atomic Energy **Corp. Author:**

Source: Moscow, n.d. 16pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** **Language:** Russian

Category: Damage probability **ID:** 726

Abstract:

Title: FACTORS AFFECTING THE SENSITIVITY TO CRACKING OF WELDS IN LARGE DIAMETER PIPES.

Author: Rudolph-W **Corp. Author:**

Source: 3R International. Nov./Dec.1977, vol.16, no.11/12, 656-659.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1977 **Language:** English

Category: Research/theoretical **ID:** 727

Abstract:

Title: DEFECTS AND FAILURES IN PRESSURE VESSELS AND PIPING.

Author: Thielsch-H **Corp. Author:**

Source: rev. ed. New York, R.E. Krieger, 1977. 443 pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1977 **Language:** English

Category: Experience/events **ID:** 728

Abstract:

Title: MID AMERICA PIPELINE SYSTEM ANHYDROUS AMMONIA LEAK, CONWAY, KANSAS, DECEMBER 6,

Author: National Transportation Safety Board **Corp. Author:**

Source: NTIS, 1974. (PB 238 158) (NTSB-PAR-74-6) 29 pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1974 **Language:** English

Category: Experience/events **ID:** 729

Abstract:

Title: PRESSURE AND LEAK TESTING OF PRESSURE VESSELS AND PIPELINES.

Author: Sweden. Arbetarskyddsstyrelsen **Corp. Author:**

Source: 1978. (Meddelanden 1978 : 21) 6 pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1978 **Language:** German

Category: inspection methods **ID:** 730

Abstract:

Title: PIPELINE ACCIDENT REPORT: MID-AMERICA PIPELINE SYSTEM LIQUEFIED PETROLEUM GAS PIPEL

Author: National Transportation Safety Board **Corp. Author:**

Source: Washington, 1979. (NTSB-PAR-79-1) 41 pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1979 **Language:** English

Category: Experience/events **ID:** 731

Abstract:

Title: Investigation of Cause of Cracking in Austenitic Stainless Steel Piping.

Author: Klepfer-HH; General Electric Co. **Corp. Author:** General Electric

Source: San Jose, California, 1975. (NEDO-21000) 2 vols. various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1975 **Language:** English

Category: Research/theoretical **ID:** 732

Abstract:

Title: FINAL REPORT ON METALLURGICAL INVESTIGATION OF FAILURE IN 8-5/8 x 219 INCH API SRD 5LX,

Author: Lare-PJ; Hermann-RA; and-others; National Transportation Safety Board Artech Corp. **Corp. Author:**

Source: Virginia, Artech Corp., 1975. 29pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1975 **Language:** English

Category: Experience/events **ID:** 733

Abstract:

Title: HOW FAR WILL HEAT FLOW DOWN A DEAD PIPELINE?

Author: Anonymous **Corp. Author:**

Source: Loss Prevention Bulletin. 1983, no.49, 28.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: Experience/events **ID:** 734

Abstract: A brief comment on a carbon steel pipe rupture.

Title: THE PREVENTION OF MAJOR LEAKS - BETTER INSPECTION AFTER CONSTRUCTION?

Author: Kletz-TA **Corp. Author:**

Source: Plant/Operations Progress. Jan.1984, vol.3, no.1, 19-24.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: Experience/events **ID:** 735

Abstract: Review of the reasons why leaks occur in the oil and chemical industries and how to prevent pipe failure with summaries of some major pipe failures and the most effective means of prevention. (13 refs.)

Title: WHY PIPES FAIL III.

Author: Needham-D; Howe-M; British Gas. Engineering Research Station **Corp. Author:**

Source: Newcastle-upon-Tyne, 1984. (ERS E.419) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: Experience/events **ID:** 736

Abstract:

Title: WHY PIPES FAIL - AGAIN!?

Author: Needham-D; Howe-M; British Gas Corp. Engineering Research Station **Corp. Author:**

Source: Killingworth, 1982. (ERS E.315) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1982 **Language:** English

Category: Experience/events **ID:** 737

Abstract:

Title: WHY PIPES FAIL.

Author: Needham-D; Howe-M; British Gas Corp. Engineering Research Station **Corp. Author:**

Source: 1979. (ERS E 244) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1979 **Language:** English

Category: Experience/events **ID:** 738

Abstract:

Title: PIPELINE ACCIDENT REPORT : CONTINENTAL PIPE LINE COMPANY PIPELINE RUPTURE AND FIRE,

Author: National Transportation Safety Board **Corp. Author:**

Source: Washington, 1986. (PB86-916501) (NTSB/PAR-86/01) 26pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Experience/events **ID:** 739

Abstract:

Title: PIPELINE ACCIDENT REPORT : TEXAS EASTERN GAS PIPELINE COMPANY RUPTURES AND FIRES A

Author: National Transportation Safety Board **Corp. Author:**

Source: Washington, 1987. (NTSB/PAR-87/01) (PB87-916501) 57pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Experience/events **ID:** 740

Abstract:

Title: FINAL STAFF REPORT ON INVESTIGATION OF TENNESSEE GAS TRANSMISSION COMPANY PIPELIN

Author: United States. Bureau of Natural Gas **Corp. Author:**

Source: Washington, 1965. (Docket no.CP65-267) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1965 **Language:** English

Category: Experience/events **ID:** 741

Abstract:

Title: INVESTIGATION REPORT OF THE HOT OIL PIPELINE FAILURE AT BROMBOROUGH ON SATURDAY 1

Author: Southgate-DA; Department of Energy **Corp. Author:**

Source: 1990. HMSO, various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Experience/events **ID:** 742

Abstract:

Title: BOILER REHEAT LINE EXPLOSION. (S. WEST UNITED STATES POWER STATION).

Author: Anonymous. **Corp. Author:**

Source: National Board of Boiler and Pressure Vessel Inspectors Bulletin. Oct. 1985, vol.43, no.2, 6-7.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: Experience/events **ID:** 743

Abstract: Report on the hot reheat pipe that ruptured at a large power station in the southwestern United States, where six people were killed in the sudden blast, and twelve seriously injured. The comments of the National Board of Boiler and Pressure Vessel Inspectors who were involved in the investigation, are described.

Title: FLASHING-LIQUID FLOW CALCULATIONS FOR USE IN RISK ASSESSMENT.

Author: Carter-DA **Corp. Author:**

Source: Loss Prevention Bulletin. 1986, no.70, 31-35.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Other **ID:** 744

Abstract: COPTERA (Calculation of pipework two-phase emission rates) is the HSE's risk assessment programme for predicting the effects of a major pipework failure. A look is taken at the calculation method. The author is a member of staff of the Health and Safety Executive.

Title: RADIOLOGICAL ISSUES OF PRIMARY COOLANT PIPE REPLACEMENT AT FIVE BWR NUCLEAR PLAN

Author: Parkhurst-MA; Harty-R; and-others **Corp. Author:**

Source: Health Physics. June.1986, vol.50, supplement 1, S71.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Other **ID:** 745

Abstract: Addresses the radiological issues of pipe replacements after the discovery of intergranular stress corrosion cracking in the primary coolant system of five boiling water reactors. Measures taken to reduce doses to the inspection and repair personnel, actual doses incurred, and lessons learned are analysed.

Title: INVESTIGATION OF A FAILURE PROBLEM IN COLD-BENT BOILER RISER AND SUPPLY PIPES.

Author: Shibli-IA **Corp. Author:**

Source: International Journal of Pressure Vessels and Piping. 1986, vol.24, no.4, 303-336.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Experience/events **ID:** 746

Abstract: Discusses the small number of failures reported at the extrados of cold-bent carbon manganese steel riser and supply pipes operating at elevated temperatures in power plant boilers world wide. These failures occurred at a very low frequency and are generally associated with minor surface defects such as laminations or hammer marks. Detailed study shows that the failure mechanism is creep crack growth and suggests that this type of failure can be eliminated by controlling the pipe chemical composition and hardness level together with adequate control of pipe bending, installation and in-service inspection practices. 15 refs

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Title: ON THERMOFRACTURE BEHAVIOUR OF LEAKING THIN-WALL PIPES.

Author: Hsu-TR; Chen-GG; and-others **Corp. Author:**

Source: International Journal of Pressure Vessels and Piping. 1986, vol.24, no.4, 269-281.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Methods **ID:** 747

Abstract: Discusses the dangers of pipeline failures due to runaway cracks, and assesses the fracture behaviour of the pipeline due to the local cooling effect on the crack surface from the Joule-Thomson expansion effect or a throttling process. Describes briefly the throttling process due to leakage of the pressurized medium. The analytical procedure described can be used to predict the critical loading conditions for runaway cracks in leaking pipelines. 20 refs

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Title: ANOTHER PIPE FAILURE.

Author: Kletz-T **Corp. Author:**

Source: Chemical Engineer. Jan.1987, no.432, 32.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Experience/events **ID:** 748

Abstract: Looks at the causes and consequences of pipe failure, and possible preventive measures. Gives details of the report on the flood at the Victoria and Albert Museum in March 1986, which was caused by pipe failure.

Title: LIST OF INCIDENTS 1985.

Author: Institution of Chemical Engineers **Corp. Author:**

Source: Loss Prevention Bulletin. 1986, no.72, supplement.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Experience/events **ID:** 749

Abstract: Presents a chronological table of over 200 accidents and incidents worldwide during 1985 under the headings of date, location, company involved, details of incident, death and injury figures, cost of damage, chemical involved and equipment involved. The incidents include fires, explosions, spills and leaks, and cover pipelines, tankers, road, rail and sea transport, refineries, hoses, valves.

Title: BRITISH NUCLEAR INDUSTRY IN TROUBLE AGAIN.

Author: Johnstone-B **Corp. Author:**

Source: Nature. 5/11 Feb.1987, vol.325, no.6104, 471.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Experience/events **ID:** 750

Abstract: Suggests that disclosures of a design fault in the advanced gas-cooled reactor (AGR) and of a leak from a pipe at the Sellafield reprocessing plant have further undermined the credibility of the report by Sir Frank Layfield on the building of the Sizewell B PWR.

Title: IS PIPEWORK GIVEN THE ATTENTION IT DESERVES.

Author: Towndrow-RF **Corp. Author:**

Source: Loss Prevention Bulletin. Feb.1987, no.73, 13-18.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Experience/events **ID:** 751

Abstract: Some extracts and main conclusions from the report by the Major Hazards Assessment Panel Working Party on Pipework Failures are summarised here although the report has yet to be published.

Title: SCOTTISH TESTS FOR PRESSURISED WATER REACTORS.

Author: Milne-R **Corp. Author:**

Source: New Scientist. 29 Oct.1987, vol.116, no.1584, 41.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Test/analysis **ID:** 752

Abstract: First experiments on the world's only full scale test rig designed to stimulate loss-of-coolant accidents in a pressurised-water reactor (PWR) have just begun in northern Scotland. The aim of the tests is to see how well the emergency core cooling systems perform if the primary cooling circuit of a PWR springs a leak because of a ruptured pipe or a faulty valve. The rig is at the Royal Navy Vulcan Research Establishment.

Title: SPILLS FROM LARGE CRUDE-OIL-CARRYING TRANSMISSION PIPELINES : AN ANALYSIS BY CAUSE,

Author: Hall-SM **Corp. Author:**

Source: Pipes and Pipelines International. Jul./Aug.1988, vol.33, no.4, 15-20.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Experience/events **ID:** 753

Abstract: Presents for one particular type of pipeline a probabilistic assessment of the likelihood of unwanted failures, and the likely spill volumes associated with each of the main failure causes. The failure frequencies shown in this analysis are low when compared with other forms of transportation, with the mean volume of oil spilt in these failure incidents being some 100 cubic metres. The environmental, humanitarian, and economic risks presented by any pipeline system will depend on particular circumstances, for example the pipeline routeing, and must be compared against those risks associated with any alternative. The analysis presented here provides the type of information which can assist this comparison. The analysis also highlights those areas where maximum benefit could be achieved if it was thought necessary to expend resources on any further risk-reduction measures.

Title: VALIDITY OF WATER LEAK RATE PREDICTION METHODS.

Author: Friedel-L; Westphal-F; and-others **Corp. Author:**

Source: Journal of Loss Prevention in the Process Industries. Oct.1988, vol.1, no.4, 213-220.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Test/analysis **ID:** 754

Abstract: Models for the calculation of leakage rates through cracks in vessel walls and pipes due to overpressure are analysed. For evaluation of predictive quality, the models were applied to all the accessible (water) leakage rate data for real and model cracks. The statistical description of the differences between model predictions and experimental results reveals that none of the models can be considered universally valid, even for experiments with water, when specific data from different sources are compared. 13 refs.

Title: THE EFFECT OF THE VARIOUS SYSTEM STRESSES ON THE INTEGRITY OF A CRACKED PIPING SYST

Author: Smith-E **Corp. Author:**

Source: International Journal of Pressure Vessels and Piping. 1989, vol.37, no.5, 321-329.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Damage probability **ID:** 755

Abstract: Describes the analysis of a simple model which allows for the effect of the various stresses: weight, thermal, pressure and earthquake to be incorporated within a stability analysis for a cracked piping system.

Title: PIPELINE LEAK DETECTION.

Author: Ellul-I **Corp. Author:**

Source: Chemical Engineer. Jun.1989, no.461, 39-44.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Inspection methods **ID:** 756

Abstract: Looks at how computers can help in detecting leaks from pipelines. Applications include acoustic monitoring, line volume balance, pipeline modelling and the deviation method.

Title: LEAKAGE THROUGH AN IRREGULAR CRACK IN A PRESSURISED COMPONENT.

Author: Smith-E **Corp. Author:**

Source: International Journal of Pressure Vessels and Piping. 1989, vol.38, no.5, 333-339.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Research/theoretical **ID:** 757

Abstract: When a stress corrosion crack or fatigue crack propagates across the wall thickness of a pressurised component (tubing, pipe or vessel) it is possible, because of residual stresses or textural effects, that the crack growth rate is greater in the axial or circumferential directions than in the radial (through the thickness) direction. As a consequence of this so-called tunnelling effect, when the crack reaches the outer surface of the component so that there is leakage of the pressurised fluid, the crack length at the outer surface will be less than its length at the inner surface. This paper investigates the leakage through the tunnelled crack, and uses a simple analysis to give the inner-surface crack size required to give a prescribed amount of leakage for a given outer-surface crack length. The results provide quantitative underpinning for the importance, and potentially adverse effects, of the tunnelling phenomenon on the leak-before-break methodology for pressurised components.

Title: HIGH TEMPERATURE BEHAVIOUR OF FERRITIC PIPE WELDS : EXPERIENCE OF LONG-TERM TESTIN

Author: Coleman-MC **Corp. Author:**

Source: International Journal of Pressure Vessels and Piping. 1989, vol.39, no.102, 109-118.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Test/analysis **ID:** 758

Abstract: For the safe extension of the life of power plants it is necessary to establish and apply procedures that ensure the integrity of welded components operating at elevated temperatures. This paper describes the experience gained in this area from testing welded components, concentrating on creep crack initiation and growth observations made in pipe to pipe welds, involving correct and incorrect materials, and pipe to end cap welds. The cracking modes are described and explained in terms of the stress distributions in the weldments and implications for operating plants are discussed.

Title: Leak-Before-Break Application in Light-Water Reactor Plant Piping.

Author: Beaudoin-BF; Hardin-T; and-others **Corp. Author:**

Source: Nuclear Safety. Apr./Jun.1989, vol.30, no.2, 189-200.

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: LBB justification **ID:** 759

Abstract: Methodology and criteria for a leak-before-break (LBB) program on high-energy nuclear piping are described. The LBB program can be applied to any operational LWR or any reactor plant under construction to minimize the number of pipe whip restraints and jet impingement shields and to discount the consideration of remaining pipe rupture dynamic effects. A candidate system must be carefully screened to verify that it is not subject to failure by cracking mechanisms that would adversely affect the accurate evaluation of flaws and loads, such as water hammer, erosion-corrosion, fatigue, creep, and brittle fracture. The general methodology and criteria used in an LBB application are described. The relationships between LBB and nuclear plant accident analysis are addressed. Discusses the impact that LBB-based licensing decisions have on environmental and equipment qualification issues. 37 refs.

Title: THE INSTABILITY CRITERION FOR A CIRCUMFERENTIAL THROUGH-WALL CRACK IN A BEND IN A

Author: Smith-E **Corp. Author:**

Source: International Journal of Pressure Vessels and Piping. 1989, vol.40, no.2, 151-159.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Research/theoretical **ID:** 760

Abstract: Motivated by the problem of intergranular stress corrosion cracking of type 304 stainless steel in boiling water nuclear reactor piping systems, the paper examines the stability of a circumferential through-wall crack in a bend in a piping system that is subjected to simulated accident loading conditions. The instability criterion is expressed in terms of the material tearing modulus TMAT and the applied tearing modulus TAPP (or the effective pipe length LEFF). The main thrust of the paper is the modelling of the bend profile and an assessment of the effect of bend angle on the instability criterion.

Title: PROBABILITY OF VOID COALESCENCE IS SPHEROIDIZED PRESSURE VESSEL STEEL.

Author: Strnadel-B; Mazancova-E; and-others **Corp. Author:**

Source: International Journal of Pressure Vessels and Piping. 1989, vol.40, no.4, 303-314.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Research/theoretical **ID:** 761

Abstract: Presents a statistical model of the coalescence of voids nucleated by the effects of plastic deformation ahead of a crack tip on carbide particles. Statistical analysis of the microstructure has been used to define the nature of the dependence of the coalescence probability upon the distance from the crack tip. Application of this model to a spheroidised steel has confirmed that the coalescence probability declines as the distance from the crack tip increases. As the coalescence of voids ahead of a crack tip governs the onset of stable propagation of the main crack, this model is particularly suitable for statistical prediction of the upper shelf of fracture toughness in the steels utilised for the fabrication of pressure vessels and pipings. 15 refs.

Title: FLASHING FLOW THROUGH RELIEF LINES, PIPE BREAKS AND CRACKS.

Author: Morris-SD **Corp. Author:**

Source: Journal of Loss Prevention in the Process Industries. Jan.1990, vol.3, no.1, 17-26.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Methods **ID:** 762

Abstract: Discusses practical methods for sizing relief lines under compressible flashing flow conditions. The emphasis is on relief lines having tortuous geometries but also includes relief networks, multicomponent mixtures and multiple choking. Although attention is focussed on flow downstream of relief devices, reference is made to methods for estimating discharge rates through pressure relief valves and/or rupture discs. The applicability of the methods to the analysis of hypothetical pipe break scenarios is demonstrated by example and compared with some experimental data. Finally, the state-of-the-art regarding two-phase flow through cracks is briefly reviewed. 23 refs.

Title: NO SLICK ANSWERS FOR SHELL.

Author: Pendrous-R **Corp. Author:**

Source: Engineer. 8 Mar.1990, vol.270, no.6989, 22.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Experience/events **ID:** 763

Abstract: Discusses Shell's Mersey oil pipeline failure of August 1989 and looks in particular at the causes of the pipeline rupture which was due to corrosion from outside.

Title: HYDROCARBON SENSING FOR DETECTING LEAKS IN UNDERGROUND STORAGE TANKS AND PIPES

Author: Koppitsch-H **Corp. Author:**

Source: Bulletin : Journal of the Association for Petroleum and Explosives Administration. Feb.1990, vol.28, no.1, 14-18.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Inspection methods **ID:** 764

Abstract: Describes the process of hydrocarbon sensing for detecting leaks in underground storage tanks, compared with alternatives for leak detection.

Title: A STRATEGY FOR PLANT MANAGEMENT TO PREVENT LOSS: 7 WAYS FOR MANAGERS TO CUT INC

Author: Dunford-N **Corp. Author:**

Source: Loss Prevention Bulletin. Jun.1990, no.093, 25-31.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Other **ID:** 765

Abstract: Outlines a method of management control designed to prevent pipework and in-line equipment failure. Preventive action is divided into four categories: hazard study; human factors; task checking and routine checking. By examining immediate and underlying causes and preventive actions in combination, management can prioritise a detailed accident plan for prevention. Three case studies, involving GRP pipeowrk flange failure; a logging fire caused by leaking oil and a scaffolding fire caused by leaking organics are also described. The author is a member of staff of the Health and Safety Executive.

Title: Organisation, Management and Human Factors in Quantified Risk Assessment: A Theoretical and Empirical Basis for

Author: Hurst-NW; Bellamy-LJ; and-others **Corp. Author:**

Source: Elsevier, 1990.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Other **ID:** 766

Abstract: Describes work sponsored by the Health and Safety Executive which is designed to give a better understanding of the effects of organisational and managerial factors on the levels of risk at industrial and major hazard plants. Failures of pipework and vessels are analysed and classified, using descriptions of both the underlying causes of accidents and their immediate causes. The implications of these classifications on the values of generic failure rates are discussed with a view to modifying risk calculations in the light of different standards at nominally identical plants. N.W. Hurst is a member of staff of the Health and Safety Executive. 17 refs.

Title: RISK MANAGEMENT FOR WATER AND ENERGY PIPELINES.

Author: Kulkarni-RB; Patwardhan-AS **Corp. Author:**

Source: Journal of Occupational Accidents. Sep.1990, vol.13, nos.1-2, 121-133.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Other **ID:** 767

Abstract: Presents a methodology which uses economic and risk exposure evaluations to make repair versus replace decisions, that is, should a particular pipe segment be replaced in the planning year, or should it be maintained and repaired, as necessary, for at least one more year? The methodology provides an efficient screening tool to rapidly analyse an entire network of piping segments and identify the critical segments for replacement during the planning year based on probabilities of breaks and leaks, replacement and repair costs, and possible consequences of pipe failure.

Title: SIMPLE TRANSIENT RELEASE RATE MODELS FOR RELEASES OF PRESSURISED LIQUID PETROLEUM

Author: Tam-VHY; Higgins-RB **Corp. Author:**

Source: Journal of Hazardous Materials. Oct.1990, vol.25, nos.1/2, 193-203.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Test/analysis **ID:** 768

Abstract: Large scale experimental data have been used to compare and derive simple mathematical models to describe the time varying release rate of pressurised liquid petroleum gas (LPG) from a ruptured pipeline. The models studied consisted of a single box, a single-node slip-flow and an empirical model. The empirical model is based on data obtained using 100 metre long pipelines of internal diameters of 50 mm and 150 mm. The empirical model was developed to describe the observed characteristics of the mass history of commercial liquid propane inside the pipe, namely the mass reduced approximately exponentially with time. While the single box model did not compare well with observed data, the single-node slip-flow model was found to produce exponentially time varying release rates.

Title: ASPECTS OF RISK ASSESSMENT FOR HAZARDOUS PIPELINES CONTAINING FLAMMABLE SUBSTAN

Author: Carter-DA **Corp. Author:**

Source: Journal of Loss Prevention in the Process Industries. Jan.1991, vol.4, no.2, 68-72.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Other **ID:** 769

Abstract: The Major Hazards Assessment Unit has developed a computerised method for the quantified risk assessment of hazardous pipelines. Describes the method in general terms, with a more detailed description of three important event models: PROFIT, a transient flowrate model for ruptured pipelines which includes compressibility and thermodynamic effects; MAJESTIC, a multiple point source jet flame thermal radiation hazard model; and DISPI, an integrating model for the risks from dispersing flammable vapour clouds using elliptical trigonometry. The application of the method to a typical pressure pipeline is described. The author is a member of staff of the Health and Safety Executive. 13 refs.

Title: SAFETY EVALUATION OF PIPELINE.

Author: Bryce-DJ; Turner-MJ **Corp. Author:**

Source: Cranfield, British Hydromechanics Research Association, 1979.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1979 **Language:** English

Category: Experience/events **ID:** 770

Abstract: Describes how hazards and risks presented by a cross-country pipeline to populations or installations in the area may be evaluated. Details of pipeline incidents involving flammable materials are given. Shows how information on rates of release and atmospheric dispersion may be combined with pipeline failure rate data to predict the frequency with which populations are likely to be affected. The authors are members of staff of the Health and Safety Executive.

Title: ELASTIC-PLASTIC FRACTURE ANALYSIS OF CARBON STEEL PIPING USING THE LATEST CEGB R6 A

Author: Kanno-S; Hasegawa-K; and-others **Corp. Author:**

Source: International Journal of Pressure Vessels and Piping. 1991, vol.45, no.1, 89-99.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Methods **ID:** 771

Abstract: The elastic-plastic fracture of carbon steel piping having various pipe diameter and circumferential crack angle subjected to a bending moment is analyzed using the latest CEGB R6 approach. The elastic-plastic fracture criterion must be applied instead of the plastic collapse criterion with increase of the pipe diameter and the crack angle. A simplified elastic-plastic fracture analysis procedure based on the R6 approach is proposed.

Title: Study on Crack Opening Area and Coolant Leak Rates on Pipe Cracks.

Author: Matsumoto-K; Nakamura-S; and-others **Corp. Author:**

Source: International Journal of Pressure Vessels and Piping. 1991, vol.46, no.1, 35-50.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: LBB justification **ID:** 772

Abstract: The study was executed to support the establishment of leak before break (LBB) standards for high energy piping, by examining crack opening shape on the pipe surface and crack opening area. Results show that the middle crack opening area between the inner and outer surfaces may be used in the analysis. The analytical leak rate calculated from the Tada-Paris equation and Moody's critical flow model was in agreement with the measured one obtained from the leak test. 17 refs.

Title: Selected Safety-Related Events

Author: Murphy, G.A. **Corp. Author:**

Source: Nuclear Safety, Vol. 32:121-123.

SKI Project File: Ja **Transfer:** Ja **Publ year:** 1991 **Language:** English

Category: Operating experience **ID:** 773

Abstract: Discusses a pipe rupture at Millstone-3, a Westinghouse 4-loop PWR near New London, Connecticut, on 31 December 1990. Two moisture separator drain lines ruptured releasing secondary water and steam into the turbine building. Note: Event description appears in SLAP data base, EID 498.

Title: ENVIRONMENTALLY INDUCED CRACKING.

Author: Scott-PM **Corp. Author:**

Source: Industrial Corrosion. Dec.1990/Jan.1991, vol.9, no.1, 8-14.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Research/theoretical **ID:** 774

Abstract: Presents a review of current research topics in the United Kingdom on environmentally induced or assisted cracking with particular emphasis on how these research interests are related to current industrial applications. Marine structures, oil/gas production, chemical processes, power generation and aluminium alloys are discussed. Discusses corrosion fatigue of offshore structural materials, pipeline cracking, high strength fastners and deaerator cracking. 25 refs.

Title: Finite Element Validation Studies of the Revised PD6493/CEGBN R6. Part 1 - Failure Assessment Methodologies Ap

Author: Finch-DM; Burdekin-FM **Corp. Author:**

Source: International Journal of Pressure Vessels and Piping. 1992, Vol.49:187-211.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Methods **ID:** 775

Abstract: Systematic investigations have been carried out for the validations and verifications of the FADs (failure assessment diagrams) and the plasticity correction factors for taking into account the residual stress effects. FADs, for a wide range of welded structures under the effect of mechanical loading and combined mechanical loading and residual stresses, have been established using fracture mechanics data generated using finite element techniques. Safety margins and non-conservatism of the FADs and the plasticity correction factor have been quantitatively examined and modifications have been recommended where necessary and possible for these geometries. 13 refs.

Title: Probability of Pipe Failure in the Reactor Coolant Loops of Babcock and Wilcox Pressurized Water Reactor Plants : V

Author: Ravindra-MK; Campbell-RD; and-others; Lawrence Livermore National Laboratory United States. Nuclear Regulatory Commission **Corp. Author:**

Source: Washington, D.C., USGPO, 1985. (NUREG/CR-4290, vol.2) (UCRL-53644, vol.2) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: Damage probability **ID:** 776

Abstract:

Title: Leak Before Break: HM NII's Present View

Author: Creswell-SL; Health and Safety Executive. Nuclear Installations Inspectorate **Corp. Author:** NII

Source: Washington, D.C., USGPO, 1986.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: LBB justification **ID:** 777

Abstract: Discusses the progress of the Pipe Break Working Group (PBWG). Topics include the duties of the Nuclear Installations Inspectorate; responsibilities of licensees; safety assessment principles; a definition of leak before break and NII safety principles. The author is a member of staff of the Health and Safety Executive.

Title: Radioactive Leakage at Nuclear Reactor, Ignalina Lithuania (Miscellaneous)

Author: Anonymous **Corp. Author:**

Source: Lloyds Casualty Week. 30 Oct.1992, vol.290, no.4, 86.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Operating experience **ID:** 778

Abstract: Briefly outlines details of a leak of radioactive water from a small crack in a narrow pipe at the Chernobyl-type nuclear reactor on 15 October 1992. The plant would not be re-opened before 23 October to allow repair work to take place. Two other reactors at the plant have been closed for routine maintenance.

Title: Short Cracks in Piping and Piping Welds. Semi Annual Report April -September 1991

Author: Wilkowski-GM; Brust-F; and-others; **Corp. Author:**

Source: Washington, D.C., USGPO, 1991. (NUREG/CR-4599) (BMI-2173) (Vol.2, no.1) various paging.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Methods **ID:** 779

Abstract:

Title: A Review of Fatigue Failures in LWR Plants in Japan.

Author: Iida-K **Corp. Author:**

Source: Nuclear Engineering and Design. Dec.1992, vol.138, no.3. 297-312.

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Experience/events **ID:** 780

Abstract: A review was made of fatigue failures of nuclear power plant components in Japan, which were experienced in service and during periodical inspection. No case has been recently reported of a service fatigue failure of a reactor pressure vessel itself, excluding nozzle corner cracks, that occurred many years ago. But, service fatigue failures have been occasionally experienced in piping systems, pumps, and valves, on which fatigue design seems to have been adequately applied. The causes of fatigue failures can be divided into two categories: mechanical-vibration-induced fatigue and thermal-fluctuation-induced fatigue.

Title: Procedure of Crack Shape Determination by Reversing DC Potential Method

Author: Hashimoto-Y; Urabe-Y; and-others **Corp. Author:**

Source: Nuclear Engineering and Design. Dec.1992, vol.138, no.3. 259-268.

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Inspection methods **ID:** 781

Abstract: On-line monitoring of a crack and evaluation of component integrity are needed for maintaining the safety of a plant. Describes the system Reversing DC Potential Method (RDCPM) developed by the authors. Discusses the simplified method for determining the crack shape and its application to a pipe is shown.

Title: Analysis of Leak and Break Behavior in a Failure Assessment Diagram for Carbon Steel Pipes.

Author: Kanno-S; Hasegawa-K. et al **Corp. Author:**

Source: Nuclear Engineering and Design, Vol.138:251-258.

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: LBB methodology **ID:** 782

Abstract: The leak and break behaviour of a cracked coolant pipe subjected to an internal pressure and a bending moment was analysed with a failure assessment diagram using the R6 approach. Examines the conditions of the detectable coolant leakage without breakage.

Title: Understanding Pipeline Failures Using Discriminant Analysis: The North Sea Application.

Author: Mare-RF-de-la; Bakouros-YL; and-others **Corp. Author:**

Source: Reliability Engineering and System Safety, Vol. 39:71-80.

SKI Project File: Ja **Transfer:** Ja **Publ year:** 1993 **Language:** English

Category: Failure probability **ID:** 783

Abstract: This paper describes a novel approach for modelling offshore pipeline failures. Using data for pipelines in the North Sea, a methodology has been developed for explaining the effects of several, factors on the reliability of pipeline systems. Discriminant analysis forms the basis of this methodology, which can accommodate the manifold variables affecting such failure and predict the probability of any pipeline failing. In this respect, the proposed methodology is superior to the conventional approach, which is based on average failure rates. 22 refs.

Title: Breakthrough in Pipeline Safety

Author: Anonymous **Corp. Author:**

Source: Health and Safety in Industry. Feb.1993, vol.16, no.2, 1.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Inspection methods **ID:** 784

Abstract: Describes a system for tracing leaks from oil and gas pipelines which has been developed by Shell.

Title: Hairline Cracks Found in Brunsbuettel Piping .

Author: Anonymous **Corp. Author:**

Source: Nuclear News. Mar.1993, vol.36, no.3, 76-77.

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Experience/events **ID:** 785

Abstract: About 110 hairline cracks have been discovered in pipework of the emergency core cooling systems and water purification plant at the Brunsbuettel power station in Germany.

Title: NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping.

Author: United States. Nuclear Regulatory Commission **Corp. Author:**

Source: Washington, D.C., USGPO, 1992. (Generic letter 88-01) (Supplement 1) 5 pp.

SKI Project File: Nej **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Other **ID:** 786

Abstract:

Title: Example of a PSA-Based Analysis of an Occurred External Pipe Break at TVO I

Author: J. Holmberg & P. Pyy **Corp. Author:** VTT Automation, Espoo, FIN-

Source: NKS/SIK-1(1990-93), Work Report No.: nks/sik-1(93)17, 11/25/95

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Rupture / PSA **ID:** 787

Abstract: This report presents a methodology to analyze incidents by the help of PSA. The methodology is a combination of qualitative root cause analysis and a quantitative analysis of an event sequence. An external pipe break that occurred at TVO-I during the commissioning in 1979 is analyzed using the methodology. The incident appeared to have little safety significance since the leakage size was rather small. Further, the leakage can be isolated in several ways, and the operators can balance the situation by other means. In light of the analysis experience gained, a rough quantitative analysis of incident contributors and the interpretation of its results seems to be an appropriate way to link incident investigation with PSA. A successful follow-up analysis of an incident requires that the information concerning the case has been recorded through documentation or interviews within a reasonable time after the incident. The chosen example was rather old (15 years), and thus, not all the details were reproducible. The investigation showed that it should be possible to model incidents such as initiating events in the probabilistic sense. A quantitative evaluation can give a fruitful view on the safety significance of the incident. However, the results of such an analysis have to be interpreted with care.

Title: Review of Main Degradations Observed on Reactor Internals of Operating Belgian PWRs

Author: P. Briegleb & P. Mignot **Corp. Author:** Vincotte A.S.B.L., Brussels (B

Source: Proceedings of NEA/CSNI - UNIPEDE Specialist Meeting on Regulatory and Life-limiting Aspects of Core Internals and Pressure Vessels, Vol. 1:61-91, Published by Swedish Nuclear Power Inspectorate, Stockholm (Sweden)

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: IGSCC / Mech. Wear **ID:** 788

Abstract: The purpose of this papaer is to describe some typical degradations experienced by reactor core internals in operating Belgian PWRs, and to review the investigations carried out to determine the cause and the extent of the problems and the corrective actions taken. The degradations described are attributed either to IGSCC of inconel alloys, or to mechanical wear resulting from flow-induced vibrations or from fretting of moving pieces. The components affected are the bolts clamping the hold down springs on top of fuel assemblies, the fixtures on top of upper guide tubes, the control rod guide tube support pins, the rod cluster control assembly (RCCA) rodlets and the incore instrumentation thimbles.

Title: Pipe Cracking Experience in Light-Water Reactors

Author: L. Frank, W.S. Hazelton, R.A. Hermann, V.S. Noonan, A. Taboada **Corp. Author:** U.S. Nuclear Regulatory Com

Source: NUREG-0679; 31 pages

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1980 **Language:** English

Category: IGSCC / Fatigue Cracks **ID:** 789

Abstract: Commercial LWRs have experienced pipe cracking since 1965. This report summarizes pipe cracking experience in LWRs as reported in LERs from 1967 through 1979, other licensee and vendor reports, and Office of Inspection and Enforcement Bulletins. Pipe cracks which were environmentally induced, such as stress corrosion cracking of metal sensitized by welding and heat treatment, were most prevalent. Feedwater pipes experienced fatigue cracking from thermal stress and many small lines developed leaks as a result of fatigue caused by vubration. Cracking incidents are separated into generic categories and listed by reactor type, pipe size, and systems affected.

Title: Reliability and Defect Sizing

Author: M. Aaltio and K.P. Kauppinen **Corp. Author:** VTT, Espoo, Finland

Source: Periodic Inspection of Pressurized Component, IMechE Conference Publications 1982-9, Paper C151/82, London (UK), pp 283-290

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1982 **Language:** English

Category: Detection reliability **ID:** 790

Abstract: The theoretical basis for inspection reliability studies is reviewed. Focus is on the experimental program set up at VTT to evaluate the reliability of ordinary X-ray and UT-examination. The results of the experimental program are summarized.

Title: Seabrook Station Risk Management and Emergency Planning Study

Author: Fleming, K.N. et al **Corp. Author:** PLG, Inc.

Source: PLG-O432

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: ISLOCA **ID:** 791

Abstract: The purpose of this report is to present the results of a technical evaluation of emergency planning options and other risk management actions under consideration for Seabrook Station. These results include an update of the Seabrook Station Probabilistic Safety Assessment (SSPSA) to account for new insights regarding radioactive release source terms and the progression of sequences involving loss of coolant events that bypass the containment. Note: Section 3.1.4 addresses RHR piping and heat exchanger strength; i.e., probability of pipe/tube failure at 2,250 psia. The RHR system design pressure is 600 psig. The system piping is composed of Schedule 40, Type 304 stainless steel.

Title: Assessment of ISLOCA Risk-Methodology and Application to a Babcock and Wilcox Nuclear Power Plant

Author: Galyean, W.J. and Gertman, D. I. **Corp. Author:** EG&G Idaho, Inc.

Source: EGG-2608 (NUREG/CR5604)

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: ISLOCA **ID:** 792

Abstract: This document presents information essential to understanding the risk associated with inter-system loss-of-coolant accidents (ISLOCAs). The methodology developed and presented in this document provides a state-of-the-art method for identifying and evaluating plant-specific hardware designs, human performance issues, and accident consequence factors relevant to the prediction of the ISLOCA risk. This ISLOCA methodology was developed and then applied to a Babcock and Wilcox (B&W) nuclear power plant. The results from this application are described in detail. For this particular B&W reference plant, the assessment indicated that the probability of a severe ISLOCA is approximately 2.2E-6/reactor-year. Note: This study developed a method for estimating pipe rupture probabilities based on an structural analysis. The basic analysis process involved: (a) estimating the pressure capacities for the components in the interfacing systems, (b) estimating the local system pressure generated in the interfacing system as a result of an ISLOCA sequence, and (c) combining these two estimates in a stress/strength comparison to calculate a rupture probability for both the individual components and the entire interfacing system.

Title: Reliability of High Energy Pipework. Presentation of a new Research Project.

Author: **Corp. Author:** Swedish Nuclear Power Inspect

Source: SKI/RA-019/94

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Pipe reliability **ID:** 793

Abstract:

Title: The Risks of Nuclear Power Reactors. A Review of the NRC Reactor Safety Study WASH-1400 (NUREG-75/014)

Author: HUBBARD, R.B., MINOR, G.C. **Corp. Author:** Union of Concerned Scientists

Source: Union of Concerned Scientists, Cambridge (MA), pp 39-52

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1977 **Language:**

Category: Piping and RPV reliability **ID:** 794

Abstract: Chapter 4 includes a summary of critical views on piping and RPV reliability, and the use of small probabilities.

Title: Prospects and Problems in Risk Analyses:Some Viewpoints

Author: Levine, S. and Vesely, W. E. **Corp. Author:** Society for Industrial and Appli

Source: Nuclear Systems Reliability Engineering and Risk Assessment pp 5-21

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1977 **Language:** English

Category: PSA methodology **ID:** 795

Abstract: Present problem areas in risk analysis are outlined and promising utilizations are described. Some of the specific problems involve lack of standardization of models, data, and quantitative approaches. The promising areas include performances of generic analyses and sensitivity evaluations. Specific applications being performed within the U.S. Nuclear Regulatory Commission are described [auth]. Note: This 1977 paper discusses fundamental issues related to passive component failures, design/construction errors, RPV reliability, etc.

Title: Erosion-corrosion of parallel feed water discharge lines in Loviisa WWER 440

Author: Hietanen, O. and P. Korhonen **Corp. Author:**

Source: Specialist Meeting on Erosion-Corrosion of Nuclear Power Plant Materials, OECD/GD(95)2, pp 75-85

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** English

Category: Erosion-corrosion **ID:** 796

Abstract: Two guillotine pipe breaks of feed water system piping have occurred in 1990 and 1993 at Loviisa Unit 1 and Unit 2, respectively. These two pipe breaks and inspections have revealed that wall thinning can be very local in nature, and wall thinning of similar components in parallel lines can be completely different. In order to find out the factors that might explain this behaviour, perational data, such as process data and water chemistry, were evaluated and compared with chemical analysis and actual dimensions of the replaced components from the feedwater system both at Unit 1 and Unit 2. However, no unambigous correlation between these parameters and different wall thinning of similar components in the parallel feedwater lines was found.

Title: Future Energy Choice in Norway. A Critique of the Application of Probability Theory in Report by Nuclear Commissi

Author: Elster, J. **Corp. Author:** Department of Mathematics, U

Source:

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1979 **Language:** Norwegian

Category: **ID:** 797

Abstract:

Title: Stress corrosion cracking studies on ferritic low alloy pressure vessel steel-water chemistry and modelling aspects

Author: Tipping, P., Ineichen, U., Cripps, R. **Corp. Author:**

Source: Specialist Meeting on Erosion and Corrosion at Nuclear Power Plant Materials, OCDE/GD(95)2, pp 271-280

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** English

Category: SCC **ID:** 798

Abstract:

Title: Erosion-corrosion in wet steam and single phase lines in nuclear power plants

Author: Tanarro, A.; Gonzalez, E. **Corp. Author:**

Source: Specialist Meeting on Erosion and Corrosion at Nuclear Power Plant Materials, OCDE/GD(95)2, pp 295-304

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** English

Category: Erosion-corrosion **ID:** 799

Abstract:

Title: Estimates of Rupture Probabilities for Nuclear Power Plant Components: Expert Judgement Elicitation

Author: Vo, T.T. et al **Corp. Author:**

Source: Nuclear Techology, Vol. 96:259-270

SKI Project File: Ja **Transfer:** Ja **Publ year:** 1991 **Language:** English

Category: Failure probability estimation **ID:** 800

Abstract: As part of the nondestructive evaluation reliability program sponsored by the U.S. NRC, Pacific Northwest Laboratory (PNL) developed a risk-based method for establishing inspection priorities for systems and components at nuclear power lants. In this method, the results of probabilistic risk assessment (PRA) are used to estimate the safety consequences of component failures. The method also requires estimates of the probabilities of structural failures. Since sufficient operating experience data and detailed fracture mechanics analyses are not available, an expert judgment elicitation is conducted to estimate component rupture probabilities. (An expert judgment process is generally adapted from the NRC severe accident risk program.) The plant selected for the detailed evaluation is the Surry-1. Systems selected for analysis are the reactor pressure vessel, the reactor coolant, the low-pressure injection including the accumulators, and the auxiliary feedwater. Additional technical information is gathered regarding the elicited issues. The data appear to be reasonable, and they generally agree with and reflect Surry-1 plant operating experience. Typical areas of concern correspond to such factors as high stresses (e.g., places where mixing of fluids with large temperature differences occurs) and places where erosion or corrosion effects are active. These results will be used by PNL in an ongoing pilot study based on the PRA results and other relevant information in determining the inspection priorities for systems and components at the Surry power plant.

Title: Pipe and Vessel Failure Probability

Author: Thomas, H. M. **Corp. Author:**

Source: Reliability Engineering, Vol. 2:83-124

SKI Project File: Ja **Transfer:** Ja **Publ year:** 1981 **Language:** English

Category: Pipe failure probability **ID:** 801

Abstract: This generalized approach to the estimation of failure probability is based on a pragmatic and scientific analysis of actual service failure statistics. Approximation strategies have been devised in order to estimate failure probability at the leakage level and for rupture. The leakage probability is estimated from global statistics for leakage failure by using an observed correlation that a geometric proportionality measure of size and shape and weldments gives a direct measure of failure probability. This is the most powerful single influence of all in the determination of leakage probability, but the influence of plant age is also worth considering. The estimate may then be scaled for other factors if their influence is known. The rupture probability may be estimated given a leakage probability, partly by using a fracture mechanics model which gives a carpet of rupture/leakage curves. Observed statistics are also used.

Title: Probabilistic Fracture Mechanics

Author: Harris, D.O. **Corp. Author:** The American Society of Mech

Source: Pressure Vessel and Piping Technology 1985 A Decade of Progress, pp 771-791

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: PFM methodology **ID:** 802

Abstract: A review of probabilistic fracture mechanics is provided. The close tie with deterministic fracture mechanics is emphasized, and the relationship between determinsitic and probabilistic models is discussed. The information on distribution of input variables is reviewed, and techniques for the generation of results from a probabilistic model are discussed. Examples of applications to pressure vessels and piping, aircraft and civil structures are provided, and the future directions of probabilistic fracture mechanics are discussed. In spite of the current inaccuracies in predicting absolute values of failure probabilities, current models are capable of providing definitive answers to questions regarding the relative influence of various factors on component reliability.

Title: Probabilistic Assessment of Pressure Vessel and Piping Reliaility

Author: Sundararajan,C. **Corp. Author:**

Source: Journal of Pressure Vessel Technology, Vol. 108:1-13

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1986 **Language:** English

Category: Piping reliability **ID:** 803

Abstract: This paper presents a critical review of the state-of-the-art in probabilistic assessment of pressure vessel and piping reliability. First the differences in assessing the reliability directly from historical failure data and indirectly by a probabilistic analysis of the failure phenomenon are discussed and the advantages and disadvantages are pointed out. The rest of the paper deals with the latter approach of reliability assessment. Methods of probabilistic reliability assessment are described and major projects where these methods are applied for pressure vessel and piping problems are discussed. An extensive list of references is provided at the end of the paper.

Title: Deutsche Risikostudie Kernkraftwerke Phase B: Eine Untersuchung zu dem durch Storfalle in Kernkrafwerken verurs

Author: **Corp. Author:** Gesellschaft fur Reaktorsicherh

Source: Verlag TUV Rheinland GmbH, Koln

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1989 **Language:** German

Category: **ID:** 804

Abstract:

Title: A Survey of Defects in the UK for the period 1962-1978 and Its Relevance to Nuclear Primany Cricuits

Author: Smith, T. A., Warwick, R. B. **Corp. Author:** United Kingdom Atomic Energ

Source: SRD R203

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1981 **Language:** English

Category: **ID:** 805

Abstract:

Title: A Remark on Data for Defects used in Probabilistic Analyses of Failure of Nuclear Pressure Vessels

Author: Ostberg, G. **Corp. Author:**

Source: Reliability Engineering and System Safety, Vol. 35:77-82

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: ISI reliability **ID:** 806

Abstract: In a fracture mechanical probabilistic analysis of the failure of nuclear pressure vessel, data are needed for the presence of defects that may have escaped detection during non-destructive examination. At present such statistical data can be obtained only by subjective estimates. A review has been made of the data on the effectiveness of defect detection on which the most widely cited probabilistic analyses of the safety of nuclear pressure vessels has been made. It seems justified to use considerably lower values for the effectiveness. Correspondingly, the calculated probabilities for failure of nuclear pressure vessels should be raised. Consequently, this type of failure would become of greater concern than presently assumed considering the risks associated with nuclear power plants.

Title: Overpressurization of Emergency Core Cooling Systems in Boiling Water Reactors

Author: Lam, P. **Corp. Author:** UlS. Nuclear Regulatory Com

Source: Preliminary Case Study Report, Office for Analysis and Evaluation of Operational Data (AEOD)

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: ISLOCA, passive component failure **ID:** 807

Abstract:

Title: Some Thoughts on the Use of PSA Experts and the Need to Break the Rules

Author: Stetkar, J. van Otterlo, R. **Corp. Author:** International Atomic Energy A

Source: Advances in Reliability Analysis and Probabilistic Safety Assessment, IAEA-J4-TC-606.4

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: PSA and passive component failures **ID:** 808

Abstract: This paper gives an example from a PSA study where a piping component failure contributed in a significant way to the overall results. The paper makes a case for need to address passive component failures in PSA to generate realistic plant safety insights.

Title: Technological Risk Analysis. Foundations of Quality Risk Analysis: The PSA and QRA Domains

Author: Lydell, B. **Corp. Author:** RSA Technologies

Source: Manuscript of book in preparation

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1996 **Language:** English

Category: PSA / QRA analytical consideration **ID:** 809

Abstract: Presents a summary quality data considerations and the analytical steps needed to derive plant-specific equipment failure data. Extracts from the manuscript were used in developing SKI Report 95:61 on pipe failure data.

Title: Tackling erosion-corrosion in nuclear steam generating plant

Author: Bignold, G.J. et al **Corp. Author:**

Source: Nuclear Engineering International, Vol. 26, No. 314, pp 37-41

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1981 **Language:** English

Category: Erosion-corrosion **ID:** 810

Abstract: There is growing international interest in the phenomenon of erosion-corrosion, which has occurred under both single- and two-phase flow conditions in carbon and low-alloy steel plant. Erosion rates can be several millimetres per year necessitating costly outages and repairs. But the prevention or minimization of erosion-corrosion damage is relatively straightforward. For example it can be achieved by choosing designs which avoid highly turbulent flow, by choosing erosion-resistant materials and by adopting an appropriate water chemistry regime.

Title: Guidelines for Preventing Human Error in Process Safety

Author: Embrey, D. **Corp. Author:** American Institute of Chemical

Source: Center for Chemical Process Safety, ISBN: 0-8169-0461-8, pp 41-44

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Operating experience **ID:** 811

Abstract: Overview of human error causes of process incidents; makes distinction between active and latent errors. See also SKI Report 95:58, Section 3.4.

Title: What Went Wrong? Case Histories of Process Plant Disasters

Author: Kletz, T. A. **Corp. Author:**

Source: ISBN:0-87201-919-5, pp 49-65

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: Operating experience **ID:** 812

Abstract: Case studies from the chemical process industry. Includes overviews of causes of piping failures and pipe failure prevention. Chapter 3 of the book discusses human errors and how they have caused piping system failures.

Title: Prevent Pipe Failures Due to Human Errors

Author: Geyer, T. A. W. **Corp. Author:**

Source: Chemical Engineering Progress, No. 11, pp 66-69

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Operating experience **ID:** 813

Abstract: Summary of survey of British chemical industry experience with piping failures. Emphasis of survey was on human factors and human reliability aspects of pipe failures. A detailed classification scheme for identifying the underlying causes of human error induced pipe failures.

Title: Evaluation of Water Hammer Events in Light Water Reactor Plants

Author: Uffer, R. A. **Corp. Author:** EG&E Idaho

Source: EGG-2203 (NUREG/CR-2781)

SKI Project File: Ja **Transfer:** Ja **Publ year:** 1982 **Language:** English

Category: **ID:** 814

Abstract: This document presents the results of an evaluation of water hammer events in LWR power plants. The evaluation was based upon reports of actual events, typical plant design drawings and operating procedures. Included in this report are design and operating recommendations for the prevention or mitigation of water hammer occurrence.

Title: Failure Data, Appendix III to Reactor Safety Study

Author: **Corp. Author:** U.S. NRC

Source: WASH-1400 (NUREG-75/014), pp III-74-78

SKI Project File: Ja **Transfer:** Ja **Publ year:** 1975 **Language:** English

Category: Pipe failure data **ID:** 815

Abstract: This appendix summarizes the basis for pipe failure rate estimates used in the Reactor Safety Study (WASH-1400). Both nuclear and nonnuclear operating experience acknowledged. The nuclear data was based on 150 reactor-years of U.S. NPP experience.

Title: The Probability of Catastrophic Failure of Reactor Primary System Components

Author: Holt, A.B. **Corp. Author:**

Source: Nuclear Engineering and Design, Vol. 28:239-251

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1974 **Language:** English

Category: Pipe failure probability **ID:** 816

Abstract: Using recent information about failures in NPPs, the author has derived the following estimates of the probability of failure events in prime piping systems of carbon steel and low-alloy steel: severance prior to service: 5.7E-2 events/plant, and severance during service: 5.0E-3 events/plant.yr. These numbers are higher, by one or two orders of magnitude, than the corresponding numbers used in calculations pertaining to nuclear safety. To estimate the probability of severance of a reactor vessel (which here includes the large nozzles), the author uses the logic of the Warner diagram. In the case of piping, it is noted that the ratio: probability of severance prior to service to probability of severanve during ten years of service is approximately 1. Over the last ten years there have been at least four failures of heavy section steel vessels (UK and US). Although there are good reasons to assert that vessels are much less prone to severance than piping, the Warner diagram speaks its logic. Statistics gathered from hardware populations in existing plants will never be directly applicable to hardware populations in future plants because we are continuously introducing changes in the main parameters: design, materials, manufacture and operation. Wilson, of General Electric, have developed methods for a priori calculations of the probability of failure events in piping systems, based on first principles as he sees it. He considers that failures are due to fatigue, and bases his methods on the Paris formula. In Wilson's scheme, each component slides down its path to failure, passing a specified series of stages where the parameters of his formulae are subjected to random variations. The result is that a crack of initial size under influence of a stress may grow to a depth exceeding the wall thickness, resulting in a leak. Or the crack may grow to critical size, and result in a severance triggered by a high stress (random). Wilson's results compare favorably with the observed probabilities of leaks and severances during service, but they are too low, by an order of magnitude, for the probability of severance prior to service.

Title: Reliability of Piping in Light-Water Reactors

Author: Bush, S. H. **Corp. Author:**

Source: Nuclear Safety, Vol. 17:568-579

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1976 **Language:** English

Category: Operating experience **ID:** 817

Abstract: This article assesses the reliability of piping in LWRs based on nonnuclear failure data, conditional failure probabilities, the role of periodic inspections, and a review of nuclear system failures. Failure statistics confirm rates of E-4 to E-6 per reactor-year in large pipes., with higher rates as the size decreases. Periodic inspection, a critical factor, enhances the reliability by factors of 10 to 10,000. Nuclear failures are classed into two statistical categories: (1) those due to IGSCC; and (2) all others due to construction, design, operational errors. The spectrum of pipe sizes inflenced by IGSCC differs from that influenced by other mechanisms.

Title: Statistics of Pressure Vessel and Piping Failures

Author: Bush, S.H. **Corp. Author:** The American Society of Mech

Source: Pressure Vesel and Piping Technology 1985. A Decade of Progress, pp 875-893

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: Operating experience **ID:** 818

Abstract: An overview is given of available statistics pertaining to nondisruptive and disruptive failures of both pressure vessels and piping. For pressure vessels applicable data is limited to nonnuclear vessels. With piping, the emphasis is on nuclear systems. Probabilities of disruptive failure of vessels, primarily steam drums at 99% confidence upper bound, are less than 1.0E-5 per vessel year. This number appears applicable internationally. Factors related to the low failure rate in the U.S. include the ASME Boiler and Pressure Vessel Codes, Sections I and VIII, periodic inservice inspection, and the hydro testing. Factors influencing failure rates are the welding process and operator error; both require special attention. With piping, failure rates will vary with size. Large pipe failure rates are inferred from vessel failure rates. Intermediate and small sizes of piping have substantially higher rates; these may be two to

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Title: A Study of Piping Failures in U. S. Nuclear Power Reactors

Author: Janzen, P. **Corp. Author:** Atomic Energy of Canada Limi

Source: AECL-Misc-204

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1981 **Language:** English

Category: Pipe failure rate estimation **ID:** 819

Abstract: A study of piping failures in nuclear power generating plants was undertaken in support of the study of pipe rupture in the Primary Heat Transport System of CANDU stations. Because of the limited operating experience of CANDU stations and the availability of documentation of the much longer history of performance of U.S. LWRs, this latter data was chosen as the initial subject of analysis. The analysis involves calculation of pipe failure rates and classification, manipulation and correlation of data according to severity of failure, pipe size, process system in which pipe is located, location of failure, cause of failure, effect of failure on reactor conditions, date of occurrence and plant age at time of occurrence.

Title: Pipe Failure Study, Probabilistic Risk Analysis and Licensing

Author: Petersen, K. E. **Corp. Author:**

Source: NKA/SAK-1-D(82), Proceedings of Seminar 2, pp 129-149

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1982 **Language:** English

Category: Pipe failure probability **ID:** 820

Abstract: This paper describes the status of the analysis of pipe failures performed at Riso. The first part of the paper contains the background for the analysis of th classification system used in the evaluation of the incident reports. The classification system is based upon systems used in similar analyses, the system used in the Swedish ATV0data base and the system used in the half-year reports by SKI. The second part of the paper contains the results from a preliminary analysis of pipe failures performed in the U.S.A. The third part of the paper describes the preliminary results of the analysis of Nordic reactors based upon incidents reported in the Swedish ATV data base and upon the safety related occurrences reported to the Swedish Nuclear Power Inspectorate. Some data from the two Finnish TVO reactors will also be taken into account.

Title: Precracked pipe under waterhammer action

Author: Brosi, S. et al **Corp. Author:**

Source: Nuclear Engineering and Design, Vol. 158:177-189

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** English

Category: Water hammer **ID:** 821

Abstract: We numerically simulate a full scale test in several computational steps with the finite element method and compare all calculated data with the experimental findings. First, we compute the deflection under static loading and the spectrum of eigenfrequencies of an integer piping, attached to a nuclear reactor pressure vessel (RPV). Then we consider a sudden pipe break at some distance from the vessel, immediately followed by an undamped closure of a check valve close to the break on the RPV side, and calculate the elastic and plastic transient dynamic response of the integer piping part between the RPV and the break. Finally we consider a circumferential internal surface crack, fairly close to the vessel; after extensive testing of our fracture mechanics calculation procedure we investigate the stress in the crack regipn under the waterhammer action.

Title: Piping Performance in Canadian CANDU NGS

Author: Janzen, P. **Corp. Author:** Atomic Energy of Canada Limi

Source: AECL-Misc-252

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: Pipe failure rate estimation **ID:** 822

Abstract: Information on pipe failure events in operating commercial CANDU nuclear generation stations (NGS) in Canada was collected and analysed. The analysis comprises (i) failure rate calculations, (ii) classification of failure event aspects, and (iii) determination of significant correlations among the classifications. Results of the analysis are then examined and compared with available corresponding published results, particularly as reported in a recent analogous study of light water reactors in the United States. In their history of operation to 1981 June, no incidents of large pipe severance occurred in the primary heat transport system of the Canada CANDU-NGS.

Title: The Probability of Leakage in Piping Systems of Pressurized Water Reactors on the Basis of Fracture Mechanics and

Author: Beliczey, S. and Schulz H. **Corp. Author:**

Source: Nuclear Engineering and Design, Vol. 102:431-438

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Pipe failure probability **ID:** 823

Abstract: Probabilities of leakages in piping systems as used in risk studies up to now do not represent the present state-of-the-art. The goal of this investigation is to formulate a new set of probabilities of leakages in piping systems of German pressurized water reactors for the whole range of pipes which are of interest using the operating experience, the principles of the basic safety approach and fracture mechanics studies.

Title: Piping and Component Replacement in BWR Systems, Safety Assessment and Licensing Decisions

Author: Schulz, H., Mueller, W. **Corp. Author:**

Source: Nuclear Engineering and Design, Vol. 85:177-182

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: Operating experience **ID:** 824

Abstract: For a successful operation of NPPs it is important to demonstrate that major problems can be handled efficiently in terms of technical as well as regulatory actions to be taken. In the years 1973 to 1975 some cracks have been detected during the construction of piping systems in some German BWR plants. The materials used for the piping was 17-MnMoV-6-4 which is a precipitation hardening ferritic steel of higher strength. To evaluate the safety implications of the problems encountered, a thorough reassessment of all BWR plants under construction or in operation has been performed by the responsible state licensing authority and the "Reactor Safety Commission" on behalf of the Federal Ministry of the Interior. Furthermore, the effects of cracks and degraded material conditions on the load carrying capability of the components were investigated by supplementary research programs. The problems have been solved by reinspection and repair and to the major part by the replacement of the piping and components affected. The replacement has been performed in a very successful manner on a narrow time-scale due to the close cooperation of all parties involved. The quality of the piping and components achieved resulted in a considerable improvement of the whole system. Secondary safety measures like pipe restraints which have some potential for a negative impact on flexibility and accessibility could be removed in cases where the licensee applied for it.

Title: Integrity of Feedwater and Main Steam Piping in KWU Light Water Reactor Plants

Author: Bieselt, R. W. **Corp. Author:**

Source: Light Water Reactor Structural Integrity, Elsevier Applied Science Publishers, ISBN:0-85334-295-4, pp 285-302

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: Operating experience **ID:** 825

Abstract: The design and manufacture of the feedwater and series under construction were based on high quality requirements and involved considerable ependiture. A great deal of effort was particularly invested in improving the quality of those sections of the piping which are located inside the containment, and of the containment penetrations, as these are vital to plant safety. These efforts led to the solutions described in this paper which are designated to ensure system and component integrity both during norman operation and in the event of unlikely, but postulated, accidents. The high quality of the piping has raised the level of inherent safety such that, under certain conditions, pipe whip restraint no longer need be probided for postulated pipe breaks. Additional examples are taken from the fields of feedwater piping and piping supports to intriduce the newly developed component catalogue for the PWR Convoy Series with which considerable standardization of piping components and their installation can be achieved. Past experience from the installation of piping systems has led to the requirement of new quality standards regarding the installation of safety-related piping systems such as the main steam and feed water piping systems of light water reactor plants.

Title: A Study of Pipe Failures in U.S. Commercial Nuclear Power Plants

Author: Jamali, K. **Corp. Author:**

Source: Halliburton NUS Corporation, Unpublished report

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Pipe failure rate estimation **ID:** 826

Abstract: This study was undertaken principally to provide a nuclear plant pipe failure data base reflecting recent experience, and to provide an updated assessment of pipe failure rates. A by-product of the data analysis effort for the quantification of the failure rates was the generation of additional quantitative information on: plant aging effects on pipe failures; break-before-leak probabilities; plant outage times due to pipe failures; and qualitative information on plant systems involved in pipe failures (referred to as system effect), failure causes, failure locations, effect of materials, discovery methods, and corrective actions. Note, this work eventally led to the EPRI TR-100380 series reports.

Title: Pipe Failures in U. S. Commercial Nuclear Power Plants

Author: Jamali, K., Sursock, J.P. **Corp. Author:** Electric Power Research Institu

Source: EPRI TR-100380

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Pipe failure rate estimation **ID:** *827*

Abstract: Recent NRC mandates require utilities to perform probabilistic risk assessments as part of their individual plant examinations (IPEs). To date a significant number of IPEs have identified small-break loss-of-coolant accidents (LOCAs) as a major contributor to nuclear power plant risk. Most existing databases that address pipe failure rates have been based on judgmental estimates from industry experts. EPRI has developed a methodology and database that uses actual experiences to support failure rate calculations on a plant- or system-specific basis. This 1993 editon of EPRI TR-100380 includes about 100 actual pipe failures in U.S. plants whereas the 1992 edition included about 40 such failures.

Title: Risk Management of Petrochemical Facilities. Basic Concepts of Risk Analysis, Risk Assessment & Risk Reduction/C

Author: Lydell, B. **Corp. Author:** RSA Technologies

Source: RSA-R-95-05

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** English

Category: PSA / QRA passive component evaluations **ID:** *828*

Abstract: This document summarizes the general model structure for PSA and QRA applications. In the latter passive component failures are key risk contributors, and the potential piping, vessel, tank, etc. failures must be explicitly modeled. Most QRAs continue to rely on "old" failure data sets such as those of WASH-1400.

Title: Risk-Based Inspection - Developmet of Guidelines., Vol 2 - Part 1: Light Water Reactor (LWR) Nuclear Power Plant

Author: Balkey, K. R. **Corp. Author:** American Society of Mechanic

Source: ISBN 0-7918-0658-8

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: **ID:** *829*

Abstract:

Title: Water Chemistry and Materials Degradation in LWRs

Author: Torronen, KP, Hanninen, H. **Corp. Author:** OECD Nuclear Energy Agency

Source: OCDE/GD(95)2, Committee on the Safety of Nuclear Installations, pp21-36

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** English

Category: **ID:** *830*

Abstract:

Title: Short-term Degradation Mechanisms of Piping

Author: Morel, A.R., Reynes, L.J. **Corp. Author:**

Source: Nuclear Engineering and Desing, Vol. 133:37-40

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Operating experience **ID:** 831

Abstract: The operation of EDF's PWR plants has shown that components have been subjected to loadings higher than the design basis loads. For example, localized degradations on nuclear system pipes were found after relatively short times (10 - 10,000 hours). The main damage mechanisms involved are: erosion-cavitation; vibrational fatigue; corrosion fatigue. This paper addresses these damage modes and the mitigating steps taken to cope with them, together with the initiatives taken for future reactors.

Title: Erosion by Cavitation on Safety-related Piping Systems of French PWR Units

Author: Thoraval, G. **Corp. Author:** IAEA

Source: Corrosion and Erosion Aspects in Pressure Boundary Components of Light Water Reactors, IWG-RRPC-88-1, pp 31-40

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Erosion by cavitation **ID:** 832

Abstract: At Fessenheim-1 in 1982, September 9th, the technicians who were working on the RHRS to connect a new manual valve, downstream from a butterfly control valve, cut the pipe and noticed traces of erosion inside. After this first detection, this damage was discovered in pipes of several other units. After a first analysis, we hoped that only this system of the oldest units (Fessenheim and Bugey) was concerned. Nevertheless, we launched studies and further inspections to analyze the risks on the other units, and on other systems. The present conclusions are that many systems may be concerned, in many units, depending on the shape of the lines and on operating conditions. So we had to set up a specific maintenance program, to recommend particular operating procedures, and to study modifications for long term solutions.

Title: Vibration Induced Failures in Nuclear Piping Systems

Author: Weidenhammer, GH **Corp. Author:** 7th International Conference on

Source: ppD1/1:1-6

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: **ID:** 833

Abstract: A survey of existing documents was conducted to ascertain the extent of pipe crack problems that have been encountered in industry that are attributable to fatigue. The work reported herein is part of a continuing study that the NRC has undertaken regarding crack growth in nuclear components. This paper documets these problems and identifies the reactor plants and the piping system in which the crack(s) occurred. The information was taken from Licensee Event Reports (LER's) from 1969 to October 1982, from Nuclear Regulatory Commission (NRC) Office of Inspection and Enforcement (IE) Bulletins, and from other licensee and vendor reports. References (5) and (6) have been useful as background documents for this report.

Title: Failure Mechanisms in Nuclear Power Plant Piping Systems

Author: Bush, S.H. **Corp. Author:**

Source: Journal of Pressure Vessel Technology, Vol 114:389-395

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Operating experience **ID:** 834

Abstract: This paper will review how nuclear piping has failed in the pasr, suggest corrective measures to eliminate or mitigate such failures, and assess the value of current design procedures to predict such failures. An important first step is to define failure. Two definitions will be used. The first covers cases of cracking plus limited leakage rates, e.g., no more than a few gallons per minute. The second class of failure covers complete severance (double-ended guillotine break), gross fish-mouth failures where leaking is in hundreds of gallons per minute and large long cracks where leakage exceeds 50 gpm. The first class of failures complies with the assumptions inherent in leak-before-break; the second class of failures does not, and occurs with no advance warning.

Title: Thermal Fluctuations in Mixing Tees. Experience, Measurements, Prediction and Fixes.

Author: Nordgren, A. **Corp. Author:** Trans. 7th International confere

Source: Trans. 7th International conference on Structural Mechanics in Reactor Technology, pp D1/2:7-14

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: Thermal fatigue **ID:** 835

Abstract: During the 1980 annual refueling shutdown at the Barseback-2 BWR plant (570 MWe)), ultrasonic inspection of the feedwtare pipes indicated cracks on the inner surface of two pipe branch connections, where feedwater at 20-180 C is mixed with water at 270 C returning from the reactor coolant clean-up system. Metallographic examinations confirmed that the cracks were caused by thermal fatigue resulting from turbulent mixing of hot and cold water in the pipe branch connection. The paper briefly describes the development of a thermal mixer and the various investigations which were carried out.

Title: How to Analyze Reliability Data

Author: Nelson, W. **Corp. Author:**

Source: ISBN:0-87389-018-3

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: **ID:** 836

Abstract:

Title: Operational Monitoring in German Nuclear Power Plants

Author: Seibold, A., Bartonicek, J. and Kockelmann, H. **Corp. Author:**

Source: Nuclear Engineering and Design, Vol. 159:1-27

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** English

Category: ISI / leakage monitoring **ID:** 837

Abstract: The Atomic Energy Act requires that measures made feasible by state of the art technology be adopted to avoid damage that could be caused as the result of the construction and operation of a nuclear plant. This stipulation constitutes the basis for deriving requirements for planning, design, construction, operation and decommissioning. Ensuring the function and integrity of those components and systems that are relevant to plant safety is of major significance with regard to operation of a NPP. The basis for ensuring these features is laid in planning, design and construiction. Important as these foundations may be, it is absolutely essential to monitor the quality originally planned and achieved in an object as undeniably complex as a NPP. The RSK-Leitlinien fuer Druckwasserreaktoren incorporate fundamental requirements for design, materials, manufacturing, testing and examination, and operation. Meeting these requirements makes it possible to exclude catstrophic rupture of the components in the RCS pressure boundary (primary system), as has been demonstrated in detailed research and development work. The principle of plant monitoring and documentation (operational monitoring) implements redundancy in a significant manner within this concept. The monitoring techniques used in Germany have reached an advanced state of development and are still being optimized. Thus, operational monitoring is a major contributory factor in the safety and high availability of NPPs. It also provides a means of expanding our knowledge of life time expectation.

Title: Statistical Forecasting of Trends in Tubular Pressure Part Forced Outage Rates for Fossil Boilers

Author: Gallucchi, R., Moelling, D., Talbot, K. **Corp. Author:**

Source: Journal of Pressure Vessel Technology, Vol. 114:389-395

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Failure rate estimation **ID:** 838

Abstract: Statistical models for calculating age-dependent component failure rates and system unavailabilities have been combined into a flexible procedure to forecast trends in tubular pressure part forced outage rates for fossil boilers as a function of their ages. These models have been computerized, and the forecasting procedure has been applied to predicting trends at six fossil units of a specific utility. The analytical procedure is described, and its application to the example study is discussed.

Title: Strength of Materials and the Weibull Distribution

Author: Lindquist, E. **Corp. Author:**

Source: Probabilistic Engineering Mechanics, Vol. 9:191-194

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: **ID:** 839

Abstract:

Title: Experience of erosion and erosion-corrosion in nuclear steam turbines

Author: Hedstroem, M. **Corp. Author:**

Source: Corrosion and Erosion Aspects in Pressure Boundary Components of Light Water Reactors, IAEA, IWG-RRPC-88-1, 66-69

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Erosion-corrosion **ID:** 840

Abstract: An overview of erosion-corrosion experience at Swedish NPPs.

Title: Erosion/Corrosion Data Handling for Reliable NDE

Author: Bridgeman, J Shankar, R **Corp. Author:**

Source: Nuclear Engineering and Design, Vol. 131;285-297

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: **ID:** 841

Abstract:

Title: Acceptance Criteria For Structural Evaluation of Erosion-Corrosion Thinning in Carbon Steel Piping

Author: Gerber, T **Corp. Author:**

Source: Nuclear Engineering and Design, Vol.133:31-36

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: **ID:** 842

Abstract:

Title: Reactor Safety Study. An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants

Author: **Corp. Author:** U.S. Nuclear Regulatory Com

Source: WASH-1400 (NUREG/CR-75/014)

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1975 **Language:** English

Category: **ID:** 843

Abstract:

Title: Light Water Reactor Safety

Author: Pershagen, B **Corp. Author:**

Source: ISBN:0-08-035915-9,pp170-190

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1989 **Language:** English

Category: **ID:** 844

Abstract:

Title: Probabilistic Analysis of te Interfacing System Loss-of-Coolant Accident and Implications of Design Decisions

Author: Leverenz, F. **Corp. Author:**

Source: Nuclear Technology, Vol. 37:5-12

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1978 **Language:** English

Category: Passive component reliability **ID:** 845

Abstract: One of the important findings of the Reactor Safety Study (RSS) was the identification of the risk due to an interfacing system loss-of-coolant accident (LOCA); i.e., failure of interfaces between the high-pressure primary system and the low-pressure injection system (LPIS). Because equivalent interfaces exist in all PWRs (although not necessarily with the LPIS), yje U.S. NRC has included in its Standard Review Plan three equally acceptable

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Title: Austenitic steel piping testing exercises in PISC

Author: Doctor, S.R., Lemaitre, P. and Crutzen, S. **Corp. Author:**

Source: Nuclear Engineering and Design, Vol. 157:231-244

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** English

Category: ISI reliability **ID:** 846

Abstract: In this paper capability and reliability studies of NDT procedures for the inspection of wrought and cast stainless steel piping used in nuclear power plants will be presented. The capability study was designed to identify procedures that have the potential to detect and size defects and to discriminate between flawed and unflawed material. The reliability study was undertaken to quantify on real and realistic flaws in-service inspection performance (detection and false call capability) under realistic field conditions. Furthermore parametric studies were performed to complement the capability and reliability studies by evaluating the effect of important material and flaw variables. The specimens used in these studies were cast-to-cast, cast-to-wrought, and wrought-to-wrought pipework welds. The evaluation methods used to quantify the inspection performance were selected to be as comparable as possible to the PISC-II methods. These were adapted to allow also the evaluation of the effect of false calls. During the PISC-II screening exercise for the cast-to-cast stainless steel round robin test and other piping round robin studies, it was indeed found that false call probabilities were large and could not be ignored in the evaluation of the inspection performance. The matrix of samples has also been designed to allow the implementation of specific statistical analysis procedures for the evaluation of results such as for example the relative characteristic analysis.

Title: Swedes repair BWR thermal fatigue cracks

Author: Burkhart, D. **Corp. Author:**

Source: Nuclear Engineering and Design, Vol. 26, No. 314, pp 25-27

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1981 **Language:** English

Category: Thermal fatigue **ID:** 847

Abstract: The discovery of cracks in the feedwater and shutdown cooling systems of Sweden's Barseback 2 BWR last summer led to investigations in other Swedish nuclear power stations. Similar cracks were found and the defective parts repaired or replaced before being returned to service. The cause of the cracks has been evaluated and efforts are being made to prevent a recurrence.

Title: Source Terms and Frequency Estimates for Selected Accidental Hydrofluoric Acid Release Scenarios in the South Coa

Author: Beychok, M **Corp. Author:** PLG Inc.

Source: PLG-0787 (Rev. 1)

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Pipe failure probability **ID:** 849

Abstract: This study developed pipe failure probabilities for application to a refinery risk analysis. The analysts applied the Thomas Model and combined it with Bayesian analysis framework to generate process unit specific pipe failure rates and failure probabilities.

Title: Pressurized Therman Shock Evaluation of the Calvert Cliffs Unit 1 Nuclear Power Plant

Author: Selby, D. **Corp. Author:** Oak Ridge National Lab

Source: ORNL/TM-9408 (NUREG/CR-4022

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: PFM evaluation **ID:** 850

Abstract: An evaluation of the risk to the Calvert Cliffs-1 NPP due to pressurized thermal shock (PTS) has been completed by ORNL with the assistance of several other organizations. This evaluation was part of a NRC program designed to study the pTS risk to three nuclear plants, the other two plants being Oconee-1 and Robinson-3. The specific objectives of the program were to (1) provide a best estimate of the frequency of a through-wall crack in the pressure vessel at each of the three plants, together with the uncertainty in the estimated frequency and its sensitivity to the variables used in the evaluation; (2) determine the dominant overcooling sequences contributing to the estimated frequency and the associated failures in the plant systems or in operator actions; and (3) evaluate the effectiveness of potential corrective masures.

Title: Seminar on the Safety of Reactor Pressure Vessels, SKI Technical Report Dnr 647/86

Author: Swedish Nuclear Power Inspectorate **Corp. Author:**

Source: SKI Technical Report Dnr. 647/86

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1986 **Language:** Swedish & English

Category: RPV reliability **ID:** 851

Abstract: On Friday, September 19, 1986, SKI organized a seminar on the reliability of RPVs. A group of Nordix experts presented papers on failure mechanisms and the probability of a vessel failure. The seminar proceedings summarize the then available state-of-knowledge about RPV reliability.

Title: Midland Nuclear Plant Probabilistic Risk Assessment

Author: **Corp. Author:** PLG, Inc

Source: Prepared for Consumers Power Company

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: PSA / passive components **ID:** 852

Abstract: The systems analysis procedure demonstrates how the impact of piping failure on safety system unavailability is addressed in PSA. This procedure was adopted by all PSAs performed by PLG.

Title: Aging Effects on Time-Dependent Nuclear Plant Component Unavailability: An Investigation of Variations From Stat

Author: Radulovich, R.D., W.E. Vesely, T. Aldemir **Corp. Author:**

Source: Nuclear Technology, Vol. 21:21-40

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** English

Category: Aging analysis **ID:** 853

Abstract: In the nuclear industry, aging effects have been traditionally incorporated into PRA studies by using a constant (static) unavailability averaged over time. However, recent work shows that because of aging, substantial deviations may occur in time-dependent nuclear plant component unavailability from that predicted by static models well within the plant lifetime. A methodology based on the standard extension of the classic renewal equation when repair is explicitly considered is used to investigate (a) trends in the effects of aging on time-dependent component unavailability as a function of changing first failure density (FFD) and test parameters and (b) the circumstances for which static approximations may be inadequate to describe these effects. The investigation uses several point- and time-averaged unavailability measures based on time-dependent unavailability, such as before-test unavailability (BTU), average-interval unavailability (AIU) and year-average unavailability (YAU), and is restricted to periodically tested components whose FFDs satisfy the Weibull distribution with aging threshold. The results show that while point measures (e.g., BTU) can substantially differ from static unavailability and while all measures are sensitive to changes in the Weibull shape parameter, aging threshold time, and time between tests, the differences between the time-averaged measures used (e.g., AIU, YAU) and the static unavailability were only found to be relatively significant for one case among more than 100 combinations of Weibull parameters that were investigated. The differences are a factor of < 2 for all other cases, which is within the uncertainty margin on the data used in the study. The results also show that the static unavailability may be an adequate measure for shape parameters < 2 and high test intervals (> 18 months) and may describe the late effects of aging on component unavailability irrespective of shape parameter or test interval (i.e., beyond 25 yr of component age for the data under consideration.

Title: Weld Repairs in Swedish Nuclear Power Plants. Results from TUD Data Search.

Author: **Corp. Author:** Swedish Nuclear Power Inspect

Source: SKI/RA-004/95

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** Swedish

Category: Operating experience **ID:** 854

Abstract: Survey of TUD data base for weld repairs of pressure boundary components yielded about 800 reports.

Title: Development of a Leak-before-break Methodology

Author: Munz, D. **Corp. Author:**

Source: Structural Mechanics in Reactor Technology: Advances 1987, A.A. Balkema, ISBN:90 6191 738 7, pp 155-174

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: LBB methodology **ID:** 855

Abstract: Pressurized components have to be designed against different failure modes: buckling, excessive plastic deformation by exceeding of the yield strength or by creep or due to ratchetting, creep rupture, fatigue. Very often fatigue is the most critical failure mode. For this type of failure two different modes of behaviour are possible. One is the occurrence of a leak after a crack has penetrated the wall. If this leak can be detected, the component can be repaired or replaced without any severe consequence for the environment. If the component is not replaced the crack may grow further subcritically until a rapid unstable crack extension occurs, when the crack has attained a critical length. If under all possible circumstances there is sufficient margin between leak occurrence and final rupture and the leak can be detected, this behavious is called leak-before-break. The second possible behaviour is an unstable crack extension immediately after the crack has penetrated the wall or after some further stable extension before the appearnce of leak can be detected. This may be called break-before-leak behaviour. In this paper the general procedure of a leak-before-break analysis is outlined. It is based on a document prepared by a German working group and additional discussion within the European working group "Flaw Evaluation" for fast breeder components.

Title: An Overview of the Leak-before-Break Concept in Relation to Nuclear Power Plant

Author: Darlaston, B.J. **Corp. Author:**

Source: Nucleon, No 3, pp 4-6

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: LBB methodology **ID:** 856

Abstract: With high technology plant such as nuclear units, safety is usually ensured by good design, high standard of fabrication and inspection coupled with well controlled and maintained operation. Leak-befire-break concept provides a further level of safety. The paper describes the LBB concept, the regulatory requirements and the application of the concept to the plant to ensure that safety care requirements are met.

Title: Lessons Learned from Application of the LBB Concepts to NPPs with VVER 440 Type 213 Reactors

Author: Pecinda, L, Zdarek, J. **Corp. Author:** Nuclear Reaseach Institute

Source: Nuclear Reaseach Institute, Czech Republic.

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: LBB methodology **ID:** 857

Abstract: As part of safety enhancement of nuclear power plants with VVER Type 213 reactors the leak-before-break concept has been applied to all NPPs operated in Czech and Slovak Republics.

Title: The Reliability of Ultrasonic Inspection for Thick Section Welds: Some Views and Model Calculations

Author: Coffey, J. **Corp. Author:**

Source: Periodic Inspecction of Pressurized Components, I Mech E Conference Publications 1982-9, pp 273-282

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1982 **Language:** English

Category: ISI reliability **ID:** 858

Abstract: The paper is concerned with the possibilities of giving a quantified, statistical measure of the reliability of manual and automatic ultrasonic inspections, Some views are expressed concerning the relative possible benefits to be gained from repeating manual and automated examinations. Also a view is presented of the types of error and failure which might occur in defect detection and sizing; this points to the value of trying to anticipate the unexpected so as to prevent any significant errors remaining undetected. The paper then presents two lines of reasoning, both involving calculations of the ultrasonic responses of cracks, by which certain factors contributing to the overall reliability can be quantified. The first underwrites the capability of the inspection procedure which was used successfully by the CEGB in the UKAEA Defect Detection Trials to detect all significant defects. The second is a preliminary analysis of the circumstances in which unexpectedly large errors in defect sizing could possibly occur. The chance of such errors occuring in a properly conducted fully automated inspection is believed to be very small.

Title: Effectiveness and Reliability of U.S. In-Service Inspection Techniques

Author: Doctor, S.R., Becker, F.L. and Selby, G.P> **Corp. Author:**

Source: Periodic Inspection of Pressurized Components, I Mech E Conference Publications 1982-9, pp 291-294

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1982 **Language:** English

Category: ISI reliability **ID:** 859

Abstract: The work presented is from an on-going program directed toward measuring the effectiveness and reliability of inservice inspection (ISI) of LWR systems (primary piping and pressure vessel). Extensive round robin and parametric evaluations have been conducted in 10" Sch 80 stainless steel as well as centrifugally cast stainless steel and clad ferritic main coolant pipe welds. The results from these measurements will be viewed in relationship to U.S. regulations and ASME Section XI Code requirements.

Title: Refinery Pipework Reliability Study

Author: Lydell, B. **Corp. Author:** RSA Technologies

Source: RSA-R-94-04:1

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: Failure rate estimation **ID:** 860

Abstract: Review and interpretation of petroleum refinery work order information and inspection records dealing with carbon steel piping repair and replacements. Primary failure mechanism addressed is wall thinning due to corrosion. Analysis results were used for a risk assessment (QRA) of an HF Alkylation processing unit that considered impact of ISI.

Title: Assesment and avoidance of erosion-corrosion damage in PWR feedpipework

Author: Woolsey, I.S. **Corp. Author:**

Source: Corrosion and Erosion Aspects in Pressure Boundary Components of Light Water Reactors, IAEA, IWG-RRPC-88-1, pp 60-65

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Erosion-corrosion **ID:** 861

Abstract: Following the main feedline rupture at the Surry-2 PWR, CEGB undertook an evaluation of the possibility of similar damage in the feedpipework of other PWRs including future UK designs. The assessment method was based on an extensive body of experimental erosion-corrosion data accumulated during investigations of possible single phase erosion-corrosion in the low temperature sections (100 to 200 C) of UK-AGRs. The analysis focussed on the materials' specification required to avoid significant erosion-corrosion damage throughout the feedpipework, taking account of pipework configuration, flow rates, temperature and water chemistry. It allowed identification of locations which would be potentially vulnerable to unacceptable erosion-corrosion damage over the operational life of the plant. However, significant damage could be avoided by adopting a minimum chromium specification for the carbon steel pipework, and a sufficiently high operational feedwater pH. For the majority of feedpipework it should not be necessary to use a chromium alloy steel. By adopting these measures, it is considered that the UK PWR currently under construction at Sizewell will not suffer significant erosion-corrosion damage of the main feedpipework over the full period of its operational life.

Title: Pipeline Risk Management Manuel

Author: Muhlbauer, W. Kent **Corp. Author:**

Source: Gulf Publishing Company, Houston (TX), ISBN 0-88415-035-6

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Loss prevention / risk assessment **ID:** 862

Abstract: The text includes a proposed approach to pipeline risk management. The book is organized to serve as a guide for persons performing pipeline risk assessments. Risk assessment does not have to be calculation-intensive exercise in probabilistic theory. Such calculations are, after all, based upon probabilities that are of questionable benefit in rare-occurrence scenarios. A false precision is often assigned to numbers that are the result of detailed calculations. In reality, the margin of uncertainty is quite high because of the large number of assumptions required in such analyses. The approach used in this book is to deviate from strict scientific procedure in building this risk model. In many situations, some risk aspects are based as much upon hard evidence. Rather than being seen as a detraction, theauthor believes that this approach strengthens the risk management process.

Title: In-Service Reliability Data for Underground Cross-Country Oil Pipelines

Author: Blything, K. W. **Corp. Author:** UKAEA

Source: Safety and Relaibility Directorate, UKAEA, SRD R 326

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1984 **Language:** English

Category: Failure data **ID:** *863*

Abstract: This study is concerned with the integrity of proposed cross-country underground pipelines in the UK and the hazard these may present to the community. It is probable that constructors of pipelines for the transport of potentially hazardous flluids will have to prepare a safety evaluation as part of their application for construction authorization under the Pipelines Act 1962. In this study the aim has been to specify mean failure rates for different pipelines categories, which have been classified by failure cause, fluid carried and pipe size, also to specify mean defect sizes for the various failure cause categories. Armed with this data it should be possible to carry out a meaningful review of the safety evaluation for individual pipelines. A data base has been prepared which specifies mean failure rate with 95% confidence limits for four cause categories over a range of diameters. This is supported by a review of other factors in the analysis and discussion sections and, where possible, their effect of failure probability is quantified.

Title: Interpretation of Probabilistic Structural Analysis of an Aging Passive Component

Author: Phillips, J et al **Corp. Author:**

Source: Journal of Pressure Vessel Technology, Vol. 116, pp 295-301

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: PSA / PFM evaluation **ID:** *864*

Abstract: This article describes a technique to calculate the risk from failure of passive components over time, and demonstrates the technique by applying it to a weld in the auxiliary feedwater (AFW) system. It uses a modified version of the PRAISE computer code to perform a probabilistic structural analysis to calculate the probability that crack growth due to aging would cause the weld to rupture. It then uses the weld rupture probability as input to a modified existing PRA to calculate the change in plant risk with time. The results show an insignificant effect on plant risk because of the low calculated rupture rate of the weld in this particular calculation over 48 yr of service. A decreasing yearly rupture rate for this weld is calculated. This results from infant mortality; that is, most of those initial flaws that will eventually lead to rupture will do so early in life.

Title: Cracking in a Reducing Pipe from a Pressurized Water Reactor

Author: Czajkowski, C. **Corp. Author:**

Source: Handbook of Case Histories in Failure Analysis, Vol;. 2, pp163-167

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: **ID:** *865*

Abstract: Three ASME SA106 grade B carbon steel feedwater piping reducers from a pressurized water reactor showed indications of flaws near welds during ultrasonic testing. Further examination and testing indicated that the cracks resulted from a low-cycle corrosion fatigue phenomenon.

Title: Interpretation of Risk Significance of Passive Component Aging using Probabilistic Structural Analysis

Author: Phillips, J. H., Atwood, C. **Corp. Author:** American Society of Mechanic

Source: Reliability and Risk in Pressure Vessels and Pipe, PVP-Vol. 251:153-162

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: Aging analysis **ID:** 866

Abstract: The PRAs generally focus on the prossible failure of active components. Except as initiating events, the possible failure of pasive components is given litle consideration. The NRC is sponsoring a project at INEL to investigate the risk significance of passive components as they age. For this project, we developed a technique to calculate the failure probability of passive components over time, and demonstrated the technique by applying it to a weld in the auxiliary feedwater (AFW) system. The selection of this component was based on expert judgement of the likelihood of failure and on an estimate of the consequence of component failure to plant safety. We used a modified version of the PRAISE computer code to perform a probabilistic structural analysis to calculate the probability that crack growth due to aging would cause the weld to rupture. We modified an existing PRA (NUREG-1150 plant) to include the possible rupture of the AFW weld, and then we used the weld rupture probability as input to the modified PRA to calculate the change in plant risk with time. The results showed an insignificant effect on plant risk because of the low calculated rupture rate of the weld in this particular calculation over 48 years of service. However, the most interesting observation was the rupture rate trend for this 48 years. A decreasing yearly rupture rate for this weld was calculated instead of the increasing rupture rate trend one might expect. We attribute this result to infant mortality; that is, most of those initial flaws that will eventually lead to repture will do so early in life. This means that although each weld in a population may be wearing out, the population as a whole can exhibit a decreasing rupture rate. This observation has implications for passive components in commercial nuclear plants and other facilities where aging is a concern. For the population of passive components that exhibit a decreasing failure rate, risk increase is not a concern. The next step of the work is to identify the attributes that contribute to this decreasing rate, and to determine any attributes that would contribute to an increasing failure rate and thus to an increased risk.

Title: Study on Life Extension of aged RPV Material Based on Proabilistic Fracture Mechanics: Japanese Round Robin

Author: Yagawa, G. and others **Corp. Author:**

Source: Journal of Pressure Vessel Technology, Vol. 117:7-13

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** English

Category: Life extension **ID:** 867

Abstract: Round-robin analyses of PFM problems of aged RPV material. A plate with a semi-elliptical surface crack subjected to various cyclic tensile and bending stresses is analyzed . A depth and an aspect ratio of the surface crack are asumed to be probabilistic variables. Failure probabilities are calculated using Monte Carlo methods with the importance sampling or the stratified sampling techniques. Material properties are chosen from the Marshall report, the ASME Code Section XI, and the experiments on a Japanese RPV material carried out by the Life Evaluation (LE) subcommittee of the Japan Welding Engineering Society (JWES), while loads are determined referring to design loading conditions of PWRs. Seven organizations participated in the study. At first, the procedures for obtaining reliable PFM solutions with low failure probabilities are examined by solving a unique problem with seven computer programs. The seven solutions agree very well with oe another,i.e., by a factor of 2 to 5 in failure probabilities. Next, sensitivity analyses are performed by varying fracture toughness valves, loading conditions, and pre and in-service inspections. Finally, life extension simulations based on the PFM analyses are performed. It is demonstrated that failure probabilities are so sensitive to the change of fracture toughness values that the degree of neutron irradiation significantly influences the judgement of plant life entension.

Title: Vibration Induced Failures in Nuclear Piping Systems

Author: Weidenhamer, G.H. **Corp. Author:**

Source: Trans. 7th International conference on structural Mechanics in reactor Technology, North Holland Physics Publishing, pp 1-6

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: Operating experience **ID:** 868

Abstract: A survey of existing documents was conducted to ascertain the extent of pipe crack problems that have been encounterd in industry that are attributable to fatigue. The work reported herein is part of a continuing study that the NRC has undertaken regarding crack growth in nuclear components. This paper documents these problems and identifies the reactor plants and the piping system in which the cracking occurred. The information was taken from the LERs from 1969 to 1982, from NRC Office of Inspection and Enforcement (IE) Bulletins, and from other licensee and vendor reports, plus NUREG-0679 and NUREG-0691.

Title: Results of reliability test program on light water reactor piping

Author: Shibata, K. and others **Corp. Author:** JAERI

Source: Nuclear Engineering and Design 153 (1994) 71-86

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: LBB methodology **ID:** 869

Abstract: The Japan Atomic Energy Research Institute has conducted a piping reliability test program to demonstrate the safety and reliability of light water reactor primary piping. In this program, pipe fatigue test, leak-before-break (LBB)verification test and pipe rupture test were carried out to examine the integrity of piping, to verify the LBB and to demonstrate the effectiveness of proective measures against jet impingement and pipe whip loads under a pipe rupture event. In the pipe fatigue test, a procedure to predict the fatigue crack growth was developed, and the integrity of piping during the plant service life was evaluated. In the LBB verification test, the pipe fracture test and the leak rate test were performed to verify the LBB in the primary piping. In the pipe rupture test, the influence of jet impingement on the target disk and the deformation behavior of whipping pipe and retraint were investigated. Using the test results, the jet impingement behavior and the effectiveness of pipe whip restraint were demonstrated.

Title: Service Experience with Corrosion Problems in LWRs

Author: Stahlkopf, K. & others **Corp. Author:** Electric Power Research Institu

Source: Trans. 8th International Conference on Structural Mechanics in Reactor Technology, North Holland Physics Publishing, pp 327-332

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: Operating experience **ID:** 870

Abstract: Corrosion damage of a wide variety of structural materials used in LWRs has caused significant forced outages with associated high costs for repairs and replacement power. The most prevalent and costly form of corrosion degradation in service is stress corrosin cracking. This paper reviews service incidents of corrosion cracking in components of the LWR pressure boundary, with emphasis on pipe cracking in BWRs and steam generator problems in PWRs. Remedial actions are identified that address stress reduction, material improvements, or control of the environment, as appropriate to each category of service application.

Title: Survey of Operating Experience to Identify Structural Degradation of Nuclear Power Plant Components

Author: Murphy, G. et al **Corp. Author:** Oak Ridge National Lab

Source: Trans. 8th International Conference on Structural Mechanics in Reactor Technology, North Holland Physics Publishing, pp 327-332

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: Operating experience **ID:** 871

Abstract: An assessment of the information available in NPP-LERs pertinent to identifying failures due to age-related degradation was performed by the Nuclear Operations Analysis Center (NOAC) staff at Oak Ridge National Laboratory. LERs from commercial power plants submitted from 1969 to 1982 were surveyed yielding 3098 events considered age-related failures. Wear, corrosion, fatigue, vibration, stress corrosion, and erosion were the identified structural failure cause mechanisms in over half of the events. The study contains data on failed components, the age-related failure mechanisms responsible, the severity of the failure, and the failure detection methods of failures from possible age-related causes.

Title: Stress-Corrosioin Cracking Experience in Piping of Light Water Reactor Power Plants

Author: Shao, L. & Burns, J. **Corp. Author:**

Source: Nuclear Engineering and Design, Vol. 57:133-140

SKI Project File: Ja **Transfer:** Ja **Publ year:** 1980 **Language:** English

Category: IGSCC **ID:** 872

Abstract: Cracking has been observed in the heat-affected zones (HAZ) of welds that join small diameter austenitic steel piping and associated components in BWRs. It was concluded that much of this was caused by IGSCC. In 1975 the U.S. NRC established a pipe crack study group to investigate this cracking in order to minimize and curtail this phenomena. In 1978 IGSCC was observed for the first time in large-diameter piping. A second pipe crack study group was formed, with an expanded charter, to continue investigations and answer specific questions concerning pipe cracking. This paper summarizes the results of the two study group investigations and present the major conclusions and recommendations regarding the causes, detection and control of such pipe cracking. Also discussed is the history of observed cracking, metallurgy associated with IGSCC, the effects of the primary coolant chemistry, developed stress levels in the HAZ of piping, methods of crack detection, and the importance of leak detection.

Title: Experimental Investigation of the stratified flow in the horizontal pipework of nuclear reactors

Author: Jud, E. & others **Corp. Author:**

Source: Nuclear Engineering and Design, Vol. 153:173-181

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** English

Category: Operating experience **ID:** 873

Abstract: Laboratory tests have been performed to study the behaviour of a stratified fluid flow in horizontal piping. Concentrated calcium chloride brine and fresh water at room temperature were used to model the density difference due to thermal effects in nuclear reactors. Flow phenomena relating to the interface and mixing of the two laters were observed through the plexiglass piping by means of a flow visualization technique involving a longitudinal laser beam section through the flow. The measurements have established that, at the representative flow conditions modelled by the experiments, despite the presence of strong surface waves on the interface, little mixing occurs between the two layers. Quantitative correlations for the depth of the interface between the two layers, its surface slope and the height of surface waves have been established by the definition and use of two dimensionless Froude numbers.

Title: Acceptance criteria for structural evaluation of erosion-corrosion thinning in carbon steel piping

Author: Gerber, T. et. al **Corp. Author:**

Source: Nuclear Engineering and Design, Vol. 133:31-36

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Operating experience **ID:** 874

Abstract: This paper provides an overview of recently developed acceptance criteria for the evaluation of carbon steel piping erosion-corrosion wall thinning. Criteria are based on code design requirements. They define the depth and extent of wall thinning that can be safely left in service.

Title: Piping and component replacement in BWR Systems; Safety assessment and licensing decisions

Author: Schulz, H. & Mueller, W. **Corp. Author:** GRS

Source: Nuclear Engineering and Design, Vol. 85:177-182

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: Operating experience **ID:** 875

Abstract: For a successful operation of NPPs it is important to demonstrate that major problems can be handled efficiently in terms of technical as well as regulatory actions to be taken. In the years 1973 to 1975 some cracks have been detected during construction of piping systems in some German BWRs. The materials used for the piping was 17MnMoV 6 4 which is a precipitation hardening ferritic steel of a higher strength. To evaluate the safety implications of the problems encountered, a thorough reassessment of all BWRs under construction or in operation has been performed by the responsible state licensing authority and the Reactor Safety Commission on behalf of the Federal Ministry of the Interior. Furthermore, the effects of cracks and degraded material conditions on the load carrying capability of the components were investigated by supplementary research programs. The problems have been solved by reinspection and repair and to the major part by the replacement of the piping and components affected. The replacement has been performed in a very successful manner on a narrow time-scale due to the close cooperation of all parties involved. The quality of the piping and components achieved resulted in a considerable improvement of the whole system. Secondary safety measures like pipe restraints which have some potential for negative impact on flexibility and accessibility could be removed in cases where the licensee applied for it.

Title: Corrosion and Erosion Aspects in Pressure Boundary Components of Light Water Reactors

Author: International Working Group on Reliability of Reactor Pressure Components (IWG-RRPC) **Corp. Author:** IAEA

Source: IWG-RRPC-88-1

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Operating experience **ID:** 876

Abstract: These proceedings include 12 papers on the erosion-corrosion experience at U.S., Russian and European NPPs.

Title: Primary Coolant Leak at KOLA-2 NPP Due to rupture of a make-up pipe

Author: Stueck, R. et al **Corp. Author:** IAEA

Source: WWER-SC-112 (Draft)

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** English

Category: Incident report **ID:** 877

Abstract: An in-depth study on the "Primary System Coolant Leak at NPP Kola-2" was conducted from 28 November to 2 December at the IAEA Headquarters. The specific objectives of this IRS (Incident Reporting System) meeting were to (1) discuss in detail information on the "Kola event", provided by the Russian experts; (2) to evaluate actions to prevent recurrence of similar events; and (3) to draw generic lessons for improving WWER safety.

Title: Integrity of Feedwater and Main Steam Piping in KWU Light Water Reactor Plants

Author: Bieselt, R. et. al **Corp. Author:** KWU

Source: Light Water Reactor Structural Integrity, ISBN 0-85334-295-4, pp 285-302

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: Operating experience **ID:** 878

Abstract: The design and manufacture of the feedwater and main steam piping used in the recently backfitted German BWR plants and in the standardized PWR plants of the KWU Convoy Series under construction were based on high quality requirements and involved considerable expenditure. A great deal of effort was particularly invested in improving the quality of those sections of the piping which are located inside the containment, and of the containment penetrations, as these are vital to plant safety. These effort led to the solutions described in this paper which are designed to ensure system and component integrity both during normal operation and in the event of unlikely, but postulated, accidents. The high quality of piping has raised the level of inherent safety such that, under certain conditions, pipe whip restraints no longer need be provided for postulated pipe breaks.

Title: A Review of Recent Incidents of BWR Pipe Cracking

Author: Danko, J. & Stahlkopf, K. **Corp. Author:**

Source: Light Water Reactor Structural Integrity, ISBN 0-85334-295-4, pp 381-381-392

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: Operating experience **ID:** 879

Abstract: During the last eighteen months incidents of intergranular stress corrosion cracking of Types 304 and 316 stainless steel piping systems in boiling water reactors haave shown a significant increase. These incidents have occurred in 305 mm (12 in.) discharge risers, 559 mm (22 in.) end caps to manifold, manifold to sweepolet and 710 mm (28 in.) diameter 316 stainless steel recirculation piping. To minimize effects on scheduled outages and radiation exposure of personnel in the repairs, a weld overlay process and flawed pipe analysis was successfully used. For the replacement of piping systems, 316 Nuclear Grade stainless steel was used.

Title: On the failure probability of pipings

Author: Schueller, G., A. Tsurui and J. Nienstedt **Corp. Author:**

Source: Nuclear Engineering and Design, Vol. 128:201-206

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Pipe failure probability **ID:** 880

Abstract: Various methods for determining the structural reliability analysis of piping systems of NPP's are discussed in view of their accuracy, efficiency and possibility of practical applications. Ultimate load as well as fatigue failure modes are considered in the analysis. The time variant reliability problem, e.g., due to fatigue and/or corrosion is solved by utilizing advanced simulation procedures.

Title: Inspection of Piping, Tubing, Valves, and Fittings

Author: **Corp. Author:** API, Washington (DC)

Source: Recommended Practice 574, First Edition, June 1990

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1990 **Language:** English

Category: Pipe inspection guidelines **ID:** 881

Abstract: This recommended practice includes pipe inspection guidelines for oil refinery operations. Primary concern is monitoring of wall thinning due to corrosion and erosion-corrosion.

Title: Risk-Based Inspection-Development of Guidelines

Author: Balkey, K., et al **Corp. Author:** ASME

Source: Volume 1 General Document, CRTS-Vol.20-1, ISBN 0-7918-0618-9

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1991 **Language:** English

Category: Risk-based inspection **ID:** 882

Abstract: This general document, volume 1, describes and recommends appropriate processes and methods using risk-based information to establish inspection guidelines for facilities or structural systems. This general document is to be used in conjunction with supplemental volumes that apply to specific types of systems and which are currently under preparation for several applications. All of these documents may be employed by users in the development and implementation of their inspection programs. These guidelines may be used by code groups to prepare new or revised codes and standards. However, further results from pilot applications may be required to provide the technical basis for actual codes and standards changes.

Title: New Initiating Event Frequencies for loss of Coolant Accidnets and Steam Generator Tube Rupture for KCB

Author: Zadel, A. **Corp. Author:** KEMA Nuclear

Source: 40198-NUC 93-4424, update 931112

SKI Project File: Ja **Transfer:** Ja **Publ year:** 1993 **Language:** English

Category: PSA, IE-frequencies **ID:** 883

Abstract: In this report new initiating event frequencies for Loss of Coolant Accidents (LOCAs) and Steam Genertor Tube Rupture (SGTR) for the Borssele Nuclear Power Plant (KCB) are determined. The LOCA and SGTR frequencies that were used in the KCB PSA up to now came from generic data sources (GRS-B, NUREG/CR-4550). In order to improve the PSA, plant specific analyses are performed generating new LOCA and SGTR initiating event frequencies.

Title: Lessons Learned from Application of the LBB Concept to the NPPs with VVER 440 Type 213 Reactors

Author: Pecinka, L. & Zdarek, J. **Corp. Author:**

Source: Division of Integrity and Materials, Nuclear Research Institute, Czech Republic

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** Englsih

Category: LBB methodology **ID:** 884

Abstract: As the part of safety enhancement of Nuclear power plants with VVER type213 reactors the leak before break concept have been applied to all NPPs operated in Czech and Slovak Republics.

Title: The Probability of Leakage in Piping Systems of Pressurized Water Reactors on the basis of Fracture Mechanics and

Author: Beliczey, S. & Schulz, H. **Corp. Author:**

Source: Nuclear Engineering and Design, Vol. 102:431-438

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Pipe failure probability **ID:** 885

Abstract: Probabilities of leakages in piping systems as used in risk studies up to now do not represent the present state of the art. The goal of this investigation is to formulate a new set of probabilities of leakages in piping systems of German pressurized water reactors for the whole range of pipes which are of interest using the operating experience, the principles of the basis safety approach and fracture mechanics studies.

Title: Comments on the Probability of Leakage in Piping Systems as used in PRAs

Author: Schulz, H. **Corp. Author:**

Source: Nuclear Engineering and Design, Vol. 110:229-232

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Pipe failure probability **ID:** 886

Abstract: In risk analysis of power reactors the leakage or failure of piping structures has to be taken into account as a possible cause of loss of coolant. As part of the SMiRT post conference seminar on "PRA of NPP for External Events" the present practice of selecting pipe failure rates as initiating or related events for PRA's has been discussed. For pipe failures as initiating events an approach has been developed in the framework of the risk study for German PWR's. As compared to NUREG-1150 some significant differences are identified. For external events the effect of seismic induced loads on pipe failure has been a subject of considerable effforts in research. Several studies have demonstrated that for moderate siting conditions the effect of seismic induced loads on pipe failure rates of large diameter high pressure piping does not lead to significant contributions to the overall risk. The main subject for future research on pipe failure mechanisms is the detailed asessment of the influence of the water chemistry conditions.

Title: Vibration Induced Failures in Nuclear Piping Systems

Author: Weidenamer, G. H. **Corp. Author:**

Source: Trans. 7th International SMiRT Conference

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: Operating experience **ID:** 887

Abstract: The results of earlier work show that a very small crack can grow to exceed acceptance standards if the crack is subjected to vibrating loadings with large usage factors. This is particularly true for a surface crack directly influenced by a hot water environment of a Light Water Reactor (LWR). A thick wall pressure vessel was the component considered in the calculatios of this earlier work. The work reported in this paper supplements this earlier work. Specifically, this paper reports on the results of a survey that was conducted to determine the extent of cracks that have occurred in piping systems and that are attributable to fatigue.

Title: Piping Failures in United States Nuclear Power Plants: 1961-1995

Author: S.H. Bush, M.J. Do, A.L. Slavich, A.D. Chockie **Corp. Author:**

Source: SKI Report 96:20, Swedish Nuclear Power Inspectorate, Stockholm (Sweden)

SKI Project File: Ja **Transfer:** Ja **Publ year:** 1996 **Language:** English

Category: Pipe failure data **ID:** 888

Abstract: Over 1500 reported piping failures were identified and summarized based on an extensive review of tens of thousands of event reports that have been submitted to the U.S. regulatory agencies over the last 35 years. The process of locating and assessing these event reports was made difficult due to the fact that the reports are distributed among a number of data systems and document storage centers. The data base contain only piping failures; failures in vessels, pumps, valves, and steam generators, or any cracks that were not through-wall are not included. The data base contains publicly available data for events from December 1961 through October 1995.

Title: Risk Evaluations of Aging Phenomena: The Linear Aging Reliability Model and Its Extensions

Author: Vesely, W.E. **Corp. Author:** SAIC

Source: NUREG/CR-4769 (EGG-2476)

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Aging analysis **ID:** 889

Abstract: A model for LWR safety system component failure rates due to aging mechanisms has been developed from basic phenomenological considerations. In the treatment, the occurrences of deterioration are modeled as following a Poisson process. The severity of damage is allowed to have any distribution, however, the damage is assumed to accumulate independently. Finallt, the failure rate is modeled as being proportional to the accumulated damage. Using this treatment, the linear aging failure rate model is obtained. The applicability of the linear aging model to various mechanisms is discussed. The model is also extended to cover nonlinear and dependent aging phenomena. The implementation of the linear aging model is demonstrated by applying it to the aging data collected in the U.S.NRC's Nuclear Plant Aging Research Program. Appendix A of the report includes an evaluation of aging in SWS piping.

Title: Applications of Probabilistic Fracture Mechanics to Light Water Reactor Pressure Vessels and Piping

Author: Lidiard, A.B. **Corp. Author:**

Source: Nuclear Engineering and Design 60 (1980) 49-56

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1980 **Language:** English

Category: Pipe reliability / PFM **ID:** 890

Abstract: This paper reviews recent calculations of the statistical reliability of LWR reactor vessels and piping. The broad theoretical principles of these calculations are well estalished and it is therefore possible to compare the physical assumptions made in different calculations. Such a comparison shows that certain functions are not known at all well; for example, (i) the frequency of occurrence of cracks in weld-regions, (ii) the size distribution of cracks, (iii) the efficiency of methods of non-destructive examination and (iv) the transient loadings that the system experiences in service. On the other hand, relevant materials properties (toughness, crack growth characteristics) appear to be known adequately if not completely. Despite these quantitative uncertainties in the input, it seems possible to draw several broad conclusions from the results of these calculations. These concern (i) the low absolute rates of failure, (ii) the way these depend upon time in service, (iii) the effect upon them of in-service inspection and (iv) their sensitivity or otherwise to the physical assumptions which are made.

Title: Research for the Rationalization of Nuclear Power Plant Pipe Break Criteria

Author: Ayres, D.J. **Corp. Author:**

Source: Nuclear Engineering and Design 59 (1980) 117-126

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1980 **Language:** English

Category: Design basis / LOCA **ID:** 891

Abstract: The need for a new design basis for pipe break criteria is demonstrated by noting the potential deleterious effect of present cirteria in piping during normal operation. Recent advances in fracture mechanics and stress analysis permit development of rational, realistic and conservative criteria that will make possible significant imporvements in piping system design. Research needed to form the basis for new criteria is suggested and the nuclear indusry is encouraged to work towards this goal.

Title: Analysis of Pipe Failures in Swedish Nuclear Plants

Author: Petersen, K. E. **Corp. Author:**

Source: Proc. 4th EuReData Conference, Venice (Italy), March 23-25, Session 8

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1983 **Language:** English

Category: Pipe failure data **ID:** 892

Abstract: The paper presents an analysis of pipe failures in Swedish nuclear power plants performed at Riso. The two main goals of the analysis are: (i) to estimate the probability of a severe pipe failure which has the potential to cause a severe accident, (ii) to estimate the probability of apipe failure which causes a repair, which again has an influence on the operation of the plant or the reliability of a safety system. The task is done by performing a detailed evaluation of the incident reports involving pipe failures in Swedish nuclear power plants. The paper comprises a description of the classification systme used in the evaluation of the incident reports. Furthermore, the choice of the statistical method is discussed. The results of the qualitative and the quantitative analysis are presented and compared to experience in USA. Finally, the conclusions of the analysis are listed besides a discussion of the limitations of the analysis with special reference to the collection of data.

Title: A Study of Piping Failures in US Nuclear Power Reactors

Author: Janzen, P. **Corp. Author:** AECL

Source: AECL-Misc-204

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1981 **Language:** English

Category: Operating experience **ID:** 893

Abstract: A study of piping failures in nuclear power generating plants was undertaken in support of the study of pipe rupture in the Primary Heat Transport System of CANDU stations. Because of the limited operating experience of CANDU stations and the availability of documentation of the much longer history of performanceof US-LWRs, this later data was chosen as the initial subject of analysis. The analysis involves calculation of pipe failure rates and classification, manipulation and correlation of data according to severity of failure, pipe size, process system in which pipe is located, location of failure, cause of failure, effect of failure on reactor condition, date of occurrence and plant age at time of occurrence.

Title: Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants

Author: Pipe Crack Study Group **Corp. Author:** U. S. NRC

Source: NUREG-0531

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1979 **Language:** English

Category: Operating experience / SCC **ID:** 894

Abstract: This report covers the investigation of the possible intergranular stress corrosion cracking (IGSCC) of large diameter piping. During 1978, IGSCC was reported for the first time in large diameter piping (>20 in.) in a BWR in Germany. This discovery, together with the reported questions concerning the interpretation of ultrasonic inspections, led to the activation of a new Pipe Crack Study Group (PCSG). The chapter of the new PCSG was expanded 1) to review the potential for IGSCC in PWRs and BWRs., 2) to examine operating experiences in foreign reactors relevant to IGSCC, and 3) to specifically address five PCSG charter questions. The specific areas considered by the PCSG and summarized in this report are PWR and BWR cracking experience, metallurgy associated with pipe cracking, reactor coolant chemistry, pipe configuraton and stress levels, Duane Arnold safe-end cracking, methods of detecting significance of cracks, and recent developments relevant to conttrol and detection of IGSCC. In the report conclusions and recommendations by the PCSG are presented.

Title: Canvey: Hazard Models and Risk Estimates

Author: Lees, F.P. **Corp. Author:**

Source: Loss Prevention in the Process Industries, Volume 2, Butterworth-Heinemann, Ltd., Oxford (UK), ISBN 0-7506-1523-0, pp1017-1022

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1980 **Language:** English

Category: Pipe failure probability **ID:** 895

Abstract: Failure of pressure piping is considered as a possible initiating event for releases of LPG and LNG. Lees provides a summary of how the Canvey Risk Assessment report addresses the failure of LPG pressure piping.

Title: U. S. BWR Internals Aging Mitigation

Author: Stancavage, P. P. **Corp. Author:** GE Nuclear Energy

Source: CSNI Report No. 146

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1987 **Language:** English

Category: Life extension **ID:** 896

Abstract: Fatigue and stress corrosion cracking affect the service life of BWR internal structures. Continued attention to understanding age-related degradation, to monitoring the equipment condition, to reducing loads and maintaining high water quality and to making durable refurbishments. where necessary will contribute to extending the operating period of major internals beyond the 40-year design life. This paper summarizes the major issues relating to BWR life extension with an emphasis on the actions which can be taken to assure structural integrity over the long term.

Title: Review of Main Degradations Observed on Reactor Internals of Operating Belgian PWRs

Author: Briegleb. P. & Mignot, P. **Corp. Author:** Vincotte A.S.B.L.

Source: CSNI Report No. 146

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Operating experience **ID:** 897

Abstract: The purpose of this paper is to describe some typical degradations experienced by reactor core internals in operating Belgian PWRs, and to review the investigatins carried out to determine the cause and the extent of the problems and the corrective actions taken. The degradations described are attriuted either to IGSCC of inconel aloys, or to mechanical wear resulting from flow induced vibrations or from fretting of moving pieces. The components afected are the bolts clamping the hold down springs on top of fuel assemblies, the flexurex on top of upperguide tubes, the control rod guide tube support pins, the rod cluster control assembly (RCCA) rodlets and the incore instrumentation thimbles.

Title: Probability of Pipe Failure in the Reactor Coolant Loops of Westinghouse PWR Plant

Author: Holman, G.S. & Chou, C. K. **Corp. Author:** Lawrence Livermore National

Source: NUREG/CR-3660, UCID-19988

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1985 **Language:** English

Category: PFM evaluation **ID:** 898

Abstract: As part of its reevaluation of the double-ended guilloting break (DEGB) of reactor coolant loop piping as a design basis event for nuclear power plant, the U.S. Nuclear Regulatory Commission (NRC) contracted wit the Lawrence Livermore National Laboratory (LLNL) to estimate the probability of occurence of a DEGB, and to assess the effect that earthquakes have on DEGB probability. This report describes a probabilistic evaluation of reactor coolant loop piping in PWR plants having nuclear steam supply systems designed by Westinghouse. Two causes of pipe break were considered: pipe fracture due to the growth of cracks at welded joints ("direct" DEGB), and pipe rupture indirectly caused byfailure of component supports due to an earthquake ("indirect" DEGB). The probability of indirect DEGB was estimated by estimating support fragility and then convolving fragility and seismic hazard. The results of this study indicate that the probability of a DEGB from either cause is very low for reactor loop piping in these plants, and taht NRC should therefore consider eliminating DEGB as a design basis event in favor of more realistic criteria.

Title: Pipe Failures in U. S. Commercial Nuclear Power Plants

Author: Jamali, K. **Corp. Author:** Halliburton NUS

Source: EPRI TR-100380

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1992 **Language:** English

Category: Failure rate estimation **ID:** 899

Abstract: Recent NRC mandates require utilities to perform probabilistic risk assessments as part of their individual plant examinations (IPEs). To date, a significant number of IPEs have identified small-break loss-of-coolant accidents (LOCAs) as a major contributor to nuclear power plant risk. Most existing databases that address pipe failure rates have been based on judgement estimates from industry experts. EPRI has developed a methodology and database that uses actual experiences to support failure rate calculations on a plant-or system-specific basis.

Title: A Simplified Leak-Before-Break Evaluation Procedure for Austenitic and Ferritic Steel Piping

Author: Gamble, R., Zahoor, A. & Ghassemi, B. **Corp. Author:** Novetech Corporation

Source: NUREG/CR-6281

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1994 **Language:** English

Category: LBB methodology **ID:** 900

Abstract: A simplified procedure has been defined for computing the allowable circumferential throughwall crack length as a function of applied loads in piping. This procedure has been defined to enable LBB evaluations to be performed without complex and time consuming analyses. The development of the LBB evaluation procedure is similar to that now used in Section XI of the ASME Code for evaluation of part-throughwall flaws found in piping. The LBB evaluation procedure was bench marked using experimental data obtained from pipes having circumferential throughwall flaws. Comparisons of the experimental and predicted load carrying capacities indicate that the method has a conservative bias, such that for at least 97% of the experiments the experimental load is equal to or greater than 90% of the predicted load.

Title: Study on Life Extension of Aged RPV Material Based on Probabilistic Fracture Mechanics: Japanese Round Robin

Author: Yagawan, G. et al **Corp. Author:** University of Tokyo

Source: Journal of Pressure Vessel Technology, Vol. 117

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1995 **Language:** English

Category: Aging analysis **ID:** 901

Abstract: This paper is concerned with round-robin analyses of probabilistic fracture mechanics (PFM) problems of aged RPV material. Analyzed here is a plate with a semi-elliptical surface crack subjected to various cyclic tensile and bending stresses. A depth and an aspect ration Failure probabilities are calculated using the Monte Carlo methods with the importance sampling or the stratified sampling techniques. Material properties are chosen from the Marshall report, the ASME Code Section XI, and the experiments on a Japanese RPV material carried out by the Life Evaluation (LE) subcommittee of the Japan Welding Engineering Society (JWES). while loads are determined referring to design loading conditions of pressurized water reactors (PWR). Seven organizations participate in this study. As first, the procedures for obtaining reliable PFM solutions with low failure probabilities are examined by solving a unique problem with seven computer programs. The seven solutions agree very well with one another, i.e., by a factor of 2 to 5 in failure probabilities. Next, sensitivity analyses are performed by varying fracture toughness values, loading conditions, and pre and in-service inspections. Finally, life extension simulations based on the PFM analyses are performed. It is clearly demonstrated from these analyses that failure probabilities are so sensitive to the change of fracture toughness values that the degree of neutron irradiation significantly influences the judgement of plant life extension.

Title: Reliability and Risk in Pressure Vessels and Piping

Author: Phillips, J. & Atwood, C. **Corp. Author:** Tenera, L.P. & Idaho National

Source: The Pressure Vessels and Piping Division, ASME PVP-Vol. 251

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1993 **Language:** English

Category: PSA methodology **ID:** 902

Abstract: The PRAs being developed at most NPPs to calculate the risk of core damage generally focus on the possible failure of active components. Except as initiating events, the possible failure of passive components is given little consideration. The NRC is sponsoring a project at INEL to investigate the risk significance of passive components as they age. For this project, we developed a technique to calculate the failure probability of passive components over time, and demonstrated the technique by applying it to a weld in the auxiliary feedwaer (AFW) system. The selection of this component was based on expert judgement of the likelihood of failure and on an estimate of the consequence of component failure to plant safety. We used a modified version of the PRAISE computer code to perform a probabilistic structural analysis to calculate the probability that crack growth due to aging would cause the weld to rupture. We modified an existing PRA (NUREG 1150 plant) to include the possible rupture of the AFW weld, and then we used the weld rupture probability as input to the modified PRA to calculate the change in plant risk with time. The results showed an insignificant effect on plant risk because of the low calculated rupture rate of the weld in this particular calculation over 48 years of service. However, the most interesting observation was the rupture rate trend for this 48 years. A decreasing yearly rupture rate for this weld was calculated instead of the increasing repture rate trend one might expect. We attribute this result to infant mortality; that is, most of those initial flaws that will eventually leadd to rupture will do so early in life. This means that although each weld in a population as a whole can exhibit a decreasing rupture rate. This observation has implications for passive components in commercial nuclear plants and other facilities where aging is a concern. For the pupolation of pasive components that exhibit a decreasing failure rate, risk increase is not a concern. The next step of the work is to identify the attributes that contribute to this decreasing rate and to determine any attributes that would contribute to an increasing failure rate and thus to an increased risk.

Title: Pipework Failures - A review of historical incidents

Author: Blything, K. & Parry, S. **Corp. Author:** United Kingdom Atomic Energ

Source: SRD R441

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Operating experience **ID:** 903

Abstract: Historical incident data has been gathered form different sources and classified into the four plant categories - Chemical, Refinery, Nuclear and Steam. However, the available world-wide data was found to be surprisingly limited and it should be regarded an indicative of typical problems rather than statisticaly significant. The incident data has been analysed to determine failure cause and the underlying reasons for failure defined as root causes. Data conderning leak severity has been gathered from some sources and this has been classified as leaks or ruptureswith the number of incidents in each category. Brief descriptions are given for a selection of incidents to illustrate the types of failure and their consequences.

Title: Review of Erosion-Corrosion in Single-Phase Flows

Author: Cragnolino, G., Czajkowski, C. & Shack, W. **Corp. Author:** Argonne National Laboratory

Source: NUREG/CR-5156

SKI Project File: Ja **Transfer:** Nej **Publ year:** 1988 **Language:** English

Category: Erosion-corrosion **ID:** 904

Abstract: This report contains two literature reviews (prepared by Brookhave National Laboratory and Argonne National Laboratory, respectively) on the available data and current mechanistic understanding of erosion-corrosion, and a a failure analysis (prepared by Brookhaven National Laboratory) of a tee-elbow joint from the Surry Unit 2 reactor that failed by erosion-corrosion in December 1986. It also includes suggestions for additional research that should be performed by the USNRC to increase the capability to rank plants and0or locations within plants in terms of susceptibility to erosion-corrosion and to ensure that proposed inspection and mitigation programs are soundly based.

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