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Organisation de Coopération et de Développement Economiques
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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

NEA/CSNI/R(2005)8
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CSNI INTEGRITY AND AGEING WORKING GROUP

THERMAL CYCLING in LWR COMPONENTS in OECD-NEA MEMBER COUNTRIES

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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

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

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

NUCLEAR ENERGY AGENCY

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

The mission of the NEA is:

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

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

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

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of senior scientists and engineers, with broad responsibilities for safety technology and research programmes, and representatives from regulatory authorities. It was set up in 1973 to develop and co-ordinate the activities of the NEA concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations.

The committee's purpose is to foster international co-operation in nuclear safety amongst the OECD member countries. The CSNI's main tasks are to exchange technical information and to promote collaboration between research, development, engineering and regulatory organisations; to review operating experience and the state of knowledge on selected topics of nuclear safety technology and safety assessment; to initiate and conduct programmes to overcome discrepancies, develop improvements and research consensus on technical issues; to promote the coordination of work that serve maintaining competence in the nuclear safety matters, including the establishment of joint undertakings.

The committee shall focus primarily on existing power reactors and other nuclear installations; it shall also consider the safety implications of scientific and technical developments of new reactor designs.

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

FOREWORD

In 2002, the Committee on the Safety of Nuclear Installations (CSNI) requested the working group on the integrity of reactor components and structures (IAGE WG) to prepare a program of work on thermal cycling to provide information to NEA member countries on operational experience, regulatory policies, countermeasures in place, current status of research and development, and to identify areas where research is needed both at national and international levels.

The working group proposed a 3 fold program that covered:

- Review of operating experience, regulatory framework, countermeasures and current research;

- Benchmark to assess calculation capabilities in NEA member countries for crack initiation and propagation under a cyclic thermal loading, and ultimately to develop screening criteria to identify susceptible components; results of the benchmark were published in 2005; Report referenced NEA/CSNI/R(2005)2;

- Organisation of an international conference in cooperation with the EPRI and the USNRC on fatigue of reactor components. This conference reviews progress in the areas and provides a forum for discussion and exchange of information between high level experts. The conference is held every other year to follow the progress and to direct research to key aspects. The last edition was held on October 3-6, 2004. Report referenced NEA/CSNI/R(2004)21.

The current report covers the first point.

The complete list of CSNI reports, and the text of reports from 1993 onwards, is available on http://www.nea.fr/html/nsd/docs/

ACKNOWLEDGEMENTS

Gratitude is expressed to the delegates of the CSNI Working Group on the Integrity of Components and Structures for providing the answers to the questionnaire. Special thanks to Mr Claude Faidy (EdF, FR), Dr Stephane Chapuliot (CEA, FR) and Mr Eric Mathet (OECD-NEA) for synthesizing the responses.

EXECUTIVE SUMMARY

Thermal cycling is a widespread and recurring problem in nuclear power plants worldwide. Several incidents with leakage of primary water inside the containment challenged the integrity of NPPs although no release outside of containment occurred. Thermal cycling was not taken into account at the design stage. Regulatory bodies, utilities and researchers have to address it for their operating plants. It is a complex phenomenon that involves and links thermal hydraulic, fracture mechanic, materials and plant operation.

Thermal cycling is connected either to operating transients (low cycle fatigue) or to complex phenomenon like stratification, vortex and mixing (low and high cycle fatigue). The former is covered by existing rules and codes. The latter is partially addressed by national rules and constitutes the subject of this report.

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The current report covers the first point. The IAGE WG prepared a questionnaire that was completed by NEA member countries in 2002-2003. This questionnaire addressed:

- ✓ Regulatory requirements and codes;
- ✓ Practical experience and incidents;
- ✓ Countermeasures related to stratification and mixing;
- ✓ Research on thermal fatigue;
- ✓ Low versus High cycle fatigue curves.

The following general conclusions can be drawn from answers to the questionnaire:

Thermal cycling degradations have the potential to be an important safety and economical issues. It strongly affects aging management program of safety components.

Various incidents occurred in different systems and countries. However only few leaks have been observed and there was no release outside the containment.

Analysis carried out showed three loading modes connected to thermal cycling: (a) stratification; (b) dead legs and vortex; (c) mixing tees.

Screening criteria and guidelines development are now proposed to review potential locations. In-service inspection programs have been modified to integrate monitoring and valve leak tests. In addition, design changes have been implemented in different systems.

Nevertheless large uncertainties remain on quantitative damage estimation. An important R&D effort must be pursued to confirm screening criteria values and to decrease uncertainties with the objective to optimize maintenance action.

TABLE OF CONTENTS

I. INTRODUCTION

Thermal cycling is widespread and recurring problems in nuclear power plants worldwide. Several incidents with leakage of primary water inside the containment challenged the integrity of the primary system of nuclear power plants although no release outside of containment occurred. Some complex thermal loads are not taken into account at the design stage of some operating plants (i.e. stratification, mixing). Regulatory bodies, utilities and researchers have to address them for operating plants and design of future plants. There are complex phenomena that involve and link thermal hydraulic, fracture mechanic, materials and plant operation.

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1. Review of operating experience, regulatory framework, countermeasures and current research;
2. Benchmark to assess calculation capabilities in NEA member countries for crack initiation and propagation under a cyclic thermal loading, and ultimately to develop screening criteria to identify susceptible components; results of the benchmark were published in 2005 [1];
3. Organization with the EPRI and the USNRC of the international conference on fatigue of reactor components. This conference reviews progress in the areas and provides a forum for discussion and exchange of information between high level experts. The conference is held every other year to follow the progress and to direct research to key aspects. The last edition was held on October 3-6, 2004 [2,3,4]

In addition a large number of NEA member countries are participating in the OECD Piping Failure Data Exchange Project (OPDE) to collect field experience on piping degradation [5,6,7]

The current report covers the first bullet. The IAGE WG prepared a questionnaire that was completed by NEA member countries in 2002-2003 (Appendix A). This questionnaire addressed:

- ✓ Regulatory requirements and codes;
- ✓ Practical experience and incidents;
- ✓ Countermeasures related to stratification and mixing;
- ✓ Research on thermal fatigue;
- ✓ Low versus High cycle fatigue curves.

The questionnaire addressed several plant designs and is limited to pressure retaining components in safety-class systems.

A synthesis of the answers is provided hereafter. Complete answers are provided in appendix B.

II. THERMAL CYCLING PHENOMENON

Complex thermal cycling phenomena are mainly connected to:

- ✓ Stratification;
- ✓ Vortex penetration in tees, dead legs, valve leaks;
- ✓ Mixing tees.

Two fluids at different temperatures stratifying in a pipe cause temperature non-linearity at the interface. This non-linearity induces deformation and important bending stresses in the pipe. In addition flow conditions can create displacement of the interface that can be superposed to fluctuations of great amplitude. All are leading to thermal cycling damages.

As opposed to the relatively low number of cycles in thermal stratification, fatigue in vortex and mixing areas is more of a high cycle fatigue nature. It occurs in pipes where flows at different temperature and different flow rate ratio mix in a turbulent manner. Local or global temperature fields resulting from this turbulent mixing lead to stresses that may cause fatigue damages.

Both problems are complex and involve expertises that have their own specificities and limits:

- ✓ Thermal hydraulic with regard to thermal loading (turbulent mixing or stratification);
- ✓ Mechanical expertise with regard to calculation of structure response to complex thermal loading on the inner surface including fracture mechanic;
- ✓ Material science with regard to material strengths and fatigue resistance.

These thermal loads are difficult to predict and quantify at the design stage. Degradations observed in nuclear power plants are mainly elephant-skin type of damage and in few occurrences through wall cracks. Phenomena leading to the degradation are now understood. Operating condition and design changes may greatly reduce the frequency of degradation. However, in some cases changes are not possible and strategies must be in place accounting for large uncertainties in parameters controlling initiation, propagation and kinetic. Definition of screening criteria is then needed and efforts should be pursued.

III. REGULATORY REQUIREMENTS AND CODES

In all countries fatigue has to be considered at the design stage and during operation.

For complex thermal loads, no specific regulatory requirements at the initial design stage (before 1988) were implemented. Codes used at this time (e.g. US ASME III, Soviet PNAE codes, French RCCM, German KTA) did not specifically mention these loads.

After 1988, some countries have issued specific regulatory requirements (e.g. USNRC Bulletins 88-08 and 88-11, YVL 3.5 in Finland) and in two cases, the regulation was amended in 2001 (Japan and France) along with respective codes.

Many countries have developed their own in-service inspection codes (e.g. USA ASME section XI, France RSEM, Soviet PK-15-14, Japan JEAC4205, Canada CSA/CAN-N285.4, Sweden SKIFS1994:1)

IV. OPERATING EXPERIENCE AND INCIDENTS

An extensive and educative list of incidents along with root causes analysis is available in Appendix B covering BWR, PWR, VVER and CANDU reactors.

IV.1 *Incidents and root causes*

The three thermal load types have affected some BWRs to different extents. The main locations concerned are:

- ✓ Feedwater system piping (i.e. nozzles and spargers)
- ✓ Control Rod Drive return nozzles
- ✓ Reactor Coolant System
- ✓ Residual Heat Removal System
- ✓ Reactor Water Clean-up system
- ✓ Auxiliary Feedwater System
- ✓ Pressure Relief System

The three thermal load types have affected some PWRs to different extents. The main locations concerned are:

- ✓ Feedwater system piping
- ✓ Pressurizer surge lines (large displacement and mixing of fluids at different temperature were observed)
- ✓ Emergency Core Cooling System injection line (leaking valves)
- ✓ Residual Heat Removal Piping in mixing tees
- ✓ Discharge line (Dead leg)
- ✓ Reactor Coolant System sampling line
- ✓ Make-up/High Pressure Injection nozzle
- ✓ Drain lines
- ✓ Reactor Coolant Pump thermal barriers
- ✓ Valves (inner surface)

IV.2 *Countermeasures*

Countermeasures are obviously linked to root causes.

For stratification, new design of piping system including new thermal sleeves along with modified operating conditions was implemented. In some cases potential degradation areas have been protected by a liner and/or helical shaped devices were installed (PWR feedwater system).

For dead legs and vortex, major concern is the leak tightness of valve. Countermeasures used are either optimization of the in service valve leak test procedure or leak collection monitoring.

For mixing tees, countermeasures consist mainly in decreasing of the amplitude of temperature difference, and/or the duration at large ΔT by changing operating procedures. A mixing device is installed in some cases. Component replacement with more fatigue resistant component (e.g. material, surface finish, thickness variation, weld locations) is performed.

In all countries, a specific in-service inspection program is implemented. Monitoring devices have also been considered and installed in few countries. Training of operators (both plant and ISI operators) is also an important aspect to increase awareness of thermal cycling issues at the plants.

IV.3 *Economic impact*

In overall forced outages typically consume a little more than five percent of total power availability, but losses associated with forced outages can represent as much as 30 to 40 percent of annual profits. Degradations due to thermal cycling have resulted in forced outages or in extension of outage from few days to several months. Definition of the repair and repair time, root cause analysis, life evaluation of replaced components, modification and validation of operating conditions are the main contributors.

V. RESEARCH PROGRAMS

V.1 *Completed research projects*

In the past R&D projects mainly addressed low cycle thermal cycling: stratification in piping system and fatigue of components like pumps, valves or components with thickness variations. In this frame, various types of materials and geometrical configurations were analyzed either through analysis or experimentation. Number of guidelines was published to optimize operating management, flaw detection by non-destructive examination for stratification, and changes in inner surface shape of valves.

Very little work was performed on high cycle thermal cycling in mixing areas as it was supposed to mainly concern fast breeder reactors sensitive to this type of load.

V.2 *On going and planned research projects*

Currently few works are ongoing on stratification and major programs are now devoted to thermal cycling in mixing areas and vortex in dead legs. Three major disciplines are involved in these programs:

- ✓ First discipline is the thermo hydraulic experimentation and calculation. This area is developed because of the difficulty to understand the real thermal load applied to the inner surface of the structure. Thus, number of geometrical or flow configurations are studied to try to derive generic parameters, screening values or recommendations.
  This work is supported by the recent models and numerical capability development (CFD calculation on large capability computers).

- ✓ The second is the thermo mechanical component testing and calculation. The objective of these studies is to reproduce, in laboratory, a load equivalent to a thermal cycling load in mixing area on analytical mock-ups. Two main difficulties are then encountered: reaching a high number of cycles (corresponding to a mixing area) and having a good knowledge of the real imposed loading (this last point has in general to be determined by finite element calculation).
  At the end, the final goal of these studies is to apply the fatigue evaluation procedures and criteria to validate or not applications on a configuration as close as possible to a reactor case.

- ✓ The third is the material study in which effects of temperature, surface finish, mean stresses or environment on fatigue resistance constitute the most important part of the work performed. In this field, low and high cycle fatigue is investigated, but the main difficulty is on the determination of the material endurance limit (high cycle fatigue) under applied strain.

There are basically three poles of research: USA, Europe and Japan. These programs are merged into integrated projects. As said before, this aspect is mainly due to the multi disciplinary aspect of high cycle thermal fatigue.

The extensive US research program was carried out by the EPRI, the USNRC and NSSS vendors. It mainly focused on stratification, dead legs and vortex. Japan works on large scale tests, environmental effects and mixing tees. Europe combines national programs and the EC/THERFAT project on mixing tees, dead legs and vortex.

In parallel of this R&D effort, programs are continuing to propose or improve thermal cycling management (in stratification or mixing zones) or general fatigue assessment procedures.

V.3 *Research needs*

Because stratification is correctly understood, major R&D needs for the future are mainly dealing with the mixing and vortex issues. Following main needs are mentioned by experts:

- ✓ Definition of screening values in terms of simple parameters such as the difference of fluid temperature or stresses. Theses screening values have to cover a large number of geometrical, flow velocity or temperature range to be applicable to the industrial case and constitute the first step to improve thermal cycling management. Screening criteria can be determined by thermo hydraulic experiments or CFD calculations with parametric studies. However in each case improvements are still needed in transfer from mock-up to reactor case and CFD models.
- ✓ In mechanical field, needs concern guidelines to account for complex loading like random biaxial loads, with or without mean stresses or strains. In addition, fracture mechanics thresholds need to be improved so that they could be included in fracture assessment procedures and help to predict large crack growth.
- ✓ On material aspects, fatigue curves need improvements to take into account surface finish, mean stresses, residual stresses, welds, environments, stress biaxiality, random loads… In this regard, high cycle fatigue under imposed strain is the key research topic. A major question is still under discussion: do we need to modify fatigue curves?

On the stratification problem, expressed needs are mainly focused on thermal cycling management and more precisely on damage evaluation by measuring temperatures on piping systems or feed water nozzles.

VI. CONCLUSIONS – RECOMMENDATIONS

Thermal cycling degradations have the potential to be an important safety and economical issues. It strongly affects aging management program of safety components.

Various incidents occurred in different systems and countries. However only few leaks have been observed and there was no release outside the containment.

Analysis carried out showed three loading modes connected to thermal cycling: (a) stratification; (b) dead legs and vortex; (c) mixing tees.

Screening criteria and guidelines development are now proposed to review potential locations. In-service inspection programs have been modified to integrate monitoring and valve leak tests. In addition, design changes have been implemented in different systems.

Nevertheless large uncertainties remain on quantitative damage estimation. An important R&D effort must be pursued to confirm screening criteria values and to decrease uncertainties with the objective to optimize maintenance action.

REFERENCES

[1] NEA/CSNI/R2005(2) OECD-NEA Benchmark on Thermal cycling (to be published in 2005)

[2] NEA/CSNI/R(2000)24 Proceedings of the International Conference on Fatigue of Reactor Components, August 2000, Napa, California

[3] NEA/CSNI/R(2003)2 Proceedings of the EPRI/USNRC/OECD International Conference on Fatigue of Reactor Components July 2002 Snowbird, Utah

[4] NEA/CSNI/R(2004)21 3rd International Conference on Fatigue of Reactor Components- Seville, octobre 2004 (to be published in 2005)

[5] PIPING SERVICE LIFE EXPERIENCE IN COMMERCIAL NUCLEAR POWER PLANTS: PROGRESS WITH THE OECD PIPE FAILURE DATA EXCHANGE PROJECT - Proceedings of ASME PVP-2004 Conference:2004 ASME Pressure Vessels and Piping (PVP) Conference July 25-29, 2004, San Diego, California, USA (Presentation)

[6] A Framework for International Cooperation in Piping Reliability: OECD Pipe Failure Data Exchange Project: Proceedings of the 25th ESReDA Seminar, Paris, France, November, 2003

[7] OECD PIPE FAILURE DATA EXCHANGE PROJECT (OPDE) - 2003 STATUS REPORT paper presented at ICONE-12: 12th International Conference on Nuclear Engineering April 25-29, 2004, Arlington, Virginia, U.S.A.

APPENDIX A: QUESTIONNAIRE

OECD/NEA QUESTIONNAIRE ON THERMAL FATIGUE DUE TO STRATIFICATION AND MIXING OF HOT AND COLD WATER

INTRODUCTION

Thermal fatigue due to stratification and mixing of hot and cold water is a recurring phenomenon. It is obvious that it was not addressed enough in the design phase of the plants. One purpose of this questionnaire is to find out how widespread a problem thermal fatigue is. The second one is to learn what kind of countermeasures has been taken by countries. Overall objectives are to obtain a detailed view of countries' actions and regulations, if any, and identifying adequate corrective actions.

The questionnaire is restricted to pressure retaining components in safety classified systems.

Simple screening rules would be useful but these can be developed only for specific cases like maximum temperature difference in Tee-joints. Usually solving a practical problem requires experts from several disciplines.

Thermal mixing is defined as the mixing of cold and hot water. Stratification is understood as layers of cold and hot water in pipes with low flow velocity. Cyclic stratification can cause a rapid fatigue damage. Stratification is usually an unforeseen event whereas mixing is part of plant normal operation. Both of these phenomena sometimes coincide. In those cases classification can be made based on the fraction of fatigue damage.

GUIDANCE

Please use the numbering when answering questions.

Text in italic indicates general requests to be applied to plant types/systems listed below.

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QUESTIONS

I. REGULATORY REQUIREMENTS AND CODES

I.1 Are there any regulatory requirements specific to thermal fatigue? Have these requirements been applied during the design phase of the nuclear power plant?

I.2 What regulatory requirements or guides have been published after incidents caused by the thermal fatigue?

I.3 Codes

A) What Design & Fabrication Code and version has been generally used?

B) What kind of fatigue analysis has been generally done at the design level?

C) Do you consider other thermal fatigue mixing areas in large components? (Like spray of cold water in pressurizer or auxiliary cold feedwater in steam generators.)

D) What In-Service Inspection Code do you use?

II. PRACTICAL EXPERIENCE AND INCIDENTS

How many incidents due to thermal fatigue have occurred in each of the following items (II1 to II3) (and how many of these after 1.1.1994)?

1. Differentiate between

cracks,
small leaks in piping (LBB),
medium leaks in piping (crack opening is 1 mm or more) and
total breaks of piping.

2. Provide a short description of root causes with details on (if relevant)

Stratification
In leakage of cold (or hot) water through a valve (intermittent leakage?)
other
Mixing
Continuous mixing during power operation
Limited operating time, like heat up and cool down
other
Other phenomenon

II.1 BWR

A) Feedwater nozzle and adjacent piping

B) Other piping (please indicate the system, the diameter of the piping and the possibility of isolation from the reactor)

C) Other large components

II.2 PWR

A) Feedwater nozzle and adjacent piping

B) PWR pressurizer surge line

C) Other piping (please indicate the system, diameter of the piping and possibility of isolation from reactor)

D) Other large components

II.3 Other plants than BWR or PWR

II.4 Estimate of whole forced shut down time for all thermal fatigue related incidents?

II.5 Other comments on practical experience and incidents

III. COUNTERMEASURES RELATED TO STRATIFICATION

For items III 2 to III 4 please describe countermeasures taken considering items listed below:

a) Identification of locations with potential risk, particularly regarding cyclic stratification. Screening criteria. What disciplines are represented in the review team?
b) Use of plant normal instrumentation
c) Temperature measurements
d) Fatigue meters
e) Inspections (methods, inspected area, frequency)
f) Testing of leak tightness of valves
g) Optimising operating conditions, training of operators
h) Changes in the systems
i) LBB
j) Other measures

III.1 General approach and policy

III.2 BWR

A) Feedwater nozzle and adjacent piping

B) Other piping

III.3 PWR

A) Feedwater nozzle and adjacent piping

B) PWR pressurizer surge line

C) Other piping

III.4 Other plants than BWR or PWR

III.5 If not addressed in items III 2 to III 4, please describe countermeasures, if any, on:
The risk of cold water in-leakage through a "closed" valve (as a common phenomenon)

Countermeasures for other stratification phenomena

III.6 Other comments

IV. COUNTERMEASURES RELATED TO MIXING

Same list as in paragraph III

IV.1 General approach and policy

IV.2 BWR feedwater nozzle and other systems

IV.3 PWR feedwater nozzle and other systems

IV.4 Other plants than BWR or PWR

IV.5 Tees (common geometry)

- additional questions: Identification of locations with potential risk:

✓ What are the temperature difference limits for austenitic stainless steel and carbon steel (including low alloy steel) in continuous mixing during power operation?

✓ What are the temperature difference limits for austenitic stainless steel and carbon steel (including low alloy steel) in limited operating time (like heat-up and cool-down)?

✓ What are the restrictions for the geometrical configuration in piping, like horizontal and vertical orientation, nearby elbows etc.

IV.6 Other comments

V. RESEARCH ON THERMAL FATIGUE

V.1 Completed research projects

Please indicate references

V.2 On-going and planned research projects

V.3 Research needs

In among others: (please specify what research)

✓ Thermal mixing
✓ Stratification
✓ CDF calculations, parametric studies
✓ Fatigue curves, low cycle (stratification), high cycle (mixing)
✓ Other

VI. HIGH CYCLE FATIGUE CURVE

VI.1 List reference (i.e. Code) of curves currently used for design and for expertise, if relevant.

VI.2 Were those curves used for assessment of failure? With what success?

VI.3 Do you feel the need for new or more appropriate curves for expertise and/or design? (For welded, fabricated, polished materials, environment, frequency etc.)

VI.4 If yes, what step or action did/do you undertake?

APPENDIX B: COMPILATION OF ANSWERS

In this compilation answers from different countries are listed after each question. The following countries have answered: Belgium, Czech Republic, Finland France, Hungary, Japan, Korea, Slovakia, Spain, Sweden and United States of America.

GUIDANCE

- **Questions are at the top of a page, in bold and in a box**
- Answers are underneath, listed by country in alphabetical order

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QUESTIONS AND ANSWERS

I REGULATORY REQUIREMENTS AND CODES

I.1 Are there any regulatory requirements specific to thermal fatigue? Have these requirements been applied during the design phase of the nuclear power plant?

Belgium:

There was no regulatory requirement specific to thermal fatigue at the time when the Belgian NPP's (PWR's) were designed. However consideration of the thermal stratification in the Feedwater system has been required in the framework of the Steam Generator Replacements.

Czech Republic:

There are no requirements specific to T.F. but only general requirements. Thermal fatigue must be analyzed for new NPP design and also for old one during operation

Finland:

There are no regulatory requirements specific to thermal fatigue by mixing or stratification, only the normal ASME III Code requirements or equivalent norm. Thermal fatigue was party taken into account but not in all vulnerable places.

France:

No specific requirements for thermal fatigue; fatigue consideration is normally request for all class 1 components; no specific requirements for class 2 and 3 piping systems (except feedwater and steam lines up to the first anchor point outside the containment).

The other request is: to formally take into account national and international field experience.

Hungary:

There are no regulatory requirements specific to thermal fatigue by mixing or stratification, only the normal ASME III Code type fatigue cycle assessments are available. In the original assessments thermal fatigue is party taken into account.

Japan:

In nuclear power plant component design, fatigue assessment was conventionally conducted according to the Technical Standards for Construction of Nuclear Power Plant Components (MITI Notification No. 501, 1980) (Notification No. 501 hereafter).

The following article was added to the Technical Standards of Facilities for Nuclear Power Plants (MITI Ordinance No. 62, 1965) (MITI Ordinance No. 62 hereafter) in relation to I.2 below. (June 2001)

Prevention of damage due to thermal fatigue:

Article 6 – 2

Vessels, tubes, pumps and valves of nuclear facilities (auxiliary boilers excluded) must be facilitated to prevent damage due to thermal fatigue caused by the mixture of fluids of different temperatures.

Korea:

Yes, we have some regulatory requirements specific to thermal stratification. These requirements have been applied to new plants design since 1988.

Slovakia**:**

There are no regulatory requirements specific to thermal fatigue by mixing or stratification, only the normal Design Code requirements. Thermal fatigue was partly taken into account but not in all vulnerable places.

Spain:

Yes. Spanish utilities (PWRs) are required to follow USNRC Bulletins 88-08" Thermal stresses in piping connected to reactor coolant systems" and 88-11 "Pressurizer surge line thermal stratification". These requirements come after an analysis on the applicability of these Bulletins. For one BWR, the requirements are those established in NUREG-0619"BWR feedwater nozzle and control rod drive return line nozzle cracking". These requirements are then developed in a case by case basis in their in-service inspection programs and the areas determined to be susceptible of thermal fatigue are placed in this program and examined according to specific procedures. These requirements have not been applied during the design phase. They have been applied once these NRC Bulletins and NUREG were published in view of the ocurrence of several incidents due to thermal fatigue in several NPPs.

However, one PWR has implemented, in its design phase, countermeasures to avoid thermal fatigue during operation, consisting on the installation of thermal sleeves in: a) pressurizer surge line: transition ring to the pressurizer nozzle and connecting nozzle to RCS, b) water connections to the three steam generators: main feedwater and auxiliary feedwater nozzles and c) class 1 piping: RHRS connections to the RCS (hot and cold legs), CVCS connections to cold legs of RCS and in the pressurizer spray nozzles. All these areas are included in the in-service inspection program and are required for inspection in each interval of 10 years by volumetric examination (ultrasonic).

Sweden:

The general requirement is that a mechanical device may not be exposed to more or greater thermal load variations than those stipulated in the design basis. Cases there the number of such a load variations is exceeded, require that necessary safety measures are adopted immediately.

United States of America:

The Nuclear Regulatory Commission, through 10 CFR 50.55a, currently requires the use of the ASME Boiler and Pressure Vessel Code (BPV) in the design of nuclear power plants. Use of the addenda to the BPV is required through Regulatory Guide 1.84. However, many older vintage nuclear power plants have components of the reactor coolant pressure boundary that were designed to industry codes that did not require the explicit fatigue analysis presently required by the ASME Code. In addition, thermal fatigue events that are not part of the design basis were not required to be evaluated.

(The responses to questions I and VI are provided by the staff of the U.S. Nuclear Regulatory Commission. The responses to questions II – V were forwarded to the NRC from EPRI by Stan Rosinski.)

I. REGULATORY REQUIREMENTS AND CODES

I.2 What regulatory requirements or guides have been published after incidents caused by the thermal fatigue?

Belgium:

No regulatory requirements or guides have been published in Belgium after incidents caused by the thermal fatigue.

Czech Republic:

No regulatory requirements or guides after incidents.

Finland:

The requirements on mapping of vulnerable areas were sent to the power companies in the letters by STUK. In the first phase mixing points were mapped (1980s) and in the second phase stratification areas (1990s). In a new Guide YVL 3.5 on the integrity of Nuclear Pressure Equipment thermal mixing and stratification is included.

France:

No specific new requirements at the regulation level; only request to enlarge thermal fatigue review of all safety class components through "safety authority generic letters".

Hungary:

The requirements on mapping and occasional monitoring of vulnerable areas are written in the Aging Management Regulatory Guidelines.

Japan:

The Guideline for Practising Technical Standards for High-cycle Thermal Fatigue has been established in December 1999 by the Nuclear Power Safety Administration Division, Public Utilities Department, Agency of Natural Resources and Energy.

The following article was added to MITI Ordinance No. 62 in June 2001 (Prevention of damage due to thermal fatigue).

Article 6 – 2: Vessels, tubes, pumps and valves of nuclear facilities (auxiliary boilers excluded) must be facilitated to prevent damage due to thermal fatigue caused by the mixture of fluids of different temperatures.

Korea:

The regulatory requirements are as follows:

Evaluation of thermal fatigue for Class 1 components including thermal stratification and turbulent penetration phenomena

Enhanced ISI for the pipings susceptible to thermal fatigue such as pressurizer surge line, ECCS, and RHR pipings

Slovakia:

No special regulatory requirements or guides have been published in Slovakia after incidents caused by the thermal fatigue. But according to our regulatory body each NPP has to submit fatigue damage calculation of all critical component and piping parts based on actual load history.

Spain:

In Spain no regulatory requirements or guides have been published after incidents caused by thermal fatigue. Licensees are required to analyse the applicability to them of USNRC requirements, through 10 CFR 50, Generic Letters, Bulletins, Information Notices and to follow the requirements established in these documents in case they applied to them. CSN verifies this applicability through evaluations and then follows it through inspections.

After an incident caused by thermal fatigue in a BWR, the feedwater spargers, the terminal-ends and the thermal sleeves of the feedwater nozzles were replaced and the CSN required this unit a specific inspection program, consisting on the inspection, each refuelling outage, of a least two feedwater nozzles. This aspect is done through a Technical Instruction issued by the CSN and addressed to the licensee, without any need of publishing a guide. These requirements follow those established in Bulletin 88-08; however, additional inspection requirements are also included in the in-service inspection program due to operating experience in other plants, in particular, the piping upstream the last check valve of the ECCS to the cold leg of RCS.

Sweden:

None

United States of America:

The NRC published Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Cooling Systems" in response to an event at Farley 2 in December of 1987. This bulletin required licensees to:

Review systems connected to the RCS to determine whether unisolable sections of piping connected to the RCS can be subjected to stresses from temperature stratification or temperature oscillations that could be induced by leaking valves and that were not evaluated in the design analysis of the piping.

For any unisolable sections of piping connected to the RCS that may have been subjected to excessive thermal stresses, examine non-destructively the welds, heat-affected zones and high stress locations, including geometric discontinuities, in that piping to provide assurance that there are no existing flaws.

Plan and implement a program to provide continuing assurance that unisolable sections of all piping connected to the RCS will not be subjected to combined cyclic and static thermal and other stresses that could cause fatigue failure during the remaining life of the unit.

Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," was issued after events at Trojan. This bulletin required licensees to:

Conduct a visual inspection (ASME, Section XI, VT-3) of the pressurizer surge line at the first available cold shutdown to determine any gross discernible distress or structural damage in the entire pressurizer surge line, including piping, pipe supports, pipe whip restraints, and anchor bolts.

Demonstrate that the pressurizer surge line meets the applicable design codes (fatigue analysis should be performed in accordance with the latest ASME Section III requirements incorporating high cycle fatigue) and other FSAR and regulatory commitments for the licensed life of the plant, considering the phenomenon of thermal stratification and thermal striping in the fatigue and stress evaluations. This was to be accomplished by performing a plant specific or generic bounding analysis.

For further details, both of these documents can be found on the Nuclear Regulatory Commission's web site.

I. REGULATORY REQUIREMENTS AND CODES
I.3 Codes

A) What Design & Fabrication Code and version have been generally used?

Belgium:

ASME B &PV Code, Section III

Czech Republic:

Soviet Code and standards were used for original design. Additionally, also partly Czech standards were partly used.

Finland:

Loviisa 1 and 2: Soviet norms at the time for the main components and for other components conventional Finnish norms augmented with additional requirements.

Olkiluoto 1 and 2: ASME III for Class 1 components and for other components conventional Finnish norms augmented with additional requirements.

France:

ASME III up to 1984 – RCCM from 1984 (last edition 2000- next addenda July 2002)

Hungary:

Soviet norms at the time of design for most of the Class 1 components and for some other components ASME III or KTA norms were used.

Japan:

As described in I.1, fatigue assessment is conducted according to Notification No. 501. The 1999 version is the latest edition.

Korea:

PWR: ASME Code Sec. III

CANDU : CSA/CAN-N285.0 & ASME Code Sec. III

Slovakia:

Soviet norms and standards at the time of design for most of the Class 1 components were applied. For some other components national standards were applied along with ASME III.

Spain:

In Spain we follow the standards of the country of origin of the technology. In this sense different Editions of ASME III (1971, 1974 and addendas) were used for class 1, 2 and 3 components and piping. One unit was designed also according to KTA standards, as the country of origin is Germany. For non-nuclear components and piping, the standard ANSI/ASME B31.1 is followed.

Sweden:

ASME III and Swedish piping codes.

United States of America:

The ASME Boiler and Pressure Vessel Code Section III was initially issued in the early 1960's and later revised. As a result, currently 102 of 104 reactor vessels at plants licensed to operate were designed to Section III. The remaining two plants, Nine Mile Point 1 and Oyster Creek, were designed to Section I of the ASME Code. Safety-related piping systems of 61 plants were designed in accordance with either USA Standard (USAS) B31.1, "Power Piping," or USAS B31.7, "Nuclear Power Piping," or a combination of them with portions of piping designed to Section III. The safety-related piping systems of the remaining 43 plants were all designed to Section III.

The specific design codes used for the Reactor Vessel, RCS Piping and the Remaining Safety-Related Piping for each individual plant is listed in SECY-97-159, "Staff Requirements Memorandum (SRM) Dated February 21, 1997, Re: Briefing for Commission on Codes and Standards, January 22,1997." This document can be found on the Nuclear Regulatory Commission's web site, and contains (in Appendix 1) a listing of all the applicable design codes for (a) the vessel, (b) the safety-related, and (c) the non-safety related piping for all of the US reactors. Appendix 2 of SEWCY-95-129 contains an interesting dissertation on the status of the older vessels designed before Section III came into existence.

I. REGULATORY REQUIREMENTS AND CODES
I.3 Codes

B) What kind of fatigue analysis has been generally done at the design level?

Belgium:

Fatigue analyses are done in accordance with the requirements of the ASME B &PV Code, Section III.

Czech Republic:

Fatigue analyses have been done according Soviet and Czech standards.

Finland:

Normal ASME Code fatigue analysis or equivalent.

In some areas like feed water nozzle and adjacent horizontal pipe the stratification was part of the original stress analyses (BWR).

France:

Low cycle fatigue for level A and B transients for all class 1-2-3 components (including crack like defects); no fatigue analysis for class 2-3 piping systems.

High cycle fatigue considerations are included in 2000 edition of RCCM.

Hungary:

Normal Soviet PNAE Code fatigue analysis or equivalent was performed for safety related equipment. (CUF is less or equal 1.0)

Japan:

Total stress (primary, secondary and peak stress intensities) is calculated for the design transient event under operation phases I and II. With the design fatigue curve, it is then designed wherein the allowable number of cycles or the Usage Factor does not exceed 1.0.

Korea:

Fatigue Analysis based on Cumulative Usage Factor (CUF).

Slovakia:

Normal Soviet PNAE Code fatigue analysis or equivalent was performed for safety related equipment. (CUF is less or equal 1.0)

Spain:

Those established in ASME III. These analyses include primary and secondary stresses where thermal fatigue is considered.

Sweden:

ASME III and Swedish piping codes.

United States of America:

See response to Question I.1. The answer generally depends on the age of the plant, and ranges from not at all, to whatever is required by Sec. III and its applicable (at the time) appendices and code cases.

I. REGULATORY REQUIREMENTS AND CODES
I.3 Codes

C) Do you consider other thermal fatigue mixing areas in large components? (Like spray of cold water in pressurizer or auxiliary cold feedwater in steam generators.)

Belgium:

Yes

Czech Republic:

Yes

Finland:

Yes, these should be part of the design. The problem is, that in some cases it is difficult identify sensitive areas.

France:

Yes this large review is on going following the CIVAUX 1 mixing tee event.

Hungary:

Yes, thermal fatigue at these critical sites was taken into account in the design (There is an allowable number of cold water injections.)

Japan:

As described in I.1, I.2, I.3 A) and I.3 B).

Korea:

Thermal fatigue mixing is only considered in the unisolable section piping connected to reactor coolant line such as ECCS line and RHR line up to the first isolation valve. No consideration in large components.

Slovakia:

Yes, thermal fatigue at these critical sites was taken into account in the design (There is an allowable number of cold water injections.)

Spain:

The areas mentioned in this question, spray of cold water in pressurizer and auxiliary cold feedwater in steam generators, are included in the in-service inspection programs of Spanish NPPs. Other areas also included in the in-service inspection programs are those referenced in USNRC Bulletin 88-08 and its supplements: ECCS to hot and cold legs of RCS, RHR suction line, normal charging of CVCS, alternative charging of CVCS, pressurizer auxiliary spray and ECCS from the accumulators. Also those areas mentioned in I.1 and I.2.

Sweden:

Such areas have been considered at the design basis.

United States of America:

Yes, the thermal mixing areas in large components are considered in the fatigue analyses.

I. REGULATORY REQUIREMENTS AND CODES

I.3 Codes

D) What In-Service Inspection Code do you use?

Belgium:

ASME B & PV Code, Section XI

Czech Republic:

Former Soviet standard PK-15-14 and partly ASME XI as recommended.

Finland:

ASME XI In-Service Inspection Code is applied.

France:

RSEM last edition 1999+addenda 2000, next addenda 2003)

Hungary:

In Hungary we mainly follow the original soviet in-service inspection requirements and there are plans to investigate the possibility of adaptation of ASME XI in the next years.

Japan:

The private sector code "In-service Inspection for LWR Components (JEAC4205-1996 edition: The Japan Electric Association Code 4205-1996)" is used.

Korea:

PWR: ASME Code Sec. XI

CANDU: CSA/CAN-N285.4

Slovakia:

In Slovakia we mainly follow the original Soviet in-service inspection requirements. For each NPP unit special requirements for in-service inspections (time intervals, methods etc.) were developed

Spain:

In Spain we follow Section XI of the ASME Code for in-service inspection activities and all other normative related with in-service inspection (Generic Letters, Bulletins, Information Notices) coming from the USA (NSSS supplier of 8 out of 9 units actually in operation). Actually, ASME/ANSI OM Standards for testing of valves, pumps, snubbers, etc. are used as referenced in Section XI of the ASME Code.

Sweden:

SKIFS 1994:1 (The Swedish Nuclear Power Inspectorate Regulations concerning Mechanical Components)

United States of America:

The In-Service Inspection Code to be used is specified in 10 CFR 50.55a. Generally, the flaw inspection provisions found in 50.55a refer to the inspection requirements of Appendix VIII of Section XI of the BPV.

References (I.1 - I.3):

Code of Federal Regulations, Title10, Part 50 (Energy), available for viewing or download at: http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0055a.html

NRC Regulatory Guide 1.84, Revision 32, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III" (June 2003); available for viewing or download at: http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/active/01-084/ml030730417.pdf

NRC Bulletins are available for viewing or downloading at:

http://www.nrc.gov/reading-rm/doc-collections/gen-comm/bulletins/

The SECY documents may be found at:

http://www.nrc.gov/reading-rm/doc-collections/commission/secys/

The ASME Boiler and Pressure Vessel Code, and the American National Standards Institute requirements (the ANSI B31.x standards) are available from ASME and ANSI, respectively.

II. PRACTICAL EXPERIENCE AND INCIDENTS

How many incidents due to thermal fatigue have occurred in each of the following items (II1 to II3) (and how many of these after 1.1.1994)?

a) Differentiate between

cracks,
small leaks in piping (LBB),
medium leaks in piping (crack opening is 1 mm or more) and
total breaks of piping.

b) Provide a short description of root causes with details on (if relevant)

Stratification
In leakage of cold (or hot) water though a valve (intermittent leakage?)
other
Mixing
Continuous mixing during power operation
Limited operating time, like heat up and cool down
other
Other phenomenon

II.1 BWR

A) Feedwater nozzle and adjacent piping

Belgium:

No BWR

Czech Republic:

No BWR

Finland:

Four (4) cracked areas.

Continuous mixing takes place during operation in the feedwater pipe (nozzle for the pipe from purification system). ΔT is about 100 C during normal operation and up to 220 C during start-up and shut-down of the reactor. Shallow cracks were found in all four similar places 1985 (Olkiluoto 1 and 2). The reason was improper mixing devices. The diameters of the pipes are 400 and 150 mm (austenitic steel). The mixer was changed to the better design. The check valve is between the Tee and the reactor.

France:

Not applicable

Hungary:

No BWR

Japan:

Plant & time of occurrence: Fukushima Daiichi Unit 1, 02/1977

Location of failure: Reactor pressure vessel feedwater nozzle

Conditions: Crack found in feedwater nozzle inner surface by liquid penetrant test. Two cracks (approx. 120mm& 40mm long) on A nozzle and two cracks (approx. 30mm & 40mm long) on C nozzle confirmed.

Causes: Thermal fatigue due to mixing of hot and cold water. Cold feedwater leakage flow through the clearance between nozzle and thermal sleeve mixed at the feedwater nozzle corner during reactor operation and fluctuated at high frequency, thus causing thermal fatigue at the nozzle corner.

Major counter-measures: Application of structure modification design to prevent hot and cold water mixing. Thermal sleeve outside diameter was made larger by 0.254 mm than nozzle and installed to nozzle by shrink-fit to eliminate clearance between the nozzle and thermal sleeve.

Korea:

No BWR

Slovakia:

No BWR

Spain:

The only known incident due to thermal fatigue consisted in some cracks found in the spargers of the main feedwater system, inside the vessel. The root cause was the difference in temperature between the incoming feedwater and the water inside the vessel, producing turbulent mixing capable of initiating cracks in this area of the spargers. Also, the thermal sleeves of this incoming piping presented some leaks which produced erosion (lack of material) in the safe-ends, which were replaced together with the thermal sleeves. The inner radii of the nozzles were no affected.

Sweden:

43 cracks, 2 small leaks. Root causes: 1a, 2a and 1b (error in design)

United States of America:

Cracking in the feedwater nozzles of boiling water reactors is an issue that occurred mainly prior to 1980 (no incidents of cracking have been noted after 1993.) The cracking was due to leakage past the feedwater sparger inlets, since the early plants did not have sufficient flow resistance to prevent a small amount of relatively colder feedwater from bypassing the inlets. The relatively colder feedwater interacted with hot water in the reactor, causing rapid thermal cycling at the feedwater nozzle bore and vessel blend radius regions. The presence of cladding in the nozzles and the cyclic behaviour of the feedwater controllers contributed to the potential for crack initiation. This cracking is well documented in NUREG-0619.

II. PRACTICAL EXPERIENCE AND INCIDENTS

II.1 BWR

B) Other piping (please indicate the system, the diameter of the piping and the possibility of isolation from the reactor)

Belgium:

No BWR

Czech Republic:

No BWR

Finland:

Case 1, mixing: Shallow cracks were found in four similar places in the shutdown cooling systems (outside the containment) of Olkiluoto 1 and 2 units in 1983. During normal operation temperature difference between cold and hot water is about 200 C. Diameters of the austenitic pipes are 250 and 50 mm. Thermal sleeve in the smaller pipe was replace by the longer sleeve, with the purpose of making waters with different temperatures mix in the centre of the larger pipe.

Case 2, mixing: Shallow cracks were found in a Tee-peace, which was in the purification system (outside the containment). During normal operation ΔT is 30 C and during increased purification flow 85 C (typically about 15 days/year, probably more during commissioning). Diameters of the pipes are 150 and 25 mm (austenitic steel). The mixer was added (1991).

Case 3, mixing: The medium size leak was in the purification system of Olkiluoto 1 unit (outside the containment). There was continues mixing during operation in a Tee-peace. Temperature difference was 150 C and rupture (medium leak) took place after 4000 h of operation (1979). Nominal diameters of pipes are 150 mm (austenitic steel). Operating instructions were unclear and a three-way valve was in the middle position to adjust the water temperature bypassing the heat exchanger. The leak was isolated automatically from the reactor.

France:

Not applicable

Hungary:

No BWR

Japan:

Plant & time of occurrence: Fukushima Daiichi Unit 1, 02/1977; Shimane Unit 1, 02/1977; Tsuruga Unit 1, 05/1977; Fukushima Daiichi Unit 3, 06/1977; Hamaoka Unit 1, 11/1977

Location of failure: Return water nozzle for control rod drive

Conditions: Several cracks found in return water nozzle inner surface by liquid penetrant test.

Causes: Thermal fatigue due to mixing of hot and cold water: Mixing of hot reactor water (approx. 280°C) and cold water (approx. 40°C) flowing through return water nozzle of control rod drive caused relatively high frequency temperature fluctuation, thus generating cracks on metal surface due to thermal fatigue.

Major counter-measures: Application of structure and system modification to prevent hot and cold water mixing:

Control rod drive hydraulic system is changed to prevent the control rod drive water returning to the reactor nozzle, thus preventing thermal fatigue.

Remarks: Structure design changed.

Korea:

Not applicable

Slovakia:

No BWR

Spain:

Not applicable

Sweden:

Reactor Coolant System: 17 cracks. Root causes: Mostly 1a (valves).

Residual Heat Removal System: 11 cracks. Root causes: 2a

Reactor Water Clean-up system: 10 cracks. Root causes: 2a

Auxiliary Feedwater System: 7 cracks, 1 small leak. Root causes: 2c (design error)

Pressure Relief System: 12 cracks. Root causes: 1a

United States of America:

At about same time as the feedwater nozzle cracking noted above, cracking was also noted in the CRD return nozzles on some of the early BWR reactor vessels. This is also addressed in NUREG-0619. There have been no cracking incidents after 1993.

II. PRACTICAL EXPERIENCE AND INCIDENTS

II.1 BWR

C) Other large components

Belgium:

No BWR

Czech Republic:

No BWR

Finland:

No incidents

France:

Not applicable

Hungary:

No BWR

Japan:

Plant & time of occurrence: Tsuruga Unit 1, 09/1983

Location of failure: Reactor recirculation pump casing cover

Conditions: Leakage of reactor recirculation pump mechanical seal purge water to chiller system assumed.

Pump casing cover disassembled for penetrant test of all potential parts for leakage including the casing cover and cooling water coil welds (casing cover lower) and cooling tubes. Some indications recognized at casing cover lower surface.

Causes: Thermal fatigue due to mixing of hot and cold water: Insufficient sealing at suction cover seal welds provided a passage resulting in the mixture of hot and cold water. Thus, high cycle thermal fatigue caused cracks on casing cover. Fluctuation of seal purge water developed cracks, and propageted due to low cycle thermal fatigue.

Major counter-measures:

Application of structure to prevent hot and cold water mixing: Pump casing cover was replaced with new one.

Remarks: Structure design changed.

Korea:

Not applicable

Slovakia:

No BWR

Spain:

No incidents

Sweden:

RPVs internal parts: 18 cracks, 7 through-thickness cracks. Root causes: Mostly 2b and a few cases 1a.

United States of America:

In one early BWR (Nine Mile Point, a BWR 2 plant), thermal fatigue cracking and leakage occurred in the isolation condenser return line to the recirculation system. This was a typical valve leakage-type thermal fatigue event, in that the isolation valve leaked a small amount of water back toward the reactor primary coolant boundary. Interaction with a check valve was attributed to producing cyclic behaviour that resulted in growth of a crack though the pipe wall. This event is described in USNRC Information Notice 92-50.

A couple of years later, this plant had thermal fatigue cracking in the isolation condenser heat exchanger tubes/tubesheet intersections. The cause of this thermal fatigue cycling was draindown of the level in the inlet pipe to the tube side of the isolation condenser such that there was unstable condensation in the tubes near the surface of the water. There have been no other cracking events attributed to thermal fatigue.

II. PRACTICAL EXPERIENCE AND INCIDENTS

II.2 PWR

A) Feedwater nozzle and adjacent piping

Belgium:

No incidents

Czech Republic:

No incidents

Finland:

No incidents

France:

Just few small cracks (2 or 3) for 58 plants

Hungary:

No incidents

Japan:

No incidents

Korea:

No significant crack due to thermal fatigue

Slovakia:

No incidents

Spain:

No incidents

Sweden:

No incidents

United States of America:

Events (a)

A number of cracking incidents have occurred in the PWR feedwater piping at the steam generator nozzles. Most of these have occurred in the counterbore region of the weld between the pipe and the nozzle safe end. These cracks are attributed to thermal stratification and fatigue. The root causes for these events are described in part b. The following are known incidents of feedwater nozzle cracking:

At D.C. Cook Unit 2, in May 1979, leaking of feedwater from circumferential cracks in two feedwater lines occurred at the 16-inch (406.4 mm) elbows adjacent to the steam generator nozzles. The largest crack was 3.5 inches (88.9 mm) long at the outside surface and originated at an inside surface corner in the weld prep counterbore, away from the heat-affected zone. The crack was located at the top of the pipe. Additional cracks were found in all steam generator feedwater lines in similar locations in both units.

As a result of inspections in response to the Cook incident, in 1979, San Onofre Unit 1 reported finding indications of cracks in the feedwater nozzle to piping welds on two of the three steam generators. Palisades found cracking in the weld heat affected zone. Cracking was also identified in 1979 at Beaver Valley 1, Ginna, Kewaunee, Millstone 2, Point Beach 1 and 2, Robinson 2, Salem 1, Surry 1 and 2, Turkey Point 3 and 4, and Zion 1 and 2. In most of the cases, the cracks were located at the upstream corner of the counterbore, away from the weld. However, there were a few cases of cracking in the weld or in the heat affected zone. The cracks were most commonly located at the side of the pipe, 90° from the top.

In 1983-1987, another group of plants found feedwater nozzle cracking. Farley 1 found cracking in the reducer to nozzle weld in all three steam generators. Another crack was found in 1986 in one nozzle. Farley 2 found cracking in a pipe to nozzle joint. Trojan found counterbore region cracking and thermal sleeve erosion. The largest cracks were 50% through-wall. Beaver Valley 1 found additional cracking in all three steam generator nozzles. A large top to bottom temperature gradient was measured during periods of auxiliary feedwater injection, which resulted in pipe movements that have closed rupture restraint gaps. At Turkey Point 4, after replacing the feedwater reducers that had cracking in 1979, crack indications were again found in the reducer base metal after only one operating cycle. St. Lucie also found some cracking.

In 1983, Maine Yankee had a through wall crack with leakage. The crack was circumferential, 11 inches (279.4 mm) long on the outside surface and 35 inches (0.89 m) long on the inside, and was located at the bottom of the pipe. All three nozzles had some cracking, but only one propagated through wall. It was determined that the leak was caused by a water hammer event exacerbating a pre-existing thermal fatigue crack.

Indian Point 2 found cracking in 1989 in the feedwater nozzle blend radius, the nozzle bore region under the thermal sleeve, as well as in the pipe to nozzle weld region. Also, cracking was noted in the steam generator upper shell to transition girth weld. It was noted that the thermal sleeve had eroded and there was flow in the gap under the sleeve. The cracking was in the lower portion of the nozzle, and was shallow, less than 0.2 in. (5.1 mm)

In 1992, Sequoyah 1 found leakage from a through wall crack in one of the feedwater nozzles. The leaking crack was 2 inches (50.8 mm) long on the outside surface and 7 inches (177.8 mm) long on the inside, in the nozzle-to-transition piece weld region, 90° from the top of the pipe. The crack originated at a geometric discontinuity associated with the inside surface of the nozzle-to-transition piece weld. Sequoyah 2 found cracks in two of the four nozzles, with a maximum depth of 60% of pipe thickness. All four nozzles in both units were replaced. One fuel cycle later, cracking was again found in five of the eight replaced nozzles, the largest crack having a length of 8.625 in. (219 mm) and a depth of 25% of pipe thickness.

In 1992, Diablo Canyon 1 found cracking in the four steam generator feedwater nozzles. The cracking was located in base metal in the nozzle weld counterbore region, and was 360° around the circumference in some locations. In addition, there was significant erosion in the thermal sleeve. The gap between the nozzle and the thermal sleeve had increased from 0.020 (.05 mm) to a maximum of 0.222 in. (5.6 mm). The four Unit 1 nozzles and one in Unit 2 were replaced with a new design that included a longer thermal sleeve that protects the counterbore region. Subsequently it was found that the crack sizing, which was done by UT, had overestimated the crack depth by a factor of 10.

In 1992, Salem 1 found feedwater nozzle cracking and thermal sleeve erosion. Unit 2 has also had some cracking. The nozzles were replaced along with the installation of an improved thermal sleeve design. Beaver Valley 1 and both units of Prairie Island also found cracking in 1992. In 1993, Turkey Point 3 found counterbore region cracking as deep as 80% of through-wall. Connecticut Yankee found circumferential crack indications in three of the four nozzles in 1993. The largest was 18 inches (457 mm) long and 40% through wall. San Onofre 3, Robinson, and Farley 2 also found some cracking indications in 1993. Millstone 3 had minor longitudinal cracking detected in 1994.

Root Causes (b)

The root cause of most of the cracking incidents was thermal fatigue caused by thermal stratification cycling. In these plants, during heatup, hot standby, and low power operation, the main feedwater system is not used to supply water to the steam generators. This is because main turbine steam is not available to supply heat to the feedwater heaters. Also, main feedwater flow control is not accurate enough at low flows to be relied upon for steam generator level management. Most plants use the auxiliary feedwater system to supply feedwater during these periods. The source of auxiliary feedwater is the condensate storage tank, which is cold, typically at 100°F (37.8°C). The auxiliary feedwater tees into the main feedwater line upstream of the steam generator nozzle. During heatup, the amount of auxiliary feedwater needed is often less than 200 gpm (757 l/min) and does not fill the cross-section of the feedwater nozzle. Steam generator liquid is drawn out of the steam generator and travels upstream into the nozzle. This produces a top to bottom thermal gradient of up to 300°F (148.9°C) between the steam generator fluid and the auxiliary feedwater. Plants that have a separate auxiliary feedwater nozzle in the steam generator do not have stratification because the nozzle diameter is smaller and is located higher up in the steam generator.

Due to the nature of the feedwater level controllers, auxiliary feedwater flow is not supplied at a constant rate, but rather it fluctuates between near zero and near maximum. Since at maximum flow, stratification can be eliminated, each variation in auxiliary feedwater flow can produce a stratification stress cycle. Many of the automatic controllers vary the flow continuously, producing numerous stress cycles. Even small variations in flow affect the elevation of the hot-to-cold interface level, which can cause a local point on the pipe to be exposed to alternating hot and cold temperatures. Plants that control auxiliary feedwater flow manually have significantly fewer stratification cycles.

The weld between the steam generator feedwater nozzle and the connecting pipe contains a counterbore weld preparation end. The counterbore was originally provided for placement of a backing ring, and the geometry was also a function of the difference in required pipe wall thickness between the two ends of the joint due to the higher strength of the nozzle material. The counterbore geometry often contains a sharp corner at each end. These sharp corners act as geometric discontinuities that result in a point of stress concentration. Most of the cracking initiated at these points. In addition, many of these joints are to reducers or elbows, which already are locations of geometric stress concentration. Later designs either removed the counterbore or employed a gradual transition to reduce the stress concentration to reduce the potential for cracking.

In a few cases, such as at Beaver Valley, there is a long horizontal pipe run attached to the steam generator nozzle. With this geometry, large global bending moments occurred due to stratification. This caused high stresses at the nozzle, and high reactions at the pipe supports.

Cracking at the nozzle bore and inner radius regions such as occurred at Indian Point was additionally attributed to the flow erosion of the thermal sleeve allowing cold feedwater under the sleeve and onto the base metal. In addition, poor chemistry control allowed high oxygen water to exacerbate the flow-assisted corrosion.

References:

The USNRC has published several information notices and reports on the issue of feedwater nozzle cracking. These include the following:

IE Bulletin 79-13, Cracking in Feedwater System Piping, 1979

NUREG-0691, USNRC Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors, 1980

NUREG/CR-5285, Closeout of IE Bulletin 79-13, Cracking in Feedwater System Piping, 1991

Information Notice 91-38, Thermal Stratification in Feedwater System Piping, 1991

Information Notice 93-20, Thermal Fatigue Cracking of Feedwater Piping to Steam Generators, 1993

II. PRACTICAL EXPERIENCE AND INCIDENTS

II.2 PWR

B) PWR pressurizer surge lines

Belgium:

No incidents

Czech Republic:

No incidents

Finland:

No incidents

France:

No incidents

Hungary:

No incidents

Japan:

No incidents

Korea:

No significant crack due to thermal fatigue

Slovakia:

No incidents. Replacement of surge line elbow at Unit 3 of Bohunice NPP has been done in 2003 due to a high calculated value of cumulative fatigue usage factor. More detail information is in chapter II.5.

Spain:

No incidents

Sweden:

No incidents

United States of America:

Events

No cracking events have occurred on the pressurizer surge line or at the nozzles. The Trojan plant reported unexpectedly high piping displacements due to thermal stratification, which resulted in crushed insulation, closing of gaps at rupture restraints, and increased pipe support loads. The displacements caused plastic stresses and permanent deformation. Beaver Valley 2 also found larger than expected piping displacements, which caused snubbers to stroke out.

Root Causes

Thermal stratification in the pressurizer surge line occurs in significant magnitude primarily during plant heatup. The surge line connects at one end to the reactor coolant loop hot leg, and at the other to the pressurizer. During the early stages of a heatup, the pressurizer temperature increases faster than does the temperature of the hot leg. Typically, the pressurizer is initially water solid, and after reactor coolant loop flow in the connecting loop begins circulating, a bubble is drawn in the pressurizer. As the pressurizer level decreases, an outsurge of pressurizer water at a temperature of approximately 425°F (218°C) travels down the pipe, but generally at a flow rate that is not high enough to fill the entire cross section of the pipe. The bottom of the pipe contains water from the hot leg, at a temperature of about 150°F (65.6°C). A top-to-bottom temperature gradient of 300°F (166.7°C) has been measured at several plants. As the heatup progresses, the magnitude of the stratification is proportional to the difference in temperature between the hot leg and the pressurizer, which does not decrease significantly until entering Mode 2 (start of power ascension). During the heatup, changes in letdown and charging flow, and actuations of pressurizer main or auxiliary spray cause partial insurges or outsurges in the line, that result in thermal stratification cycling. Bowing of the lines occurs due to global stratification moments; stress cycling occurs at the hot leg and pressurizer nozzles due to the resisting of these moments.

References (USNRC Documents):

IE Bulletin 88-11, Pressurizer Surge Line Thermal Stratification, 1988

II. PRACTICAL EXPERIENCE AND INCIDENTS

II.2 PWR

C) Other piping (please indicate the system, diameter of the piping and possibility of isolation from reactor)

Belgium:

Tihange Unit 1 :Leak (of about $1m^3/h$) in an elbow of an Emergency Core Cooling System injection line in a pipe section (6-inch) directly connected to the RCS hot leg downstream of the first isolation valve. (Before 01.01.1994)

Ref.: NEA/CSNI/R(98)8 "Specialists Meeting on Experience with Thermal Fatigue in LWR Piping Caused by Mixing and Stratification, pp 103-114

Czech Republic:

No

Finland:

2 leaks and 2 cracks.

A small leak was discovered during an inspection walk-down in the pressuriser auxiliary spray line valve (Loviisa 2, 1994). Small heating flow is continuous though the valve (1 mm hole). Nominal diameter of the pipe is 50 mm. Isolation is not possible. The valve was changed and the pipe modified. A crack was also found by UT in the same valve of Loviisa 1 unit. The main reason is stratification.

A small leak was discovered in the so-called cross-tie line between the hot and cold leg. The line is closed by one valve during normal operation. A first indication of a reactor primary circuit leak at Loviisa 2 came from the steam generator compartment air activity measurement system on January 1997. Short vertical (height 400 mm) pipe (diameter 50 mm) is connected to the horizontal pipe (50 mm). Hot turbulence water can reach the horizontal pipe, cools down there and causes stratification (pour thermal insulation). A crack was found by UT in another cross over pipe in the same unit. The pipes were repaired and the valves tightened. Later on the cross-tie lines have been removed as unnecessary.

France:

Leak and few cracks on Safety Injection Systems, 10" diameter, class 1 but after first isolation valve on BUGEY 2; chemical pollution, stratification and support problems.

2 Leaks on Safety Injection Systems, 10" diameter, connected to primary loops for 2 plants DAMPIERRE 1 and 2 (class 1, not isolable), one deep crack after 9 months of operation of one repaired solution; vortex and stratification connected to continuous valve leaking, ΔT fluid greater than 150°C.

Leak on Reactor Heat Removal System, 10" and 4" tees, class 2 part of the system, (2 trains damaged, one leak) on CIVAUX 1; mixing tees, around 1500 hours at 140°C ΔT fluid, during intermediate shutdown.

Many Cracks on Reactor Heat Removal System, 10" and 4" tees, more than 100 tees damaged on 58 plants, a lot of small cracks (2-3 mm) and few deep cracks (less than 10, greater than 50% of the thickness); mixing tees from 300 to 3000 hours at 140°C ΔT fluid, during intermediate shutdown.

Hungary:

No incidents

Japan:

Plant & time of occurrence: Gengai Unit 1, 06/1988

Location of failure: Welds on horizontal piping (8B) between RHR line No.1 isolation valve and hot leg (circumferential direction)

Outline: A 97mm length crack (approx. 1.5mm on outer surface) on welds inner surface in circumferential direction, crack starts at inner surface and propagates through along the boundary of weld metal and base metal penetrating to the external surface through the weld metal.

Striation recognised on the fracture surface. (0.3-1.0μm intervals)

Causes: Thermal fatigue due to stratification caused by valve gland leakage:

Hot water leakage to isolation valve packing generates stratification at RHR piping horizontal part where cold water constantly stays. Repetition of cycle (valve expansion due to hot water leakage) → closure of leakage path → leakage stop → valve contraction due to heat discharge & cooling → leakage start) results repetition of formation & elimination of stratification, thus generating high-cycle thermal fatigue.

Major counter-measures:

Reinforced monitoring of stratification occurrence: Thermometer installed to detect stratification.

Reinforced monitoring of unexpected flow generation: Leak off line insulated to improve gland leakage detecting performance.

Plant & time of occurrence: Mihama Unit 2, 04/1999

Location of failure: Base metal at the bend of excess extracted water discharge piping (2B&2DR)

(at rear)

Outline: Penetrating crack (approx. 24mm on inner surface & approx. 7mm on external surface) at approx. 45° toward tube axial direction plus approx. 14.5mm crack & several fine cracks in inner surface of opposite side of tube axis.

Benchmark on fracture surface. Precise observation recognized striation patterns.

Valve downstream from leak area is usually closed.

Causes: Thermal fatigue due to cavity-flow type stratification: Since cold water stays in the tube where flow does not occur during the normal operation, inflow of hot water (primary cooling water) generates temperature boundary. As the boundary situates at the piping bend, thermal stress fluctuates in response to the fluctuation of temperature boundary. With the residual stress, this generated/ developed the crack. (The stratification fluctuates periodically at approx. 100 seconds causing approx. 100°C temperature difference.)

Major counter-measures:

Elimination of high stress part from the hot/cold water fluctuation region: Bend of the subject piping relocated to prevent impact of stratification.

Reduction of mean stress: Subject piping replaced with member of less residual stress. Where the impact of stratification is expected, location of bend and impacts of material on residual stress are reflected on the design standards.

Plant & time of occurrence: Tsuruga Unit 2, 07/1999

Location of failure: Connecting piping of regenerative heat exchanger (extraction piping elbow). Regenerative heat exchanger shell

Outline: Twelve (12) large and small cracks were recognized in the axial and circumferential directions in the connecting piping. Cracks were also recognized in five (5) regions in the shell internal surface of the heat exchanger.

Cracks in both the axial and circumferential directions originated from several points on the internal surface and developed therefrom. The cracks in the shell internal surface were hexagonally patterned.

Striation pattern and beach marks were confirmed on the fracture surface. The beach mark intervals become narrower according to the crack depth.

No abnormalities regarding the materials or general corrosion were recognized.

Causes: Thermal fatigue caused by overlapping of long cycle thermal fluctuation and short cycle thermal fluctuation due to hot/cold water mixing:

Since the regenerative heat exchanger is of an inner shell structure, the flow pattern changes at the junction where the hot bypass flow outside of the inner shell and the main cold flow meet. Therefore, the temperature distribution at the connecting piping and shell main body changes cyclically. The temperature fluctuation in a relatively long cycle due to this flow pattern change and the temperature fluctuation in relatively short cycle due to mixing of hot bypass flow and cold main flow overlapped. This generated cyclic stress exceeding the fatigue strength on the connecting piping and shell main body, thereby generating thermal fatigue cracks.

Major counter-measures:

Regenerative heat exchanger replacement: The regenerative heat exchanger was replaced with a model that has no inner shell.

<u>Korea</u>:

15 operating PWRs in Korea. We have 2 leakage accidents due to thermal fatigue in PWR after 1.1.1994 in Korea.

Case 1: RCS sampling line in 2001
a) Small leak in piping
b) Thermal transient
Case 2: RCS sampling line Heat Exchanger in 2001
a) Small leak in piping
b) Thermal transient

<u>Slovakia</u>:

No incidents

<u>Spain</u>:

No incidents

<u>Sweden:</u>

No incidents

<u>United States of America</u>:

High Pressure Safety Injection Piping (events and root causes)

Through-wall leakage has occurred in unisolable high-pressure safety injection piping. A leak occurred at Farley 2 in December 1987 and was located in the heat-affected zone of the weld between the first elbow and the horizontal run upstream of the RCS cold leg nozzle. The pipe is 6-inch (152.4 mm) in diameter, Schedule 160, made of 304 stainless steel. The line rises vertically from the cold leg, then turns horizontally for 3.5 feet (1.07 m) to the check valve. The crack location was about 3 feet (0.9 m) from the ID of the RCS cold leg. The orientation of the crack was circumferential, in the bottom third of the pipe, 120° around the pipe on the inside diameter and about 1 inch (25.4 mm) long on the outside. The leak rate was estimated to be 0.7 gpm (2.6 l/min) when the crack was discovered, which was while the plant was at 33% power returning from a refueling outage. Farley is a Westinghouse design, 3 loop plant which had been operating for 6.5 years at the time of the failure.

The cause of the crack was turbulence penetration of hot RCS fluid interacting with cold fluid from valve leakage that had stratified at the bottom of the pipe. High pressure injection is supplied by the charging pumps, operating at about 2450 psia (16.9 MPa), which is higher than RCS pressure nominally at 2250 psia (15.5 MPa). This pressure difference enables cooler, stagnant fluid upstream of the isolation valve to potentially enter the unisolable piping when the valve leaks. It was originally believed that the failure was caused by thermal cycling due to the check valve motion; testing demonstrated that it was instead caused by turbulence penetration of the RCS fluid. Temperature monitoring performed subsequent to the cracking indicated that downstream of the check valve, the difference in temperature between top and bottom of the pipe was 215°F (101.7°C). The bottom temperature was not constant; the amplitude of the cycling was 70°F (38.9°C) with a cycle period between 2 and 20 minutes. On the upstream side of the check valve, the top to bottom temperature difference was 128°F (71.1°C) as a result of heat transfer through the valve. The leaking valve was a 1in. (25.4 mm) manual globe valve on the Boron Injection Tank (BIT) bypass line. The leak rate was estimated at 0.5 gpm (1.9 l/min).

No other U.S. plants have found cracking in this line. Some plants have measured thermal stratification at times when not all of the reactor coolant pumps are in operation, which was caused by check valve back-leakage.

Residual Heat Removal Piping

A number of power plants measured significant amounts of thermal stratification in the RHR piping near the reactor coolant loop, but no cracking was found. Some plants have noted top-to-bottom gradients just downstream of the RHR heat exchanger, where the bypass flow mixes with the heat exchanger discharge, but cracking has not been found. At Sequoyah, a 70% thru wall crack was found on the upstream side of the first check valve off of the reactor coolant loop in the RHR return line, which is an isolable location. The crack was attributed to IGSCC but thermal fatigue was considered a possible contributing cause.

Makeup / High Pressure Injection Nozzle

In January 1982 a crack in a makeup / high pressure injection nozzle occurred at the Crystal River plant. The plant had been operating for five years; the leak occurred during normal operation and measured one gpm. The nozzle is 2.5 in. (63.5 mm), Schedule 160 carbon steel and is shop welded to a safe end made of 316 stainless steel. The safe end is then field welded to the 2.5 in. (63.5 mm), Schedule 160 type 316 stainless steel makeup line. A thermal sleeve of type 316 stainless steel was installed in the nozzle by a mechanical press fit. The crack was in the safe end to pipe weld. The orientation was circumferential, 140° around on the outside surface. There were actually two separate cracks; one initiated on the inside due to thermal fatigue caused by turbulent mixing of the hot reactor coolant and the cold makeup water; the other originated on the outside and was believed to be caused by mechanical vibration. Other cracks were found in the safe end, check valve, and the thermal sleeve.

In the B&W plant design, makeup does not pass through a regenerative heat exchanger and thus the flow temperature is much colder than the RCS temperature. A thermal sleeve is provided in the nozzle to protect the immediate area from thermal fatigue. However, if there is any problem with the sleeve, turbulent mixing of fluids with a large temperature difference can result. Also, in-leakage stratification can potentially occur in the makeup / HPI line. At Crystal River, the thermal sleeve was found to have been loosened, and there was a gap between the sleeve and the safe end. The gap allowed cold makeup flow to mix with the RCS fluid on the nozzle surface, and the resulting flow induced vibrations caused additional wear of the thermal sleeve. However, there was no thermal stratification. Five of the other B&W plants also found cracking at this location although it was not through wall.

In April 1997, a 2 gpm (7.6 l/min) leak occurred at Oconee 2 in the makeup/HPI nozzle. The leak was through a circumferential crack in the weld between the safe end and the makeup line. The crack was 360° around on the inside surface and 77° around on the outside, centered about 30° off top dead center. Cracking was also found in the thermal sleeve, and around the connection of a constant flow "warming" line to the makeup line. Examination of the thermal sleeve found that it was loose and had a significant gap. The gap had been present for several years but had not been detected. The thermal sleeve was of the original design and had not been replaced after the Crystal River event.

The cause of the cracking was also turbulent mixing of the cold makeup flow with the hot RCS flow. The gap behind the sleeve allowed the RCS fluid to penetrate back past the safe end. Variations in the makeup flow rate plus high frequency turbulence effects resulted in thermal cycling at the crack location. It was postulated that the sleeve had been loosened by differential thermal expansion cycling due to back flow with partial pump operation, and evidence suggested that further wear occurred due to flow induced vibrations.

One of the replacement HPI nozzle thermal sleeves at Oconee was found to be cracked again in 2001. The crack was located at the reactor coolant loop end of the sleeve, in the A1 nozzle of unit 3. The crack was oriented axially, about 2 in. (50.8 mm) long, and was through-wall. A small crack was also found in one of the Unit 2 thermal sleeves.

The cracking was attributed to high cycle thermal fatigue caused by turbulence penetration. Previous computational fluid dynamics analyses had determined that a minimum required makeup line flow of 30 gpm (113.6 l/min) is required to push the turbulence penetration out of the thermal sleeve. Two nozzles provide makeup flow during operation, supplied by the makeup/HPI pumps. Monitoring of flow rates indicated that a mismatch was occurring, with one nozzle receiving 75% of the total flow of 37 gpm (140 l/min) and the other 25% during four pump operation, but during three pump operation, these percentages reversed. The variations in flow rate resulted in varying turbulence penetration and consequently, cyclic thermal stresses. The thin thermal sleeve was not effective in preventing crack initiation and propagation.

Davis-Besse also reported finding axial cracks in two makeup/HPI nozzle thermal sleeves in 2002. The cracks were part through-wall, and were attributed to the same mechanism as the original thermal sleeve cracking issue.

Pressurizer Spray Piping

Several plants measured significant thermal stratification in the upper pressurizer spray piping during plant heatup and cooldown when the reactor coolant pumps are not in service. At such times, main spray and bypass spray flow is not available. Pressurizer steam can re-enter the upper spray piping in between actuations of auxiliary spray, causing stratification and thermal cycling. Stratification cycles also occur at initiation and termination of main spray flow. No cracking has been found to date.

Reactor Coolant Loop Drains

Each of the reactor coolant loop crossover legs normally contains a drain line, typically 1.5 in. (38.1 mm) to 2.5 in. (63.5 mm) in diameter. Westinghouse plants also have an excess letdown line, which is used primarily during startup to facilitate drawing a steam bubble in the pressurizer. The excess letdown line is similar in size and orientation to the cold leg drain lines. The lines are generally oriented vertically down from the reactor coolant loop, followed by a horizontal run to an isolation valve. In recent years there have been leakage events in these lines.

In September 1995, a leak occurred in a cold leg drain line at Three Mile Island 1. The crack was located in the weld between the first elbow downstream of the reactor coolant loop nozzle and the horizontal pipe run. At the time of the leak, the plant had been operating for 21 years and was at 0% power, beginning a cooldown. The leak rate was 20 “drops” per second. The pipe routing is vertically down 14 in. (355.6 mm) from the cold leg, then 7.3 ft. (2.23 m) horizontally to the first valve. The vertical run is 1.5 in. (38.1 mm) diameter pipe, the horizontal run is 2 in. (50.8 mm) diameter, and the elbow between them is a reducing elbow. The elbow and horizontal run are type 316 stainless steel, and the vertical run included an Inconel safe end. The location of the crack was in the weld between the elbow and the horizontal pipe, near the top of the pipe. The distance from the crack to the cold leg inside diameter was 14 in. (355.6 mm), or about 10 pipe diameters. The crack was circumferential, 2 in. (50.8 mm) long on the inside and 0.55 in. (14 mm) long on the outside surface, centered at the 11 o’clock position. The drain line was not insulated.

The cause of the cracking was thermal fatigue, attributed to turbulence penetration of the hot RCS fluid extending into the horizontal pipe. The horizontal pipe, being uninsulated, allowed heat to escape to the surroundings, and when the turbulence penetration reached the horizontal run, this produced thermal stratification in the line. Fluctuations in the extent of turbulence penetration caused local thermal cycling at the elbow weld, a point of stress concentration.

There were other contributing causes as well. Two improperly installed pipe support U-bolts restricted the free thermal expansion of the pipe and produced a 37 ksi (255.1 MPa) stress at the elbow. Also, the placement of the pipe supports caused the horizontal pipe to slope upward away from the elbow, which facilitated thermal stratification and cycling in this pipe run. In addition, the toe of the cracked weld had a pre-existing notch.

A similar crack occurred at Oconee 1 in February 2000. During normal operation, a small leak of 0.04 gpm (.15 l/min) was discovered during a walkdown. It was located in an RCS cold leg drain line, at the first elbow downstream of the cold leg. The crack was near the center of the extrados of the elbow, at the elevation of the top of the horizontal pipe. The pipe was 1.5 in. (38.1 mm) Schedule 160, type 316 stainless steel. The pipe routing was 11 in. (279.4 mm) vertically down from the cold leg, then 5.5 ft. (1.7 m) horizontally to a downturn, then 2 ft. (0.61 m) to the first valve. The line was uninsulated. The distance from the RCS inside surface to the crack was 13 in. (330.2 mm), or about 9 pipe diameters. The crack orientation was 45° off vertical, 0.5 in. (12.7 mm) long on the inside and 0.2 in. (5 mm) long on the outside surface. There were two major cracks that originated in the ID, plus crazed cracking along the top surface of the horizontal pipe. The cracking was all in base metal, and had propagated slowly over time.

The crack was attributed to turbulence penetration from the RCS intermittently extending into the horizontal run. The lack of insulation of the pipe made it easier for the horizontal portion to stratify. The vertical pipe length was short enough for the hot fluid to periodically extend into the horizontal run, but not so short as to keep the horizontal pipe warm all the time. A contributing cause was that the Post Accident Liquid Sampling system is connected to this line, and samples taken quarterly introduced a temperature fluctuation of 104°F at the elbow.

At Callaway, a leak occurred in October 1995 in the excess letdown line in the first elbow downstream of the RCS crossover leg. The line is 2 in. (50.8 mm) Schedule 160, type 304 stainless steel, and the crack was located 13 in. (330.2 mm) from the reactor coolant loop inside surface, or about 8 pipe diameters. The crack was in the center extrados of the elbow, oriented 45° from the pipe centerline. The leak was 2 gpm (7.6 l/min) and the plant was at full power operation at the time. The crack was located in the heat affected zone of the weld between the first elbow downstream of the RCS crossover leg and the horizontal pipe. The orientation was circumferential, at the bottom of the pipe, 210° around on the inside surface and 10° around on the outside. The geometry of the Callaway line is a 2 in. (50.8 mm) diameter vertical run, 13 in. (330.2 mm) down, followed by a horizontal run that is 12 ft. (3.66 m) long to a vertical upturn. About 2.5 ft. (0.76 m) from the first elbow is a tee to a ¾ in. (19 mm) Schedule 160 drain pipe. The line was fully insulated.

The ¾ in. (19 mm) line contained a flange, which was found to be interfering with a pipe support plate on the floor. As the RCS heated up, the flange scraped the plate from crossover leg temperatures of 310°F (154.4°C) to 520°F (271°C), then completely hung up the pipe from 520°F (271°C) to 560°F (293.3°C). The cause of the crack was attributed to low cycle fatigue due to high bending stresses resulting from the thermal interference. Subsequent metallographic examinations of the crack concluded that it had initiated on the inside surface and was typical of mid to high cycle fatigue. It was postulated that the crack was due to the combined effects of the thermal interference and high cycle fatigue, possibly from flow induced vibrations, attributed to turbulence of the RCS flow into the branch being close to the resonant frequency of the piping. This location is also potentially susceptible to thermal stratification cycling due to reactor coolant turbulence penetration periodically entering the horizontal piping, as described in the above events.

References (USNRC Documents)

Information Notice 82-09, Cracking in Piping of Makeup Coolant Lines in B&W Plants, 1982

Generic Letter 85-20, Resolution of Generic Issue 69: High Pressure Injection/ Makeup Nozzle Cracking in Babcock & Wilcox Plants, 1985

Information Notice 88-01, Safety Injection Pipe Failure, 1988

Bulletin 88-08 and Supplements 1-3, Thermal Stresses in Piping Connected to Reactor Coolant Systems, 1988-89

Information Notice 97-46, Unisolable Crack in High Pressure Injection Piping, 1997

II. PRACTICAL EXPERIENCE AND INCIDENTS

II.2 PWR

D) Other large components

Belgium:

Thermal fatigue cracking in thermal barrier of primary pump

Czech Republic:

No incidents

Finland:

No incidents

France:

1 deep crack (17mm) and a lot of small cracks on Reactor Coolant Pump thermal barriers, no leak, not directly the pressure boundary part of the pump; mixing zone, continuous operation with small ΔT (around 40°C) few small cracks in valve inner surface on high fatigue load systems: CVCS, auxiliary pressuriser spray line…, low cycle fatigue, high thermal shocks (up to 200°C), small inner surface radius

Hungary:

High cycle thermal fatigue cracking in the rotor and drive wheel of the main circulation pumps in the area of mixing flows with temperature difference of 120 degree C. Root cause: design deficiency, partial change of material was completed

Japan:

No incidents

Korea:

No significant crack due to thermal fatigue

Slovakia:

No incidents

Spain:

No incidents

Sweden:

Only a few cases on Reactor coolant pumps and 1 case on Vessel Head Penetration. Root causes: 1b (Generic, design error).

United States of America:

There has not been any cracking in other components that has been attributed to thermal fatigue or thermal stratification.

II. PRACTICAL EXPERIENCE AND INCIDENTS

II.3 Other plants than BWR or PWR

Belgium:

Not applicable

Czech Republic:

Not in NPP, but some case in coal power plant

Finland:

Not applicable

France:

Some thermal fatigue cracks on collectors of French fossile plants, low cycle fatigue, high thermal shocks, cold water injection, no significant effect on plant availability due to low number of hours of use

Some on SPX FBR (now shut down)

Hungary:

Not applicable

Japan:

No incidents

Korea:

CANDU (4 operating CANDUs in Korea)

We have 1 leakage accident due to thermal fatigue in CANDU after 1.1.1994 in Korea.

A) Feedwater nozzle and Adjacent piping
No significant crack due to thermal fatigue
B) Purification piping in 2001
a) Medium leak in piping
b) Thermal transient

Slovakia:

Not applicable

Spain:

No incidents

Sweden:

Not applicable

United States of America:

Not applicable

II. PRACTICAL EXPERIENCE AND INCIDENTS

II.4 Estimate of whole forced shut down time for all thermal fatigue related incidents?

Belgium:

Tihange Unit 1: about 2 weeks (from June 18 to June 30, 1988)

Czech Republic:

None

Finland:

Days: 25

France:

(1): negligible, use of steam generator replacement operation for inspections and destructive examinations (II.2.A)
(2): few weeks for one PWR- 900MWe (II.2.C)
(3): few months on 2 PWRs- 900MWe (II.2.C)
(4): 1 year for 4 PWRs- 1450MWe (II.2.C)
(5): few days on outage duration for each of the 52 remaining plants (II.2.C)
(6): few weeks of some plants (II.2.D)
(7): few weeks on limited number of different plants (II.2.D)

Hungary:

No forced shut downs due to incidents related to thermal fatigue

Japan:

No answer

Korea:

A) RCS sampling line in 2001: None

B) RCS sampling line Heat Exchanger in 2001: 2 day
C) CANDU Purification Piping in 2001: 10 days
Total Time: 12 days

Slovakia:

No forced shut downs due to incidents related to thermal fatigue

Spain:

No forced shut downs due to incidents related to thermal fatigue

Sweden:

About 6 months.

United States of America:

It is estimated that each of the through-wall crack events caused an average of one month of forced outage or outage extension time. The total can be estimated at six months.

II. PRACTICAL EXPERIENCE AND INCIDENTS

II.5 Other comments on practical experience and incidents

France:

Different other locations are under surveillance: pressuriser spray and auxiliary spray lines, RHR system and RCL noozle; without any crack detected anywhere

United States of America:

Many plants have instrumented piping systems and measured thermal stratification profiles. Thermal cycling has also been measured. Analysis of this data has provided many insights, which are helping to develop screening criteria to identify potential locations and mechanisms for thermal fatigue damage. Analytical efforts are currently underway to better quantify turbulence penetration effects.

Several BWR plants have experienced pipe hanger damage due to stratification in the feedwater piping beyond the primary coolant isolation valves. This stratification occurs as a result of relatively hot reactor water cleanup flow entering long horizontal sections of feedwater piping, following feedwater flow termination with relatively cold feedwater temperature. The returning reactor cleanup water flow tended to stratify in the tops of the isolated lines. There has been no leakage observed due to this occurrence.

Slovakia

Based on thermal stratification monitoring, a high cyclic load has been identified on the elbow No.6 of pressurizer surge line at Unit 3 of Bohunice NPP. A regular evaluation of a fatigue usage factor has shown that it reached an important value. Following countermeasures have been implemented:

- installation of temporary monitoring system (32 strain gauges and 16 termocouples) to determine and to verify respectively the temperature loads equivalent (measurement in the period of two fuel cycles)

- based on the annual evaluation of a fatigue usage factor the preparation of a replacement of the elbow have been initiated by the utility and the replacement took place in 2003.

- preparation and experimental testing of proposed modification of operational regimes, especially during the start-up of the unit in order to reduce the thermal loads on the pressurizer surge line.

- implementation of wide material research programme including surface and volumetric NDT, mechanical and low fatigue tests, metallographic analyses, etc.

- re-evaluation of a new elbow section from the point of view of validity of LBB criteria after the replacement.

Remark: This case has been reported at the “Third International Conference on Fatigue of Reactor Components”, Seville, Spain, 3-6 October 2004.

III. COUNTERMEASURES RELATED TO STRATIFICATION

For items III 2 to III 4 please describe countermeasures taken considering items listed below:

a) Identification of locations with potential risk, particularly regarding cyclic stratification. Screening criteria. What disciplines are represented in the review team?
b) Use of plant normal instrumentation
c) Temperature measurements
d) Fatigue meters
e) Inspections (methods, inspected area, frequency)
f) Testing of leak tightness of valves
g) Optimising operating conditions, training of operators
h) Changes in the systems
i) LBB
j) Other measures

III.1 General approach and policy

Belgium:

As for other degradation phenomena, operational experience related to stratification from other countries is analysed. If incidents appear in foreign plants, their applicability to the Belgian plants is evaluated. As such, actions have been undertaken regarding stratification in the feed water system (see III.2.A) and stratification in the PWR pressuriser surge line (see III.2.B).

In general, stratification is addressed by performing short-term or long-term external wall temperature measurements, followed by stress and fatigue analyses based on measured stratification transients.

Czech Republic:

Periodic temperature measurements on outer surfaces and periodic fatigue analyses for selected locations. In case of cold injection or mixing of medium with significant ΔT, non destructive testing has to be performed.

Finland:

After the 1987 and 1988 incidents in foreign PWR's mapping of the potential stratification locations was started. Power companies did not fully understand the seriousness of the problem, which resulted into slow progress. After the incidents in Loviisa the work intensified.

Training: Designers, operators and plant personnel should be aware about thermal mixing and thermal stratification. Revised operating practices can decrease the risk of thermal fatigue.

The responsibility is always in the power company and the authority review there work.

France:

detailed root cause analysis / comparison with international field experience
repair-replacement analysis to re-start the plant

life evaluation of repaired-replaced components, and definition of construction specifications for replacement (surface finish, residual stress, system modification, new layout, device development…)

define surveillance (temperature measurements, valve leak tightness…) and ISI (performance and periodicity) of the new solution

define condition and screening criteria to review all similar locations (same systems / other systems, same plants / other plants)

define complementary surveillance or ISI, specifically for these potentially damaged areas

anticipate some repair – replacement programs

define and develop a complementary R&D program, on thermal-hydraulic load evaluation, material properties, stress and damage analysis

review and improve existing codes requirements

report periodically to French Safety Authority

Hungary:

In the 2000 mapping of the potential stratification locations was started. As a pilot study the monitoring of the surge line stratification phenomena was implemented the assessment of the results are being performed. Not only the stationary temperature measurements are used but additional measurements are installed as well.

Japan:

Establishment and verification of the standards for high-cycle thermal fatigue shall be carried out as described below.

(1) Revision of MITI Ordinance No. 62

Regulations to prevent damage due to high-cycle fatigue shall be introduced into MITI Ordinance No. 62.

(2) Verification of explanation

The utility companies have voluntarily practiced certain damage prevention measures in which overseas and domestic events are reflected. Concerning their engagement with the prevention of damage caused by high-cycle thermal fatigue, as a part of these measures, ‘Explanation of Impact of High-cycle Thermal Fatigue’ shall be attached to the construction plan for renovation/ replacement. Since the construction plan must be applied for permission and notification, the aforesaid Explanation shall be verified in the review of the said construction plan.

(3) Establishment of private sector standards

Promoting the activities of the private sectors to develop standards for high-cycle thermal fatigue in the future, and the enacted private sector standards shall be considered in the interpretation of MITI Ordinance No. 62.

Korea:

Consideration of thermal stratification in piping design

Measurement of temperature during hot functional test

Enhanced ISI for the piping susceptible to thermal fatigue

Slovakia:

After the incidents in foreign PWR’s and as a result of R&D activities in Slovakia a mapping of the potential stratification locations was started and first measurement systems were installed (since 1993). An example of concrete specific countermeasures- also see in chapter II.5.

Spain:

The policy that we follow in Spain is to analyse the operational experience in other countries, mainly USA, as 8 out of 9 of our units are of W or GE design. If incidents appear in foreign plants, then they are analysed to determine the applicability of the countermeasures adopted or proposed.

Sweden:

Detected cracks may be removed without any repair of the material provided that safety margins are maintained and necessary measures are adopted to prevent the occurrence of new damage. If the damage is of such an extent that the safety margins cannot be maintained, the component or part of the component shall be replaced or repaired. Such components should be included in a revised ISI-programme.

United States of America:

The U.S. Nuclear Regulatory Commission issues generic letters and information notices to alert nuclear facilities of thermal fatigue events that have occurred and their root causes, and requires preventative actions. The Electric Power Research Institute (EPRI) has sponsored a number of research programs, including the development of thermal fatigue management guidelines. The guidelines include screening criteria to identify locations of potential thermal fatigue damage, and methodology for mitigating or preventing damage. Thermal fatigue operating experience and corrective actions taken have also been compiled in order to help plants avoid damage. EPRI has sponsored training seminars at each nuclear facility, in which the screening criteria are applied to the specific plant geometries, and the plant personnel are instructed in the application of the thermal fatigue management guidelines. This training has increased the awareness of thermal fatigue issues at the plants.

III. COUNTERMEASURES RELATED TO STRATIFICATION

III.2 BWR

A) Feedwater nozzle and adjacent piping

Belgium:

No BWR

Czech Republic:

No BWR

Finland:

Nozzle
improvement of feedwater control in low flow conditions
temperature measurements in the nozzle using thermocouples on the outer surface (difficulties to interpret the results)
monitoring of plant normal temperature measurements

France:

Not applicable

Hungary:

No BWR

Japan:

See II.1. A)

Korea:

Not applicable

Slovakia:

No BWR

Spain:

Following the recommendations established in NUREG-0619:

Replacement of the safe-ends of the incoming feedwater piping.

New design of the thermal sleeves of the incoming feedwater piping, improving its design to prevent from leaks of the incoming cold water to the nozzle avoiding its erosion.

Sweden:

-

United States of America:

Cracking in BWR feedwater nozzles has been managed using a number of different countermeasures:

All plants have installed an improved feedwater sparger inlet. For most plants, the inlet is a double piston-ring, triple sleeve sparger that routes any leakage past the primary seal between an inner set of sleeves, such that leakage does not impinge directly on the nozzle inside diameter. This configuration is standard on all relatively newer plants.

Plants with nozzle ID cladding have removed the cladding. This configuration is standard on all relatively newer plants.

Plants have implemented improved feedwater flow controllers to avoid on-off flow cycling.

Many plants have re-routed the reactor water cleanup flow so that it enters both feedwater loops. This is standard on all relatively newer plants.

Some plants have implemented temperature monitoring on the feedwater nozzles as an approach to justifying a longer inspection interval.

Many plants have performed nozzle flaw tolerance evaluations to justify an increased interval for nozzle examination. These analyses show that cracks will not grow to an ASME Section XI allowable flaw size even if crack initiation were to occur.

III. COUNTERMEASURES RELATED TO STRATIFICATION

III.1 BWR

B) Other piping

Belgium:

No BWR

Czech Republic:

No BWR

Finland:

The main focus was on valves, which are closed but not tight enough. The area was considered vulnerable if there was only one valve (control valve is supposed to leak), the temperature is high enough and pressure difference is possible.

In the review teams there were people from different disciplines (stress analyses, thermal hydraulic, systems and operation).

Temperature measurements of vulnerable areas using thermocouples on the outer surface

France:

Not applicable

Hungary:

No BWR

Japan:

See II.1. B)

Korea:

Not applicable

Slovakia:

No BWR

Spain:

-

Sweden:

-

United States of America:

CRD return nozzle cracking has been eliminated by terminating any flow to the nozzles. Most plants have cut the line to the nozzle and capped the nozzle. More recent plants do not have the nozzle. The flow that was previously routed to this nozzle is now routed through the CRD nozzles in the bottom of the reactor.

For the plant that experienced cracking in the Isolation Condenser return line, to mitigate the valve-leakage induced cracking, valve modifications were made to keep the isolation valve (which is a globe valve) as leak-tight as possible. However, there is still a small amount of leakage. The isolation valve is also used for containment isolation and has a more stringent leakage acceptance criteria to help keep the leakage in the reverse direction to a minimum. The isolation condenser return piping is instrumented with thermocouples to monitor for thermal stratification and cycling to assure that the temperatures are below an acceptance criteria based on piping stress analyses. To mitigate the heat exchanger cracking, a small amount of CRD return flow has been routed to the heat exchanger as a keep-full system. This combination of mitigation approaches has prevented further thermal fatigue in the heat exchanger tubes and assures stratification/cycling in the return piping remains below acceptance limits.

Stratification in feedwater piping has been handled by optimizing the piping supports so that stratification movements can be accommodated.

III. COUNTERMEASURES RELATED TO STRATIFICATION

III.2 PWR

A) Feedwater nozzle and adjacent piping

Belgium:

Short-term or long-term detailed external wall temperature measurements have been performed on the feed water lines of almost all Belgian NPPs. At the same time, data from normal plant instrumentation, such as auxiliary feed water flow rate, are recorded in order to facilitate the interpretation of measured stratification transients.

From the measurements, typical stratification phenomena have been identified, and for each phenomenon, a representative transient has been derived. Some of the phenomena are generic, others are really plant specific. All representative transients are transformed into a set of 'design' stratification transients that are used in the stress and fatigue analyses of the different components of the feed water system (steam generator feed water nozzle, feed water lines, feed water line reactor building penetrations). The number of occurrences of the stratification transients is determined in a conservative way from the measurements. The stress and fatigue analyses are performed using the in-house developed code THERMAXS (see V.1).

In the framework of steam generator replacement projects, stratification has been considered in a detailed way by:

- ✓ including stratification load-cases in the design specification of the replacement steam generators;
- ✓ defining envelope stratification load-cases for the ASME Cl.2 and/or Cl.1 analyses of the feed water lines;
- ✓ providing different types of anti-stratification devices in the feed water nozzles of the replacement steam generators;
- ✓ adapting the lay-out of the feed water lines when necessary in order to eliminate particular types of stratification transients;
- ✓ adapting the supports of the feed water lines when necessary in order to accommodate un-avoidable stratification transients;
- ✓ performing external temperature measurements after steam generator replacement in order to validate the hypotheses made during the analyses and to judge the effect of the anti-stratification devices. In some cases, measurements had to be continued for several years in order to obtain enough data to establish a set of typical and representative transients that served as a basis for new analyses;
- ✓ non-periodic inspections in specific heavily loaded areas.

Czech Republic:

Periodic fatigue analyse including stratification based on thermocouple measurements on outer surfaces. Optimization of locations where thermocouple are installed after four years of measurement of stratification.

Finland:

Nozzle
Continues temperature measurements
Estimated crack growth in the geometrical crack
NDT

France:

large ISI program
French and USA field experience collection
Root cause analysis (leak and deep cracks in USA, practically nothing in France)
fatiguemeter development
temperature measurements on some typical plants, stratification and fluctuations
mock-ups developments (plexiglass and metallic), stratification and fluctuations
load evaluation through CFD Codes for stratification
stress-strain evaluation and corresponding damage evaluation without and with cracks
critical crack size and ISI performance
repair-replacement pre-analysis

Hungary:

Continues temperature measurements

ISI (4 yearly)

Japan:

-

Korea:

e) Periodic ISI (UT in ASME Code, Welds near elbow, once per 10 years inspection interval)

Slovakia:

Special system measuring continuously temperatures on outer surface of SG feed water line nozzle is installed. Periodic ISI (4 year period). Stress and fatigue periodic evaluation.

Spain:

-

Sweden:

United States of America:

There are several countermeasures that plants have taken to prevent cracking in the steam generator feedwater nozzles and connecting piping. These are the following:

Increased inspections – The feedwater nozzles are inspected more frequently than is required by the ASME Section XI inspection plans. Typically, at least one nozzle will be nondestructively examined at each outage, rather than once every ten years as was previously the case.

Change in counterbore geometry – The sharp corners at each end of the weld preparation counterbore at the nozzle to pipe weld have been removed. On the nozzle side, the counterbore inside diameter is built up to match the nozzle diameter; on the pipe side, a gradual taper such as 10° is applied from the pipe inside diameter to the root of the weld. This reduces the geometric stress concentrations in this region.

Installation of a protective sleeve – The thermal sleeve has been replaced by a new design that does not allow any flow between the sleeve and the pipe, and extends over the nozzle counterbore region to shield this area from thermal stratification cycling. This also prevents sleeve erosion and nozzle bore cracking. There are two designs that have been used – one extends to the counterbore area, and the other extends all the way upstream through the connecting elbow.

Fatigue monitoring – Thermocouples have been installed on the piping to measure thermal stratification and correlate it to plant evolutions. Based on these correlations, software has been developed and installed that calculates the fatigue usage due to thermal stratification using existing plant instrumentation.

Changes in auxiliary feedwater flow control – Feedwater flow is regulated manually rather than automatically, which results in fewer changes in flow rate and consequently reduces the number of thermal fatigue cycles.

Preheating feedwater – Some plants have installed additional feedwater heaters that preheat the water when the main turbine extraction steam is not available. During plant startup, or other periods of low feedwater flow, an auxiliary steam supply is available to the startup feedwater heater, which allows feedwater heating during periods when the turbine extraction steam is unavailable. Feedwater heating reduces the temperature gradients within the steam generator feedwater nozzle metal that occur due to the low flows during operation at 3 percent reactor power and below. During periods of low feedwater flow, feedwater temperature is maintained at above 250°F (121.1°C) to minimize thermal cycling of the metal and associated metal fatigue. Such conditions may occur at plant hot stand-by where full no-load steam conditions exist in the steam generators, the main turbine is not passing out heating steam, and feedwater is being supplied from a cold source such as the condenser or condensate storage tank.

Direct injection of auxiliary feedwater – In some cases, the auxiliary feedwater piping has been rerouted so that it can inject into the steam generators through a dedicated auxiliary feedwater nozzle. The advantage of this design is that the auxiliary feedwater nozzle is of much smaller diameter and consequently is much less susceptible to stratification at low flows. Also, it discharges at a higher elevation in the steam generator, which prevents steam generator liquid from exiting through the nozzle.

Installation of anti-stratification device – A helical shaped device has been inserted into the feedwater nozzles to prevent thermal stratification. Primarily installed during steam generator replacement, the helical device prevents the steam generator liquid from exiting the steam generator, and also facilitates mixing of the incoming colder fluid.

Installation of "goose neck" – An upward bend and loop was installed downstream of the feedwater nozzle, primarily in replacement steam generators. The goose neck provides a loop seal that prevents the steam generator fluid from travelling upstream, this preventing thermal stratification in the nozzle.

Oxygen control – Hydrazine is added to the condensate system to scavenge oxygen during operation. The reduction in oxygen content in the water reduces wall thinning in the feedwater nozzle thermal sleeve, which helps to prevent cold feedwater from travelling under the sleeve. This helps avoid thermal fatigue cracking in the nozzle bore and transition regions.

III. COUNTERMEASURES RELATED TO STRATIFICATION

III.2 PWR

B) PWR pressurizer surge line

Belgium:

The Belgian Safety Authorities imposed the actions requested by the US-NRC Bulletin No. 88-11. As such, visual inspection walk-downs and short term temperature measurement campaigns were performed in all Belgian NPPs, showing the generic character of stratification in the surge lines.

Long-term external wall temperature measurements have been performed in two of the seven units, that were considered to cover the other five. Based on the measurements, detailed fatigue analyses have been performed of the surge lines and their nozzles on the hot leg and pressuriser of all units. The analyses have been performed without modification of the plant heat-up method and without modification of the maximum temperature difference between pressuriser and hot leg.

Czech Republic:

Periodic fatigue analysis based on thermocouple measurements on outer piping surface

Finland:

Continues temperature and strain gauge measurements

NDT

France:

large ISI program

French and international field experience collection
fatiguemeter development
temperature measurements on some typical plants, stratification and fluctuations
mock-ups developments (plexiglass and metallic), stratification and fluctuations
load evaluation through CFD Codes for stratification
stress-strain evaluation and corresponding damage evaluation without and with cracks
critical crack size and ISI performance
repair-replacement pre-analysis

Hungary:

Continuous temperature monitoring of several locations of the surge line
Stress analysis
ISI(every 4 years)

Japan:

-

Korea:

c) Temperature Measurement

e) Enhanced ISI (Advanced UT, all Welds including nozzle area, once per 10 years inspection

interval)
i) LBB
j) Stress and fatigue analysis including thermal stratification

Slovakia:

Continues temperature measurements and periodic stress and fatigue evaluation.

ISI (4 years)

Spain:
-

Sweden:
-

United States of America:

The following countermeasures have been taken to prevent damage to the pressurizer surge line and nozzles from thermal stratification effects:

Fatigue monitoring – Thermocouples have been installed on the piping to measure thermal stratification and correlate it to plant evolutions. Based on these correlations, software has been developed and installed that calculates the fatigue usage due to thermal stratification using existing plant instrumentation.

Control of temperature differential – A restriction is put on the temperature difference between the pressurizer liquid and the reactor coolant loop hot leg temperature during heatup and cooldown. The limit is generally 200°F (93.3°C). This limits the magnitude of the thermal stratification loading.

Constant outsurge – Thermal stratification is minimized or eliminated in the surge line by establishing a continuous outflow of fluid from the pressurizer to the RCS hot leg during heatup and cooldown. This is done by turning on the pressurizer heaters and spraying in the pressurizer continuously, while running at least one of the reactor coolant pumps for the loops that supply main pressurizer spray.

Modified heatup method – Instead of heating up and cooling down the plant using the standard steam bubble method, the water solid heatup method is used. In the water solid method, the pressurizer remains filled while the reactor coolant system is pressurized to 350 psig (2413 kPa). This allows all reactor coolant pumps to be started before drawing a bubble. Then the pressurizer heaters are energized and main spray is actuated, and pressurizer liquid is circulated through the RCS, resulting in a uniform temperature between the hot leg and the pressurizer. Then the bubble is drawn and the heatup progresses, but the temperature difference between the pressurizer and the hot leg does not exceed 100°F (37.8°C). This greatly reduces the potential thermal stratification magnitude as compared with the standard heatup and cooldown method.

Avoiding insurges – inflow through the surge line is avoided by varying the charging / letdown flow match to compensate for shrink and swell effects resulting from heat input to the reactor coolant inventory. This reduces thermal stratification cycles in the surge line.

III. COUNTERMEASURES RELATED TO STRATIFICATION

III.2 PWR

C) Other piping

Belgium:

⇒ In Tihange 1, following the incident referred to in II.2.C, hardware modifications were performed on the Emergency Core Cooling System injection line in order to eliminate in-leakage definitely.

⇒ In Tihange 3, thermal stratification was discovered and monitored in the junction between the auxiliary feed water system and the normal feed water system. Due to the presence of stress raisers in the auxiliary, fatigue analyses yielded unacceptable fatigue usage factors for a short operation time. Therefore, siphons were installed that eliminate stratification in the auxiliary feed water system.

Czech Republic:

Thermocouple periodic measurements on outer surfaces (main circulation loops, charging systems, high pressure ECCS, low pressure ECCS, passive ECCS, cold water injection to pressurizer)

Finland:

In Loviisa 1 there are over 150 thermocouples (continuous recording). In one measurement point there are 2 or 7 thermocouples. After some years thermocouples are moved to another location (questions A+B+C).

The main focus was on valves, which are closed but not tight enough. The other configuration was short vertical pipe connected to horizontal pipe with no flow. In the review teams there were people from different disciplines (stress analyses, thermal hydraulic, systems and operation).

Temperature measurements in several places.

NDT of the most affected areas have been done.

France:

- ✓ assure absence of pollution (chloride/resine in stainless steel piping systems or oxygen in carbon steel piping systems)
- ✓ check leak tightness requirements of valves and periodic leak test procedures
- ✓ modify some piping layouts
- ✓ reduce number of welds (mainly longitudinal and un-flushed welds)in high fatigue areas
- ✓ increase surface finish requirements
- ✓ develop high compressive residual stress on the inner surface
- ✓ different system modification: lower ΔT, different flow rate ratio, decreased duration at high ΔT, define pressure drop complementary systems
- ✓ develop specific devices: like mixing devices or thermal sleeves
- ✓ increased inner surface radius in valves

<u>Hungary:</u>

There are plans to map other critical locations in the piping connected to the main circulation lines and to install additional temperature measurements as well.

<u>Japan</u>:

See II.2. C)

<u>Korea:</u>

Unisolable section piping connected to reactor coolant line such as ECCS line and RHR line up to the first isolation valve

e) Enhanced ISI (Advanced UT, all Welds including nozzle area, once per 10 years inspection interval)
i) LBB
j) Stress and fatigue analysis including thermal stratification and turbulent penetration

<u>Slovakia</u>:

Spray piping nozzle of pressurizer, RPV primary piping nozzles and RPV ECC piping nozzles.

Continues temperature measurements and periodic stress and fatigue evaluation.

ISI

<u>Spain</u>:

-

<u>Sweden</u>:

-

<u>United States of America</u>:

Thermal fatigue cracking events in unisolable piping attached to the reactor coolant loop have been addressed by a variety of preventative actions:

Research programs – the EPRI Materials Reliability Program (MRP) has sponsored research to improve understanding of the thermal fatigue phenomena, provide tools for evaluation, develop screening criteria to identify susceptible locations, and recommend countermeasures. These research programs are described in Question V.

Screening criteria – the EPRI MRP has developed the Interim Thermal Fatigue Management Guidelines. Final guidelines are in preparation. This guideline presents evaluation and inspection recommendations for detecting potential cracking that might be occurring in normally stagnant piping systems attached to main reactor coolant piping. The guideline also provides criteria to identify lines that should not be susceptible to cracking. The specific locations recommended for evaluation and/or inspection in this guideline are those where cracking and leakage have been identified in domestic and similar foreign PWRs, and are not currently part of other augmented inspection programs as a result of specific cracking issues (e.g. B&W plant high pressure injection nozzle/thermal sleeves). The guideline provides the specific recommendations for near-term assessment and possible volumetric examination. The piping systems and locations considered are those like drain lines and safety injection lines subject to valve inleakage. It describes how assessments may be performed to determine if volumetric examination and/or monitoring should be considered. Guidelines for conducting effective examinations and monitoring are also provided.

Training – the EPRI MRP has also sponsored a training program. Member utilities have received training in the thermal fatigue management guidelines. This training included the application of the guidelines to the plant specific piping system geometry to identify potential locations for thermal fatigue damage. The plant personnel obtained a better understanding of the thermal fatigue phenomena and the actions that can be taken to prevent damage.

Fatigue monitoring – the locations with the greatest potential for thermal fatigue damage have been identified, and software has been developed (FatiguePro) and installed that tracks the actual fatigue usage at these locations. Fatigue calculations are done using transfer functions that take plant instrument data and calculate stress and strain, based on logic that identifies significant plant evolutions.

Thermocouple installation – a number of plants have measured thermal stratification levels by installing thermocouples on the pipe. The temperature profiles obtained from this instrumentation have been analyzed to better understand the phenomena taking place.

Leakage monitoring – on the high pressure safety injection lines, valve leakage is monitored to minimize the potential for thermal stratification. Maximum allowable leakage levels are determined by calculation and are tracked. In addition, some plants have installed pressure monitoring systems that determine if the higher pressure from the charging system is present downstream of the isolation valve, which would indicate a valve leak. Pressure is also monitored to prevent backleakage through the check valves in the low pressure safety injection systems.

Added valves – some plants have added a second isolation valve to the safety injection line to prevent leakage into the unisolable portion of the piping.

Improved valve maintenance – improvements in valve packing design, and more frequent preventative maintenance actions are taken to prevent leakage in reactor coolant system boundary valves. In some cases, the valve was replaced with a different type that is less prone to leakage.

On makeup / HPI lines (Babcock and Wilcox plants), a double wall thermal sleeve design is being developed for installation in the RCS nozzle. Flow velocity will be increased to prevent turbulence penetration from entering the upstream region. Better thermal sleeve fabrication and inspection methods have been implemented.

Isolation valves have been relocated to provide for a vertical piping section downstream of the valve, which assists in mixing any possible leakage, thus avoiding stratification.

In the pressurizer spray line, in order to prevent draindown of the upper spray piping and the associated thermal stratification cycling, bypass spray flow is increased. In a cooldown, auxiliary spray is started before shutting down the reactor coolant pumps to prevent the upper piping from draining.

Drain lines attached to the reactor coolant loop are insulated to prevent long horizontal runs from losing sufficient heat set up a natural convection thermal stratification gradient. This gradient can be subject to thermal cycling due to turbulence penetration effects.

Thought is given to the pipe support arrangement so as to prevent an adverse slope in the pipe. Sufficient flexibility is also provided in the piping layout to prevent high stresses from global stratification moments.

The residual heat removal system is preheated before entering into service during a cooldown. Flow initially bypasses the RHR heat exchanger in order that the RCS nozzle is not thermally shocked.

III. COUNTERMEASURES RELATED TO STRATIFICATION

III.4 Other plants than BWR or PWR

Belgium:

No plants other than BWR or PWR in Belgium

Czech Republic:

Not applicable

Finland:

Not applicable

France:

- ✓ change operation procedure
- ✓ repair-replacement techniques
- ✓ ISI

Hungary:

Not applicable

Japan:

-

Korea:

CANDU(4 operating CANDUs in Korea)

A) Feedwater nozzle and adjacent piping

e) Periodic ISI (UT in ASME Code, Welds near elbow, once per 10 years inspection interval)

B) Other systems: No action for thermal stratification

Slovakia:

Not applicable

Spain:

-

Sweden:

Not applicable

United States of America:

Not applicable

III. COUNTERMEASURES RELATED TO STRATIFICATION

III.5 If not addressed in items III 2 to III 4, please describe countermeasures, if any, on:
The risk of cold water in-leakage through a "closed" valve (as a common phenomenon)

Belgium:
-

Czech Republic:
-

Finland:

See above

France:
-

Hungary:
-

Japan:

See II. 1. and 2.

Korea:
-

Slovakia:

See above

Spain:
-

Sweden:

United States of America:

Addressed in III.3 above. BWR plants do not have the potential for cold water inleakage toward the reactor coolant system. There is no alternate path for CRD return water or reactor water cleanup water to be injected toward hot primary coolant boundary components.

III. COUNTERMEASURES RELATED TO STRATIFICATION

III.5 If not addressed in items III 2 to III 4, please describe countermeasures, if any, on:

Countermeasures for other stratification phenomena

Belgium:
-

Czech Republic:
-

Finland:

See above

France:
-

Hungary:
-

Japan:

See II. 1. and 2.

Korea:

Slovakia:

See above

Spain:
-

Sweden:

United States of America:

III. COUNTERMEASURES RELATED TO STRATIFICATION

III.6 Other comments

Belgium:

Feedwater baffle plate system installed in Doel Units1/2 to fix a water hammer problem was also found to be an effective countermeasure for thermal stratification.

Ref.: NEA/CSNI/R(98)8 “Specialists Meeting on Experience with Thermal Fatigue in LWR Piping Caused by Mixing and Stratification, pp 377-395

Czech Republic:

Periodic temperature measurements on outer surfaces and periodic fatigue analyses for selected locations. In case of cold injection or mixing of medium with significant ΔT, non destructive testing has to be performed.

Finland:

Pressurizer, bottom nozzle: Protection (thermal insulation) against rapid temperature changes has been improved. Changes in the water level and temperature difference between the pressurizer and the surge line causes fatigue. The crack growth in the geometrical crack is estimated.

Steam generator: Auxiliary feedwater can be pumped to the bottom of the horizontal steam generator. The water has been warmed up due to stratification and thermal shock in thick wall vessel.

France:

The more sensitive location are:

- ✓ a small cold water flow in a small hot water flow, with deep gradient through the height of the pipe
- ✓ a stratified situation frequently push by a large flow rate of cold water

The countermeasures are different in these 2 cases.

Hungary:

In the 2001 there were fatigue cracks in 4 Steam Generator nozzles connecting the auxiliary feedwater. The possible root cause is under investigation. (Mixing?)

Japan:

Regarding “Research regarding Assessment of High-cycle Thermal Fatigue due to Stratification of Stagnant Water Part” and ‘Research into Cavity-flow type Stratification Phenomena (PWR)” in V.2: On-going and Planned Research Projects on Thermal Fatigue, the concept of thermal stratification related damage preventive measure shall be regulated as below:

The structure shall be designed wherein the stratification boundary is not situated in the elbow. The main piping flow velocity, vertical branch piping length and branch piping diameter shall be designed wherein the stratification boundary stays sufficiently within the vertical piping or sufficiently goes inside the horizontal piping. Tests are presently being conducted to confirm the relationship between the main piping flow velocity, the length and diameter of the vertical branch piping, and the depth of cavity flow intrusion.

Korea:

Slovakia:

Spain:

-

Sweden:

Approach: According to a) and e)

Operation: According to b), c) and f).
Mitigation: According to g) and h).
Note: No difference made between BWR / PWR or type of components.

United States of America:

A number of plants have done detailed fatigue analyses to demonstrate that the effects of thermal stratification are acceptable, based on monitored leakage levels or thermocouple measurements.

IV COUNTERMEASURES RELATED TO MIXING
Same list as in paragraph III

IV.1 General approach and policy

Belgium:

As for stratification, operational experience related to mixing from other countries is analysed. If incidents appear in foreign plants, their applicability to the Belgian plants is evaluated. As such, the applicability of the Civaux incident on the Belgian RHR systems has been evaluated.

Czech Republic:

No

Finland:

After the 1979 leak in Olkiluoto (Case 1 in II.1 A) the main systems in Loviisa and Olkiluoto were reviewed for thermal mixing. The mapping was based on process and instrumentation (P&I) drawings and operating instructions.

The affected areas are inspected every three years.

France:

- ✓ try to reduce the load: temperature difference or duration at large ΔT
- ✓ try to justify large margins in term of crack growth rate versus critical crack sizes
- ✓ define specific ISI program and objectives for performance demonstration
- ✓ prepare replacement, and define specification to assure the life of the new component

Hungary:

-

Japan:

See III.1.

Korea:

1) Consideration of turbulent penetration in piping design

2) Enhanced ISI for the piping susceptible to thermal fatigue

Slovakia:

-

Spain:

Same as for III.1

Sweden:

In such cases mixing devices should be installed.

United States of America:

There have not been any incidents of damage occurring due to thermal mixing phenomena in U.S. plants. Therefore, no preventative actions have been implemented beyond basic good design.

The findings from the Civaux event have been considered in development of risk-informed inspection programs to assure that the regions downstream of the RHR heat exchangers are considered for detailed inspection. Although not specifically a countermeasure, it has been found that U.S. plants do not operate for any significant time with hot water bypass around the RHR heat exchangers.

IV COUNTERMEASURES RELATED TO MIXING

IV.2 BWR feedwater nozzle and other systems

Belgium:

-

Czech Republic:

Not applicable

Finland:

See the general approach.

Mixing devices in the feedwater lines.

France:

-

Hungary:

No BWR

Japan:

See II. 1.

Korea:

N/A

Slovakia:

Not applicable

Spain:

Following the recommendations established in NUREG-0619: Replacement of feedwater spargers.

Sweden:

United States of America:

IV COUNTERMEASURES RELATED TO MIXING

IV.3 PWR feedwater nozzle and other systems

Belgium:

-

Czech Republic:

After-Outer surface temperature measurement, FE flow analysis and fatigue evaluation on mixing tee (Temelin NPP), new control valve will be designed on surge lines by-pass piping.

Finland:

See the general approach

France:

– detailed analysis of all the existing data of each locations: geometry, material, load history, fabrication, ISI, transient monitoring…
– load evaluation through mock-ups, computation and direct measurements on the plant
– stress evaluation and damage analysis
– usage factor / crack growth rate / critical crack size
– ISI program and performance
– replacement pre-analysis

Hungary:

Feedwater nozzle and adjacent piping: No action for mixing

Pressurizer surge line: No action for mixing

Japan:

See II. 2.

Korea:

A) Feedwater nozzle and adjacent piping: No action for mixing

B) PWR pressurizer surge line: No action for mixing

C) Unisolable section piping connected to reactor coolant line such as ECCS line and RHR line up to the first isolation valve

e) Enhanced ISI (Advanced UT, all Welds including nozzle area, once per 10 years inspection interval)

i) LBB

j) Stress and fatigue analysis including thermal stratification and turbulent penetration

D) Other large components: No Action for mixing

Slovakia:

See the general approach

Spain:

None

Sweden:

-

United States of America:

IV COUNTERMEASURES RELATED TO MIXING

IV.4 -Other plants than BWR or PWR

<u>Belgium</u>:

Not applicable

<u>Czech Republic</u>:

Not applicable

<u>Finland</u>:

Not applicable

<u>France</u>:

-

<u>Hungary</u>:

Not applicable

<u>Japan</u>:

-

<u>Korea</u>:

A) Feedwater nozzle and adjacent piping: No action for mixing

B) Other systems: No action for mixing

<u>Slovakia</u>:

Not applicable

<u>Spain</u>:

None

<u>Sweden</u>:

Not applicable

<u>United States of America</u>:

IV COUNTERMEASURES RELATED TO MIXING

IV.5 Tees (common geometry)

- **additional questions: Identification of locations with potential risk:**

✓ **What are the temperature difference limits for austenitic stainless steel and carbon steel (including low alloy steel) in continuous mixing during power operation?**

✓ **What are the temperature difference limits for austenitic stainless steel and carbon steel (including low alloy steel) in limited operating time (like heat-up and cool-down)?**

✓ **What are the restrictions for the geometrical configuration in piping, like horizontal and vertical orientation, nearby elbows etc.**

Belgium:

-

Czech Republic:

No limits, no restrictions on geometry

Finland:

Temperature difference limits in continuous mixing: Screening limit for austenic steel is 30 C and for carbon steel 50 C. We consider these as conservative values. Higher values are allowed, when inspected (NDT) periodically.

Temperature difference limits in limited operating time: Not specified for limited operation, case by case

Restrictions for the geometrical configuration in piping: There are no formal restrictions and seriousness is estimated case by case. It would be useful to have recommendations concerning favourable configurations and warnings on vulnerable configurations.

France:

Temperature difference limits in continuous mixing: To-day RCCM 2000 proposes: mixing fluide temperature difference 80°C for SS and 50°C for CS (it's base on existing RCCM fatigue analysis procedure RCCM fatigue curves)

Temperature difference limits in limited operating time: 110°C between pressuriser and hot leg, 140°C between RHRS or CVCS and RCL during intermediate hot stand-by (180°C)

Restrictions for the geometrical configuration in piping: Some are under preparation for future plants or operating plant modifications.

Hungary:

These limits are not established

Japan:

See II.1.and 2.

Korea:

-

Slovakia:

These limits are not established

Spain:

These limits are not established, except as described in IV.6 .

Same answer as before.
Standard configurations are used, that's to say: 90º or 45º elbows.

Sweden:

United States of America:

IV COUNTERMEASURES RELATED TO MIXING

IV.6 Other comments

Belgium:

A screening of the branches connected to the reactor coolant system with respect to thermal fatigue susceptibility, including the thermal mixing issue, is going on in the framework of the 10-yearly safety reassessments of the Belgian NPPs.

Czech Republic:

-

Finland:

-

France:

-

Hungary:

-

Japan:

Regarding ‘Research regarding Assessment of High-cycle Fatigue at Hot and Cold Water Junction (BWR & PWR) in Table V-2: On-going and Planned Research Projects on Thermal Fatigue, the concept of mixing related damage preventive measure shall be regulated as below:

Assessment shall be made by step-by-step screening. The necessity for such assessment shall be screened in the first step by the temperature difference. Simple assessment shall be conducted when the temperature difference is small. This assessment shall be gradually, forwarded into more detailed assessment according to the level of temperature difference. Tests are presently being conducted to confirm the effect of reducing the temperature differences by mixing or by disturbance, and to confirm the coefficient of thermal conductivity under conditions wherein the flow velocity conditions of an actual plant are enveloped.

Korea:

-

Slovakia:

-

Spain:

Spanish utilities have restrictions in their operational procedures related to the speed of valve opening in certain lines, and with the speed they can heat-up and cold-down, especially for the presurizer surge line and for the presurizer sprinklers, which are the locations with potential risk.

Sweden:

See under III.6

United States of America:

V RESEARCH ON THERMAL FATIGUE

V.1 Completed research projects

Please indicate references

Belgium:

Development of a set of computer codes (THERMAXS) to perform thermo-elastic transient calculations.
Ref:

Inverse and Direct Transfer Functions for the Fatigue Follow-up of Piping Systems Submitted to Stratification, Proceedings of the 12th International Conference on Structural Mechanics in Reactor Technology, 1993, Volume D, pp 339-344

Justifying fatigue Induced by Severe Thermal Stratification Transients in Feedwater Systems, ASME Pressure Vessel and Piping Conference, 1997, PVP Vol.350, pp 441-445

Therma-mechanical Analysis of Components Subjected to Thermal Stratification Transients, 7th International Conference on Nuclear Engineering, Tokyo, 1998, Paper ICONE-7444

Czech Republic:

No research projects, only periodic evaluations

Finland:

-

France:

Analytical tests on specimens:
yes, thermal fatigue on pipe thickness variation
Test on components (but welds, socket welds, piping thickness variation, valves, stratified feedwater lines):
yes, on SS but welds, SS SW, SS and CS (with typical water chemistry) valves for low cycle fatigue
yes, for stratified CS FW line
SS: stainles steel, CS: carbon steel

Hungary:

None

Japan:

Project title: Research on Improvement of Recirculation Pump Reliability

Period: 1981 - 1985

Outline of the technical development: The purge water is supplied to the reactor recirculation pump shaft seal. By simulating the temperature fluctuation occurring on the pump shaft external surface and the casing internal surface when the purge water partially flows into the casing and mixes with hot water, the structural integrity was examined and confirmed. The structural improvement plan was also studied and confirmed. Reference: Shhiina, Nakamura, Mizushima, Endo, Takehara, Narabayashi, and Kato, “Heat Transfer Characteristics of Fluid Flow in an Annulus with an Inner Rotating Cylinder having a Labyrinth Structure”, Heat Transfer Japanese Research Scripta Publishing Co.,USA,1997

Korea:

1) Pressurizer Surge Line Thermal Stratification Program
Pressurizer Surge Line of Korean Standard Reactor
Performed by KAERI during1990~1994
Stress analysis and measurement of temperature distribution in piping
2) Pressurizer Surge Line Thermal Striping Program
Pressurizer surge line of Korean Standard Reactor
Performed by KINS during 1991~1994
Stress analysis and fatigue analysis

Slovakia:

None

Spain:

None

Sweden:

None (supported by SKI)

United States of America:

Research on thermal fatigue has been undertaken primarily by the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP). Additional work has been sponsored by the U.S. Nuclear Regulatory Commission. The NSSS vendors (Westinghouse, Combustion Engineering, and Babcock and Wilcox (Framatome)) owners groups have also carried out research programs.

The following research reports on the topic of thermal fatigue have been published by EPRI over the past three years:

NDE Technology for Detection of Thermal Fatigue Damage in Piping, MRP-23 (1000152), 10/10/2000

Mitigation of Thermal Fatigue in Unisolable Piping Connected to PWR Reactor Coolant Systems, MRP-29 (1001017), 12/4/2000

Operating Experience Regarding Thermal Fatigue of Piping Connected to PWR Reactor Coolant Systems, MRP-25 (1001006), 12/4/2000

Interim Thermal Fatigue Management Guidelines, MRP-24 (1000701), 1/16/2001

Thermal Fatigue Monitoring Guidelines, MRP-32 (1001016), 4/24/2001

Computer Based NDE Training for Thermal Fatigue Cracking, MRP-36 (1001317), 6/18/2001

Proceedings of 2000 International Conference on Fatigue of Reactor Components, MRP-46 (1006070), 6/25/2001

Identifying Thermal Cycling Mechanisms in Two Piping Configurations,MRP-54 (1003081), 11/20/2001

EDF Thermal Fatigue Monitoring Experience on Reactor Coolant System Auxiliary Lines, MRP-69 (1003082), 4/8/2002

Interim Report on Thermal Cycling Model Development for Representative Un-Isolable Piping Configurations, MRP-81 (1003527), 11/27/2002

Lessons Learned From PWR Thermal Fatigue Management Training, MRP-83 (1003666), 12/5/2002

Proceedings of 2002 International Conference on Fatigue of Reactor Components, MRP-84 (1003536), 3/5/2003

Operating Experience Regarding Thermal Fatigue of Piping Connected to PWR Reactor Coolant Systems, Rev. 1, MRP-85 (1007761), 4/30/2003

In addition, EPRI published the report “Thermal Stratification, Cycling, and Striping (TASCS)” in 1994, which gives methodology for calculating thermal stratification loadings and describes the results of testing programs.

The USNRC has published some research reports related to thermal stratification. These include:

“Review of Industry Efforts to Manage Pressurized Water Reactor Feedwater Nozzle, Piping, and Feedring Cracking and Wall Thinning”, NUREG/CR-6456, 1997

“Assessment of Pressurized Water Reactor Primary System Leaks”, NUREG/CR-6582, 1998

The NSSS vendor owners groups have published a number of reports but these are proprietary and not available to the general public.

V RESEARCH ON THERMAL FATIGUE

V.2 On-going and planned research projects

Belgium:

-

Czech Republic:

Thermal fatigue of materials in chemical and Nuclear industry FT-ta/011 – MPO 2004-2006

Finland:

Tee-junction: Tests, CFD calculations and related fatigue calculation (VTT).

France:

fatigue curves for stainless and carbon steels, high cycle fatigue curves, with different material (304 / 316 SS), different temperature, different surface finish, different mean stress and different environmental situation (air and PWR water)

analytical tests on specimens (bars/plates, cylinders)

test on components: valves, thermal fluctuations on tees

Hungary:

Renewed fatigue calculations for Class 1 components in connection of life extention

Japan:

Project title (1): Research regarding Assessment of High-cycle Fatigue at Hot and Cold Water Junction (BWR & PWR)

Period: 2000 - 2001

Outline of technical development:

- Establishment of general methods for thermal fatigue assessment to verify the integrity of parts where hot and cold water meet.

- By specifying the locations where hot and cold water meet in the LWR environment and where temperature fluctuation is concerned based on the configurations, flow conditions and temperature difference, a method to assess the impact on the structural integrity shall be established. The integrity of parts where temperature fluctuation is concerned shall also be assessed.

Project title (2): Research regarding assessment of High-cycle Thermal Fatigue due to Stratification of Stagnant Water part

Period: 2000 - 2001

Outline of technical development:

Assessment of high-cycle thermal fatigue caused by stratification at the stagnant water part in BWR.

Project title (3): Research on Cavity-flow type Stratification Phenomena (PWR)

Period: 2000 - 2001

Outline of technical development:

Where branch tubes are connected to the main piping with flow inside, it is known that cavity flow (secondary flow) penetrates into branch tube due to the flow in the main tube. A bend (elbow) from right angle to horizontal angle near the boundary where the cavity flow penetrates generates stratification due to the hot water in the main tube and cold water in the branch tube. This can initiate fatigue crack after the cyclic load is repeated due to fluctuations in the penetration depth of the cavity flow and the stratification state. Therefore, an assessment method for the penetration depth of cavity flow shall be studied considering the difference in the flow conditions in the main piping and other factors.

Korea:

1) Pressurizer Surge Line Thermal Stratification Program

Pressurizer Surge Line of WH type Reactors and Framatome Type Reactor in Korea
Performed by KEPCO and KEPRI from 1997
Stress Analysis and Fatigue Analysis
Measurement of thermal distribution in piping
Enhanced ISI Program Development

2) Thermal Stratification and Turbulence Penetration Program
Unisolable Section Connected to RCS such as ECCS and RHR Lines
Performed by KEPCO and KEPRI from 1999
Flow Analysis
Stress Analysis and Fatigue Analysis
Enhanced ISI Program Development

Slovakia:

VVER 440 NPP main component ageing management research project is running till 2002 (up to 2005) which includes activities connected with fatigue damage evaluation.

Spain:

Some Spanish utilities are involved in the R&D program called THERFAT (Thermal Fatigue Evaluation of Piping Systems Tee-Connections), which is an initiative of some European institutions, in the frame of the 5th European R&D Program. The objective is to determine thermal fatigue stresses in piping in order to improve risk management.

Sweden:

None (supported by SKI)

United States of America:

The EPRI MRP is currently developing the final version of the Thermal Fatigue Management Guidelines, as well as a computer based training program for its application.

Work is ongoing to develop a better phenomenological understanding of thermal cycling, including turbulence penetration effects. The results will be used in developing the final version of a thermal fatigue screening model.

V RESEARCH ON THERMAL FATIGUE

V.3 Research needs

In among others: (please specify what research)

- ✓ **Thermal mixing**
- ✓ **Stratification**
- ✓ **CDF calculations, parametric studies**
- ✓ **Fatigue curves, low cycle (stratification), high cycle (mixing)**
- ✓ **Other**

Belgium:

⇒ Thermal mixing: screening criteria for different configurations, diameters, flow rates, temperature differences

⇒ Stratification: no particular need

⇒ CFD calculations: validation of CFD analyses with measured transients

⇒ Fatigue curves: environmental effects, low and high cycle fatigue.

Czech Republic:

Stratification, precise relations between outer measured temperature and medium temperature

Finland:

In among others:

- Thermal mixing: Screening values for different configurations, diameters, flow rates etc.
- Stratification: The collection of relevant phenomena with some screening values if possible. After the mapping the temperature measurement can be done in vulnerable areas.
- CDF calculations, parametric studies: Can support previous points.
- Fatigue curves, low cycle (stratification), high cycle (mixing): Important. For high cycle the effect of the surface (as fabricated, as welded, polished) and residual stresses.

France:

Thermal mixing: Typical temperature load spectrum for a given geometry and a given flow rate ratio

Stratification: no urgent specific need

CDF calculations, parametric studies: for stratified situations and mixing situations

Fatigue curves, low cycle (stratification), high cycle (mixing):

with corrected factors for mean stress, surface finish, temperature, different material, different welds and base metals

and cyclic stress-strain curves for different materials (SS, CS) and different temperature

Other

damage analysis guidelines for random bi-axial loads : equivalence rules to 1D sinusoidal load with mean stress

crack growth rate and threshold

Hungary:

Thermal mixing: Screening values for different configurations, diameters, flow rates etc. CDF calculations, parametric studies for supporting monitoring.

Japan:

JAPEIC has conducted the following Project under the auspices of the Ministry of Economy, Trade and Industry (METI) since 1995.

Project: Environmental Fatigue Tests of Nuclear Power Plants Materials for Reliability Verifications

Outline: To collect comprehensive S-N fatigue data for evaluating the effect of strain rate, temperature, dissolved oxygen level, and so on. under the mechanical - cyclic load conditions.

To characterise and evaluate factors to be used in assessing and formulating the environmental effects.

To develop and verify an environmental fatigues evaluation method.

Duration: 1995 JFY –2003 JFY

Korea:

1) Thermal mixing

2) Stratification

Slovakia:

Thermal mixing: Screening values for different configurations, diameters, flow rates etc.

CDF calculations, parametric studies for supporting monitoring.

Spain:

One BWR licensee is considering to analyse thermal stratification phenomenon in feedwater nozzles, by measuring temperatures on them and evaluating their impact on the structural integrity of the component.

Sweden:

Primary on thermal mixing and stratification.

United States of America:

See V.2.

VI. HIGH CYCLE FATIGUE CURVE

VI.1 List reference (i.e. Code) of curves currently used for design and for expertise, if relevant.

Belgium:

Fatigue curves of the ASME B and PV Code, Section III

Czech Republic:

Fatigue curves and equations of Soviet standard PNAE-G and Czech standard N.T.D. A.S.I.

Finland:

ASME XI and in one case interim fatigue curves in the NUREG report.

France:

(1) ASME III curve

(2) RCC-M curve

(3) RCC-MR curve

(4) FRAMATOME curve derived for CIVAUX 1

(5) new EDF curves under development (for 2003)

Hungary:

High cycle fatigue curves of the Soviet Code PNAE can be used if needed.

Japan:

The curves in VI-1 (Separate Figure 2: Design Fatigue Curve (Austenitic stainless steel and high-nickel alloy)) in the separate sheet has been added to the high-cycle side of design fatigue curve since June 1994. These curves are equivalent to those in the ASME Boiler and Pressure Vessel Code Sec. III.

Korea:

ASME Code Sec. III for the fatigue design

ASME Code Sec. XI for ISI flaw fatigue analysis

Slovakia:

High cycle fatigue curves of the Soviet Code PNAE can be used if needed.

Spain:

No high cycle fatigue curves for thermal stresses are used, neither for design nor for expertise. The only fatigue curves used are those of the ASME Code. When the incident described in II.1 appeared, corrective actions were adopted, so there was no need to use any fatigue curve to justify the presence of cracks in the spargers for continuing operation.

Sweden:

SKIFS 1994:1 (This code refers to ASME XI curves)

United States of America:

Section III of the ASME Boiler and Pressure Vessel Code provides fatigue design curves. These design curves have recently been extended to include the high cycle region. Still at issue is the treatment of the mean stress effect in the high cycle regime. For carbon and low-alloy steels, in the absence of an environmental effect, the analysis of the existing data, including material variability and data scatter, specimen size and geometry, surface finish, and cycle counting (i.e., the application of Miner's Rule), it appears that the Code design rules, with the adjustment for mean stress effects using the Goodman equation, are sufficiently conservative. Response VI.3 deals with fatigue in the presence of environmental effects.

The application of design curves in the high-cycle regime for stainless steels is a subject of current discussion in the US. Given that the difference between the Code design curve, and the best fit of engineering data in the regime N < 106 cycles is about 1.5 on stress, and 10-16 on cycles (vs. 2 and 20), it is likely that this erosion of code margins extends into the high-cycle regime. High-cycle regime, fatigue life data for the in the presence of a simulated coolant environment does not yet exist in a quantity that provides meaningful statistics.

VI. HIGH CYCLE FATIGUE CURVE

VI.2 Were those curves used for assessment of failure? With what success?

Belgium:

No

Czech Republic:

No

Finland:

No practical experience

France:

- ✓ all of them are consistent up to 104 cycles; the Ke values for plasticity effects are different
- ✓ except (4), all these curve are based on base metals data, transferability factors of 2 and 20, stress indices for welds
- ✓ the endurance limits (at 106 cycles) are slightly different
- ✓ the fatigue curves from 105 to 109 cycles are different

Hungary:

No practical experience

Japan:

As fatigue assessment is conducted to calculate the strength when applying for a construction plan permit, it is used for the assessment of failure.

Korea:

Yes, the fatigue curves in ASME Sec. III & XI were used for assessment of failure.

Slovakia:

No practical experience

Spain:

See VI.1 above

Sweden:

Comparing the calculated ΔKI-value with the threshold value.

United States of America:

To our knowledge, theses curves have not been used for assessment of failure.

VI. HIGH CYCLE FATIGUE CURVE

VI.3 Do you feel the need for new or more appropriate curves for expertise and/or design? (For welded, fabricated, polished materials, environment, frequency etc.)

Belgium:

Yes

Czech Republic:

Yes

Finland:

It would be valuable to make realistic assessments on fatigue life. Improvements in the stress concentration, surface smoothes, residual stresses etc. increase the fatigue life, but quantifying improvement is difficult.

France:

Yes: a set foe expertise with all the factors; and a simplified and more conservative one to design future components

Hungary:

It would be valuable to make realistic assessments on fatigue life to evaluate the conservativity or nonconservativity in current fatigue calculations

Japan:

We feel this not to be necessary.

Korea:

Yes, we feel the need for new fatigue curve considering environmental effect.

Slovakia:

Perhaps environmental effect on fatigue curves of some special materials should be investigated

Spain:

Sweden:

Fatigue curves in water environment for austenitic materials.

United States of America:

The short answer is “yes”. Recently acquired data suggests that, in general, code margins are reasonable, but do not contain excess conservatism that can be assumed to account for the effects of light water reactor environments. In other words, the margins of 2 and 20 are sufficient to account for material variability and data scatter, specimen size and geometry, surface finish, and cycle counting, with little margin, if any, left over for environmental effects. It seems likely that this trend will persevere into the high cycle regime.

Historically, it is generally believed that even if the design curves are not conservative, after taking into account all the factors listed in VI.3, there were enough conservatisms in the other contributors to the fatigue analysis (i.e., stress calculations, cycle counting, as examples), that the overall calculation would have a conservative result. However, with more sophisticated, and presumably more accurate, computational methodologies coming into routine use, this other conservatisms are disappearing, and the possible non-conservatism of the design curves is emerging as the possible Achilles heel of the fatigue analysis. This is an issue of vigorous discussion between the NRC and the US licensees.

VI. HIGH CYCLE FATIGUE CURVE

VI.4 If yes, what step or action did/do you undertake?

Belgium:

No action foreseen

Czech Republic:

No current action

Finland:

The work requires international co-operation.

France:

See paragraph V.2

Hungary:

Japan:

There are no specific steps or actions.

Korea:

Our actions for the fatigue design are as follows;

Step 1: Research on the environmental effect on S-N fatigue curve

Step 2: Development of Regulatory Guide on the fatigue design especially for the periodic safety review (PSR) and life extension

Slovakia:

No action foreseen

Spain:

Sweden:

No current plans

United States of America:

Currently, we have no plans to fund research in this area.