### AMP 161 High cycle FATIGUE MONITORING (VERSION 2020)

### Programme Description

This programme manages high-cycle fatigue that may be induced by fluid flow conditions having the potential for generating high-frequency cyclic thermal stresses such as:

* Mixing of hot and cold water in Tees;
* Thermal cycling due to the interaction of hot swirl penetrations into normally stagnant reactor coolant system (RCS) branch lines filled with cold water;
* Thermal cycling due to the interaction between the hot swirl penetration from the RCS and the ‘in-leakage’ of the cold water from a leaking, normally closed valve.

High cycle fatigue characteristics are:

* High frequency such that fluctuations of temperature are difficult to measure at the outside of the pipe;
* Generally, no usage factor determined, managed differently from low cycle fatigue. However, TLAA 119 is applied by some countries;
* Generally, deficiency in design or operation.

Thermal stratification transients such as those observed in PWR pressurizer surge lines, pressurizer spray lines and steam generator feed water nozzles typically induce low-cycle fatigue and are managed by AMP 101.

Based on OE, guidelines for management of high-cycle fatigue were developed, for instance in the US [1-5], Switzerland [6], France [7-8], and Japan [9].

Some countries such as Canada, consider thermal fatigue when selecting inspection locations, which has been included in national standards [10-11].

### Evaluation and Technical Basis

1. ***Scope of the ageing management programme based on understanding ageing:***

The scope of the high cycle fatigue monitoring programme includes piping systems where the following conditions on the temperature difference between two flows (ΔT) are present:

* Mixing zones with high ΔT;
* Swirl penetration into branch lines with high ΔT including leakage across valves.

1. ***Preventive actions to minimize and to control ageing degradation:***

Effective preventive and mitigative actions for high cycle fatigue damage consist of careful identification of the possible locations where this type of damage could occur. When possible locations are identified, the reduction of the temperature difference ΔT and/or the repair of leaking valves are appropriate preventive actions.

For new plants in design phase, the geometry and configuration of the pipelines could be modified based on OE to avoid this high cycle fatigue damage to occur.

1. ***Detection of ageing effects:***

High cycle thermal fatigue may lead to the initiation of cracks on the internal surface. Such cracks can be detected by appropriate in-service inspection of the suspect location.

This programme includes identification of susceptible locations, as indicated by operating conditions (ΔT) or special considerations (leaking valves in stagnant branch or mixing tee). Screening of locations and conditions can for example be found in [1].

1. ***Monitoring and trending of ageing effects:***

For high cycle fatigue susceptible locations, components to be inspected are on the basis of operational experience, risk analysis or engineering judgment. For components that have been inspected at least once, predictability of the extent of degradation is obtained through trending which means that the next inspection date is determined based on the observed fatigue cracking damage.

In all cases, inspection results are evaluated to determine if additional inspections are needed to assure that the extent of damage is adequately determined, assure that intended function will not be lost, and identify corrective actions.

1. ***Mitigating ageing effects:***

Where practical, effective mitigation methods and technology for high cycle thermal fatigue include:

1. Controlling ΔT;
2. Repair or replacement of valves which are leaking; and
3. Possible changes in design of the piping lay-out to control the mixing of hot and cold fluids. This can include modifying pipe / component configuration to reduce flow velocities.
4. ***Acceptance criteria:***

When a susceptible location has been identified, the screening is performed based on a maximum allowable temperature difference ΔT, see [1,6] for example. Another approach is to combine the allowable temperature difference in relation to the time of operation where the temperature difference occurs [7].

1. ***Corrective actions:***

Repair, replacement and/or redesign of pipework affected by high cycle thermal fatigue (see attribute 5).

1. ***Operating experience feedback and feedback of research and development results:***

This AMP addresses the industry-wide generic experience. Relevant plant-specific operating experience is considered in the development of the plant AMP to ensure the AMP is adequate for the plant. The plant implements a feedback process to periodically evaluate plant and industry-wide operating experience and research and development (R&D) results, and, as necessary, either modifies the plant AMP or takes additional actions (e.g. develop a new plant-specific AMP) to ensure the continued effectiveness of the ageing management.

In 1998, cracking appeared in mixing zones in the RHR system of Civaux-1 [12].

Fatigue susceptibility assessments related to mixing Tees and RCS branch lines in US PWR/BWR are addressed in [13-14] and [15], respectively.

A number of incidents with through-wall cracks in non-isolable branches connected to the reactor coolant system, such as the leak in the ECCS of Farley-2 [16] and Tihange-1 [17] NPP, led to the publication of NRC Bulletin 88-08 and its supplements [18-21]. Other similar experiences are Dampierre-2 [22], Dampierre-1 [23], Tsuruga-2, Mihama-2 [9] and Genkai-2 [24].

Reference [25] covers OE up to 2014 in PWR/BWR/CANDU plants related to mixing tee and branch line events.

An overview of experience, regulations, countermeasures regarding thermal fatigue in OECD member states is given in [26].

Recent US OE related to RCS branch lines and mixing tees is included in [27]. One conclusion is that UT procedures intended to detect defects due to thermal fatigue need to be further improved [28].

In the US, EPRI is preparing revisions of [1,14].

The Civaux case initiated several R&D projects such as: THERFAT (Thermal Fatigue Evaluation of Piping Systems Tee-Connections) [29], FAT3D [30] and the EDF R&D Programme [31].

On-going R&D projects are the thermal mixing Tee research at the university of Stuttgart in Germany [32] and the experimental and numerical comparison on a mixing Tee in the MOTHER project [33].

1. ***Quality management:***

### Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the different national regulatory requirements (e.g., 10 CFR 50, Appendix B [34]).

### References

1. ELECTRIC POWER RESEARCH INSTITUTE, Materials Reliability Program: Management of Thermal Fatigue in Non-Isolable Reactor Coolant System Branch Lines MRP-146 Revision 2 (EPRI 3002007853), September 2016.
2. ELECTRIC POWER RESEARCH INSTITUTE, Temperature Monitoring Data Evaluation for RCS Branch Lines Subject to Thermal Fatigue, MRP-365, Revision 1, (EPRI 3002016011), November 2019.
3. ELECTRIC POWER RESEARCH INSTITUTE, Materials Reliability Program: Thermal Fatigue Monitoring Guidelines MRP-32, Revision 1 (EPRI 1022563), EPRI, Palo Alto, CA, 2011.
4. ELECTRIC POWER RESEARCH INSTITUTE, NDE Technology for Detection of Thermal Fatigue Damage in Piping, MRP-23 Revision 3, 3002017285, December 2019,
5. ELECTRIC POWER RESEARCH INSTITUTE, Fatigue Management Handbook MRP-235 Revision 3, : 3002018246, August 2020, 2015.
6. ENSI B-01, Alterungsüberwachung, Richtlinie für die schweizerischen Kernanlagen, August 2011.
7. IRSN, Le point de vue de l’IRSN sur la sûreté et la radioprotection du parc électronucléaire français en 2013, https://www.irsn.fr/FR/expertise/rapports\_expertise/ Documents/surete/Rapport-Surete-Parc-2013\_IRSN\_201412.pdf
8. Investigations of mixing zones subjected to thermal fatigue, N. Robert et al. Fontevraud 6, sept 2006
9. The Japan Society of Mechanical Engineers, Guideline for evaluation of high-cycle thermal fatigue of a pipe, JSME S017, (2003) (in Japanese).
10. Canadian standard: CSA N285.4 ‘Periodic inspection of CANDU nuclear power plant components’.
11. Canadian standard: CSA N285.7 ‘Periodic inspection of CANDU nuclear power plant balance of plant systems and components’.
12. Chapuliot, S., Gourdin, C., Payen, T., Magnaud, J.P., Monavon, A., 2005. Hydrothermal-mechanical analysis of thermal fatigue in a mixing tee. Nuclear Engineering and Design 235, 575-596.
13. ELECTRIC POWER RESEARCH INSTITUTE, BWR Vessel and Internals Project, Assessment of Mixing Tee Thermal Fatigue Susceptibility in BWR Plants, BWRVIP-196 Revision 1 (EPRI 3002013099), Palo Alto, CA, November 2018.
14. ELECTRIC POWER RESEARCH INSTITUTE, Materials Reliability Program: Assessment of Residual Heat Removal Mixing Tee Thermal Fatigue in PWR Plants MRP-192 Revision 3 (EPRI 3002013266), November 2018 .
15. ELECTRIC POWER RESEARCH INSTITUTE, BWR Vessel and Internals Project, Evaluation of Thermal Fatigue Susceptibility in BWR Stagnant Branch Lines, BWRVIP-155 revision 1 (EPRI 3002013098), EPRI, Palo Alto, CA, November 2018.
16. INCIDENT REPORTING SYSTEM, Safety Injection Pipe Failure, IRS-851, 1987.
17. INCIDENT REPORTING SYSTEM, Through Wall Crack in a Non-Isolable ECCS Fitting on the Reactor Coolant Pressure Boundary, IRS-864, 1988.
18. UNITED STATES NUCLEAR REGULATORY COMMISSION, Thermal stresses in piping connected to reactor coolant systems, Bulletin 88-08, USNRC, June 22, 1988.
19. UNITED STATES NUCLEAR REGULATORY COMMISSION, Thermal stresses in piping connected to reactor coolant systems, Bulletin 88-08 Supplement 1, USNRC, June 24, 1988.
20. UNITED STATES NUCLEAR REGULATORY COMMISSION, Thermal stresses in piping connected to reactor coolant systems, Bulletin 88-08 Supplement 2, USNRC, August 4, 1988.
21. UNITED STATES NUCLEAR REGULATORY COMMISSION, Thermal stresses in piping connected to reactor coolant systems, Bulletin 88-08 Supplement 3, USNRC, April 11, 1989.
22. INCIDENT REPORTING SYSTEM, Non-Isolable Primary Leak of 600 l/h, IRS-1362, 1992.
23. INCIDENT REPORTING SYSTEM, Non-Isolable Primary Leak on the Hot Leg N°1 Safety Injection Pipe, IRS-7019, 1996.
24. INTERNATIONAL INCIDENT REPORTING SYSTEM (IRS), Cracks discovered in excess letdown system piping, IRS-7908, 2007.
25. ELECTRIC POWER RESEARCH INSTITUTE, Operating Experience Regarding Thermal Fatigue of Piping Connected to PWR Reactor Coolant Systems MRP-85 Rev.2, 2018.
26. ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, CSNI Integrity and Ageing Working Group, Thermal Cycling in LWR Components in OECD-NEA Member Countries, NEA/CSNI/R(2005)8, July 27, 2005.
27. M. McDevitt, M. Hoehn, T. Childress, R. McGill, Analysis and Impact of Recent US Thermal Fatigue Operating Experience, 4th International Conference on Fatigue of Nuclear Reactor Components, 28/9-1/10/2015, Sevilla, Spain.
28. J. Spanner, Improving Ultrasonic Examination Procedures for Detection of Thermal Fatigue, 4th International Conference on Fatigue of Nuclear Reactor Components, 28/9-1/10/2015, Sevilla, Spain.
29. K.-J. Metzner & U. Wilke, European THERFAT project – Thermal Fatigue Evaluation of Piping System ‘Tee’-Connections, Nuclear Engineering & Design, Vol. 235, Issues 2-4, 2005, pp. 473-484.
30. ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, FAT3D – An OECD/NEA Benchmark on Thermal Fatigue in Fluid Mixing Areas, NEA/CSNI/R(2005)2, August 1, 2005.
31. J.M. Stephan, C. Vindeirinho, S. Taheri, F. Curtit, M. Akamatsu, C. Peniguel, Evaluation of the risk of damage in mixing zones: EDF R&D programme, 10th International Conference on Nuclear Engineering, Arlington, VA, April 14-18, 2002, Paper ICONE 10-22432.
32. P. K. Selvam, R. Kulenovic, E. Laurien, " Experimental and numerical analyses on the effect of increasing inflow temperatures on the flow mixing behavior in a T-junction". International Journal of Heat and Fluid Flow, Volume 61, Part B, October 2016, Pages 323–342.
33. O. Braillard, R. Howard, K. Angele, A. Shams, N. Edh, "Thermal mixing in a T-junction: Novel CFD-grade measurements of the fluctuating temperature in the solid wall". Nuclear Engineering and Design, Volume 330, Pages 377–390, 2018.
34. UNITED STATES NUCLEAR REGULATORY COMMISSION, 10 CFR Part 50, Appendix B, Quality Assurance criteria for Nuclear Power plants, Office of the Federal, Register, National Archives and Records Administration, USNRC, Latest Edition.