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French regulation and flow-induced vibration issues for steam generator design qualification

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1. Introduction

French utility, EDF, is planning to replace a number of steam generator (SG) and to take international call for bids. SG are used in PWR power plants to transfer heat from the reactor coolant into water in secondary circuit to produce the superheated steam, used to power the electricity generating turbines. Usual French PWRs may have up to four loops, each with a SG. Each of SG covers 450 tones of weight and have interior circuit consisting of 122 km of tubes (covering 80 % of pressure boundary) for the production of steam. The steam generator (SG) will be installed at 1300MWe series of nuclear power plant (NPP). EDF planning order is part of the program for the gradual replacement of major plant components. Thus, market prospect from future call for bids would be bright.

Preliminary survey on French regulatory practices and technical issues on safety qualification of SG design would be highly necessary to investigate. This paper will support to build SG design methodology to cover international SG market requirement.

2. French regulation for SG design qualification

French government and ministerial decrees and orders under the law disciplines the lower ASN (French Nuclear Safety Authority) decision and ASN regulatory guides for NPPs construction and operation. This framework is corresponding to DOE-10CFR and US-NRC regulatory guides, leading to international consensus by the US nuclear industry. ASN issues regulatory decision (general technical rules and guideline) that able to be endorsed by the French government. Their guides explain how to consider the corresponding regulations for the general nuclear power components. Based on the French regulatory framework, upper design requirement and technical standards for RSG design can be referred to the decree N° 99-1046[1]/Directive 2014/68/UE [2] and ESPN order [3]. Then RCC-M code [4] provides specific design requirement, technical standard, and design criteria for pressure containing equipment.

The RCC-M code, "Design and Construction Rules for Mechanical Components of PWR Nuclear Islands", was first published in 1981, based on the ASME code, since then, has been updated on a regular basis to incorporate the experience feedback and the evolution of industrial practices. AFCEN program has updated with the main objective being to facilitate the demonstration of the compliance of Nuclear Pressure equipment designed and constructed according to the RCC-M code with the French Nuclear Pressure Equipment.

General requirements to dynamic loading analysis of SG components are given in RCC-M code, which have similar construction with ASME code. RCC-M code are used for integrity validation of pressure components regarding dynamic loadings, as the French regulation guidance. However, RCC-M do not explicitly describe the flow-induced vibration consideration, but share the needs of 'deserve attention' when considering technical requirements and dynamic analysis methods.

Philosophy of designing pressure retaining component of class 1 NPP components is very traditional. Classification of the service loadings is the fundamental step and the design criteria (allowable stress limit) is then defined according to the loading classification and structural material to be used. From this parallel respect, designing SG based on the RCC-M code is to ensure the integrity (structural margin for safety) of that component by comparing stress intensity (by analysis) at specific location with the design criteria or allowable stress limit, for given and/or anticipated service loading. Likewise design rule [5] for ASME Subsection NB class1 component, the requirement for the steam generator pressure boundary can be referred to the RCC-M Section B [4].

Two international codes sharing with the equivalent integrity evaluation philosophy have similar configuration, but ASME Code [5] has a limited scope based on simple assumptions (some are conservative, some are not, some are based on very old data and so on) with respect to structural design, while the RCC-M is more prescriptive and detailed with respect to analytical requirements and provides an increased understanding of margins. RCC design rule is to ensure safety margin to possible damages against imposed-loadings. Code rule aims for ensuring proper operation of pressure containing equipment. Detail rules applicable to the sizing of pressure retaining components and to the analysis of their behavior when subjected to the loads stipulated in the equipment specifications.

3. Some issues on flow-induced vibration

Frequently-problem-occurring regions in SG design are the flow entrance/exit and the U-bend region, where fluids cross-flow across the tube bundle. A combination of flow-induced turbulence and fluidelastic forces on the tubes results in excessive vibration of tubes within their supports. This vibration causes tube/support impact and sliding, leading to premature tube damage. To calculate tube vibratory motion due to forces created by flow turbulence and fluidelastic coupling, international researchers developed a finite element model of the tube and its support structure using thermal-hydraulic and structural codes [6, 7]. They incorporated the non-linearity associated with clearances at the tube support plates and AVBs in the finite element analysis. The EPRI(US) and CEA (France) developed methods to calculate the work rate and contact time.

The developers validated these methodologies by comparing analysis results with experimental data. By applying this methodology to evaluate tube wear in operating SGs, they assessed progression of tube wear during plant operation. These thermal-hydraulic and flow-induced vibration analyses used modeling software was confirmed by that had been used in nuclear steam generator designs. Using results of extensive experiments, they developed a tube wear evaluation method and put them into their code system. These design tools calculate tube vibration induced by flow turbulence and fluid-elastic forces including maximum response of a tube, calculate tube motion (for example, sliding and impact within its support structure) at tube-to-support, tube-to-tube support plate (TSP), and tube-to-AVB and evaluate their wall metal loss. The effectiveness of AVBs, tube support plate, and lattice grid (eggcrate) at preventing fluid-elastic instability was considered in the analyses. The analysis was performed for the most limiting location in the tube bundle and with consideration of the limiting tolerances.

In addition, technical issues, arising in France nuclear power industry, on the flow-induced vibration and wear failure of SG, are vibro-impact-induced work rate[8, 9], nonlinear vibration analysis of gap supported tube[10], SONGS tube failure related issues and root cause analysis[11], systematic wear prediction in EPRI(US), and other topical issues arising from French nuclear industry and RSG market.

4. Summary and Conclusion

French regulation for NPP components has a general design requirement for pressure retaining components with respect to flow-induced dynamic loadings and some specific requirements would be added to guide the analysis code for predicting potential tube damage and the safe design. There are no explicit descriptions on the flow-induced vibration and its adverse loadings effect in French nuclear code, RCC-M, but French NPP component supplier seem to usually follow long established practices and more or less US-NRC regulation guides. Vibro-impact wear work rate and nonlinear vibration analysis for gap-supported tube would be highlighted among technical issues on flow induced vibration of NPP component design qualification

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