Nuclear Engineering and Design 319 (2017) 40–47

Contents lists available at ScienceDirect

Nuclear Engineering and Design

journal homepage: www.elsevier.com/locate/nucengdes

Risk assessment of thermal striping in MYRRHA research reactor

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HIGHLIGHTS

- An assessment study of thermal striping was performed.
- The purpose was to assess the risk in MYRRHA research reactor.
- The study has given satisfying results.
- The risk incurred is globally low, due to the operating conditions.
- Few problems remain persistent.

ARTICLE INFO

Article history:
Received 21 November 2016
Received in revised form 24 April 2017
Accepted 26 April 2017

Keywords:
Thermal striping
Safety analysis
MYRRHA/LBE
Fatigue
Fatigue curve
RCC-MRx
Nuclear engineering

ABSTRACT

The thermal striping is one of the problems encountered in nuclear facilities. It may affect the reactor main components structural integrity and, sometimes, it can cause their rapid failure.

The thermal striping problem touches, in general, mixing zones of industrial facilities. While it affects several areas as the core of liquid metal reactors that is due to the larger difference between core inlet and core outlet temperature, these reactors are more vulnerable to striping compared to water reactors.

It is true that a better understanding of the induced thermal loads and associated damage mechanisms are obtained nowadays, thanks to recent developments in different domains such as CFD calculation and fatigue modelling. However, in the MYRRHA reactor project, the coolant used will be the Lead Bismuth Eutectic (LBE), i.e. a liquid-metal with a high thermal conductivity. Such conditions are broadly different compared to the most conventional coolant that is water.

The present paper presents an assessment of the consequences of thermal striping under the severe conditions of the MYRRHA reactor. The results of the study can be considered as satisfactory since the risk to structural integrity appears to be globally low. A number of aspects will however require further investigation as the reactor design progresses.

1. Introduction

The Belgian Nuclear Research Centre, SCK.CEN has been working for several years on the design of MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications), a flexible fast spectrum nuclear reactor. This project is one global first demonstration for a new type of reactor, driven by a particle accelerator – ADS (Koloszar et al., 2014).

This situation involves a continual thermal assessment in order to support the design and safety analyses of the reactor. Preliminary results of the CFD simulations show the existence of temperature gradients inside the core and thermal stratification in the hot plenum (Koloszar et al., 2014).

These phenomena are recognized as one of the important issues in fast reactor design since they cause a cyclic thermal loading, which may influence the reactor vessel structural integrity (Ohno et al., 2011).

The objective of the current work is to perform an assessment study of the thermal striping risk in the MYRRHA research reactor. The approach adopted is rather conservative as a first attempt; detailed works can be developed in future.

This study consists in the investigation of the sensitivity of the temperature inside the component's wall to some key parameters. Consequently, a better understanding is obtained on the origin of the phenomenon and the situations that promote its creation and its different stages of evolution. The risk's statement can be then appreciated. During this study, RCC-MRx code is used to evaluate the mechanical effect on structures.

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http://dx.doi.org/10.1016/j.nucengdes.2017.04.031

A simple survey about nuclear incident events can reveal the proportion taken by this phenomenon. The IAEA technical document (Iaea-Tecdoc-1318, 2002) gives the following cases: A high-cycle thermal fatigue was found to be the cause of the cracks in the connecting pipe and the middle-stage heat exchange (HE) shell at the Tsuruga-2 PWR (Japan) in 1999. An initial crack in a tee-junction zone, in the secondary loop was detected during a campaign of inspections, in the Phenix LMFR (liquid metal fast reactor). Several areas of the reactor were subjected to this problem.

Many authors reported incidents occurred in nuclear reactors. Chellapandi and Chetal (2009) reported that, in SPX-1 reactor, a sodium leak was observed on a circumferential weld downstream of a tee-junction in an auxiliary pipe of the secondary sodium circuit.

The majority of component failures due to thermal striping were observed in austenitic stainless steels (Gavrilov, 2011): in Phénix (PHX) an extensive cracking at secondary sodium pump vessel was observed. Similar problems happened in the tee-junction of an auxiliary pipe of the secondary circuit of "Super phénix" (SPX), in the control rod guide tubes of Dounreay's Prototype Fast Reactor (PFR) and in the primary cold trap of BN-600.

It is important to notice that the most kind of nuclear reactors affected by thermal striping accidents, inside the reactor core, are those using liquid metals. The causes are the high thermal conductivity of liquid metals and the high temperature difference between the input and the output of the reactor core. On the contrary, for water reactors, the coolant characteristics are less favourable and the temperature differences are low.[1] That is why the risk in the first ones is higher than in the second ones. Exception is made in water reactors, for some situations of very high temperature and pressure, in which water thermal characteristics may promote thermal striping (Bejan, 1993). The NESC database (Dahlberg et al., 2011) contains about 40 documented cases.

There have been instances of thermal fatigue damage and over the last 10 years, several recent R&D programmes have been devoted to the development of a better understanding of the induced thermal loads and associated damage mechanisms (Dahlberg et al., 2011). Gritskevicha et al. (2014), Lee et al. (1999), Kamaya (2014), Chellapandi and Chetal (2009), Dahlberg et al. (2011), and others were interested by developing new approaches to resolve the thermal striping problems.

Globally, Paranjpe (2006) & Benhamadouche (2006) claimed that the new trend is focused on the improvement of CFD calculation software in order to identify finely thermal loads. However, Muramatsu (Iaea-Tecdoc-1318, 2002) affirms that the establishment of numerical evaluation methods is desirable in support of the experimental approach.

These recent developments constitute a major advance over what was done before. However, it is interesting to consider the particular conditions in which MYRRHA is expected to incur. In addition to its operation at high-temperature, the use of the LBE brings some further difficulties. Despite its nuclear advantages, this coolant presents a number of challenging aspects. Firstly, it is a high density liquid metal, which causes additional loads; secondly, it is characterised by high heat transfer coefficients that are responsible for higher values of the Biot number; and finally it is more corrosive on steel than other coolants, which requires the reduction of the upper-limit of the cooling velocity. Such conditions make the thermal striping problem even more challenging are the starting point to establish the goals of this work. The main one of these is the assessment of the thermal striping risk incurred by MYRRHA principal components with the aim of to evaluate the thermal fatigue life of the vessel, barrel and the heat exchanger. The challenge of this work is to resolve problems set by these specific conditions.

The second objective is to establish guidelines for ensuring a good design of those components.

2. General conditions

2.1. Temperature distribution inside MYRRHA reactor

Heat transfer through the structures of the reactor is expected to have a non-negligible effect on the power balance of the different regions in the reactor, because of high temperature differences between the lower and upper plenum. In order to evaluate the temperatures of the structures, the simulation has to account for conjugate heat transfer (CHT) that is the combination of the heat conduction through the walls and the thermal boundary layers on both sides of the structures. The temperature differences between adjacent fluid regions create thermal gradients in the structures, which induce thermal stresses as illustrated in Fig. 1(b).

Fig. 1 shows that the temperature difference range is around 20–60 °C.

2.2. Study conditions

In order to investigate the thermal striping risk within the structures of the MYRRHA research reactor, a simplified model is used where a temperature fluctuation is imposed on one side of a flat plate structure with the following hypotheses:

1. a pure sinusoidal fluid temperature variation,
2. the temperatures are defined as illustrated in the following Fig. 2:
3. negligible creep as the specific environment temperature is less than the limit stated by the creep criterion of reference (AFCEN, 2012);
4. the irradiation effect is not considered (AFCEN, 2012),
5. two fluid states are defined: (i) and (j) around the time averaged temperature T_0,
6. the elastic analysis is chosen as work basis, however, the plastic case is considered by the introduction of appropriate correction factors (AFCEN, 2012),
7. the errors estimation is based on the uncertainty of the Nusselt-correlation as defined in the Lead-Bismuth handbook (Nuclear Science Committee, 2007), where the predictions are based on a numerical analysis and the minimum error of the cloud of experimental data (Nuclear Science Committee, 2007).
8. the plate is clamped on its 4 edges,
9. the material is the 316L austenitic stainless steel and the temperature is 350 °C,
10. in order to simplify the presentation of results, the temperatures mentioned in this document represent differences of temperatures.

3. Theoretical investigation

3.1. Resolving methodology

The solving methodology was inspired by the European procedure for assessment of high cycle thermal fatigue (Dahlberg et al., 2011). It was adapted to solve the simplified flat plate within liquid metals reactor as shown in Fig. 3.

[1] Most of these cases could be avoided if foreseen during the design phase. For innovative design temperature difference to be considered as unavoidable, for structural material, is the difference between core inlet and core outlet.