QUANTITATIVE MODEL FOR PREDICTING THE SUITABILITY OF CANDIDATE MATERIALS FOR REACTOR PRESSURE-VESSELS MANUFACTURING

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ABSTRACT
In this work, an integrative approach that eases the material selection stage is presented. Twenty different specifications of materials from 1st to 4th generation of pressurized water reactors have been analyzed, covering the main designs of pressurized reactors from Western Europe, Japan and Russia. Thus, the results show that steels 15Kh2MFAA, 20MnMoNi55 and SA533 Gr.B Cl.1 are the best material options to prevent the embrittlement induced by radiation.

Keywords: radiation-induced embrittlement, reactor pressure-vessels, structural materials.

INTRODUCTION
The materials of nuclear reactor pressure-vessels exposed to neutron radiation generated by nuclear fission reactions experience considerable damage at very low doses of radiation, causing embrittlement and a shift of the ductile-to-brittle transition temperature ($\Delta RT_{NDT}$). Irradiation embrittlement of steels is the principal mechanism of aging that affects the pressure vessels of light water nuclear reactors, being the chemical composition the most important parameter affecting the radiation-induced embrittlement; in particular, the percentages of copper, nickel and phosphorus are considered the most influencing parameters.

The selection of materials for the construction of the primary loop of a light water reactor is a complex process that involves great responsibility because small differences in chemical composition can adversely affect in-service behavior of the material. To address this issue, an evaluation of chemical composition and mechanical behavior based on the requirements described by the most important manufacturing codes has been performed. In this work, an integrative approach that eases the material selection stage is presented. Thus, twenty different specifications of materials from 1st to 4th generation of pressurized water reactors have been analyzed, covering the main designs of pressurized reactors from Western Europe, Japan and Russia.

METHODOLOGY
This new approach combines structural calculations, based on the mechanical requirements of materials, with the application of a stringency level (SZ) methodology (Rodríguez-Prieto et al., 2016) to analyze the content of copper, nickel and phosphorus and the transition temperature ($\Delta RT_{NDT}$) according to Regulatory Guide (R.G.) 1.99 Rev.2 (1988) prediction model. This methodology consists of a calculation process based on equations using a
deterministic algorithm. The criteria acceptance limits are provided by key research publications. Necessary data for the SL methodology have been selected by performing data mining from key scientific publications, materials handbooks and databases, manufacturing codes, technical specifications and regulatory requirements.

RESULTS AND CONCLUSIONS

It has been demonstrated, with structural calculations, that safety design margin calculated as a ratio between longitudinal or transversal membrane stresses and yield strength of materials are below than 2/3 according to requirements described by ASME B&PV (2015) and KTA 3201.2 (2011) design codes. On the other hand, Russian materials 15Kh2MFAA and 15Kh2NMFAA and the American A336 Gr.F22V and SA533 Gr.B Cl.1 provide the less ductile-to-brittle transition temperature together with more stringent chemical requirements to avoid radiation-induced embrittlement. Finally, the average results (Fig.1) show that steels 15Kh2MFAA, 20MnMoNi55 and SA533 Gr.B Cl.1 are the best material options according to the stringency level, $SL$.

Results obtained by applying this new methodology exhibit how, despite all options could be valid from a safety approach, the most suitable material options correspond to some of the specifications more widely used in the 2nd and 3rd generation of pressurized water reactors such as 15Kh2MFAA, 20MnMoNi55 and SA533 Gr.B Cl.1.

ACKNOWLEDGMENTS

This work has been financially supported by the funds provided through the Annual Grant Call of the E.T.S.I.I. of UNED of reference 2017-IFC04.

REFERENCES


