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PROBABILITY OF LARGE DIAMETER PIPING RUPTURE IN CONTEXT OF SEISMIC EVENT

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In the safety of nuclear power plants the large loss-of-coolant accident (LLOCA) still represents one of the essential issues. The so-called redefinition of large LOCA which was prepared by US NRC is based on the theory of transition break size (TBS). For Pressurized Water Reactor (PWR) the TBS is defined by the primary circuit piping with 360 mm diameter. The task for the regulatory body is to prove that the postulated circumferential crack satisfies the requirements of ASME Code, Section XI.

Keywords: large LOCA, redefinition, transition break size, circumferential break

Introduction

All the extreme effects that have to be involved into the process of nuclear power plants' (NPPs') safety assessment may be considered as exceptional or highly improbable. Nevertheless the consequences of these extreme effects can occur as vibrational or other unexpected impact. Considering many sources of risk, the experts concentrate their efforts to reduction of priority risks. There are many techniques used for the seismic hazard assessment and consequently for the seismic risk assessment that are based on different concepts and approaches.

The occurrence of the extreme effects on NPP's equipment and structures is expected with the probability less than 10^{-1}/year and more than 10^{-6} to 10^{-7}/year, i.e. the combination of unexpected events that do not eliminate each other, and decrease of NPP's system function, if the probability of appearance of such a combination is more than 10^{-6} to 10^{-7}/year. Selected combinations or separate extreme events determine the extreme cases essentials for NPP's projects.

Historically, the safety of NPPs is based on the ability to eliminate the large loss-of-coolant accident (large LOCA) which is represented by the double ended guillotine break of the primary circuit piping. The loss of coolant accident is one of the most limiting design-basis accidents that cause the loss of ability of the coolant to remove heat from the fuel. Even small losses of fluid (or loss of coolant flow) may have important consequences [1].

2. Transition Break Size

Transition break size (TBS) is a break of area equal to the cross-sectional flow area of the inside diameter of specified piping for a specific reactor.

The specified piping for a pressurized-water reactor is the largest piping attached to the reactor coolant system. The specified piping for a boiling-water reactor is the larger of the feedwater line inside the containment or the residual heat removal line inside containment.

There are two statements that describe TBS according to NUREG - 1903 [1]:

I. The current spectrum of LOCA break sizes would be divided into two regions. The division between the two regions is determined by a "transition break size" (TBS). The first region includes small breaks up to and including the TBS. The second region includes breaks larger than the TBS up to and including the double-ended guillotine break (DEGB) of the largest reactor coolant system pipe. The term, “break,” in the TBS does not mean a double-ended guillotine break; rather it refers to an equivalent opening in the reactor coolant system boundary.

II. Transition break size (TBS) is a break of area equal to the cross-sectional flow area of the inside diameter of specified piping for a specific reactor. The specified piping for a pressurized-water reactor is the largest piping attached to the reactor coolant system. The specified piping for a boiling-water reactor is the larger of the feedwater line inside containment or the residua heat removal line inside containment.

3. Seismic Risk Contributions

The goal of the analysis is to determine whether the risk associated with the direct, seismically induced failure of the primary reactor cooling piping (PLP) is significantly less than the failure risk caused by the expected loading histories considered in [2]. For any of the following three criteria satisfied at each analyzed location, the seismic risk of direct failure of PLP is considered negligible:

1. The critical flaw depths are greater than 30% of the through-wall thickness.

2. The critical flaw depths are greater than the ASME Code, Section IX, flaw acceptance criteria.

3. The ISI programs are sufficient for detecting flaws before reaching critical flaw depths calculated according to [3,] Section 2.2.2.4.2.

4. Example Calculation

This example will be used to demonstrate the principle of numerical procedure. The individual steps are enumerated in the following text.

4.1. Determine seismic hazard curve coefficients

At the very beginning of the calculation we need to find the seismic hazard of the locality, which will be labeled as Step 1.

1. The seismic hazard curve is determined by the Weibull equation fit for peak ground acceleration (PGA) versus the probability of occurrence

$$P(x) = Scale.\alpha.\beta^{-\alpha}\, x^{\alpha-1}\, e^{-(\frac{x}{\beta})\alpha}, \tag{1}$$

where parameters α and β are determined as a matter of geophysical research. The values depend on the given country and site (See Figure 1).

For the Czech NPP Temelin where the new two units may be built in near future, the seismic hazard curves (see Figure 2) will be established according to the Specific Safety Guide SSG-9: Seismic Hazards in Site Evaluation for Nuclear Installations [4]. Parameters α and β will be quantified for the locality of Temelin NPP.

Site Identification Code	Original design SSE, g	Weibul fit parameters for mean PGA probability curves			SSE probability	PGA at 10^{-6}, g	Ratio of PGA to original SSE value 10^{-6} / 1SSE
		scale	alpha	beta			
A	0.153	0.047	0.430	13.890	2.32E-05	0.633	4.135
B	0.100	0.062	0.384	12.300	5.85E-05	0.826	8.263
C	0.100	0.063	0.410	11.200	5.58E-05	0.675	6.754
D	0.100	0.068	0.395	12.280	6.37E-05	0.799	7.990
E	0.120	0.076	0.405	7.494	3.78E-05	0.574	4.785
F	0.100	0.081	0.424	11.340	7.24E-05	0.692	6.922
G	0.120	0.095	0.364	3.792	3.10E-05	0.526	4.384
H	0.104	0.098	0.391	15.270	9.65E-05	1.080	10.380
I	0.100	0.107	0.359	6.193	7.09E-05	0.780	7.798
J	0.120	0.120	0.389	18.130	1.04E-04	1.313	10.946
K	0.100	0.126	0.384	10.690	1.11E-04	0.991	9.914
L	0.104	0.127	0.379	15.100	1.24E-04	1.271	12.221
M	0.120	0.128	0.377	12.780	9.39E-05	1.165	9.709
N	0.120	0.130	0.380	13.050	9.63E-05	1.165	9.711
O	0.120	0.138	0.387	16.640	1.15E-04	1.327	11.062

Fig. 1: Example of table of scale factors, original design SSE PGA values, Weibull fit coefficients to mean PGA probability curves, and calculated PGA values at seismic event with probability 10^{-6} [3]

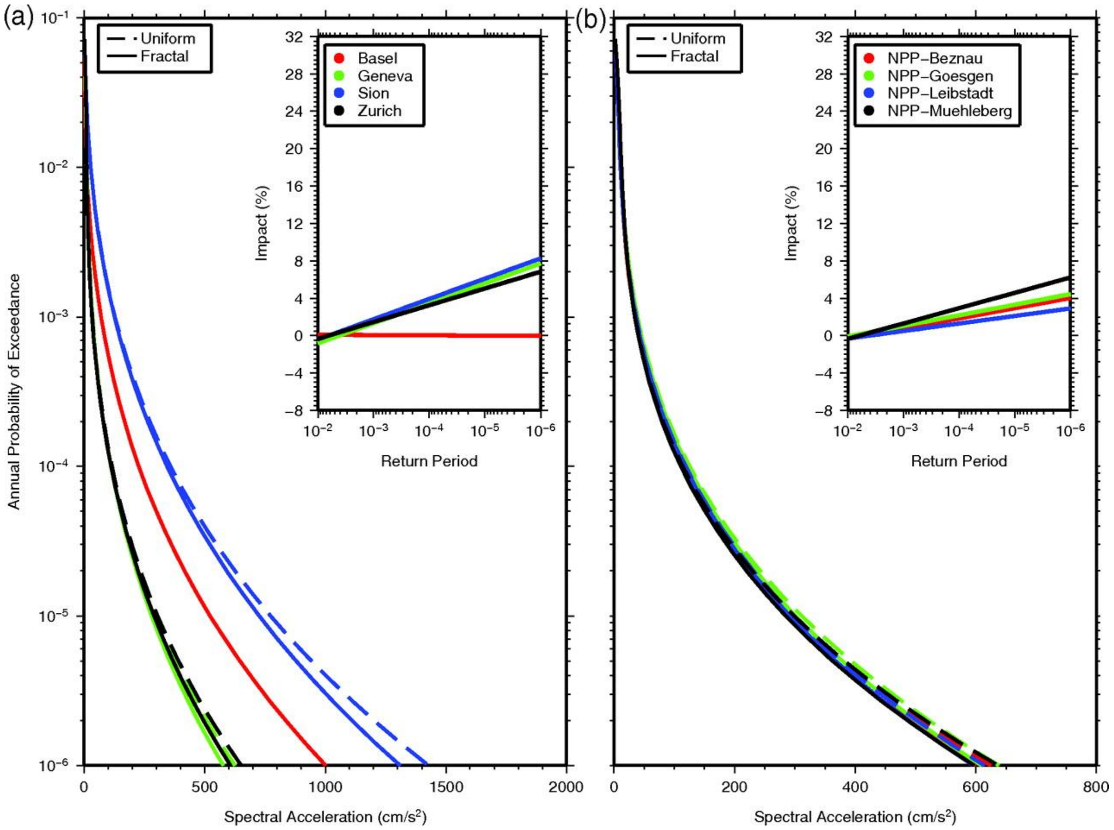

Fig. 2: Eample of seismic hazard curves

4.2. Next steps

After having determined the seismic hazard the following steps have to be proceed:

2. Obtain SSE (safe shut-down earthquake) design PGA value.
3. Solve for PGA value at $1x10^{-6}$ probability of occurrence, and obtain ratio of PGA at $1x10^{-6}$ to PGA at SSE.
4. Determine the highest SSE stress location.
   Note: For the seismic purpose stress distribution will be determined for all the primary circuit.
5. Determine the materials of interest at the critical localization.
6. Determine the pipe cross-sectional dimensions at critical location.
7. Determine normal operating conditions/stresses.
8. Determine strength values for materials of interest.
9. Determine the SSE stresses.
10. Determine the linearly scaled seismic stresses for the $1x10^{-6}$ seismic even.t
11. Apply seismic scaling factor for plant site to correct the linearly scaled stresses from Step 10 and add the normal operating conditions.
12. Apply nonlinear correction factor to the elastic N + $1x10^{-6}$ seismic stresses from Step 11 to obtain the nonlinear stress S_{NL}.
13. Determine the elastic-plastic correction factor (Z-factor) for the critical flaw size evaluation.
14. Determine EPFM-corrected stress S_{EC} for use in limit-load equations.
15. Determine the minimum critical surface flaw depth from limit-load equations.
16. Calculate the a/t value corresponding to ASME Service Level D loading.
17. Compare BE a/t value to the ASME Code a/t value from Step 16.

Conclusion

The so-called redefinition of large LOCA prepared by US NRC is based on the theory of transition break size (TBS). For the reactors of PWR type the TBS is characterized by the piping with diameter of 360 mm. In the paper a numerical example is presented to prove that the postulated circumferential crack satisfies the requirements of ASME Code, Section XI.

Literature

[1] US NRC, Seismic Considerations For the Transition Break Size. NUREG – 1903. Washington, DC., 2008

[2] US NRC, Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process. NUREG-1829. Washington, DC., 2008

[3] US NRC, Plant-Specific Applicability of 10 CFR 50.46 Technical Basis. Washington, DC., 2009

[4] IAEA, Specific Safety Guide SSG-9: Seismic Hazards in Site Evaluation for Nuclear Installations. IAEA, Vienna, 2010.

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