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Quantification of fuel rod cladding failure during LOCA accident

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ABSTRACT
The paper describes methodology for quantification of fuel cladding failure as a result of Loss of Coolant Accident. The methodology is based on external coupling of the RELAP5 code and TRANSURANUS code. The thermo-hydraulic response of the unit to the accident is simulated by RELAP5 code, providing initial and boundary conditions for the thermo-mechanical simulation by the TRANSURANUS code. Cladding failure criterion of the TRANSURANUS code, derived and implemented into the code in the framework of EXTRA EURATOM 5th Framework Programme is used. Cladding failure probability is evaluated by the Monte Carlo algorithm varying the outer cladding temperature. In the second part of the paper, an example of application of the methodology for typical maximum design basis accident of the VVER-440 is given, presenting every step of methodology and typical failure rate for this type of accident.

1. INTRODUCTION

A Loss of Coolant Accident (LOCA) is initiated by loss of integrity of the primary circuit or of the pipes and devices connected to it. It leads to the degradation of core cooling and could result in fuel and cladding overheating, increase of internal fission gas pressure and thermally induced stresses. Due to this, the integrity of the cladding can be lost and fission products could be released into the primary circuit. Thus prediction of the fuel and cladding behaviour during LOCA conditions and prediction of failure probability is of high safety importance.

One of the decisive presumptions for an applicable methodology of evaluation of the fuel rod failure is a validated code, containing algorithms of all relevant physical phenomena up to the cladding failure. During the last years there was large effort spent for validation of the TRANSURANUS code for application to VVER fuel. Recently the criterion for prediction of the fuel rod cladding failure was included into the code, as a result of EXTRA project.

TRANSURANUS is a computer program for the thermal and mechanical analysis of fuel rods in nuclear reactors. It was developed at the Institute for Transuranium Elements (ITU) [2]. It can be employed as a deterministic as well as a statistical code. The TRANSURANUS code version for VVER fuel rods is under development since mid 90's. Based on the EXTRA Project the code TRANSURANUS has been extended for specific VVER fuel and cladding type.

The EXTRA Project (Extension of TRANSURANUS Code Applicability with Nb Containing Cladding Models) has been realized in the framework of the EURATOM 5-th Framework Programme through the partnership of three research institutions, JRC Institute for Transuranium Elements (Germany), VUJE (Slovak Republic) and KFKI Atomic Energy Research Institute (Hungary). The EXTRA Project started on December 2001 and was completed by the end of 2003 [1].

The main objective of the EXTRA Project was to provide the possibility of the correct and consistent simulation of VVER fuel rods, especially at LOCA conditions. The TRANSURANUS code was extended by new models for the high temperature steam oxidation, plastic deformation and failure of the cladding tube. New VVER specific empirical correlations for the mass gain rate, the Zirconium oxide layer growth rate and the deformation rate were developed and adapted to the models. The recent mechanical model takes into account the oxygen concentration as a parameter of the deformation rate. The present state of the model validation proves the correctness of the applied approaches and indicates the reliable prediction of the Zr1%Nb cladding failure under simulated LOCA conditions. This enables application of the code for the licensing purpose safety analysis.

The complex methodology for quantification of the fuel rod failure requires not only the fuel rod analysis but also coherent definition of initial and boundary conditions both for the individual fuel rod and for variations of relevant parameters through the core and operational time. Such complex approach is presented in the following sections, together with the example application of the methodology.

The RELAP5 code is selected to be the representative of system codes for LOCA evaluation for VVER-440 reactor type. The RELAP5 code is light water reactor transient analysis code and was developed at the Idaho National Engineering Laboratory for the U.S. Nuclear Regulatory Commission (NRC) [3]. The code is well validated also for VVER-440 reactor types.

2. Methodology for quantification of the fuel rod cladding failure during LOCA accident.

The methodology was developed and is applicable for VVER-440 reactor type fuel, for the purpose of conservative safety analysis (licensing type safety analysis). It is based on application of the thermo-hydraulic RELAP5 code and thermo-mechanical TRANSURANUS code. The response of the unit to LOCA initiation is simulated by the RELAP5 code, resulting in time functions of the primary and secondary circuit parameters. The results of this thermo-hydraulic calculation represent the initial and boundary conditions for thermo-mechanical calculation with TRANSURANUS code. The TRANSURANUS code, utilising the code internal statistical Monte Carlo methodology is used for quantification of cladding failure during the postulated LOCA accident.
The methodology includes following steps.

- **Division of fuel rods in the core**.

The aim of this step is to reduce number of thermo-mechanical analyses to an acceptable number. All fuel rods in the core are split into several groups based on the fuel rod power and on time of irradiation (or burn up). The thermo-mechanical calculation is performed only for one fuel rod, which is conservatively representing all fuel of the relevant group. Based on experience with VVER-440 reactor type the core fuel rods should be divided at least into four groups based on power and into five groups based on irradiation. It means that totally twenty groups of fuel rods should to be taken into account to consider all fuel rods in the core.
Selection of the number of groups derived from power has an impact to the RELAP5 model. Number of groups must be therefore adjusted to the existing model or to possibilities of the RELAP5 model modification.

- **Adjustment of models for RELAP5 and TRANSURANUS codes**

The aim of this step is to ensure correspondence of models for TRANSURANUS and RELAP5 code. The input model for TRANSURANUS code should correctly reproduce RELAP5 results, which are initial and boundary conditions for thermo-mechanical analysis of the fuel rod. The important features and parameters to be considered in adjustment contain axial and radial discretisation of fuel rod, geometry of fuel rod (inner and outer fuel and cladding radius, cladding thickness and gap width). The RELAP5 model should address and provide all relevant data - boundary conditions - for all groups defined in the previous step, especially for all power based groups.

- **Thermo-hydraulic analysis of LOCA accident using the RELAP5 code**

This step is the defining of boundary conditions for thermo-mechanical analysis using required level of conservatism for the whole spectrum of fuel rods in the core. Initial and boundary conditions (scenario) for thermo-hydraulic analysis are selected with the aim to reach maximal cladding temperature during the analysed LOCA accident. This assumption leads to maximum number of cladding failures. Results of thermo-hydraulic analysis, being passed to the following step, to the thermo-mechanical analysis typically contain axial power distribution, pressure at the core inlet and outer cladding temperature in each axial node.

- **Data transformation from thermo-hydraulic analysis and input preparation for thermo-mechanical analysis**

Objective of this step is in formal transformation of the results of thermo-hydraulic calculation using RELAP5 code into the form suitable and directly applicable to the thermo-mechanical analysis by the TRANSURANUS code. The TRANSURANUS input file (model) has two main sections. First section contains data, which are not time dependent and are not touched by the transformation. The second section (Macro Time Step section) contains time dependent data, which describe analysed accident. This section should be generated directly, and in a validated and automated way by the transformation algorithm.

As the thermo-mechanical analysis includes steady state calculation at the nominal power with the aim to reach required burn-up for the corresponding group and the consecutive thermo-mechanical analysis of the LOCA accident, both data for steady state and transient should be generated, including also time step synchronization.

- **Statistical thermo-mechanical analysis using TRANSURANUS code**

The aim of this final step is to quantify fuel rod cladding failure. The calculations are preformed for all fuel rods, which conservatively represent each defined group based on burn-up and fuel rod power. The outer cladding temperature, evaluated by thermo-hydraulic calculation, is considered to be the decisive boundary condition from the point of view of potential cladding failure. For the statistical thermo-mechanical analysis based on Monte Carlo methods, it is assumed the values of outer cladding temperature are randomly varied with uniform statistic distribution.

Sufficient number of Monte Carlo runs needs to be performed and confirmed by sensitivity check.

Finally, the probability of fuel rod cladding failure is evaluated based on number of cladding failures, predicted by the statistical calculation.

3. Example Application of the Methodology

The methodology, described in the chapter 2 was applied to the licensing type evaluation of the maximum design basis accident of the VVER440/V213 reactor. It presents an example illustrating each individual step of application and provides results, typical for this type of reactor.

- **Division of fuel rods in the core**

All fuel rods in the core (349 fuel assemblies in the core x 126 fuel rods in the assembly = 43 974 fuel rods in the core) are divided into four groups based on fuel rod power. Average fuel rod power is 31.268 kW (1375 MW / 349 assemblies / 126 fuel rods), the maximum rod power allowed for the core is 55 kW.

There is five-year-load scheme applied, thus each of the power group of rods is split into five groups (if relevant) to reflect time of irradiation. For each of these groups the maximum burn-up within the group was applied to the analysed fuel rod.

For the analysis, a typical core load pattern was considered at the end of cycle. Split of the fuel rods into the defined groups is summarised in the Table 1.

Table 1: Grouping of the fuel rods (End of cycle, five-year load)

Calculation number	Analysed range of relative fuel rod power in the core	Power of the rod used for calculation [kW]	Time of base irradiation [year]	Number of fuel rods [-]	Max. Burn-up [MWD/kgU]
1.	(1.4 ; max>	55.000	5	0	--
2.	(1.4 ; max>	55.000	4	0	--
3.	(1.4 ; max>	55.000	3	0	--
4.	(1.4 ; max>	55.000	2	42	27.34
5.	(1.4 ; max>	55.000	1	498	13,82
6.	(1.2 ; 1.4>	43.776	5	0	--
7.	(1.2 ; 1.4>	43.776	4	0	--
8.	(1.2 ; 1.4>	43.776	3	30	38.64
9.	(1.2 ; 1.4>	43.776	2	7080	27.79
10.	(1.2 ; 1.4>	43.776	1	5256	13.82
11.	(1.0 ; 1.2>	37.522	5	84	54.16
12.	(1.0 ; 1.2>	37.522	4	1014	47.89
13.	(1.0 ; 1.2>	37.522	3	6114	38.83
14.	(1.0 ; 1.2>	37.522	2	2706	25.28
15.	(1.0 ; 1.2>	37.522	1	2790	12.68
16.	(0.0 ; 1.0>	31.268	5	6846	56.71
17.	(0.0 ; 1.0>	31.268	4	8058	46.97
18.	(0.0 ; 1.0>	31.268	3	2172	35.67
19.	(0.0 ; 1.0>	31.268	2	0	--
20.	(0.0 ; 1.0>	31.268	1	1284	12.18

- **Adjustment of models for RELAP5 and TRANSURANUS code**

Input file for thermo-hydraulic analysis should consider thermo-hydraulic channels of four rods with different rod power, with conservatively specified hydraulic conditions. Models for both RELAP5 and TRANSURANUS were modified accordingly. RELAP5 model includes models of hot rod, 1.4 x average rod, 1.2 x average rod and average rod. All the rods are in one hydraulic channel defined with conservative boundary conditions.

Nodalization of fuel rod is exactly the same for the both codes. In axial direction, the fuel is divided into ten slices. The radial discretisation of the rod covers five nodes in the fuel and three nodes in the cladding (Figure 1).

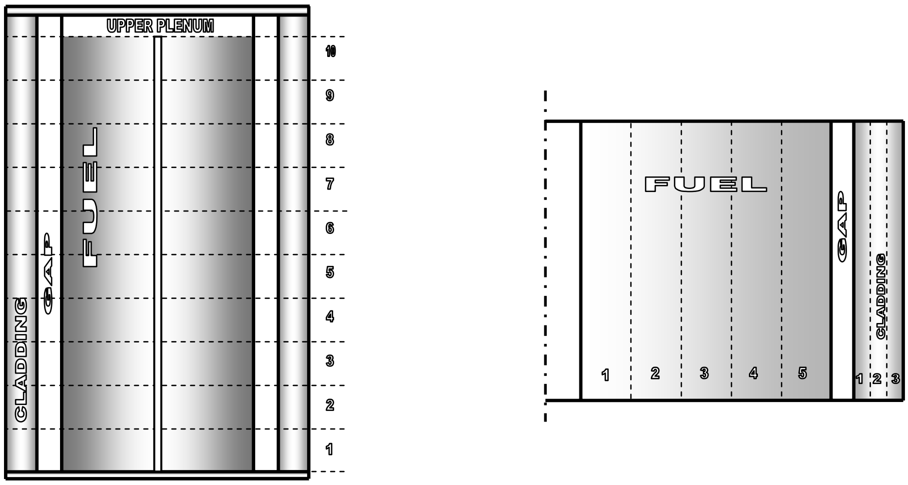

Figure 1: Axial and radial fuel rod discretisation

➢ **Thermo-hydraulic analysis of LOCA accident using the RELAP5 code**

For this analysis Large Break LOCA with equivalent diameter of 500 mm was selected, which is the maximal design basis accident for VVER440/213. Methodology of calculation and definition of initial and boundary conditions was based on guidelines for accident analysis [4] and on requirements for Safety Analysis Report in Slovak Republic.

The most conservative position of the break, resulting in the highest fuel cladding temperature in the reference analysis, was chosen on the basis of sensitivity calculations. It pointed out that a guillotine-type break in the cold leg of main circulation loop with the pressurizer results in the worst cooling conditions.

Definition of LOCA case scenario was:

- Double ended guillotine break (2 x F500) on cold leg (CL) of main circulation loop between main isolation valve and reactor inlet nozzle (cold leg with pressurizer (PRZ)). The position of the break is shown in the Figure 2.
- Two hydroaccumulators (HA) are available from four installed (one is connected to the upper plenum and second to the downcomer);
- Only one from three high pressure injection pumps (HPIP) is available
- Only one from three low pressure injection pumps (LPIP) is available
- Initiating event was assumed during a stable operation of the unit at nominal power
- No operator intervention was assumed

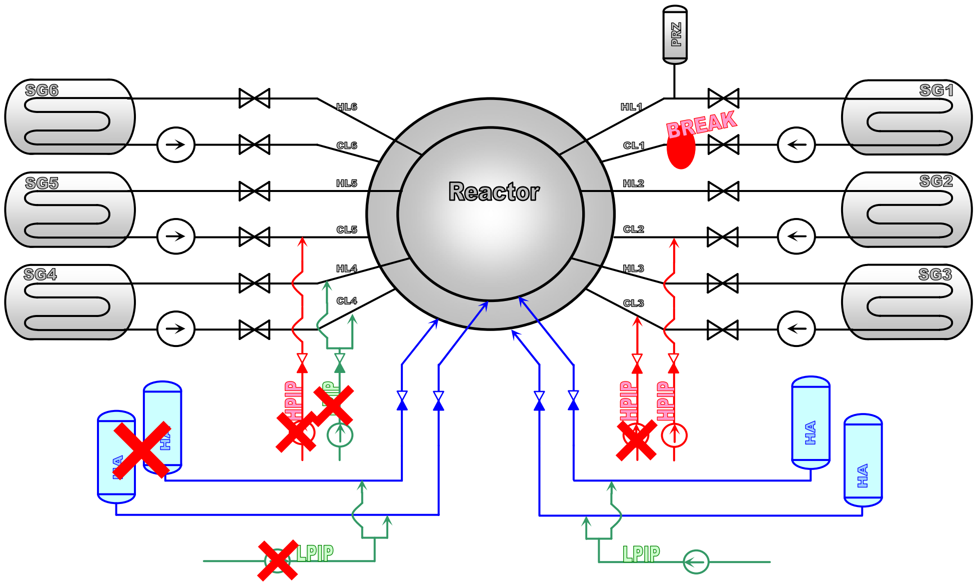

Figure 2: Break location

➢ **Data transformation from thermo-hydraulic analysis and input preparation for thermo-mechanical analysis**

The transformation of the data from thermo-hydraulic calculation was performed by newly developed tool (code). It transfers selected parameters as a function of time directly from RELAP5 output to TRANSURANUS input file (Macro Time Step Part). Data for transformation are: axial power for each axial element of the relevant fuel rod; outer cladding temperature for each axial element of the relevant fuel rod; and coolant pressure at the core inlet. Temperature is recalculated from K to °C, pressure is recalculated from Pa to MPa, power of analysed fuel rod is recalculated to W/mm with appropriate axial power distribution.

The values are written into the required format of the TRANURANUS input file. Also the data step is synchronized. For the analysed LOCA accident the important transient part of thermo-hydraulic calculation is 450 s. The thermo-mechanical calculation contains initial steady state calculation on the nominal power up to the required burn-up of the analysed fuel rod (see Table 1), following by the LOCA transient simulation, defined by the boundary conditions. The initial steady state section of the macro time step is generated based on on-line data supplied by the user, and the time specification of the RELAP5 outputs is modified to reflect the initial section.

➢ **Statistical thermo-mechanical analysis using TRANSURANUS code**

The initial conditions (power and burn-up) for statistical calculation are defined in the Table 1. The number of statistical runs was optimized to balance stabilisation of the results and hardware possibility. For this case the 2000 individual statistical runs were proved to be fully satisfactory.

Results of the calculation showed that the fuel cladding failure is predicted only for the groups No. 4 and No. 5 (see Table 2), characterized by the maximal power (55 kW) and the axial power profile including maximal linear heat rate 32.5 kW/m in the ninth node. The probability of cladding failure for calculations number 4 and 5 is 24.2 % and 27.4 % respectively. This represents less than 148 failed fuel rods.

Table 2: Quantification of failed fuel rods

Calculation number	Analysed range of relative fuel rod power in the core	Power of the rod used for calculation [kW]	Time of base irradiation [year]	Number of fuel rods [-]	Max. Burn-up [MWD/kgU]	Probability of cladding failure [%]	Number of failed rods
1.	(1.4 ; max>	55.000	5	0	0.00	0	0
2.	(1.4 ; max>	55.000	4	0	0.00	0	0
3.	(1.4 ; max>	55.000	3	0	0.00	0	0
4.	(1.4 ; max>	55.000	2	42	27.34	24.2	11
5.	(1.4 ; max>	55.000	1	498	13,82	27.4	137
6.	(1.2 ; 1.4>	43.776	5	0	0.00	0	0
7.	(1.2 ; 1.4>	43.776	4	0	0.00	0	0
8.	(1.2 ; 1.4>	43.776	3	30	38.64	0	0
9.	(1.2 ; 1.4>	43.776	2	7080	27.79	0	0
10.	(1.2 ; 1.4>	43.776	1	5256	13.82	0	0
11.	(1.0 ; 1.2>	37.522	5	84	54.16	0	0
12.	(1.0 ; 1.2>	37.522	4	1014	47.89	0	0
13.	(1.0 ; 1.2>	37.522	3	6114	38.83	0	0
14.	(1.0 ; 1.2>	37.522	2	2706	25.28	0	0
15.	(1.0 ; 1.2>	37.522	1	2790	12.68	0	0
16.	(0.0 ; 1.0>	31.268	5	6846	56.71	0	0
17.	(0.0 ; 1.0>	31.268	4	8058	46.97	0	0
18.	(0.0 ; 1.0>	31.268	3	2172	35.67	0	0
19.	(0.0 ; 1.0>	31.268	2	0	0.00	0	0
20.	(0.0 ; 1.0>	31.268	1	1284	12.18	0	0
Sum							**148**

4. Conclusions

Quantification of the fuel rod cladding failure due to the postulated accidents at nuclear power units is an important step in quantification of the consequences in release of radioactivity from fuel rod into primary circuit. It needs to be performed using a complex safety evaluation of the nuclear power plant, especially for the licensing purposes. Progress and development of tools allows enhancement of commonly used methodologies, to reach higher credibility of the predictions.

As a result of EURATOM 5th Framework Programme EXTRA Project, the TRANSURANUS code was extended and validated for VVER440 fuel rod failure prediction under LOCA conditions. This feature of the code enables it to be included into the methodology used for VVER440 units. The methodology, briefly described in the paper, relies on code RELAP5 as a widely used and outstandingly validated code for thermohydraulic analyses of the PWR accidents. This code’s role in the methodology is in proper specification of the unit response to the accident and in definition of the initial and boundary conditions for the coupled thermo-mechanical calculation.

TRANSURANUS code providing its thermo-mechanical algorithms, failure criterion and statistical feature (Monte Carlo) applied to the outer cladding temperature proves to be an effective and appropriate tool to quantify extent of fuel rod failure in the VVER440 core during LOCA accident.

The methodology applied for selected LOCA scenarios and compared with the original, several years old methodology, generally used for safety licensing evaluation of VVER units. It has proved its applicability, decrease of unnecessary conservativeness in predictions, minimising of user effect and better user friendliness.

As an example of the methodology application, the maximum design LOCA for VVER440/V213 reactor is presented in the paper, analyzed according to conservative licensing approach. It shows that even with conservative assumptions, the cladding failure will be very limited.

The development outlined in the paper clearly shows that results of highly specialized activities on validation of the codes can result in decrease on conservativeness of predictions, finally to be beneficial at licensing activities connected with reactor power uprates, changes in fuel schemes and types and other development to reach safer and more efficient operation of the existing units.

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