Study on the DLOFC accident of the GEMINI+ conceptual design of HTGR reactor with MELCOR and SPECTRA

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Abstract. The work presented in this paper was performed within the Euratom Horizon 2020 GEMINI Plus project. Behavior of the HTGR reactor under severe accident conditions was investigated and the maximum fuel temperature was observed. Due to application of the TRISO particles and SiC layers in the fuel element, no damage of the fuel is expected up to 1600°C. Under the cooperation in the project between Nuclear Research Group (NRG) and National Centre for Nuclear Research (NCBJ) a code-to-code calculations were carried out between the SPECTRA and MELCOR codes. SPECTRA code, developed by the NRG is a thermal hydraulic analysis code and MELCOR 2.1.6342 used by NCBJ developed by SANDIA National Laboratory is fast running severe accident code. Both codes have already HTGR specific models build in. The following accident was analyzed and will be presented: Depressurized Loss of Forced Circulation (DLOFC) with 65 mm break at the top of reactor vessel. The scenario was calculated applying following sets of assumptions: best estimate and conservative. Plant behavior was analyzed including primary and secondary side of the reactor. As the results of applying conservative assumptions, it was found that fuel temperature excides the acceptable limit of 1620°C. Therefore, changes in the core design were proposed by project participants. Analyses of the new core showed acceptable temperatures. In the paper the results of code-to-code comparison are presented. Both codes have shown a good agreement of presented following characteristics on maximum fuel temperature, relative power and Reactor Cavity Cooling System power, primary pressure and break flow.

1. Introduction

The work presented in this paper was performed within the GEMINI Plus project, WP1, Task 1.6: Transient Analyses. This work provides a description of analyses of selected accident scenarios for an innovative reactor design, which is considered for the Polish industry utilizing process heat in various sectors of production. Two following accidents were analyzed: Depressurized Loss of Forced Circulation (DLOFC) with 65 mm break in the primary system.

Due to the nature of the project and the investigatory character of the performed analyses presented here, both scenarios were calculated applying two sets of conditions Best Estimate (BE) and Conservative (C) and considering uniform radial peaking factor (flat).

2. HTR-GEMINI

HTR reactor specified for Polish end-users is designed as reactor which produces 251 t/h of live steam (138 bar, 540°C). Thermal power that was estimated as the most suitable for the end user corresponds to 165 MWth and 15MWth was dedicated for internal power needs.
During the project few designs were taken under investigation and the final decision was done on the basis of the extensive calculation campaign. The main interest was to keep the maximum temperature under limit of 1600°C, which in the literature is considered the fuel integrity limit for short period lasting temperature spikes and 1620°C in modern small HTRs [1].

Developed GEMINI+ HTR core (presented in the Figure 2) consists of prismatic graphite blocks, 800 mm height and 360 mm across the flat cross section. Two types of the blocks can be distinguished: one fully fueled and second one with a large channel serving as a duct for the absorber rods Figure 2 and Figure 3 respectively.

Fuel compact is produced with TRISO particles, pressed together in the graphite matrix into a small cylinder, with a diameter of 12.5 mm and a height of 50 mm. There are approximately 3760 particles in a compact which results with the packing fraction of around 0.25. For TRISO particle, the UO$_2$ fuel were used with an enrichment of 14%. Each fuel hole is filled by fuel pin which consist of 15 compacts and graphite plugs are cemented into the tops of the fuel hole enclosing the fuel compact into stacks.

The version of design, which is analyzed is called Configuration AY [2], with the active core height of 8.8 m. This translates to 11 axially prismatic blocks to create an active core region. There are two reflectors, on the top and bottom of the core, situated with dimensions of 1.2 and 1.6 m respectively. Active core is surrounded by the permanent reflector. Details are being presented in Figure 4 below.
Figure 3. Block with control rod hole.

Figure 4. Core configuration AY.

Table 1. Main parameter of the reactor for normal operating conditions.

| Parameter                          | Value                  |
|------------------------------------|------------------------|
| Thermal power                      | 180 MW                 |
| Core inlet temperature             | 325°C/598 K            |
| Core outlet temperature            | 750°C/1023 K           |
| Primary pressure                   | 6 MPa                  |
| Number of blocks in column         | 11                     |
| Number of fully fuel loaded blocks in the core | 275          |
| Number of fuel blocks with control rods in the core | 66                  |
| Number of reflector blocks with control rods in the core | 198*        |
| Number of reflector blocks in the core | 396*                  |
| Number of reflector blocks- upper  | 255                    |
| Number of reflector blocks- lower  | 340                    |
| Gap between blocks:                | 2.0 mm                 |
| Secondary steam pressure           | 13.8 MPa               |
| Feedwater inlet                   | 200°C/473 K            |
| Steam outlet temperature           | 540°C/813 K            |

*Only core region, without lower and upper permanent reflector
Figure 5 shows the main parameters and scheme of the GEMINI+ plant. The scheme presents the concept of the steam production line that would be used by the end-users of the reactor.

![Steam production line scheme](image)

**Figure 5.** Reactor reference parameters and scheme.

3. Code introduction

3.1. MELCOR

MELCOR [3] is a fully integrated, engineering-level computer code developed by Sandia National Laboratories for the U.S. Nuclear Regulatory Commission, that models the progression of severe accidents in nuclear power plants.

The structure of the code is package-based organization, which allows the user to decide upon the use of case specific physics to be modelled. This structure of the code enables the calculations to be more flexible and less time consuming in instances of extensive sizes of input.

A broad spectrum of severe accident phenomena, in both boiling and pressurized water reactors, is treated in MELCOR in a unified framework. The most important one is the ability to model thermal-hydraulic response in the reactor coolant system [3], which is the main focus for the application of the HTR in the GEMINI+ project. In terms of specific HTR designs features the MELCOR framework is also adopted, there is possibility of using specific modules, allowing the modelling of prismatic and pebble bed HTR reactors. The most interesting one is the model of the Radial Conduction in the HTGR, which involves the Radial Effective Conductivity of Graphite Blocks with the Tanaka and Chisaka expression for a continuous solid system [3]. This expression does take into account the discontinuous character of the domain (presence of the pores), which are affecting the conductive heat transfer in the block greatly. The expressions are given below:

\[ k_{eff} = k_s \left\{ A + (1 - A) \cdot \frac{\ln \left[ 1 + 2 \cdot B \cdot \left(k_{por} / k_s - 1\right) \right]}{2 \cdot B \cdot \left(1 - k_s / k_{por}\right)} \right\} \]  

\[ k_{rad} = 4 \cdot \varepsilon_r \cdot \sigma \cdot T^3 \cdot D \]  

\[ k_{er} = \frac{1}{\frac{h_{gap} \cdot D_{blk}}{k_{blk}} + \frac{1}{k_{blk}}} \]  

\[ h_{gap} = \left( k_{He} + 4 \cdot \varepsilon_r \cdot \sigma \cdot T^3 \cdot \sigma_{gap} \right) \]
\[ A = 2 \frac{(1 - \varepsilon)}{(2 + \varepsilon)}, \]
\[ B = \frac{(1 - \varepsilon)}{3}, \]
\[ \varepsilon = \text{porosity}, \]
\[ k_s = \text{thermal conductivity of solid material} \ [\text{W/m-K}], \]
\[ k_{por} = \text{thermal conductivity of pores} \ [\text{W/m-K}], \]
\[ k_{rad} = \text{radiative conductivity} \ [\text{W/m-K}], \]
\[ \varepsilon_r = \text{emissivity in pores (channels walls)}, \]
\[ \sigma = \text{Stefan-Boltzmann constant} \ [\text{W/m2-K4}], \]
\[ D = \text{effective diameter of pores} \ [\text{m}], \]
\[ k_{er} = \text{effective radial block conductivity} \ [\text{W/m-K}], \]
\[ h_{gap} = \text{gap heat transfer coefficient} \ [\text{W/m2-K}], \]
\[ D_{blk} = \text{effective radial diameter of a block} \ [\text{m}]. \]

3.2. SPECTRA
The SPECTRA code developed at NRG, the Netherlands, is a computer program designed for thermal-hydraulic analyses of nuclear or conventional power plants [4]. The code’s main applicability is in the area of Light Water Reactors, High Temperature Reactors, Liquid Metal Fast Reactors, and Molten Salt Reactors. The code can be used for thermal accident scenarios, involving loss-of-coolant accidents (LOCAs), operational transients, and other accident scenarios in nuclear power plants. The code structure is based on packages that contain models on a given topic. The available models include multidimensional two-phase flow, non-equilibrium thermo-dynamics, transient heat conduction in solid structures, and general heat and mass transfer, with natural and forced convection, condensation, boiling. SPECTRA was applicable in past to many Generation IV reactors and analyses:

- PBMR: thermal-hydraulics (TH), dust and fission product analyses, Reactor Cavity Cooling System (RCCS).
- NGNP: (pebble) TH dust and fission product analyses.
- HTR-PM: safety analyses, including [5]:
  - DLOFC (comparison with the TINTE code)
  - PLOFC (comparison with the TINTE code)
  - water ingress.

4. Calculation model and scenario description
This section will firstly focus on the calculation models developed under the GEMINI+ project, with the use of two described codes in the Section III – MELCOR and SPECTRA. It will be followed by the description of the scenario, which was performed and is the interests of this comparative study. This scenario will be the DLOFC accident, which is crucial from safety point of view for HTR reactors.

4.1. Calculation model
For the purpose of the proper investigation of the proposed design and elimination of the user effect (due to the choice of nodalization), the model for each code was prepared using the same nodalization scheme [6], [7]. The schemes are presented in the following figures: Figure 6 – Figure 9.
Figure 6. Reactor pressure vessel -SPECTRA model. Figure 7. Reactor pressure vessel – MELCOR model.

The description on the model on the example of the MELCOR code model is given in the following way:

- Reactor Pressure Vessel was divided into 34 axial and 6 radial nodes, see Figure 7.
- First two bottom levels are dedicated to the lower plenum, next three are corresponding to supporting structures zone.
- Sixth to ninth levels are the lower reflector.
- Tenth to thirty-one level is the active core region and last four are upper reflector.
- In axial levels 1-9 and 32-34 the radial meshes are the same for all rings (graphite material for the reflector).
- In 10-31 the two zones can be recognized 1-4 the active core region and 5-6 surrounding reflector.

Outer volume of the vessel represents a riser and heat structures of the vessel wall (modeled only by CVH package in MELCOR). Exterior of the vessel - reactor cavity is also modelled with the RCCS, what is important for the long-term lasting heat exchange during accident scenarios. The heat exchange at the cooling side of the RCCS was decided to be modelled with heat exchange coefficient $h=1000$ W/m$^2$-K and temperature of $T=373$ K (100 °C). Detailed scheme of the RCCS is presented in Figure 8.

Figure 9 presents steam generator (SG) nodalization, where primary and secondary side are modelled. Primary side is modelled by inlet plenum of the hot helium, SG-tubes bundle region, lower plenum, riser with upper plenum and blower, and outlet plenum. Flow of the helium is forced by helium blower located on the top of SG. Secondary side (water) consists of feedwater inlet, pump and isolation valve; inlet header, helical tubes, outlet header, steam line including the isolation valve, as well as the SG drain system, which is used in case of SG tube rupture to minimize the water ingress into the primary system by draining the SG water. The 2200 SG tubes walls are represented by the heat structures.
The SCRAM signals modelling is an important part of the logic in the developed model. During propagation from the steady state (SS) to transient (T) situations the following SCRAM signals are used in the model:

- \( \frac{dP}{dt} < -1000.0 \) (Pa/s) - Fast primary pressure drop.
- \( T > 1073 \text{ K (800°C)} \) - High gas exit temperature.
- \( \frac{W(1)}{W(2)} < 0.7 \) - Low ratio of primary mass flow to secondary mass flow.

**Figure 8.** Reactor pressure vessel – MELCOR model.

**Figure 9.** Steam Generator MELCOR and SPECTRA models.
4.2. Scenario description

In this section the transient scenario will be described. The following accident was analyzed:

Depressurized Loss of Forced Circulation (DLOFC), 65 mm break in the primary system.

The scenario was calculated for radially uniform (flat) power profile applying following sets of assumptions:

- Best Estimate (BE) and
- Conservative (C)

The following assumptions were made for the Best Estimate runs:

- Core power: 180 MW (100%)
- RV inlet gas temperature 325°C
- RV outlet gas temperature 750°C
- RV/RCCS wall emissivity 0.9
- RV/RCCS natural convection default (best estimate) models

Based on [8], the following assumptions were made for the conservative runs:

- Core power: 189 MW (105%)
- RV inlet gas temperature 355°C (+30 K)
- RV outlet gas temperature 800°C (+50 K)
- RV/RCCS wall emissivity 0.54
- RV/RCCS natural convection multiplier of 0.7 (–30%)
- Conservative decay heat curve: approximately 10% higher

Both DLOFC scenario is detected by the protective system due to the signal of “low ratio of primary mass flow to secondary mass flow”. After that the blower and feedwater pump are tripped and control rods are dropped.

5. Calculation results

In this section the results of the comparative work will be summarized for performed scenario and both codes.

5.1. DLOFC scenario

DLOFC accident is a typical Design Basis Accident (DBA) and one of the most serious for the reactor safety barriers to withstand. It receives high attention, because it results in the higher maximum fuel temperature in comparison to other DBAs. With such large 65 mm break, the reactor will depressurize quickly (as shown in Figure 10), causing coolant medium to escape the reactor coolant loop and decrease the ability of cooling heated structures of the reactor core.

Within 500 seconds the system is almost depressurized and the flow through the break is almost zero Figure 11. For both scenarios and codes, the results are very similar and the progression of the accident happens with a good agreement. The two assumptions (BE and C) show the same behavior of the system depressurization and the break mass flow characteristics.

Reactor power after SCRAM depends on the decay heat curves, which are based on data from NRG, obtained using 11-group calculations [7]. Those calculations were the effect of neutronic calculations based on the SERPENT code simulations (done by GEMINI+ project partner), which enabled to obtain the neutron precursors group constants. Later on, the constants were used to update the decay heat data available from [10] by group constants modified in SPECTRA.
After reactor shutdown, the decay heat is transferred from the core to the RPV via heat conduction, convection and radiation. In terms of the HTR cores the dominating mechanism of heat transfer is conduction and in comparison, to the water reactors the solid structures in the core are responsible for heat extraction from the core. The graphite blocks are heated up by the fuel pellets and later on they are transferring the heat through the helium gaps to the outer walls surrounding the core. Thanks to the RCCS the reactor vessel is cooled. Due to high heat capacity of the graphite, the core heat relatively slowly and the maximum fuel temperature is reached around 20-30h of simulation Figure 13.
Figure 12. DLOFC relative power - long term.

Figure 13. DLOFC maximum fuel temperatures.

Depending on the code and assumptions the maximum fuel temperature is predicted around 1450°C and 1600°C for best estimate and conservative assumptions respectively. Detailed data with the results for DLOFC scenario is presented in summarizing Table 2.
As it can be seen in the Figure 14 – Figure 15 no exceedance of the temperature limit for the both codes SPECTRA and MELCOR was reached for the case of the DLOFC scenario with best estimate assumptions. The hottest areas are determined to be in the center of the core. The following peak values are observed in MELCOR calculations:

- **BE:** 1807 K (1534°C) at 100,000 s,
- **C:** 1873 K (1600°C) at 98,600 s.

The following peak values are observed in SPECTRA calculations:

- **BE:** 1718 K (1445°C) at 80,000 s,
- **C:** 1850 K (1577°C) at 72,000 s.
The highest temperature is observed FA 1 (ring 1) - Figure 14 and Figure 15, which represents a single fuel assembly, 3.2% of the whole core fuel mass.

Figure 16. Steam Generator MELCOR and SPECTRA models.

The differences in the maximum fuel temperatures are 90 K in the BE case and 23 K in the conservative case. Investigation of the code modeling and input assumptions showed the following main differences:

- Use of separate effective conductivity correlations in fuel and reflector in SPECTRA. Sensitivity calculations showed that this gives roughly 40 K difference in the maximum fuel temperatures [9].
- Different correlations for natural convection, as well as different characteristic dimensions that are being used with these correlations. Consequently, clearly larger heat transfer to RCCS in SPECTRA (Figure 17) than in MELCOR (Figure 18). The differences in default natural convection correlation were pointed out in [7]. Differences in the characteristic dimensions are a consequence of using a single value in MELCOR and separate values for natural and forced convection in SPECTRA. The hydraulic diameter is typically applied for forced convection; surface height is applied for the natural convection. Due to those differences the convective heat transfer coefficients observed on the RV and RCCS surface were approximately twice larger in SPECTRA than in MELCOR. A sensitivity calculation was performed [9], where a multiplier of 0.5 was applied on the HTC correlations to get approximately twice smaller convective HTC on the RCCS and RV. The resulting maximum fuel temperatures were:
  - BE: 1733 K (1460°C),
  - C: 1866 K (1593°C).

In this case the difference compared to MELCOR is 74 K in the BE case and 7 K in the conservative case.
• Other differences include: in MELCOR the core is composed of 4 rings, in SPECTRA there are 5 different assembly types, which come from 1/12 core symmetry), and
• small differences in core component masses, which are a consequence of different input (SPECTRA input model is based on geometrical dimensions; MELCOR model is based on component masses).

It is considered that for the current conceptual state of the reactor design, the agreement is satisfactory. Some changes of the design were recently implemented that are not even fully implemented in both models, e.g., the gas riser channel, discussed in [9]. This will have to remain as eventual future work. For now, the agreement of 6 K on the maximum fuel temperature in the most conservative case is considered as very good.

Figure 17. DLOFC/PLOFC with BE/C SPECTRA power extracted by RCCS.

Figure 18. DLOFC with BE/C MELCOR power extracted by RCCS.

6. Summary of results
In DLOFC scenario calculated maximum fuel temperatures do not exceed the acceptable level of 1600°C and it is far from limit 1620°C specified in [1]. Any fraction of the fuel is not overheated and the design fulfil safety criteria for operation specified in [2]. Safety margin varies between assumptions from 20 to 296 degrees for MELCOR code and 57 to 452 degrees for SPECTRA code. Some result differences will need more model investigations and further results will be published.

Table 2. Summary of DLOFC results

| Assumption | BE       | C       |
|------------|----------|---------|
| Code       | MELCOR   | SPECTRA | MELCOR   | SPECTRA   |
| Thermal power | 180 MW   |         | 189 MW   |           |
| Maximum fuel | 1807K 1534°C | 1718K 1445°C | 1873K 1600°C | 1850K 1577°C |
7. Results discussion
In the two models, the peak values of the maximal fuel temperature are below the HTR TRISO fuel element temperature limitation 1620°C [1]. For MELCOR conservative calculations it is predicted as 1600°C and for SPECTRA 1577°C, which give the relatively small safety margin.

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