WHITE PAPER

Design Considerations on Reactor Core (Fuel) at Nominal Conditions

(Non-Proprietary)

Total Pages: 18

Document Number: LTR-TINN-25-012
Revision 0

(Enclosure 4)

2022-06-04

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Notes on revisions, source control and e-attachments:

Revision 0
First release.

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1 INTRODUCTION

SOLO reactor is a 1MWe Gas Cooled Reactor whose design leverages at most licensed and commercially available systems and components including the fuel. The employed fuel is the same that powers the Light Water Reactor fleet, including notably cladding material and enrichment (< 5%). **[**

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1.1 PURPOSE

The purpose of the present white paper is to describe how standard PWR fuel is integrated in the design of SOLO and under which conditions is operated.

1.2 SCOPE

The scope of the present white paper is to inform the NRC about the technical details that pertain to the SOLO fuel working conditions, the adopted design solutions and final to solicit NRC opinion on the proposed solutions. Two nominal conditions are considered in the analysis presented in the following pages, identified as “Baseline” and “Stretched” nominal conditions that are essentially characterized by different cladding temperatures.

1.3 APPLICABLE REGULATIONS AND REGULATORY GUIDANCE

Regulatory Doc	**Title**
NUREG-1537, Part 1 and Part 2	Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors
DANU-ISG-2022-01	Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap, Interim Staff Guidance, March 2024

1.4 REQUEST FOR NRC

As introduced above and detailed in the following sections, the fuel that powers SOLO is exactly the same operated in PWRs. It is described that the general environment is different from LWR, being much less aggressive as the coolant is helium and not water. Overall the operating conditions are less demanding than that of LWR technology apart for the cladding temperature which is instead slightly higher.
Because of the large operational experience, a not aggressive environment, and reduced mechanical loads we would like USNRC to pose the attention, and then receive feedback, on the following aspects:

- On the use of LWR fuel type in SOLO considering Baseline- and Stretched- Nominal conditions.
- On the possibility to rely on the LWR fuel type qualification studies.
- On the justifications provided to demonstrate the applicability of LWR fuel products pre-existing licensing basis for SOLO implementation.

2 REPORT STRUCTURE

The present white paper is organized in six main Chapters. The first chapter provides an introduction of the SOLO design and includes the scope, and purpose of the review; Chapter 2 deals with the description of the paper structure while Chapter 3 provides with information of key characteristics of the core.
Chapter 4 describes the adopted code and methods while Chapter 5 presents the obtained results and the associated analysis. Finally Chapter 6 drives some conclusions.

3 SOLO REACTOR CORE

The SOLO reactor core includes a solid moderator and reflector made of graphite. **[[**

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The Pressure Coolant Tubes are located inside the graphite beside the fuel rods. These tubes are made of Zry and allow the He, selected as coolant, to flow (from bottom to top) and remove the heat generated by the nuclear fission inside the fuel rods, **[[**

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The entire core is surrounded by a stainless-steel cylinder, the Integrated Radiological Containment (IRC), which is filled by He **[[** **]]**.

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3.1 SOLO REACTOR CORE KEY FEATURES

The active part of the SOLO core is constituted by **[[**

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In addition, the linear heat rate (LHR) is a fraction of that of PWR: the maximum predicted value by a 3D whole core Montecarlo [1] code simulation is about **[[** **]]**.

4 ADOPTED CODE AND METHODS

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The rod simulated in the presented analysis is a long fuel rod located in the center of the core and subject to the highest solicitations. Namely, both the maximum cladding temperature and the maximum linear rate are assumed as boundary conditions and kept for the entire simulation period with the aim to provide the bounding responses for the whole core. Moreover, while the SOLO fuel cycle lasts 15 years (EOL), the simulation time has been extended to 16 years to detect possible issues in longer term.

Two main simulations are discussed in this document pertaining to the Baseline- and to the Stretched-Nominal conditions, characterized by different cladding temperatures.

4.1 FUEL ROD SIMULATION MODEL

The fuel rod design considered in the present analysis is the one adopted in the US-EPR described in the Chapter 4.2 of the Safety Analysis Report [6].

The fuel performance (FP) model has been built consistently with the associated boundary conditions (see next section). It is composed of six axial slices and an upper plenum to account for the free space available on the top of the fuel rod (the presence of the spring in contact with the fuel pellet stack is taken into account). The lower plenum has not been taken into account to maximize the predicted Rod Internal Pressure (RIP).

The computational domain is radially subdivided into 5 coarse zones for the fuel region and 1 coarse zone for the cladding region. Each coarse zone is further subdivided into 10 mesh points.

Standard models applicable to PWR design are adopted, e.g. the UO_2 conductivity, the fuel swelling, the clad mechanical properties.

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4.2 ADOPTED BOUDARY CONDITIONS

The fuel performance code requires a set of proper boundary conditions (BC) to be imposed in each analysis, namely: LHR, neutron flux, cladding temperature and external pressure.

The LHR and the neutron fast flux are calculated by means of a 3D detailed whole core Montecarlo [1] code simulation from which an axial distribution (at six positions) is derived without making any optimization or improvement.

The cladding temperature has been calculated by a detailed Computational Fluid Dynamic (CFD) simulation model [7] in which the power distribution predicted by the Montecarlo simulation is imposed, while an adiabatic boundary condition is assumed at the external surface of the graphite to maximize the cladding temperature (because of the significant graphite heat conduction about 5-7% of the reactor thermal power is expected to be dispersed from the moderator outer surface). To further bring the analysis toward bounding case, the cladding temperature imposed in the FP calculation conservatively corresponds to the maximum value calculated by the CFD simulation in the length intervals covered by each axial slice of the FP model.

Figure 2 provides with the adopted boundary conditions in relation to the axial distribution of the LHR and of the cladding temperature for the Baseline Nominal Conditions. The maximum values are [[ ]] kW/m imposed at the slice 3 and [[ ]] K imposed at the slice 6. Additionally, a neutron fast flux distribution with an average value of [[ ]] n/cm^2 s and an external pressure of [[ ]] MPa are imposed. All the boundary conditions are kept constant for the entire simulation period (16 years).

Figure 3 provides with the adopted boundary conditions in relation to the axial distribution of the LHR and of the cladding temperature for the Stretched Nominal Conditions. The maximum values are [[ ]] kW/m imposed at the slice 3 and [[ ]] K imposed at the slice 6. In fact, the cladding temperature distribution is shifted up of 100 K to account for a similar He coolant temperature increase at the core inlet aimed at improving the efficiency of the power conversion system. All the other boundary conditions are exactly the same of the Baseline case.

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5 ANALYIS OF THE RESULTS

The results reported below are provided for two axial positions: the maximum linear heat rate position (slice #3) and the maximum cladding position (slice #6). The following Figures of Merit (FoM) are chosen to illustrate the fuel rod performance for the entire fuel cycle plus one year (EOL +1 yr):

- Cladding and Fuel Temperature.
- Cladding creep and strain.
- Cladding stress.
- Gap Width.
- Rod Internal Pressure.
- Rod Radial Temperature distribution at EOL+1yr.

5.1 BASELINE NOMINAL CONDITIONS

Figure 4 illustrates the cladding and fuel temperature evolution over the irradiation cycle at the two selected axial positions. **[**

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Figure 5 provides with the clad creep and strain predictions. The cladding does not suffer of any plastic deformation for the entire simulation period, while the strain is always less than [[ ]] considering thermal and irradiation contributions, with the maximum value associated with the maximum clad temperature (slice #6). **[[**

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The Cladding stresses are shown in Figure 6 for the two selected axial positions. The mechanical loads are very limited and always in compression (because the external pressure is higher than the rod internal pressure, see Figure 7 right side), showing ample margin between the equivalent stress and the cladding yield stress (greater than 100 MPa at the considered temperature for un-irradiated material).**[[**

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The gap (Figure 7 left side) is always open throughout the simulated irradiation time (EOL+1 year). The initial increase is linked to the pellet shrinking at the beginning of the irradiation period which stops at a certain burnup level reached in different moment along the fuel rod due to the axial variation of the imposed LHR. When the fuel starts to increase its volume then the gap width linearly decreases.

The rod internal pressure (Figure 7 right side) follows the gap width evolution though with opposite trend. Namely, when the gap increases the RIP decreases and vice versa. However the RIP is always less than the external pressure **[[**

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The radial temperature profile at the end of the simulation period is given in Figure 8 for the max LHR position (slice #3) and max Cladding Temperature position (slice #6). No critical values are reached, while the profile smoothly decreases from the rod center to the outer surface. It should be noted the quite low values in respect to the typical PWR fuel temperature profile. The gap opening can be appreciated by the temperature jump in correspondence of its position.

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5.2 STRETCHED NOMINAL CONDITIONS

Figure 9 illustrates the cladding and fuel temperature evolution over the irradiation cycle at the two selected axial positions. The trend is very flat, with a maximum fuel temperature below [[ ]] K pertaining to the top of the rod (driven by the highest imposed clad temperature). [[

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Figure 10 provides with the clad creep and strain predictions. The cladding does not suffer of any plastic deformation for the entire simulation, while the strain is always less than [[ ]] considering thermal and irradiation contributions, with the maximum value associated with the maximum cladding temperature (slice #6). [[

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The Cladding stresses are shown in Figure 11 for the two selected axial positions. The mechanical loads are very limited and always in compression (because the external pressure is higher than the rod internal pressure, see Figure 12 right side showing ample margin between the equivalent stress and the cladding yield stress (greater than 50 MPa at the considered temperature for un-irradiated material). [[

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The gap (Figure 12 left side) is always open throughout the simulated irradiation time (EOL+1 year). The initial increase is linked to the pellet shrinking at the beginning of the irradiation period which stops at a certain burnup level reached in different moment along the fuel rod due to the axial variation on the imposed LHR. When the fuel starts to increase its volume then the gap width linearly decreases.

The rod internal pressure (Figure 12 right side) follows the gap width evolution though with opposite trend. Namely, when the gap increases the RIP decreases and vice versa. However the RIP is always less than the external pressure [[ ]].

The radial temperature profile at the end of the simulation period is given in Figure 13 for the max LHR position (slice #3) and max Cladding Temperature position (slice #6). No critical values are reached, while the profile smoothly decreases from the rod center to the outer surface. It should be noted the quite low values in respect to the typical PWR fuel temperature profile. The gap opening can be appreciated by the temperature jump in correspondence of its position. [[

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5.3 COMPARISON BETWEEN BASELINE AND STRETCHED NOMINAL CONDITIONS

Selected time trends are provided in this section to allow a direct comparison between the two investigated nominal conditions. [[

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The differentiator between the two nominal conditions is the different cladding temperature profile, being higher of 100K in the stretched nominal conditions. Such temperature shift can be observed in Figure 14, however none of the two conditions show unexpected values even extending the fuel cycle of 1 year.

A direct consequence of the higher cladding temperature in the Stretched nominal conditions is a higher cladding strain in respect to the Baseline, although a minor difference is predicted (Figure 15). Plastic deformation is not predicted in both conditions. [[

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The comparison between equivalent stress in Stretched and Baseline nominal conditions is provided in Figure 16. The lower mechanical loads in the Stretched condition case is due to the less pressure difference between the rod internal pressure and the external pressure, however in both cases any issue on the mechanical resistance of the cladding has been identified. [[

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To complement the above comparison, Table 5-1 provides SOLO Baseline and Stretched nominal conditions fuel related parameters together with corresponding typical PWR values. As it can be seen, the higher cladding temperature in SOLO is certainly counterbalanced by the much less demanding working conditions.

Table 5-1 – Comparison between SOLO Baseline and Stretched nominal conditions and Typical PWR data

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6 CONCLUSIONS

The white paper provides an overview of the fuel working conditions in SOLO design. The same fuel adopted in PWR fleet is selected to be operated in SOLO.

Apart the slightly higher cladding temperature, all parameters, such us outer pressure, burnup, oxide thickness, fission gas release, etc., have been predicted to be less than that of PWR typical configuration. Moreover, fretting, corrosion and erosion are excluded by design in SOLO.

Taking into consideration the obtained results, in which several conservative assumptions have been made in terms of BC (notably maximizing the cladding temperature, adopting a non-optimized linear heat rate axial distribution, assuming the maximum values are imposed on the same positions for the whole irradiation, extending the irradiation for one extra year), the following can be concluded:

- The fuel rods designed for PWR conditions can be operated in SOLO.
- The working conditions in SOLO are similar or enveloped by the PWR working conditions, thus the fuel qualification process pertaining to PWR design can be considered applicable to SOLO design.
- The Stretched nominal conditions do not pose particular challenging to the fuel rods, the mechanical responses (even after 1 year extra of the nominal fuel cycle length) are well within the elastic range and showing reasonable strain value.

7 REFERENCES

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8 APPENDIX 1

Material Compatibility

Considering the different material pairs for inner and outer cladding, i.e. UO2- Zry and C-Zry, a literature research has been made for the latter case, being very different from PWR application.

A study conducted at Idaho National Laboratory related to the Fuel-Clad Chemical Interaction Evaluation of the TREAT reactor has been analyzed [8]. Experiments were conducted keeping in tight contact a Zircalloy 4 sheets on a graphite support (see Figure 17 taken from [8]).

The test, performed at high temperature (820 °C) and under high vacuum conditions (6 mPa), showed that an interaction layer of few micron was formed. Furthermore the presence of Nb in the cladding alloy decreases the chemical interactions between Zr and C.

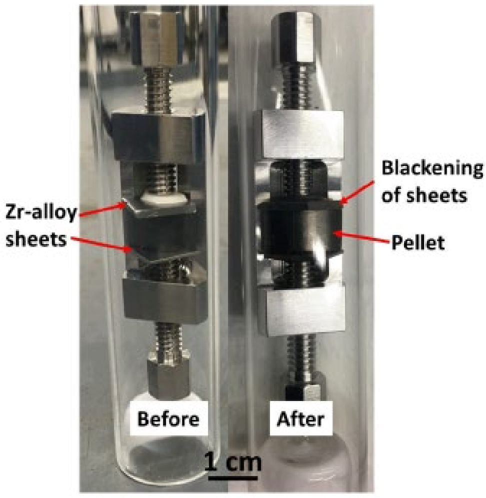

Figure 17 – Experimental set-up before and after test execution

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