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FY21 Status Report on the ART-GCR CMVB and CNWG International Collaborations

September 2021

ART - M3AT-21IN0603011

Vincent Laboure[1], Javier Ortensi[1], Andrew Hermosillo[2], Gerhard Strydom[1], and Paolo Balestra[1]

[1]*Idaho National Laboratory*
[2]*Texas A&M University*

Idaho National Laboratory

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[1]Idaho National Laboratory
[2]Texas A&M University

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Idaho National Laboratory
Advanced Reactor Technologies
Idaho Falls, Idaho 83415

http://www.art.inl.gov

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ART Program

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Technical Reviewer: (Confirmation of mathematical accuracy, and correctness of data and appropriateness of assumptions.)

Austin Carter	08/31/2021
Austin L. Carter Neutronics Analyst	Date

Approved by:

M. Davenport	*09/03/2021*
Michael E. Davenport ART Project Manager	Date
Travis Mitchell	9/3/2021
Travis R. Mitchell ART Program Manager	Date
Michelle Sharp	8/31/2021
Michelle T. Sharp INL Quality Assurance	Date

SUMMARY

The ART-GCR Methods Area includes an international collaboration work package that covers the tasks defined for the Computational Methods Validation and Benchmark (CMVB) and Civil Nuclear Energy Research and Development Working Group (CNWG) projects. This report summarizes the status of the fiscal year (FY)-21 tasks and planned FY-22 DOE contributions. The CMVB Project Arrangement (PA) is not yet formally approved by all signatories; therefore, no work has been performed at Idaho National Laboratory (INL) in FY-21 related to this activity. The CNWG activities in FY-21 consisted of the simulation of the High Temperature Test Reactor (HTTR) Loss Of Forced Cooling (LOFC) experiment with the INL codes Griffin, BISON, and RELAP-7 based on the Multiphysics Object-Oriented Simulation Environment (MOOSE). It was found that the multiphysics-coupled suite is capable of simulating all the important phenomena occurring during the LOFC experiments showing promising agreement with the measurements given the number of uncertainties and approximations introduced into the model.

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ACRONYMS

ANL	Argonne National Laboratory
ART	Advanced Reactor Technologies
ATWS	Anticipated Transients Without Scram
CFD	Computational Fluid Dynamics
CMVB	Computational Methods Validation and Benchmark
CNWG	Civil Nuclear Energy Research and Development Working Group
DOE	U.S. Department of Energy
GIF	Generation-IV Forum
HTGR	High Temperature Gas-Cooled Reactor
HTTR	High Temperature Test Reactor
INET	Institute of Nuclear and New Energy Technology of Tsinghua University
INL	Idaho National Laboratory
JAEA	Japan Atomic Energy Agency
JRC	Joint Research Centre
KAERI	Korea Atomic Energy Research Institute
LOFC	Loss Of Forced Cooling
MOOSE	Multiphysics Object-Oriented Simulation Environment
PA	Project Arrangement
PMB	Project Management Board
RPV	Reactor Pressure Vessel
SPH	super homogenization
U.S.	United States
VCS	Vessel Cooling System
WP	Work Package

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1. GENERATION-IV FORUM COMPUTATIONAL METHODS VALIDATION AND BENCHMARK PROJECT

The Generation-IV Forum (GIF) CMVB Project Management Board (PMB) focuses on ensuring the numerical models used for reactor system analysis are capable of calculating reactor system behavior during normal operational conditions, operational transients and accident scenarios. Generally, PMB members are performing numerical model and software development for use within their own national organizations and, except in very specific cases, are not likely to share development activities with other members.

The CMVB participants include the following signatories:

- Institute of Nuclear and New Energy Technology of Tsinghua University (INET) for China
- Korea Atomic Energy Research Institute (KAERI) for the Republic of Korea
- U.S. Department of Energy (DOE) for the United States (U.S.)
- Joint Research Centre (JRC) for EURATOM
- Japan Atomic Energy Agency (JAEA) for Japan

The CMVB focus areas includes:

- Identifying the key phenomena (i.e., performing phenomena identification and ranking studies)
- Identifying the data that may be available within the CMVB project member organizations to be used for validation
- Defining the standards that validation data sets must achieve before the data sets may be qualified for use in validation matrices
- Performing validation studies using data sets shared among CMVB members for that purpose.

The work of interest to the PMB in these areas is distributed within five Work Packages (WPs) as noted in Table 1. In general, Work Package 1 and 3 are being led by DOE for the U.S., while Work Packages 2, 4, and 5 are being led by INET for China. The CMVB PA is currently in the

Table 1: List of Work Packages and Lead Organizations for each Work Package.

WP No	WP Title	Lead
1	Phenomena Identification and Ranking Table (PIRT) Comparison, Evaluation, and Update	DOE (U.S.)
2	Computational Fluid Dynamics (CFD)	INET (CHINA)
3	Reactor Core Physics and Nuclear Data	DOE (U.S.)
4	Chemistry and Transport	INET (CHINA)
5	Reactor and Plant Dynamics	INET (CHINA)

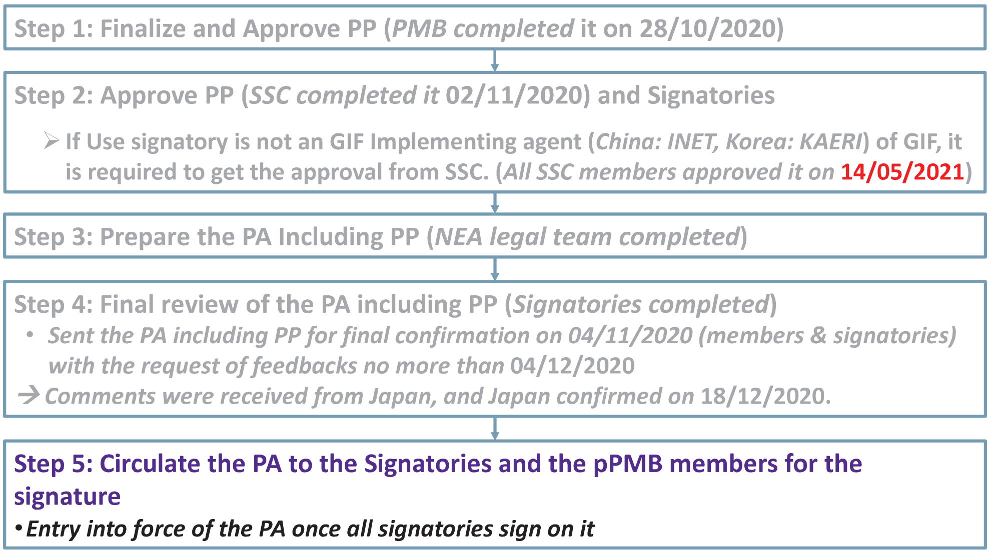


Figure 1: CMVB PA signature process and status.

process of being approved by the signatories for each participant, as listed in Table 1. At the end of August, the representatives of Japan, the Republic of Korea, and EURATOM signed the PA. The PA is currently in the final review process at DOE and the U.S. State Department; approval is not expected before Spring 2022. Two new U.S. members were nominated and confirmed by DOE in FY-21: Dr. Paolo Balestra (INL) as the CMVB Member and Dr. Rui Hu of Argonne National Laboratory (ANL) as the alternate member. The general process flow of the PA signature sequence is shown in Figure 1.

2. CIVIL NUCLEAR WORKING GROUP (CNWG) - HTTR LOFC MODELING

The High Temperature Test Reactor (HTTR) is a graphite-moderated, helium-cooled reactor originally developed by the Japan Atomic Energy Research Institute, now known as the JAEA. As a part of a cooperative effort between Japan and the U.S. under the CNWG, the Advanced Reactor Technologies (ART) Program at INL and JAEA are participating in a multi-national research project sponsored by the Nuclear Energy Agency of the Organization of Economic Cooperation and Development. Three LOFC tests are being performed at the HTTR to confirm the ability of the core to shut down and safely reject heat in the event of a circulator trip, without control rods being inserted. These events are classified as Anticipated Transients Without Scram (ATWS). The first of the experiments (LOFC#1) was completed on December 21, 2010, at 9 MW (30% of rated power) with data provided by JAEA to participating countries (including the U.S.) to be used for system code/model validation. The 30 MW LOFC#2 and the 9 MW with loss of the Vessel Cooling System (VCS) LOFC#3 experiments are currently scheduled to be completed in 2021 or early 2022, since JAEA received permission at the end of July, 2021, to restart the HTTR after the 2011 Fukushima accident. As a part of INL's contributions to the CNWG High Temperature Gas-Cooled Reactor (HTGR) activities in FY-21, the simulation of the LOFC transients were performed using the INL MOOSE-based [10] coupled codes Griffin [4, 14], BISON [15], and RELAP-7 [1] for validation of the multiphysics numerical models. These respectively simulate the neutronics, heat transfer, and thermal-hydraulics behavior of the reactor. The model also relied on Serpent [8] for cross-section generation [5] and the Griffin full-core super homogenization (SPH) technology [7, 9] to enable the use of a coarse mesh with a diffusion solver while maintaining a reasonable accuracy. The preliminary steady-state model developed in [6] was used as starting point for the transient simulations but numerous other improvements summarized in Section 2.1 were necessary to capture the main physical phenomena driving the transient behavior. Preliminary numerical results are then presented in Section 2.2, before conclusions and future work are discussed in Section 2.3.

2.1 HTTR Model Improvements

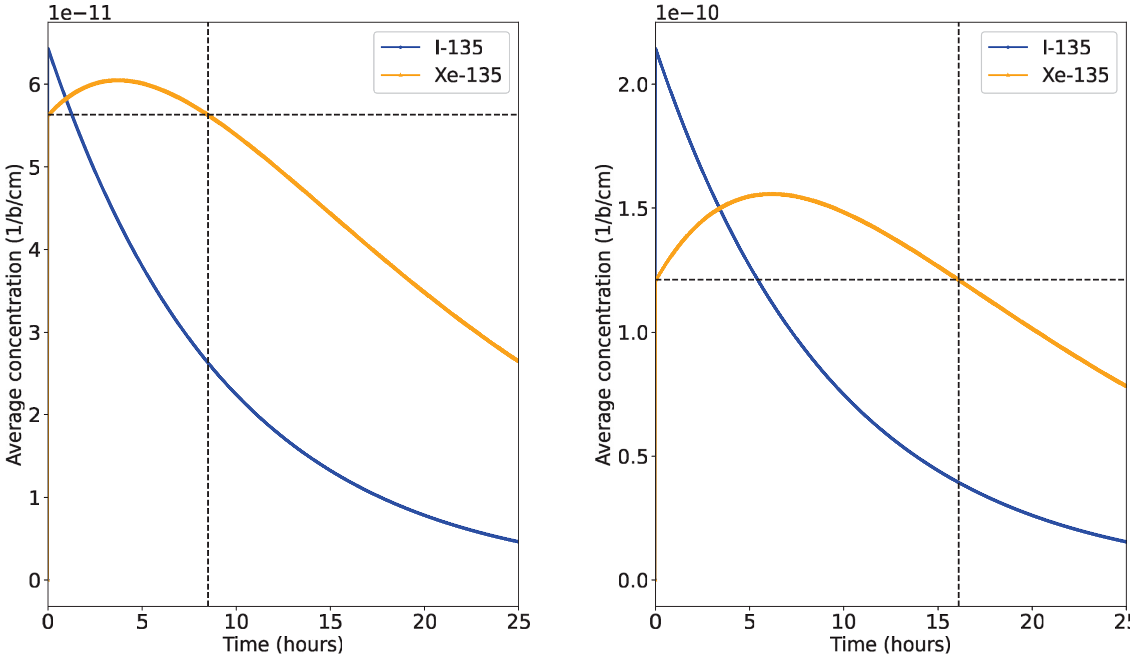
2.1.1 Neutron Source

Immediately following the initial loss of cooling, the reactor becomes sub-critical and stays in this state for many hours. If no fixed source is modeled in the reactor, the fluxes reach levels that are effectively zero (at least, numerically). When the reactor becomes super-critical again—through heat dissipation by the VCS and poison decay—the flux levels obtained would be completely unreliable and the associated uncertainty would be tremendous. Moreover, the intensity of the sources have a noticeable influence on the LOFC transient behavior [13]. HTTR contains three Cf-252 neutrons sources, which have a 2.645-year half-life and thus need to be replaced every seven years [12]. The last replacement before the 9 MW test (HTTR LOFC test run #1) seems[1] to have been conducted in 2006 [12], whereas the transient was performed in December 2010. The most recent replacement of the neutron source was conducted from July 6 to October 20 in 2015 [3], with the actual experiment scheduled for early 2022. Since overestimating the neutron sources makes the power peak after re-criticality earlier [13] and is considered to be a conservative assumption; the sources for the 9 and 30 MW transients are assumed to be 4 and 6 years old, respectively.

2.1.2 Decay Heat

Given the very low flux levels during the sub-critical phase of the transient, decay heat constitutes the primary heat source at that time and is thus very important to the model. For simplicity, the total decay heat curve as a function of time after shutdown is obtained from Serpent. It is then applied to the fuel region proportionally to the initial power distribution. This approach neglects the contribution to decay heat of the nuclides that undergo fission and their fission products during the transient, which should be a decent approximation, at least until re-criticality.

[1] On the other hand, [2] indicates between September and November 2004.



(a) 9 MW

(b) 30 MW

Figure 2: Average poison concentration in the homogenized fuel regions as a function of time after shutdown. The dotted lines help visualize the time at which the Xe-135 concentration goes back to its initial value (8.5 and 16.1 hrs for the 9 and 30 MW case, respectively).

2.1.3 Poison Tracking

Re-criticality time is significantly affected by the I-135/Xe-135 decay chain. Initially, Xe-135 builds up, inducing negative reactivity, and then starts to disappear, primarily through radioactive decay. Figure 2 shows the predicted evolution of the concentration of these isotopes assuming zero flux after the initial LOFC event. The total Xe-135 worth at the beginning of the transient is -1,600 and -2,300 pcm for the 9 and 30 MW case, respectively. The former is about 60% larger than the one reported in [13] (-980 pcm). More investigation is needed to understand this discrepancy. However, it is noted that the time at which the Xe-135 concentration goes back to its initial value is very similar and is around 8.5 hrs in both this work (see Figure 2) and [13].

In terms of cross-section generation, the contribution of Xe-135 is first removed from the macroscopic cross-section library, and another library containing its microscopic cross-section is prepared. While Serpent does provide both kinds of data, some correction is required to account for the fuel volume ratio. The Serpent documentation erroneously claims that the macroscopic cross-sections should be multiplied by the fuel volume ratio, whereas it is actually the microscopic cross-section that should be divided by it. In particular, this can be seen by the otherwise surprisingly small value (below 3×10^4 barns) for the Xe-135 absorption microscopic cross-section of the most thermal group ($E < 0.02$ eV), as opposed to almost 2.5×10^6 barns with the proper correction.

As for the fission yields for Xe-135 and I-135, the HTTR has a low enough burnup (around 12.5 MWd/kgU to date) that the effective multi-group yields can be assumed to be constant (in energy and space) and are chosen to be:

$$\gamma_I = 6.282 \times 10^{-2} \quad , \quad \gamma_{Xe} = 2.566 \times 10^{-3}. \tag{1}$$

2.1.4 Full Core Heat Transfer

During the accident, the ultimate heat sink is the VCS. For heat to reach the VCS, radiation and natural convection are the primary heat transfer mechanisms in at least two places: 1) between the core and the Reactor Pressure Vessel (RPV) and 2) between the RPV and the VCS. An issue with the `GapHeatTransfer` model in BISON was found in that the radiation component does not currently preserve energy for cylindrical and spherical geometries, with the error being proportional to the difference of the ratio of the outer to inner radii and one that can be significant in this case (greater than 10%). The preliminary results herein, therefore, do not use this model at this point. A MOOSE issue (#18585) was opened and a potential fix proposed.

2.1.5 Thermal-Hydraulics

The main modification made to the thermal-hydraulics model lies in the computation of heat transfer coefficients using Dittus-Boelter correlations. Natural convection is also modelled through heat transfer coefficients, computed using the same methodology as in [11], and summarized in Table 2. Future work should include making all the channels connected into a single input—if RELAP-7 allows it without drastically degrading convergence. This could be important to calculate the pre-heating of the coolant, as it flows upward around the permanent reflector before entering the core. Currently, that component is neglected during steady-state to avoid artificially removing heat from the overall model.

Table 2: Forced convection heat transfer coefficients used for the 9 and 30 MW steady-state calculations.

	p (MPa)	P (MW)	D_h (m)	ρ (kg/m^3)	ν (m^2/s)	Re	Pr	Nu	h (W/m^2/s)
RPV (inside)	4	30	0.9	1.71	2.93E-5	3.12E+4	0.662	76.8	33.5
RPV (inside)	2.8	9	0.9	2.266	1.42E-5	4.87E+4	0.659	110	30.7
CR	4	30	0.123	2.141	2.00E-5	7.51E+3	0.660	24.6	67.0
CR	2.8	9	0.123	2.575	1.14E-5	1.09E+4	0.659	33.1	62.2
Fuel Pin	4	30	0.007	2.141	2.00E-5	4.75E+3	0.660	17.0	817
Fuel Pin	2.8	9	0.007	2.575	1.14E-5	6.92E+3	0.659	23.0	758

2.2 Preliminary Numerical Results

It is emphasized that the results presented in this section are still preliminary and are bound to change in the near future as the model is refined.

2.2.1 Steady-State

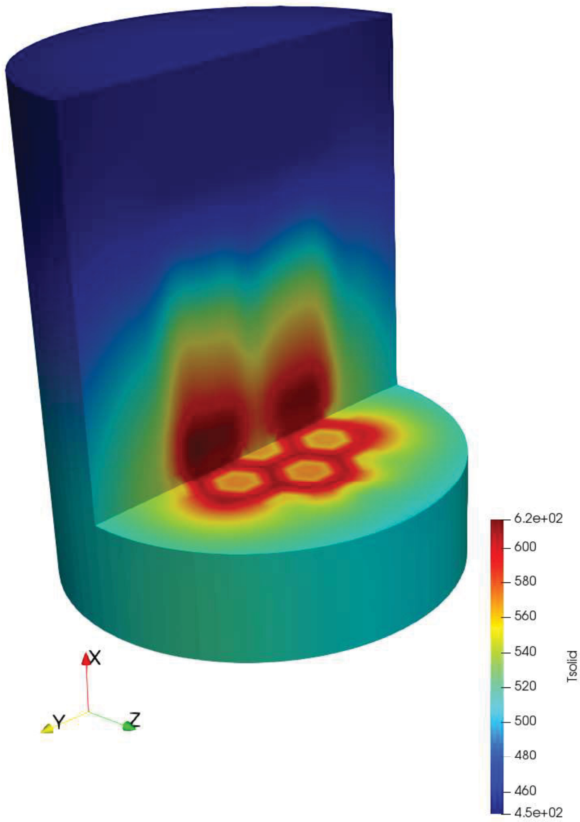
Obtaining a steady-state solution is crucial to the accurate modeling of the transient behavior. However, as explained in [6], numerical instabilities between physics often necessitate performing so-called pseudo-transient calculations to reach a steady-state solution. In addition, RELAP-7 usually requires a very gradual increase in time-step size to slowly ramp-up the wall and fluid temperatures. The coupling approach remains similar to that presented in [6].

Figure 3 shows the steady-state temperature distribution on the full-core heat transfer model for the 9 and 30 MW transients, which corresponds to the homogenized moderator temperature at the full-core level.

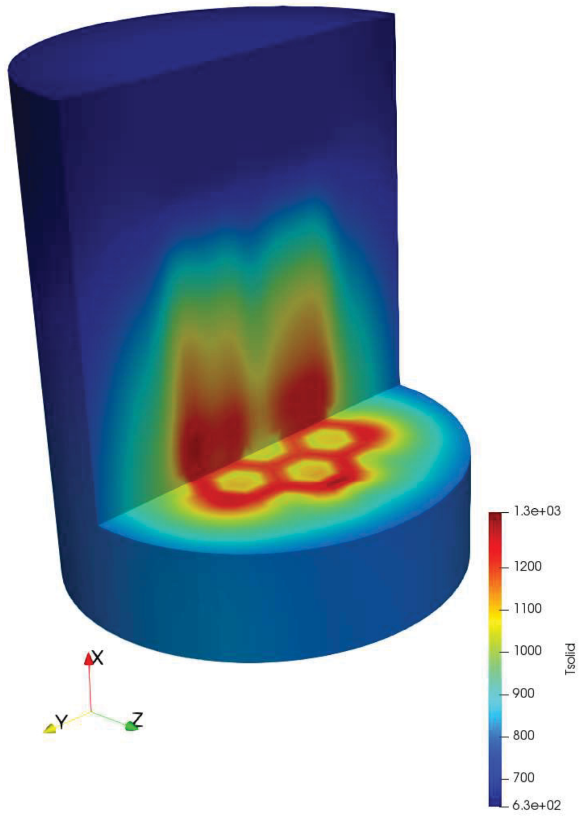
2.2.2 9 MW LOFC#1

The LOFC transients are initiated by decoupling the RELAP-7 model from the other physics to simulate the reduction on heat removal from the core caused by the circulators trips. Afer the decoupling the heat can only be removed from the core through its outer boundary.

Figure 4 gives the evolution of the numerical fission and decay powers over time and allows comparison of the former to the experimental data. The sudden initial increase in temperature leads the core to a sub-critical configuration, due to overall negative temperature reactivity



(a) 9 MW



(b) 30 MW

Figure 3: Steady-state solid temperature distribution (in K).

Figure 4: Evolution of the heat sources (fission and decay power) for the 9 MW LOFC transient along with the measured fission power.

coefficients. The number of fission events thus keeps decreasing until it becomes comparable to the neutron production produced by the neutron sources. The residual decay heat is then the only remaining significant heat source. As the core cools down and the Xe-135 decays, positive reactivity is slowly added to the core until re-criticality is reached. The current simulation predicts a re-criticality time of 9 hrs and 42 min and a subsequent power peak of 10 hrs and 50 min after the initial accident while the measurement indicate 6-7 hrs [13] and about 8 hrs, respectively. As for the fission power peak after re-criticality, the experimental is around 280 kW, whereas the predicted is about 128 kW with an additional 57 kW from decay heat.

2.2.3 30 MW LOFC#2

The main difference between this and the the 9MW LOFC case is that the core is initially operated at 30 MW, leading to a much larger axial temperature gradient: on the order of 100 K/m

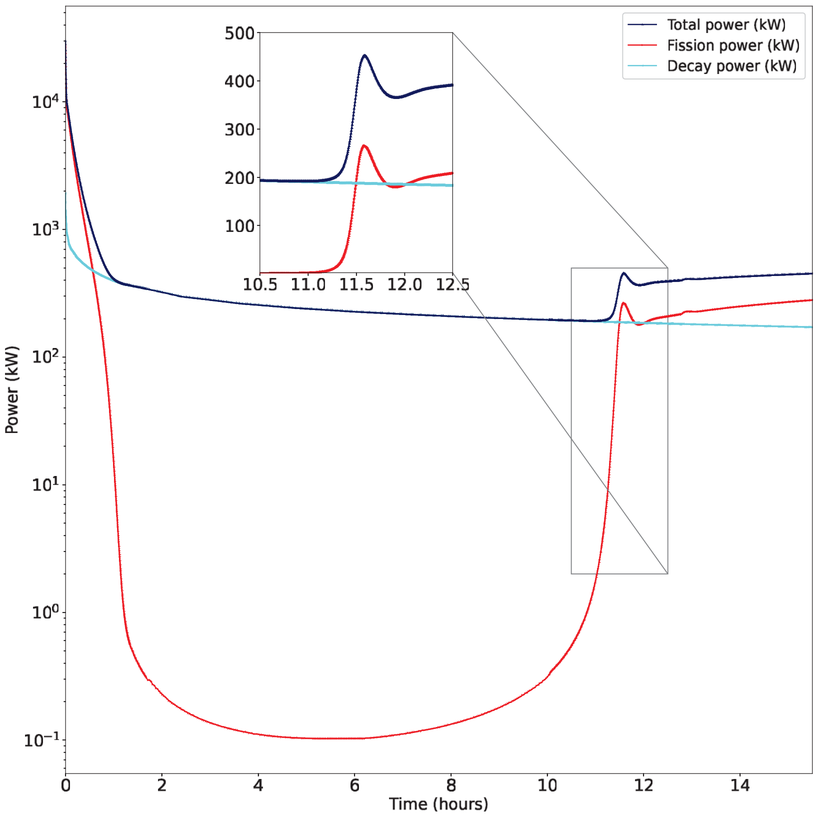


Figure 5: Evolution of the heat sources (fission and decay power) for the 30 MW LOFC transient.

for the radially-averaged fluid and moderator temperatures as compared to about 30 K/m for the 9 MW steady-state configuration. The other difference is that no experimental data is currently available, as the test is planned to be performed in early 2022.

The re-criticality happens later than in the 9 MW case, in part because the Xe-135 peak is much larger (as seen in Figure 2). The current model predicts that to happen around 10 hrs and 20 min after the initial event with the subsequent power peak being observed 1 hour and 15 min later. The fission power then reaches 260 kW with an additional 190 kW from decay heat.

2.3 HTTR Modeling Conclusions and Future Work

While the first LOFC transient attempt gave fairly promising results, with the overall behavior of the reactor successfully captured, numerous improvements are still needed to better compare with the experiment on the 9 MW case and have more confidence in the ability of the model to accurately predict the 30 MW behavior. Among them, the ability to model the VCS, especially to be able to simulate core behavior with and without VCS operation, is needed but will require resolving the issue in the BISON gap heat transfer model.

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