

THERMAL INVESTIGATIONS ABOUT THE LIQUID-SALT-COOLED HIGH TEMPERATURE REACTOR

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Abstract. A thermal model for the LS-VHTR (Liquid-Salt-Cooled High Temperature Reactor) has been developed using the RELAP5 code. The reactor operates at 2400 MW(t) and it is cooled by liquid Li_2BeF_4 (Flibe) salt. An initial thermal analysis of the LS-VHTR was performed in a previous work where the simulation of only one unit cell was considered. In the present work, the RELAP5-3D model was increased to represent the complete core with all hexagonal blocks. Also the salt recirculation has been simulated. The LS-VHTR core inlet and outlet coolant temperatures, heat structures temperature and pressure drop have been simulated. The results have been compared with the available data and demonstrate that the developed model is capable of reproducing the thermal behavior of the reactor in steady state operation.

Keywords: LS-VHTR, RELAP5-3D, FLIBE

1. INTRODUCTION

The Liquid-Salt-Cooled Very-High-Temperature Reactor (LS-VHTR) is one of the IV generation reactors and combines several new technology assets such as successful use of coated-particle graphite-matrix fuel in helium-cooled reactors; reactor plant and safety systems similar to that developed to the liquid-metal cooled fast reactors; low-pressure liquid-salt coolants studied and researched for liquid fuel reactors; and Brayton power cycles at high-temperatures. The LS-VHTR project goal is to provide an advanced design which offers the potential for higher power output, improved efficiency of electricity production, and higher operating temperatures leading to significant reduction in plant capital costs, as well as its use in high-temperature process heat applications that can economically produce hydrogen (Ingersoll et al., 2005).

The LS-VHTR core uses coated-particle graphite-matrix fuel have uranium oxycarbide fuel kernels which is coated with multiple layers of pyrolytic carbon and silicon carbide to form a microsphere of 0.8 mm diameter that prevents the release of radionuclides at very high temperatures. These microspheres fuel are frequently referred to as “TRISO” fuel and are incorporated into a graphite-matrix fuel compact, which, in turn, is loaded into a hexagonal fuel block which provides more control of the fuel and coolant volume fractions and geometry. Figure 1 shows the TRISO fuel and a typical fuel compact and prismatic graphite block, which is the fuel assembly shape utilized in the LS-VHTR.

A total of 265 columns of fuel blocks are assembled into a cylindrical geometry with nonfueled graphite reflector blocks filling in the region between the outer diameter of the core and the reactor vessel. Figure 1 provides a plan view of the core and reflector geometry. This cylindrical shape improved neutron economy, heat transfer, transport of liquid coolant and increase the total power output compared with the VHTR gas-cooled. LS-VHTR uses a closed primary cooled loop immersed in a tank containing a separate buffer salt. This design allows the use of a better salt in the primary coolant loop, that is, a salt that has better coolant properties but is not so good financially (Ingersoll et al., 2007).

The advantages of using liquid salt as coolant are mainly related to its high efficiency of heat transfer, operation at low pressures, high volumetric heat capacity compared with gas and sodium, possible optical inspection and low corrosion rate. A drawback aspect of liquid salts is the high melting temperature (between 350°C and 450°C), however, since the reactor operates at high temperature, this is not a problem. The salt used for coolant is called Flibe (66% LiF and 34% BeF_2). The Flibe has a small neutron total cross section, which makes it relatively transparent to neutrons (Davis et al., 2006).

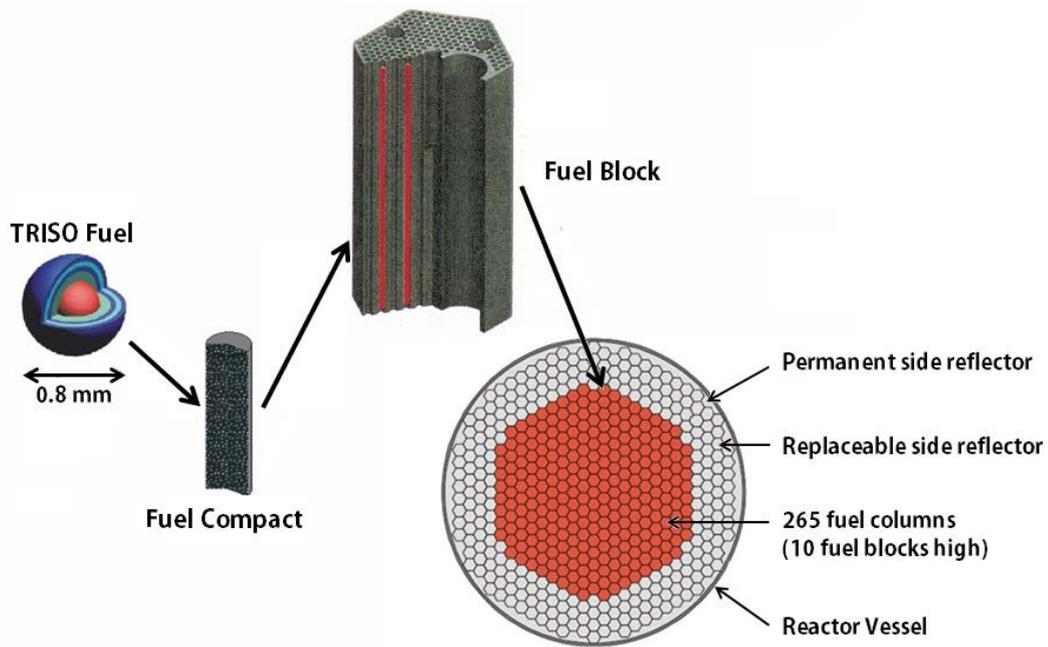


Figure 1. Fuel element - design of the LS-VHTR core.

In this study, the baseline fuel block design considered is the same as that considered in the work of Davis and Hawkes (2006). Each fuel block consists of a hexagonal element of 216 fuel channels, with diameter of 12.7 mm, 108 coolant channels with diameter of 9.53 mm and a fuel handling hole. The flat-to-flat distance of the block is 360.0 mm and the channel pitch is 18.8 mm. The baseline block is shown in Fig. 2. The geometrical parameters used in this study are in the Tab. 1.

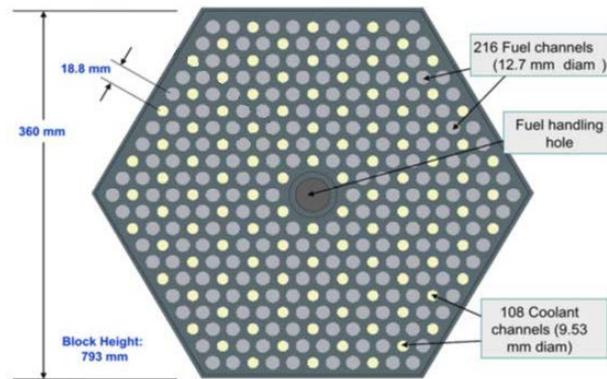


Figure 2. The LS-VHTR block.

Table 1. Geometrical parameters of the LS-VHTR.

Parameter	Value
Coolant channel diameter, mm	9.53
Fuel compact diameter, mm	12.45
Fuel channel diameter, mm	12.7
Fuel channel pitch, mm	18.8
Number of coolant channels per block	108
Number of fuel channels per block	216
Number of fuel columns	265
Flat-to-flat distance of hexagonal blocks, mm	360
Gap between hexagonal blocks, mm	1.0
Heated length, m	7.93

An initial thermal analysis of the LS-VHTR was performed in a previous work (Scari et al., 2013) where the simulation of only one unit cell was considered in the RELAP5-3D code. Each unit cell, represented as a part of the hexagonal fuel block, was modeled to represent one coolant channel, filled with Flibe, and two fuel channels, with two gaps, immersed in graphite moderator matrix (Scari et al., 2013). Since for thermal-hydraulic evaluations a 3D neutron kinetics model is required, the profile of the fission density, which is proportionally related to the power density, must be determined or estimated. Two power density distributions were estimated from different simulations performed by MacDonald (2003) with the same geometry configuration that the LS-VHTR core for steady state and control analysis calculation.

In the present work, the thermal modeling of this reactor has been improved. The core inlet and outlet coolant temperatures, the coolant mass flow, pressure drop and the temperatures along the core were simulated for two power density distributions in the core, cosine and chopped cosine profile. The results have been compared with the available data. The developed model demonstrated that the RELAP5-3D is capable of reproducing the thermal behavior of the LS-VHTR in steady state operation.

2. LS-VHTR CORE NODALIZATION

RELAP5-3D code can employ a variety of coolants in addition to water, the original coolant employed in early versions of the code. Liquid metals (sodium, potassium, NaK, lithium, Flibe) and cryogenic fluids (hydrogen, helium, nitrogen) are some of the available coolants (Qualls et al., 2006). In this way, the RELAP5-3D can appropriately simulate high temperature reactors. The RELAP5 code versions were originally designed to simulate light water reactors (LWR). The hydrodynamic model is a two-fluid model for flow of a two-phase steam-water mixture that allows non condensable components as, for example, helium, in the steam phase and/or a soluble component in the water phase. In this way, it is possible to use RELAP5 with only helium and no steam, as in the case of a HTR simulation.

In this way, the LS-VHTR core has been modeled using the RELAP5-3D code. In the developed model, 53 thermal hydraulic channels represent the 265 fuel columns of the core, that is, each thermal hydraulic channel represents 5 fuel columns with its corresponding heat structure (HS), as represented in Fig. 3.

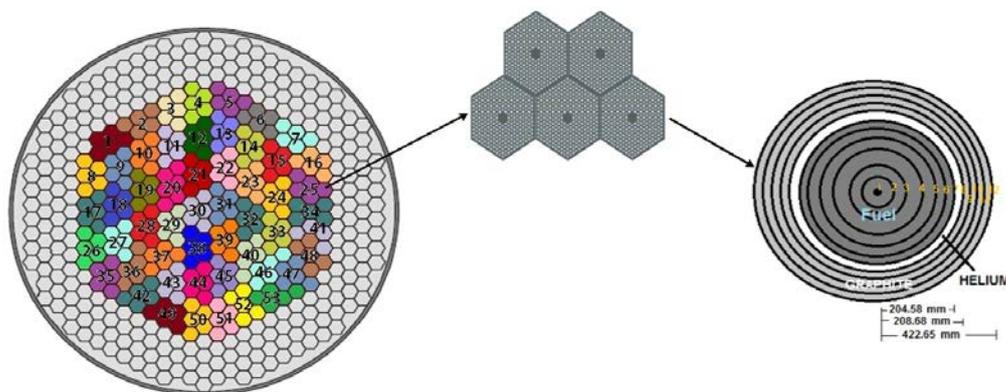


Figure 3. Heat Structure represented in RELAP5-3D model.

The HSs simulate the power source of each channel and they were divided axially according to the same quantity of the thermal channel volumes. All HS have 12 radial meshes, being 6 intervals to the fuel region, 1 interval for the helium gap and 4 intervals representing the graphite region. A radial view of the heat structure is represented in the Fig. 3. It was selected the thermodynamics properties of the LiF-BeF₂ in the calculations. The point kinetics option was used in the calculations, the data of volumetric heat capacity and heat transfer coefficient of the standard fuel compact, the helium in the gap and the graphite were considered. The initial conditions used to simulate the core are shown in Tab. 2.

Table 2. Initial conditions for the LS-VHTR

Parameter	Value
Core total power, MW	2400
Core mass flow rate, kg/s	10,264
Core inlet temperature, °C	900
Core outlet temperature, °C	1000
Average fuel temperature, °C	1093
Maximum fuel temperature, °C	1329
Core pressure drop, MPa	0.211
Vessel pressure drop, MPa	0.276

The complete core model is illustrated in Fig. 4. The coolant channels were represented by the component of the type pipe and were divided in 24 axial volumes of 0.3304 meters corresponding to the active length of 7.93 m. Two time dependent volumes, 101 and 800, represent, respectively, the inlet and outlet core plena. The components 700 and 100 are single junctions and the pipes from 201 up to 253 represent the coolant channels. The idea of to model the complete core is, in the future, to perform a multi-dimensional neutron kinetics modeling using the NESTLE code and then to simulate a 3D thermal-neutron kinetic coupled calculation allowing specific local core investigations during a transient (RELAP-3D Code Manuals, 2009).

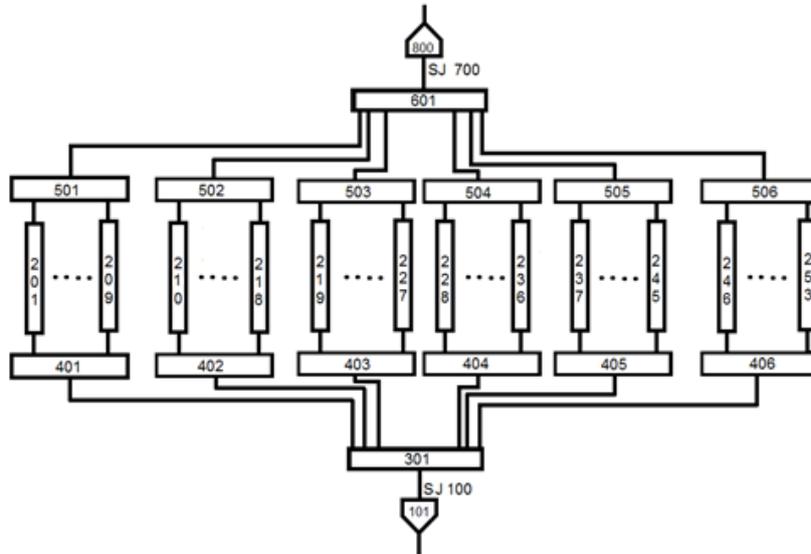


Figure 4. LS-VHTR core reactor modelled in the RELAP5-3D.

In the nodalization, the axial power profile used for the steady state calculation has a cosine shape shown in the curve cosine profile of the figure 5, corresponding to the behavior of the core reactor without any disturbance. The axial power factor is a strong function of control rod position. The axial power factors assumed for control analysis shown in the Fig.5 is a chopped cosine profile represents the time point between the middle and end of cycle when the control rods are inserted to the axial midpoint of the active core. This type of power distribution generally results in higher fuel temperatures, since more power is shifted toward the bottom of the core where coolant temperatures are also the highest. These two profiles were extracted from the work performed by MacDonald (2003).

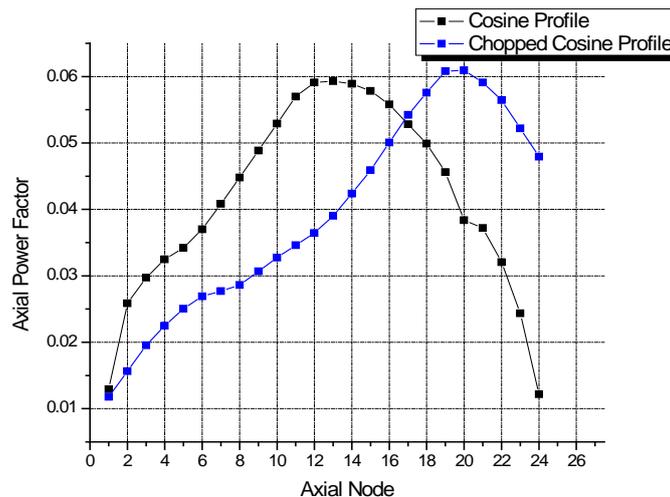


Figure 5. Axial Power Factor used for core reactor, the cosine profile corresponding to steady state and the Chopped cosine profile corresponding to control analysis.

3. STEADY STATE AND CONTROL ANALYSIS CALCULATION

The calculated temperature along coolant and the heat structure for each axial power profile are shown in the Fig. 6. Each point represents an axial node of the structure, from 1 up to 24. The temperature of each point is the average radial temperature of the corresponding axial node in the heat structure and coolant channel. Both Figures, The temperature increases along the coolant channel and heat structure reaching maximum value in the fuel, following approximately the same behavior of each axial power profile with the difference that the higher temperature regions is displaced axially due this difference is directly connected to the fact that the two models have a different axial power distribution which influence the distribution of heat in the heat structure

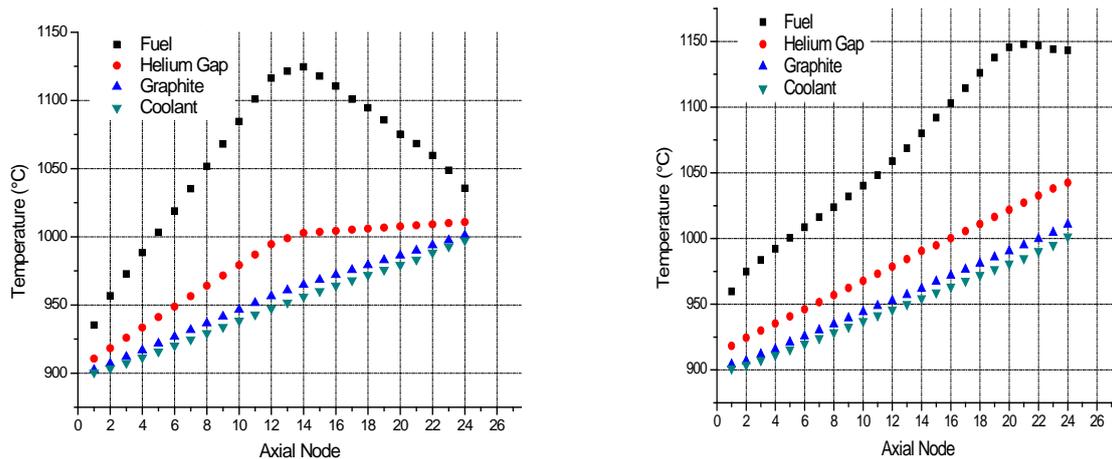


Figure 6. Temperature along coolant channel and the heat structure for each axial power profile, left is shown temperature calculation of cosine profile, right for chopped cosine profile.

In Figure 7, the temperature distribution for each axial and radial node of the heat structure is plotted, the inner region is fuel (1 to 6 radial node), middle region is Helium (only 7 radial node) and outer region is graphite (8 to 12 radial node). The maximum temperature for cosine profile is located near of the fuel central region. In the case of chopped cosine power profile, the high temperatures are located in the fuel central region displaced axially. These temperatures are directly connected with its axial power distribution.

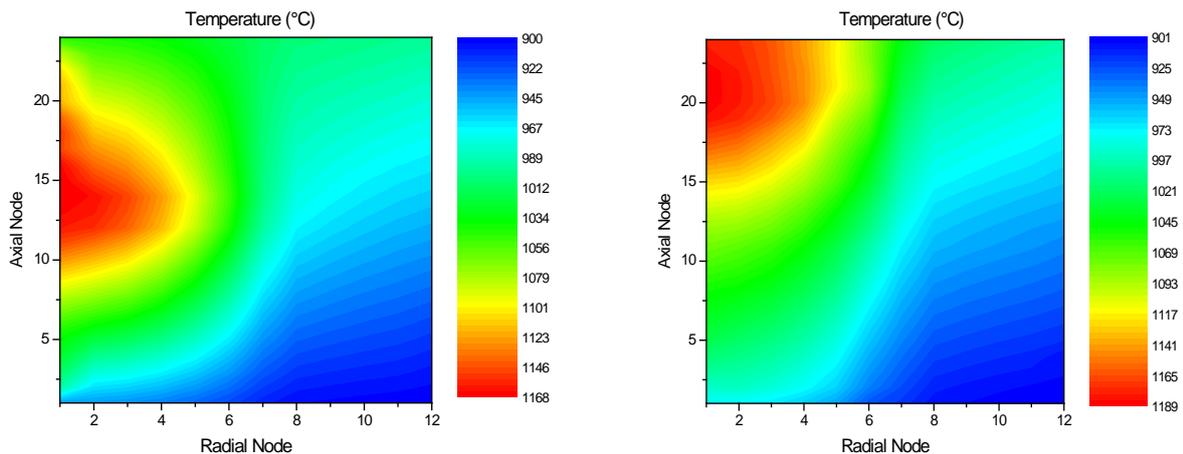


Figure 7 Temperature distribution of the heat structure for the cosine and chopped cosine power profile.

As we can see in Figure 7, the temperature in the heat structure is not uniform; it spans from 900°C for the outermost channel to 1168°C in the case when it has a cosine profile and from 901°C to 1189°C when it has a chopped cosine profile. While the temperature of HS depends on the radial position the temperature distribution is affected axially due of control rod position associated with the chopped cosine profile This type of power distribution generally results in higher fuel temperatures, since more power is shifted toward the bottom. Although the temperature of HS is

not uniform, it spans over a relatively range (~260°C) of temperatures for cosine profile and (~280°C) for chopped cosine profile.

The calculated average pressure along the 201 thermal hydraulic channels for cosine profile is shown in the Fig. 8. Each point represents an axial node of the structure, from 1 up to 24. The pressure decreases along the channel reaching minimum value in the axial node 24, approximately the expected behavior is similar for chopped cosine profile.

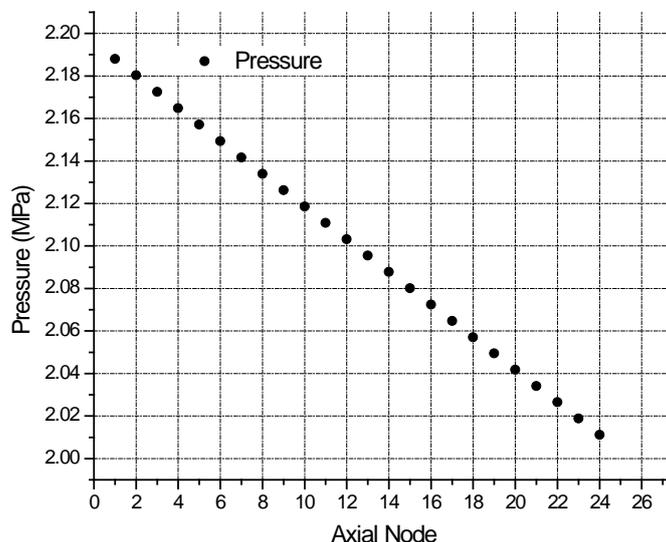


Figure 8. Pressure of 201 coolant channel for the cosine profile.

Tab. 3 presents a comparison between the calculated results with the RELAP5-3D for cosine profile with the reference. As it is possible to verify, in a general way, the parameters are in good agreement each other. Also presents the calculated results for chopped cosine profile which presents a difference due influence of axial power distribution of heat in the heat structures of LS-VHTR core.

Table 3. Calculated results with the RELAP5-3D for the LS-VHTR

Parameter	Reference ¹	profile Cosine	Profile Chopped Cosine
Core Power, MW	2400	2400	2400
Core mass flow rate, kg/s	10,264	10,132	10,110
Core inlet temperature, °C	900	900	901
Core outlet temperature, °C	1000	1000	1010
Average fuel temperature, °C	1093	1066	1070
Core pressure drop, MPa	0.21	0.20	0.18

¹Ingersoll (2005)

4. CONCLUSIONS

Core thermal analysis of the LS-VHTR reactor has been performed in this study using the RELAP5-3D code. Then, simulations of thermal parameters of the reactor cooled by liquid Li₂BeF₄ (Flibe) salt presented similar behaviour in relation to those of the reference one. The developed model demonstrated that the RELAP5-3D is capable of reproduce the thermal behavior of the LS-VHTR in steady state operation, and represent too the behavior of reactor core for different power profile. The next step of this work will be considering more details including safety mechanisms and neutronic parameters to perform a more realistic reactor simulation using the multidimensional capabilities of the RELAP5-3D.

5. ACKNOWLEDGMENTS

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6. REFERENCES

- Ingersoll, D. T., Clarno, K. T., Forsberg, C. W., Gehin, J. C., Christensen, R. W., Davis, C. B., Hawkes, G. L., Sterbenz, J. W., Kim, T. K., Taiwo, T. A., Yang, W. S., 2005. Status of Physics and Safety Analyses for the Liquid Salt-Cooled Very High-Temperature Reactor (LSVHTR), ORNL/TM-2005/218, Tennessee, USA.
- Ingersoll, D. T., Forsberg, C. W., MacDonald, P. E., 2007. Trade Studies for the Liquid-Salt-Cooled Very High-Temperature Reactor: Fiscal Year 2006 Progress Report, Oak Ridge National Laboratory, ORNL/TM-2006/140, Tennessee, USA.
- Davis, C.B. and Hawkes, G. L., 2006. Thermal-Hydraulic Analyses of the LS-VHTR, *Proceedings of ICAPP 2006*, Reno, Nevada, USA.
- Scari, M. E., Costa, A. L., Pereira, C., Veloso, M. A. F., Silva, C. A. M., Reis, P. A. L., 2013. Thermal hydraulic modeling of the LS-VHTR, *Proceedings of the 2013 International Nuclear Atlantic Conference – INAC 2013*, Recife, Brazil.
- Qualls, A. L. and Wilson Jr. T. L., 2006. Dynamic System Model of LS-VHTR to Estimate Design Parameter Impacts on Safety Margin and Reactor Economics, *Proceedings of ICAPP 2006*, Reno, Nevada, USA.
- The RELAP5-3D© Code Development Team, 2009. RELAP5-3D© Code Manuals, INEEL-EXT-98-00834, Idaho National Laboratory.
- MacDonald, P. E., 2003. NNGP Preliminary Point Design – Results of the Initial Neutronics and Thermal-Hydraulic Assessments, INEEL/EXT-03-00870, Idaho, USA.

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