

Transformational Challenge Reactor Accident Analysis¹

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INTRODUCTION

The Transformational Challenge Reactor (TCR) is a helium-cooled, yttrium-hydride moderated reactor being designed for the US Department of Energy (DOE) Office of Nuclear Energy. The purpose of the TCR is to employ advanced manufacturing techniques in a nuclear system and demonstrate its potential for revolutionizing the nuclear reactor design process.

Transient analyses were performed to assess the TCR's performance under postulated accident scenarios. The current analyses are preliminary and are under ongoing development to support the operation of the TCR under DOE regulations.

METHODOLOGY

A description of the TCR core design, system design, and operating parameters has been given previously [1, 2]. For brevity, the current discussion is limited to the analysis of accident events based on this design.

Accident Event Identification

A preliminary list of postulated accident events was developed from the available literature including published safety analyses, articles, and white papers for gas-cooled thermal reactors [3, 4, 5]. A workshop was held in January 2020, within the TCR program, to refine this list and develop a set of accident scenarios relevant to the TCR's specific core and system configuration.

Among these, the following scenarios were analyzed in the current study:

- pressurized loss of forced cooling (PLOFC), in which the helium circulator trips and coasts down to natural circulation conditions;
- circulator rotor lock-up event, assuming an abrupt seizure of the circulator with flow remaining able to continue in natural circulation mode;
- depressurized loss of forced cooling (DLOFC), in which a break in the primary loop helium boundary leads to rapid depressurization to atmospheric pressure;

- loss of heat sink (LOHS), in which the air fan trips, leading to impaired heat exchanger heat removal; and
- core partial flow blockage, in which an obstruction or geometric distortion of one or more core channels causes loss of flow in a localized region of the core.

These events range from highly to extremely unlikely during the TCR's limited operating time. However, as will be shown, the safety limits remain satisfied even for these low-likelihood occurrences. Note that though reactivity insertion accidents were also initially analyzed for TCR, owing to the inherent design features of the reactor protection and control systems they were deemed non-viable and therefore omitted here.

Safety Limits

A preliminary set of safety limits for the materials present in the TCR core and vessel is shown in TABLE I. Small batch heating tests on irradiated uranium nitride tristructural isotropic (UN TRISO) [6] particles is planned under the TCR program and will inform the final safety limit for UN TRISO; however, the current expectation is that failures will not occur below 1600°C since these particles have been tested in the as-fabricated state by heating up to 1850°C without failure. The dissociation of hydrogen from YH_{1.85} (yttrium hydride) increases with temperature; however, the equilibrium partial pressure of H₂ gas in YH_{1.85} reaches 2 atm within a sealed sheath, and therefore the degree of dissociation is expected to be minimal below 1100°C [7].

TABLE I. Preliminary safety limits for the TCR

Material	Preliminary Safety Limit	Comment
UN TRISO	1600°C	Fuel particles
SiC	1900°C	Fuel matrix
316L SS	1370°C	Moderator sheath
YH _{1.85}	1100°C	Moderator
304 H SS	673°C	Pressure vessel; limit applies for up to 5 hr at 5 MPa helium pressure

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The goal of the safety analyses, of which a preliminary set is provided here, is to demonstrate that all components remain within their acceptable safety limits during the postulated accidents. If not, improvements to the analyses (e.g., reduction in conservatism) or design modifications would be required.

Analysis Approach

Because of its extensive use in the nuclear community for advanced reactor transient analyses and its ability to model the important physics for gas-cooled thermal reactors, RELAP5-3D [8] was chosen for the transient system-level thermofluidic analyses of the TCR. As depicted in Figure 1, the primary helium and secondary air flow paths were explicitly modeled using current assumptions for component sizing and performance. Neutron point kinetic feedback was modeled, with parameters taken from 3D neutron kinetics calculations for the TCR core. Realistic scram timing was provided by use of reactor trip setpoints using current best-estimate assumptions for the reactor protection system (RPS) design. The helium circulator was assumed to trip concurrently with the reactor trip in all events to protect the circulator from high temperatures and avoid damage.

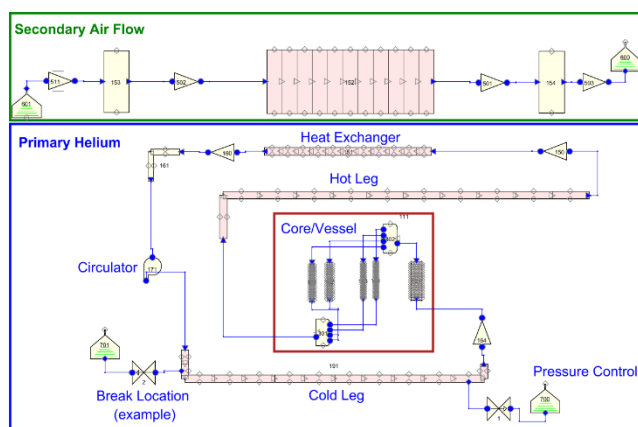


Figure 1. RELAP5-3D TCR model diagram for the primary and secondary flow paths.

For many gas-cooled thermal reactors, conductive, convective, and/or radiative heat transfer occurring radially from the core to the pressure vessel and beyond provides an important heat transfer mechanism during accident events such as loss of flow events. This heat transfer pathway was implemented as shown in Figure 2, using radiation enclosures in RELAP5-3D, as well as hydraulic volume connections where appropriate, to model the radiative and convective heat transfer between solid volumes.

Radiation view factors were calculated based on ratios of surface areas exposed between the inner hot assemblies, the remaining assemblies, the core barrel, and the vessel. Conduction was modeled within each solid volume but is not currently accounted for between volumes. This is due to a limitation in RELAP5-3D: each heat structure surface can

participate in either conduction or radiation with other heat structures, but not both. An approach was attempted to model conduction between all surfaces while approximately including the effect of radiation through a temperature-dependent surface-to-surface conductance enhancement. However, this resulted in temperatures well exceeding the results when using radiation enclosures only (and leading to code failure), indicating that the radiative heat transfer effect (with its fourth-power temperature dependence) was greatly underpredicted when using the approximated conduction-based enhancement factors. Due to this unrealistic behavior, the decision was made to use radiation enclosures only, which is expected to provide a reasonable but conservative result for peak core temperatures and a reasonable prediction of ex-vessel heat transfer.

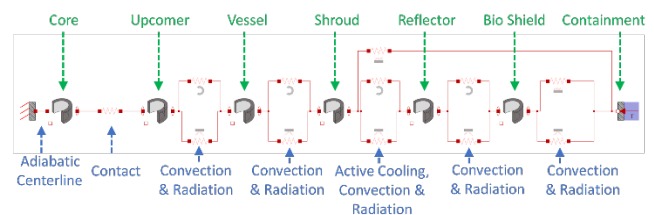


Figure 2. Diagram of the radial core, vessel, and ex-vessel heat transfer pathway used by RELAP5-3D, COMSOL, and TRANSFORM.

DLOFC was found to be the most limiting event in terms of core temperatures; other events were potentially more challenging to vessel temperatures, particularly because the 673°C vessel temperature limit only applies to at-pressure conditions. Because of its full system modeling capability, RELAP5-3D was used to analyze the non-DLOFC events. It was also used to determine appropriate core power vs. time behavior considering realistic trip timing based on the RPS primary loop pressure sensor. This was input into a 2D radial COMSOL model, which was used to provide a more accurate and detailed core temperature response during the DLOFC event. The detailed 2D radial core geometry was explicitly modeled in COMSOL, while the core barrel, vessel, shroud, reflector, and bioshield were modeled with concentric cylinders consistent with the RELAP5-3D model. The power generation distribution was taken from the peak power plane of the SCALE neutronic calculation, with the coolant temperature conservatively set to the core outlet temperature of 500 °C. This provided conservatively high solid temperature predictions by neglecting axial heat transfer out of the highest-power plane. A simplifying, conservative assumption was made in COMSOL that all heat transfer from the core to the coolant was set to zero for the duration of the DLOFC event. A depiction of the 2D radial temperature profile in the core, vessel, and external components is given in Figure 3.

Confirmatory analyses for the DLOFC were also performed with TRANSFORM [9], using similar assumptions as used in COMSOL. Both models employed a

radial heat transfer configuration as shown in Figure 2 and consistent with the RELAP5-3D model. The primary difference between the models is that TRANSFORM employed a simplified homogeneous core treatment, with thermophysical properties calculated as volume averages of the core constitutive materials, compared to the detailed 2D core geometric representation used in COMSOL.

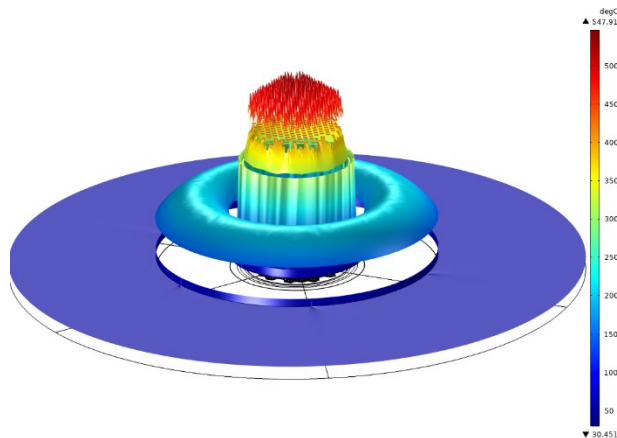


Figure 3. Steady-state temperature distribution calculated by COMSOL.

Core partial flow blockage analyses were performed with COMSOL, due to the localized thermal peaking and resulting need for detailed 2D core modeling in this scenario.

RESULTS

COMSOL and TRANSFORM Analyses

As shown in Figure 4 and Figure 5, COMSOL and TRANSFORM are in reasonable agreement in terms of DLOFC event progression and temperature trends, given the difference in core modeling fidelity in the two models. The peak core centerline temperature predicted by TRANSFORM was approximately 120°C lower than in COMSOL, while the peak vessel temperature was roughly 35°C higher. This was attributed to the homogeneous core treatment in TRANSFORM, which led to a more rapid removal of heat from the core relative to COMSOL, and therefore lower peak core temperatures but higher vessel temperatures.

COMSOL results for a limiting flow blockage scenario are shown in Figure 6. In this scenario, the helium flow rate was set to zero in the primary and secondary flow channels for each of the inner six fuel elements in the core. (These elements have the highest power peaking factor and are expected to be the most limiting for this scenario.) In reality, no specific mechanism for flow blockage has been identified, and such an occurrence is considered extremely unlikely. However, even under such conservative assumptions, the peak temperatures are calculated to be 955°C, and the core temperature limits are satisfied. An additional, less conservative scenario was modeled in which only the primary

flow channels were blocked in the inner six fuel assemblies; the peak temperatures were 591°C in this case.

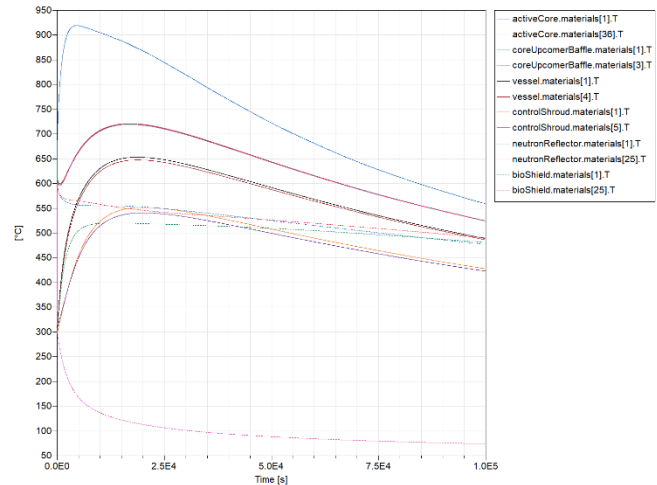


Figure 4. TRANSFORM temperature vs. time results for the DLOFC event.

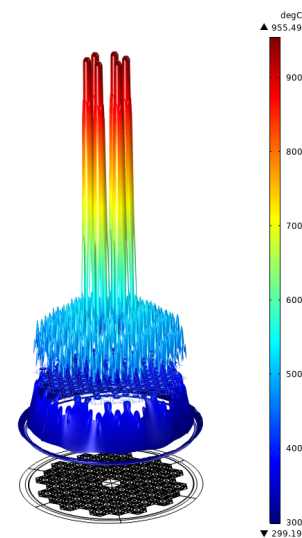


Figure 5. COMSOL partial flow blockage results, assuming complete blockage of the inner ring of fuel elements.

RELAP5-3D Analyses

RELAP5-3D results, as well as the COMSOL DLOFC results, are shown in TABLE II. Time-dependent results for the circulator lock-up event are given in Figure 7.

In the RELAP5-3D events, the peak fuel and vessel temperatures are dictated by a balance between impaired cooling capability (i.e., loss of forced helium and/or air flow) and decreased core heat generation due to negative Doppler reactivity feedback and, eventually, reactor scram. For the PLOFC and LOHS events, the latter effect quickly outweighed the former, such that the peak fuel and vessel temperatures were less than 5°C higher than their steady-state values. This was mainly because significant active helium

cooling persisted for some time following the initiating event, aided by the gradual coastdown of the helium circulator. However, for the circulator lock-up event, the driving flow was assumed to drop to zero immediately, which led to immediate impairment of heat removal from the core via helium following the initiating event. This led to elevated core and vessel temperatures, which nonetheless remained well within their respective preliminary safety limits.

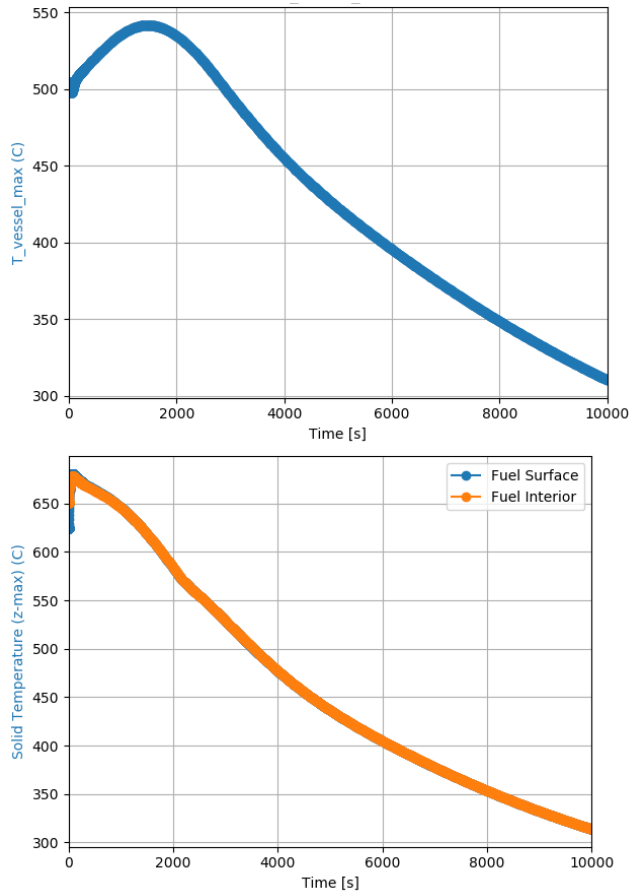


Figure 6. RELAP5-3D peak vessel temperature (top) and peak fuel interior and surface temperature (bottom) results for the circulator lock-up event.

TABLE II. Peak temperatures calculated with RELAP5-3D (PLOFC, LOHS, and circulator lock-up) and COMSOL (DLOFC)

	DLOFC	PLOFC	LOHS	Circulator Lockup
Peak vessel temperature (°C)	616	502	500	542
Peak fuel temperature (°C)	1035	654	651	678

CONCLUSIONS

The preliminary analyses shown here demonstrate the safety of the TCR under all postulated accident events.

Additional work includes assessing the impact of modeling and measurement uncertainties on these accident analyses, as well as supporting further development of RPS signals and setpoint specifications informed by RELAP5-3D and COMSOL calculated temperatures.

ACKNOWLEDGMENT

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