**OVERVIEW OF THE VERSATILE TEST REACTOR**

**SAFETY ANALYSIS**

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**Abstract**

The Versatile Test Reactor is a fast spectrum test reactor currently being developed in the United States under the direction of the US Department of Energy, Office of Nuclear Energy. Safety analysis of the conceptual VTR design has been performed using the SAS4A/SASSYS-1 fast reactor safety analysis code with a model representing the reactor core, primary and intermediate heat transport systems, reactor vessel auxiliary cooling system, and reactor protection system. The reactor protection system response following the detection of elevated plant conditions dominates the transient behaviors in these transients. Because the primary heat transport system is able to transition quickly and effectively to natural circulation and because the reactor vessel auxiliary cooling system provides sufficient heat rejection, large margins for all criteria were predicted for the analyzed transients.

## INTRODUCTION

The Versatile Test Reactor (VTR) is a fast spectrum test reactor currently being developed in the United States under the direction of the US Department of Energy, Office of Nuclear Energy. The VTR mission is to enable accelerated testing of advanced reactor fuels and materials required for advanced reactor technologies. The conceptual design of the 300 MW(th) sodium-cooled metallic-fueled pool-type fast reactor has been led by the US National Laboratories in collaboration with General Electric-Hitachi and Bechtel National Inc.

Safety analysis of the conceptual VTR design has been performed using the SAS4A/SASSYS-1 fast reactor safety analysis code [1]. This tool has been developed by Argonne National Laboratory over the course of several decades and is the state-of-the-art tool in the U.S. for fast reactor analysis. The VTR SAS4A/SASSYS-1 model represents the reactor core, primary and intermediate heat transport systems, reactor vessel auxiliary cooling system (RVACS), and an assumed reactor protection system (RPS).

A number of protected transients have been evaluated to demonstrate the system’s response to various initiating events. Ultimately, the full spectrum of event initiators and subsequent accident sequences will be defined through a probabilistic risk assessment (PRA) of the plant design following the Licensing Modernization Project guidelines. Since that process is on-going, the present analysis addresses several transients that represent the three key ways to perturb a reactor: through changes to the core inlet temperature, mass flow rate, or reactivity. These correspond to a loss of heat sink (LOHS), loss of flow (LOF) or station blackout (SBO), and transient overpower (TOP), respectively. Results of this work provide input for the Conceptual Safety Design Report, which summarizes the overall safety basis for the VTR.

The RPS response following the detection of elevated plant conditions dominates the transient behaviors in these transients. Because the primary heat transport system is able to transition quickly and effectively to natural circulation and because the RVACS provides sufficient heat rejection, large margins for all criteria were predicted for the analyzed transients. All transients analyzed in the paper were initiated at nominal full power, full flow conditions and were simulated at beginning of cycle (BOC) and end of cycle (EOC) conditions.

The work reported in the paper is the result of studies supporting a VTR conceptual design, cost, and schedule estimate for DOE-NE to make a decision on procurement. As such, it is preliminary.

## Model Description

Transient simulations for the safety analyses are performed using the fast reactor safety analysis code SAS4A/SASSYS-1, Version 5.4. Table 1 summarizes the model predictions at steady-state for BOC and EOC conditions. The SAS4A/SASSYS-1 model of the VTR core, heat transport systems, and reactor protection system is presented below.

TABLE 1. NOMINAL STEADY-STATE MODEL PREDICTIONS

|  |  |
| --- | --- |
| Parameter | Value |
| Core Power | 300 MW(th) |
| Core Inlet Temperature | 350°C |
| Core Outlet Temperature | 500°C |
| Peak Fuel Temperature at BOC/EOC | 769°C/751°C |
| Peak Cladding Temperature at BOC/EOC | 553°C/547°C |
| Peak Coolant Temperature at BOC/EOC | 528°C/520°C |
| Core Mass Flow Rate | 1565 kg/s |
| Core Pressure Drop – Total | 0.654 MPa |
| Primary Pump Head | 0.739 MPa |
| IHTS Mass Flow Rate Per Loop | 737 kg/s |
| IHX Secondary Inlet Temperature | 300°C |
| IHX Secondary Outlet Temperature | 461°C |

In SAS4A/SASSYS-1, the thermal-hydraulic performance of a reactor core is analyzed with a model consisting of a number of single-pin channels. The channel model provides input to specify a single fuel pin and its associated coolant and structure. A single-pin model represents the average pin in an assembly, and assemblies with similar reactor physics and thermal-hydraulic characteristics are grouped together. A number of channels are selected to represent all assemblies in the reactor core. The VTR core model has 5 different types of channels:

* Fuel (F),
* Experimental (Exp)
* Control rod (CR) and safety rod (SR),
* Reflector (R), and
* Shield (S).

The 313 core assemblies are modeled with 26 channels. Channels 1-9 represent 61 of the 66 fuel assemblies, Channels 10-19 represent the non-fueled assemblies, and Channels 20-26 represent the fuel assembly with the highest power-to-flow ratio (Channel 20) and its six neighboring assemblies (Channels 21-26). Fig. 1 illustrates the core loading pattern. The peak assembly for Channel 20 shown in dark red. Consistent with the neutronics calculations, the experimental positions are assumed to be empty assemblies filled with sodium.

Axial and radial power distributions for the VTR were calculated with the Advanced Reactor Codes (ARC) tools suite, which has been developed over many decades and demonstrated to be well suited for the analysis of SFRs [2] [3] [4]. Fuel assembly flow rates were calculated with the SuperEnergy2-ANL steady-state thermal hydraulics code to optimize flow orificing [5]. Flow through non-fueled assemblies was assumed to provide an assembly outlet temperature, averaged for BOC and EOC conditions, of 500°C, which is equivalent to the average core outlet temperature. Point kinetics and reactivity feedback coefficients were also calculated with the ARC tools suite. The VTR SAS4A/SASSYS-1 model calculates the following reactivity feedback effects:

* Fuel Doppler,
* Axial fuel, cladding, and structure expansion,
* Coolant density,
* Radial core expansion,
* Control rod driveline expansion, and
* Reactor vessel expansion.



Fig. . Assembly Loading Pattern.

The PRIMAR-4 module of SAS4A/SASSYS-1 simulates the thermal hydraulics of the heat transport systems other than the core. Compressible volumes, or CVs, are zero-dimensional volumes that are used to model larger volumes of coolant such as inlet and outlet plena and pools. CVs are characterized by their pressure, temperature, elevation, and volume. Compressible volumes are connected by liquid segments, which are, in turn, composed of one or more elements. Elements are modeled by one-dimensional, incompressible, single-phase flow and are used to model pumps, pipes, valves, heat exchangers, steam generators, and more. Elements are characterized by their length, inlet and outlet elevations, temperatures, form and friction pressure losses, and mass flow rate.

Fig. 2 illustrates the primary heat transport system model. Sodium enters the core, Segment 1, from the inlet plenum, CV1, and discharges into the hot pool, CV2. Segments 2 and 3 represent the primary sides of the two intermediate heat exchangers (IHX). Sodium discharges from the IHXs into the bulk cold pool volume, CV4. Sodium is then drawn up through annular flow paths within the fixed shielding by the electromagnetic (EM) primary pumps. The flow paths for the four primary pumps are represented by Segments 5 and 6 with multiplicity factors of two so each segment represents two pumps. Sodium returns to the inlet plenum via eight core inlet pipes, with two discharge pipes per pump.

The PRIMAR-4 model of the VTR also includes a preliminary model of the secondary loop, which rejects heat to the environment via sodium-to-air-heat exchangers (SAHXs).

The reactor protection system (RPS) design, both the logic and trip setpoints, has not yet been completed. A reactor protection system model has been assumed to monitor important plant parameters and initiate a series of actions to trip the reactor during abnormal or accident conditions. Thresholds have been assumed for reactor power, the power-to-flow ratio, the average core inlet temperature, and the average core outlet temperature. The main role of the RPS will be to initiate a rapid reactor shutdown when one of these thresholds are exceeded. The RPS is assumed to first scram the control and safety rods. Next the primary and secondary pumps are tripped. And finally SAHX heat rejection is terminated. RVACS, which initially rejects approximately 0.25% of the nominal core power at nominal full power, full flow conditions, is always operational so the RPS does not need to take any action to activate it.



Fig. . Primary Heat Transport System Model.

## Safety Criteria and Metrics

The three protected transient scenarios presented below are expected to fall into either the “anticipated” (>10-2 occurrences per reactor-year) or “unlikely” (10-2 to 10-4) event categories [6]. For this analysis, the protected transients are conservatively evaluated against the criteria of the anticipated operational occurrences (AOO) category. The higher assumed frequency means that these transients may be expected to occur once or more during the plant lifetime. While any one physical barrier will not fail during a single protected transient scenario, the stress of repeated AOOs over the lifetime of the plant must be low enough to ensure no damage to the plant following multiple events. The protected transient scenarios are evaluated against the following criteria:

* Cladding temperatures must remain below 649°C,
* Bulk coolant temperatures must remain below 649°C,
* Subcriticality must be established and maintained.

The first requirement ensures that cladding temperatures remain below the initial threshold for fuel-cladding eutectic formation. The second requirement ensures structural integrity of the primary coolant boundary, i.e. the reactor vessel. For the third criteria, the reactor protection system is assumed to scram the control rods when a reactor trip threshold is exceeded, terminating the fission process during the protected transient scenarios. The VTR core design team is designing the control rods to bring the reactor to subcriticality while the system is at colder refueling temperatures. Even with a single stuck rod, this criterion will be met for all protected transient scenarios.

Fuel melting will likely not be permitted for protected transient scenarios in the AOO event category. However, because of the high thermal conductivity of the metallic fuel, if the cladding temperature criterion is met for the current VTR fuel pin design, the system will have maintained large margins to fuel melting and in-core sodium temperatures will remain below 649°C. Therefore, there are no additional requirements imposed on fuel temperatures or in-core coolant temperatures for the protected transients.

## RESULTS

Results of transient simulations of the protected station blackout, protected transient overpower, and protected loss of heat sink scenarios are presented below. The results are presented first for beginning-of-cycle (BOC) conditions, with end-of-cycle (EOC) condition results summarized at the end of each section.

### Protected Station Blackout

The protected station blackout (PSBO) transient is initiated by an assumed loss of electrical power to all plant systems. Forced circulation in the primary and secondary loops is lost and heat rejection through the sodium-to-air heat exchangers is assumed to decrease to zero. The reactor protection system responds to off normal conditions by initiating a full system shutdown. The PSBO is considered to be a bounding event for perturbations of the reactor core through flow rate changes.

When the primary pumps trip, core flow decreases to 60% and coasts down with a 12s initial flow halving time thereafter, as shown by the left side of Fig. 3. The secondary pumps, which do not have a coastdown mechanism, trip with a negligible coastdown of the pump head.

The beginning of the BOC PSBO transient is driven by an increasing power-to-flow ratio after the primary sodium pumps trip. The 115% high power-to-flow ratio threshold is soon exceeded, and the reactor protection system initiates a reactor shutdown at 0.5s. All sodium pumps and SAHXs have already tripped so the only remaining action for the RPS to take is to scram the control and safety rods.

Peak fuel, cladding, and coolant temperatures increase 5°C, 46°C, and 60°C, respectively, in the brief time before the control rods are scrammed and then promptly decrease with the decreasing power-to-flow ratio. Peak in-core temperatures are presented on the right side of Fig. 3.

Once the control rods scram, fission power is quickly terminated, leaving decay heat as the sole source of heat production in the reactor. RVACS heat rejection increases with the cold pool temperature, which increases less than 100°C, as shown on the left side of Fig. 4. Power and decay heat rejection are presented on the right side of Fig. 4. Decay heat continues to decrease and at fifty-eight hours, RVACS heat rejection has increased enough to match decay heat production. Beyond this point, decay heat will continue to decrease and RVACS heat rejection will cool down the system gradually.

Because the RPS threshold for the power-to-flow ratio is exceeded so quickly after the pumps trip, the progression of the EOC PSBO is nearly identical to the BOC PSBO. More negative reactivity is introduced by the scram because the control rods are fully withdrawn at EOC conditions. But the result is the same, that fission power is terminated leaving decay heat as the only contributor to total power. Decay heat slowly decreases and again at fifty-eight hours, RVACS heat rejection has increased enough to match decay heat production.

With the control rods scramming within the first second of the transient, all criteria defined above are met with very large margins for both the BOC and EOC PSBO.

 

Fig. . BOC PSBO – (Left) Core Mass Flow Rate and (Right) Peak In-Core Temperatures.

 

Fig. . BOC PSBO – (Left) Hot and Cold Pool Temperatures and (Right) Power and Decay Heat Rejection.

### Protected Transient Overpower

The protected transient overpower (PTOP) scenario is initiated by an unintentional withdrawal of the most reactive control rod, introducing a total of 95.7¢ of positive reactivity[[1]](#footnote-2), which is illustrated by the green line on the left side of Fig. 5. The rate of the control rod withdrawal is set by the mechanical speed of the control rod drive mechanism, which has not yet been determined. For analysis of TOP events, a withdrawal speed of 2.1 mm/s is assumed, which results in an average insertion rate of 0.5 ¢/s. This speed has been assessed to be faster than is necessary for a non-power producing reactor such as the VTR. The reactor protection system responds to off normal conditions by initiating a full system shutdown. The PTOP scenario is considered to be a bounding event for perturbations of the reactor core through reactivity changes.

Due to the positive reactivity of the withdrawing control rod, power increases, exceeding the 115% high power threshold at 17s, as illustrated by the right side of Fig. 5. The reactor protection system responds by initiating a reactor shutdown, beginning with a scram of the five remaining control rods and the three safety rods. Even as the first control rod continues withdrawing and introducing the remainder of the 95.7¢ of positive reactivity, the negative reactivity introduced by the scram is much larger and quickly terminates fission power.

As illustrated on the left side of Fig. 6, the peak fuel, cladding, and coolant temperatures increase 53°C, 29°C, and 25°C, respectively, before the remaining rods scram. Because the control rods are scrammed before power increases much beyond 115%, peak in-core temperatures do not challenge the limiting temperatures defined above. After the remaining control and safety rods are scrammed, the RPS trips the primary and secondary sodium pumps. Flow in the primary system coasts down following a similar profile to the coastdown defined above for the PSBO transient. The RPS then closes the sodium-to-air heat exchanger dampers and trips the air blowers, eliminating SAHX heat rejection and transferring long-term heat rejection duties to the RVACS.

Following these RPS actions, the long-term progression of the BOC PTOP is similar to the BOC PSBO. Throughout the remainder of the transient, RVACS heat rejection increases as the cold pool temperature increases. Decay heat continues to decrease and at fifty-nine hours, RVACS heat rejection has increased enough to match decay heat production, as illustrated on the right side of Fig. 6.

Because the control rods are nearly completely withdrawn at EOC conditions, the impact of a single rod withdrawal is benign. Therefore, the EOC PTOP is not presented here.

 

Fig. . BOC PTOP – (Left) Reactivity and (Right) Short-Term Power and Decay Heat Rejection.

 

Fig. . BOC PTOP – (Left) Peak In-Core Temperatures and (Right) Long-Term Power and Decay Heat Rejection.

### Protected Loss of Heat Sink

The protected loss of heat sink (PLOHS) accident is initiated by a simultaneous trip of the secondary sodium pumps, which significantly reduces heat rejection through the IHXs. Additionally, heat rejection through the SAHXs is assumed to be lost. The mechanism for this loss of SAHX heat rejection is not specifically defined here. It is included in the transient definition to demonstrate how the system responds and progresses with diminished heat rejection, with the RVACS providing the only source of heat rejection before the RPS detects elevated system conditions. Assuming zero SAHX heat rejection is conservative because in reality, even if the dampers for all ten SAHXs were completely closed, there would still be some heat rejected to the air and steel within the SAHXs. The current SAS4A/SASSYS-1 model also does not consider heat losses from the secondary sodium loops, which, though insulated, will provide some additional heat rejection.

Once elevated system conditions are detected, the reactor protection system responds by initiating a full system shutdown. Control rods are scrammed, the primary pumps are tripped, and the RVACS is responsible for rejecting all heat in the system. The PLOHS is considered to be a bounding event for perturbations of the reactor core through core inlet temperature changes.

The beginning of the PLOHS transient is driven by the loss of forced circulation in the secondary sodium loops. The large volume of sodium in the cold pool heats up gradually, with a correspondingly slow core inlet temperature increase, as illustrated by the left side of Fig. 7. This leads to the PLOHS proceeding more slowly than for either the PTOP or PSBO transients.

The core inlet temperature continues to increase and at 36s the 370°C high core inlet temperature threshold is exceeded. The reactor protection system then initiates a reactor shutdown. The coolant density and axial expansion feedbacks provide small amounts of negative reactivity, causing power to reduce slightly before the control rod scram is initiated by the RPS. Peak in-core temperatures do not increase significantly before the RPS shutdown is initiated. Peak cladding and coolant temperatures only increase by 7°C and 9°C, respectively, during the BOC PLOHS, while the fuel temperature remains below its initial value after the transient initiates. These peak temperatures are illustrated by the right side of Fig. 7.

 

Fig. . BOC PLOHS – (Left) Core Inlet and Outlet Temperatures and (Right) Peak In-Core Temperatures.

Once the control rods are scrammed, the RPS trips the primary pumps, leaving the system to progress similarly to the long-term progression presented above for the PTOP and PSBO transients. Fission power is quickly terminated, leaving decay heat as the sole source of heat production in the reactor. The left side of Fig. 8 illustrates the hot and cold pool temperatures. Over the sixty hours of simulation, the cold pool temperature increases less than 100°C above its initial 350°C and the hot pool temperature increases back up to 487°C, just below its nominal 500°C.

Decay heat continues to decrease and at fifty-four hours, RVACS heat rejection has increased enough to match decay heat production, as illustrated by the right side of Fig. 8. Beyond this point, decay heat will continue to decrease. RVACS heat rejection will cool down the system gradually. Because the large mass of sodium in the cold pool moderates how fast the core inlet temperature increases, all criteria defined above are met with very large margins.

 

Fig. . BOC PLOHS – (Left) Hot and Cold Pool Temperatures and (Right) Long-Term Power and Decay Heat Rejection.

The progression of the EOC PLOHS is nearly identical to the BOC PLOHS presented above. This is because small differences in the reactivity feedbacks generated at BOC and EOC conditions are mitigated by the size of the hot pool, so the IHX primary-side inlet temperature progression is nearly identical for both transients, ensuring a similar core inlet temperature increase and the same timing for the RPS response.

## CONCLUSIONS

At the current stage of design, transient simulation results for the Versatile Test Reactor indicate that large safety margins exist for many event initiators. The RPS dominates the transient behaviors in the protected transients presented above. Because the primary heat transport system is able to transition quickly and effectively to natural circulation and because the RVACS, which is responsible for decay heat removal, provides sufficient heat rejection, large margins for all criteria were predicted for the transients. Tables 2 and 3 list the peak fuel, cladding, and coolant temperatures as well as key safety margins. These results confirm that all criteria defined above are met with large margins. However, several “enabling” assumptions have been made, in terms of both design features and design limits, in order to perform the analyses. These assumptions will need to be revised as the design matures.

TABLE 2. BOC Transients – Peak Temperatures and Minimum Margins

|  |  |  |  |  |
| --- | --- | --- | --- | --- |
|  | InitialValue | PTOPValue | PLOHSValue | PSBOValue |
| Peak Fuel Temp. | 769°C | 822°C | 769°C | 774°C |
| Peak Cladding Temp. | 553°C | 582°C | 560°C | 599°C |
| Peak In-Core Sodium Temp. | 528°C | 553°C | 537°C | 588°C |
| Peak Bulk Sodium Temp. | 500°C | 503°C | 501°C | 501°C |
| Fuel Melting Margin | 350°C | 297°C | 350°C | 345°C |
| Sodium Boiling Margin | 430°C | 396°C | 410°C | 383°C |
| Bulk Coolant Margin | 149°C | 146°C | 148°C | 148°C |
| Cladding Margin | 96°C | 67°C | 89°C | 50°C |

TABLE 3. EOC Transients – Peak Temperatures and Minimum Margins

|  |  |  |  |
| --- | --- | --- | --- |
|  | InitialValue | PLOHSValue | PSBOValue |
| Peak Fuel Temp. | 751°C | 751°C | 756°C |
| Peak Cladding Temp. | 547°C | 555°C | 591°C |
| Peak In-Core Sodium Temp. | 520°C | 530°C | 578°C |
| Peak Bulk Sodium Temp. | 500°C | 501°C | 501°C |
| Fuel Melting Margin | 368°C | 368°C | 363°C |
| Sodium Boiling Margin | 438°C | 411°C | 393°C |
| Bulk Coolant Margin | 149°C | 148°C | 148°C |
| Cladding Margin | 102°C | 94°C | 58°C |

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1. Based on PRA results, the 95.7¢ reactivity insertion event may occur at a frequency below the “anticipated” or “unlikely” event categories described in Section 3 but it is utilized here as a conservative treatment of reactivity insertions within these categories. [↑](#footnote-ref-2)