# SAFETY ANALYSIS OF THE ARC-100 SODIUM-COOLED FAST REACTOR

T. SUMNER

Argonne National Laboratory

Argonne, IL, USA

Email: tsumner@anl.gov

A. MOISSEYTSEV

Argonne National Laboratory

Argonne, IL, USA

**Abstract**

The ARC-100 is an innovative 100 MW(e) sodium cooled fast reactor design being developed by Advanced Reactor Concepts, LLC. The safety design philosophy of the ARC-100 relies on passive safety, leveraging the safety characteristics of the metallic-fueled fast spectrum core, pool-type configuration with sodium coolant at atmospheric pressure, passive redundant decay heat removal by DRACS and RVACS, and engineering features such as the limited free bow core restraint system. To demonstrate and characterize the safety performance of the ARC-100 reactor, extremely low probability but potentially severe unprotected transients have been analyzed using the SAS4A/SASSYS-1 fast reactor safety analysis code. In all considered transients, it is assumed that the reactor protection system fails to respond to an initiating event, such that reactor power control is achieved by inherent reactivity feedbacks only. The results for all transients show that due to the passive characteristics of the ARC-100 reactor, large safety margins are maintained during these severe accidents, even without the support of the reactor protection system.

## INTRODUCTION

The ARC-100 is an innovative 100 MW(e) sodium cooled fast reactor design being developed by Advanced Reactor Concepts (ARC), LLC [1]. The reactor employs a long-lived metallic-fueled core with low burnup reactivity swing, which facilitates citing in remote locations and/or small grids with minimal reactor maintenance during its lifetime. The safety design philosophy of the ARC-100 relies on passive safety, leveraging the safety characteristics of the metallic-fueled fast spectrum core, pool-type configuration with sodium coolant at atmospheric pressure, passive redundant decay heat removal by direct reactor auxiliary cooling system (DRACS) and reactor vessel auxiliary cooling system (RVACS), and engineering features such as the limited free bow core restraint system. The pool-type reactor configuration with a guard vessel, low-pressure coolant, and no penetrations in the reactor vessel below the vessel head precludes the entire class of loss-of-primary-coolant accidents.

To demonstrate and characterize the safety performance of the ARC-100 reactor, extremely low probability but potentially severe unprotected transients have been analyzed. The work was performed using the SAS4A/SASSYS-1 safety analysis code for transient simulation of liquid metal-cooled fast reactors [2]. The paper presents the results and analysis of severe unprotected transients, including unprotected station blackout (USBO) with loss of power to all pumps, unprotected withdrawal of the most reactive control rod (transient overpower, UTOP), and unprotected loss of heat sink (ULOHS) with a complete loss of heat removal by the balance-of-plant.

In all transients considered below, it is assumed that the reactor protection system fails to respond to an initiating event, such that reactor power control is achieved by inherent reactivity feedbacks only. The reactor protection system is a highly reliable safety grade system and its assumed failure results in a very low expected frequency of occurrence.

## Model Description

Transient simulations for the safety analyses were performed using the fast reactor safety analysis code SAS4A/SASSYS-1, Version 5.3. Table 1 summarizes the model conditions at steady-state for beginning- (BOL), middle- (MOL), and end-of-life (EOL) conditions. The SAS4A/SASSYS-1 model of the ARC-100 core, primary and intermediate heat transport systems, and DRACS and RVACS decay heat removal systems is presented below.

TABLE 1. NOMINAL STEADY-STATE MODEL CONDITIONS

|  |  |
| --- | --- |
| Parameter | Value |
| Core Power | 286 MW(th) |
| Core Inlet Temperature | 355°C |
| Core Outlet Temperature | 510°C |
| Peak Fuel Temperature at BOL/MOL/EOL | 560°C/575°C/586°C |
| Peak Cladding Temperature at BOL/MOL/EOL | 548°C/542°C/551°C |
| Peak Coolant Temperature at BOL/MOL/EOL | 547°C/541°C/549°C |
| Core Mass Flow Rate | 1450 kg/s |
| Total Intermediate Mass Flow Rate | 1450 kg/s |
| Intermediate Cold Leg Temperature | 333°C |
| Intermediate Hot Leg Temperature | 487°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. A single-pin model represents the average fuel pin in an assembly and its associated coolant and structure. Assemblies with similar reactor physics and thermal-hydraulic characteristics are grouped together in SAS4A/SASSYS-1 channels.

The ARC-100 core is represented by 18 channels. Fig. 1 illustrates the assembly types and core channel assignments. Core assemblies were assigned to the SAS4A/SASSYS-1 channels based on the orifice flow zones. Channels 1-7 represent the nominal fuel channels, each representing one flow zone. Channels 8-11 represent the non-fuel assemblies, including control, reflector, and shield. Axial and radial power distributions for the ARC-100 were calculated using the nodal transport theory option of DIF3D/VARIANT [3]. Assembly flow rates were calculated with the SuperEnergy2-ANL steady-state thermal hydraulics code [4].

Channels 12-18, one for each orifice zone in the fuel region, are peak fuel channels that were developed to account for flow and temperature variations in each fuel orifice zone. Each of the seven peak channels simulates only one pin, not entire assemblies, with power and flow for those pins specified to match the peak coolant temperatures predicted in each orifice zone with the SuperEnergy-2-ANL code. Adding one peak channel for each orifice zone ensures that the peak channels not only represent the maximum power-to-flow ratios in each zone, but also provide the possibility to track the location of peak temperatures in various transients, where power-to-flow peaking could occur at different assemblies at different times.



Fig. 1. SAS4A/SASSYS-1 Channels and Core Assembly Types.

The ARC-100 SAS4A/SASSYS-1 model calculates the following reactivity feedback effects:

* Fuel Doppler,
* Axial fuel, cladding, and structure expansion,
* Coolant density,
* Radial core expansion driven by assembly load pad and grid plate expansions,
* Control rod driveline expansion (CRDL), and
* Reactor vessel expansion.

The fuel Doppler, axial expansion, and coolant density feedbacks are channel-dependent reactivity feedbacks, calculated based on temperatures within each channels. The radial core expansion, control rod driveline expansion, and reactor vessel expansion feedbacks are channel-independent reactivity feedbacks calculated based on average core temperatures or temperatures in the primary heat transport system. Point kinetics and reactivity feedback coefficients were calculated using PERSENT [5] and DIF3D [3].

The PRIMAR-4 module of SAS4A/SASSYS-1 simulates the thermal hydraulics of the heat transport systems. Compressible volumes, or CVs, are zero-dimensional volumes that are used to model large volumes of coolant such as inlet and outlet plena and pools. 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.

Fig. 2 illustrates the primary heat transport system model. The PRIMAR-4 model also represents the intermediate loops, DRACS loops, and RVACS. NaK flow through the DRACS loops as well as air flow through the DRACS air heat exchangers and RVACS are always driven by natural circulation; there are no pumps in either system.

Sodium enters the core, Segment 1, from the inlet plenum, CV1, and discharges into the hot pool, CV2. Segments 2 and 3 represent the parallel paths of primary sides of the two intermediate heat exchangers (IHX). Sodium discharges from the IHXs into the cold pool, which is represented by two compressible volumes. Most of the cold pool is represented by CV4, the lower portion, while the upper portion is represented by CV3. The four primary pumps are represented by Segments 5 and 6 with multiplicity factors of two so each segment represents two pumps. Segments 7 and 8 represent the primary sides of the submerged decay heat exchangers. Additionally, the hot pool and upper cold pool volumes share a common cover gas. Segment 4 represents sodium flow between the two cold pool volumes.

Each of the two intermediate loops, which were modeled separately to allow for asymmetric transient simulations in the future, include the intermediate side of the IHX, a pump, hot and cold leg piping, and the sodium-side of the steam generator. The steam generators are represented by the simple tabular heat exchanger model.

For the transient scenarios that were analyzed, it was conservatively assumed that one of the three DRACS units was offline for maintenance or otherwise unavailable and therefore rejected no heat at all. The two remaining DRACS units were modeled separately and identically. During normal operating conditions, each DRACS unit has a heat rejection rate of 0.067% of the total core power. When the DRACS are activated, DRACS heat rejection increases to 0.25% per unit at the nominal cold pool temperature.

In the RVACS model, heat rejected from the reactor vessel, which is represented by the walls of the two cold pool volumes, is transferred across a gas gap between the reactor vessel and guard vessel via radiative and convective heat transfer. Cold air supplied through the inlet stacks flows down along the outside of the collector cylinder, turns around, and then flows upward between the collector cylinder and guard vessel, cooling the outside of the guard vessel via convective heat transfer. Hot air is then discharged through the outlet stacks. During normal operating conditions, the RVACS has a heat rejection rate of 0.30% of the total core power.



Fig. 2. Primary Heat Transport System Model.

## Safety Criteria and Metrics

The three unprotected transient scenarios presented below are expected to have frequencies of occurrence below 10-6 per reactor-year. This assumption is to be confirmed by a probabilistic risk assessment. However, the unprotected transient scenarios are conservatively evaluated against criteria for beyond design basis events. The primary concern for these transients is maintaining a coolable pin geometry with a minimal number of fuel pin failures. Any change to the system preventing liquid coolant from flowing over the cladded fuel pins is considered unacceptable because sodium boiling and fuel pin failures can quickly result. Significant margins must be maintained to account for modeling and design uncertainties.

Four criteria were considered for these accidents:

* No sodium boiling,
* No fuel melting,
* Maintain reactor vessel integrity, and
* No significant loss of cladding from fuel-cladding chemical interaction.

At nominal steady-state BOL conditions, the ARC-100 is predicted to have a boiling margin of 396°C relative to a saturation temperature of 944°C at the outlet of the hottest assembly and a fuel melting margin of 659°C to a melting temperature for fresh fuel of 1223°C.

For this analysis, minimum margins of 100°C to sodium boiling and 300°C to fuel melting were further imposed to provide confidence that boiling and melting do not occur when uncertainties are considered. Significant margins are required for the sodium boiling and fuel melting criteria for these transients because those phase changes can occur very quickly and lead to a rapid reactivity introduction or material relocation. Cladding eutectic penetration, on the other hand, is a reaction that occurs very slowly at temperatures near an initial threshold. The eutectic model described in Reference 2 was applied for these analyses. If cladding temperatures remain above the 715°C threshold [2] for significant periods of time, eutectic penetration induced cladding rupture can occur, which could result in fission gas released from the fuel pin. Displaced coolant could produce a local positive reactivity insertion. Cladding temperatures can remain above the threshold temperature for short periods of time, however, as long as there is not a significant loss of cladding thickness.

Finally, because these transients are evaluated against criteria for BDBEs, the key metric for ensuring the coolant boundary does not fail is the Service Level D limits [6]. According to the PRISM PSID [6], temperatures of important structural components may not exceed 704°C in order to ensure structural stability. For conservatism, instead of applying this temperature limit to the reactor vessel and redan temperatures, the 704°C limit was applied to bulk coolant temperatures in the hot and cold pool.

## RESULTS

Results of transient simulations of the unprotected transient overpower, unprotected station blackout, and unprotected loss of heat sink scenarios are presented below. The results are presented first for beginning-of-life (BOL) conditions, with middle-of-life (MOL) and end-of-life (EOL) results summarized at the end of each section. Peak temperatures and minimum margins for all transients presented below are summarized in Section 5.

### Unprotected Station Blackout

The unprotected station blackout (USBO) transient is initiated by a postulated loss of electrical power to all plant systems. Forced circulation in the primary and intermediate loops is lost and heat rejection through the steam generators is assumed to decrease to zero. Heat rejection responsibilities transfer to the RVACS and DRACS, with the DRACS air baffles opening automatically via a failsafe mechanism following the loss of electrical power. When the primary sodium pumps trip, flow through the core decreases with a ten-second initial flow halving time via the energy provided by the primary pump coastdown mechanism. Intermediate flow coasts down much faster due to the absence of any coastdown mechanisms; the flow coastdown is driven by the inertia of the sodium in the intermediate loops as well as the elevations and density differences throughout the loops. The flow coastdowns in both the core and intermediate loops are shown on the left side of Fig. 3.

The beginning of the USBO transient is driven primarily by the resulting increase of the power-to-flow ratio in the core. With flow initially decreasing faster than core power, the power-to-flow ratio reaches 218% at 28s. The system gradually transitions to natural circulation and flow through the core begins to level off. During the first minute, negative reactivity generated by radial core expansion, Doppler, and axial core expansion insert nearly negative 20¢ collectively. The reactivity feedbacks are shown on the right side of Fig. 3. Power continues to decrease and the power-to-flow ratio reduces from its peak, dropping below 100% within three minutes. While the power-to-flow ratio is above 100%, the hot pool temperature increases approximately 20°C. This results in an expansion of the control rod drivelines, driving the control rods further into the core and introducing an additional negative 29¢ of reactivity to further reduce fission power. Because of the strong negative reactivity feedback response, fission power decreases below 1% within six minutes.

After the first hour, the system continues to gradually heat up, with most of the temperature increase occurring in the cold pool. Because heat rejection through the balance of plant is assumed to be lost, heat rejection duties fall to the RVACS and DRACS. The cold pool heating up from 355°C to 494°C by the end of the transient. Although this generates positive reactivity from reactor vessel expansion, which acts to pull the core down away from the control rods, the other reactivity feedbacks are more than sufficient to maintain a negative or zero net reactivity for the remainder of the transient. Additionally, the cold pool temperature increase is beneficial for long-term heat rejection because the heat rejection rates of both RVACS and DRACS are driven by the cold pool temperature. During the same period, the hot pool temperature increases 50°C.

The left side of Fig. 4 illustrates the long-term power production and decay heat removal. At approximately twenty-five hours, total power production has decreased sufficiently to match the total available heat rejection and system temperatures begin to decrease. The simulation was terminated at this time, with power and total decay heat removal stabilizing at 1.26% The available decay heat rejection is more than sufficient such that system temperatures remain low enough for fission power to stabilize at 0.7%.

The right side of Fig. 4 illustrates the peak in-core temperatures. The peak cladding temperature remains above the slow eutectic threshold for approximately one minute, limiting eutectic penetration to 0.1% of the cladding thickness in the hottest pins. Thus, there is no risk of pin failures, other than from stochastic issues related to faulty fuel pins. Significant margins to sodium boiling and fuel melting were maintained to ensure neither occurred during this transient. In the longer-term, peak temperatures in the core stabilized at approximately 560°C. And finally, because the hot pool temperature increase was limited to just 50°C, a margin of 144°C to the Service Level D limit was maintained to ensure structure integrity of the reactor vessel and redan. All criteria defined above were successfully met for the BOL USBO scenario.

The unprotected station blackout transient during middle-of-life conditions progresses similarly to the USBO during beginning-of-life conditions. As large safety margins were maintained for the BOL USBO, so too were large safety margins maintained for the MOL USBO. The biggest differences were slightly more negative reactivity from radial core expansion, which was nearly balanced by slightly more positive reactivity from the coolant density feedback. Cladding and coolant temperatures peaked 8°C and 9°C lower, respectively, at MOL while fuel temperatures peaked 11°C higher due to the build-up of plutonium in the fuel and the small resulting decrease in the fuel thermal conductivity.

 

Fig. 3. BOL USBO – (Left) Core Mass Flow Rate and (Right) Reactivity.

 

Fig. 4. BOL USBO – (Left) Power and Decay Heat Removal and (Right) Peak In-Core Temperatures.

While the BOL and MOL USBO transient progressions were predicted to be very similar, several differences were observed for the EOL USBO. These differences were due primarily to the control rods being withdrawn to the top of the fuel at EOL conditions, which reduced the magnitude of the CRDL expansion and reactor vessel expansion reactivity feedbacks. Temperatures in the core peaked higher at EOL conditions due to the lack of negative reactivity from control rod driveline expansion; however, large safety margins were still maintained. In the long-term, with significantly less positive reactivity from reactor vessel expansion, fission power is decreases more than during the either the BOL or MOL simulations. Total heat rejection matches power production after just four hours with smaller temperature increases in the hot and cold pool. After this point, temperatures in the system gradually decrease.

Table 2 presents the peak temperatures and minimum margins predicted for the unprotected station blackout transient at BOL, MOL, and EOL conditions, respectively.

TABLE 2. USBO PEAK TEMPERATURES AND MINIMUM MARGINS

|  |  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- | --- |
|  | BOL | | MOL | | EOC | |
|  | Steady-State | Transient | Steady-State | Transient | Steady-State | Transient |
| Peak Fuel Temp. | 564°C | 755°C | 575°C | 751°C | 586°C | 805°C |
| Fuel Melting Margin | 659°C | 468°C | 640°C | 464°C | 615°C | 396°C |
| Peak Cladding Temp. | 548°C | 752°C | 544°C | 744°C | 551°C | 795°C |
| Max. Clad Penetration | 0% | 0.1% | 0% | 0.1% | 0% | 0.6% |
| Peak Sodium Temp. | 547°C | 751°C | 544°C | 742°C | 549°C | 798°C |
| Sodium Boiling Margin | 396°C | 193°C | 400°C | 201°C | 395°C | 148°C |
| Peak Hot Pool Temp. | 510°C | 560°C | 510°C | 564°C | 510°C | 538°C |
| Margin to Service Level D | 194°C | 144°C | 194°C | 140°C | 194°C | 166°C |

### Unprotected Transient Overpower

The unprotected transient overpower (UTOP) scenario is initiated when the single most reactive control rod is withdrawn from the reactor. At BOL conditions this introduces 47¢ of positive reactivity, while for MOL conditions 34¢ of positive reactivity is inserted. In both cases, an average reactivity insertion rate of 2 ¢/s was assumed. This transient was not simulated for EOL conditions because the control rods are withdrawn to the top of the core and one rod withdrawing inserts a negligible amount of reactivity. The steam generators are conservatively assumed to continue performing at their 100% pre-transient heat rejection rate, which results in slight undercooling because steam generator heat rejection efficiency will increase as the temperature difference between sodium in the intermediate loop and water in the steam generators increases.

During the BOL UTOP, when the control rod withdrawal begins, fission power begins to increase due to the introduction of positive reactivity. Although a single control rod withdrawal introduces 47¢ of positive reactivity at BOL conditions, net reactivity is predicted to reach only 19¢. While the control rod is still withdrawing, the Doppler and radial expansion reactivity feedbacks contribute the most negative reactivity. The temperature of sodium discharged from the core increases and the control rod drivelines heat up and expand, driving the control rods further into the core. The CRDL feedback becomes the dominant source of negative reactivity, and these large negative reactivity contributions more than compensate for the reactivity from the withdrawn rod. Net reactivity becomes negative at 34s, 10s after the control rod withdrawal terminates, as illustrated by the left side of Fig. 5. The right side of Fig. 5 illustrates core power, which reaches a maximum of 258% at 24s. Core power then reduces below 200% and gradually decreases to match the total available heat rejection.

The left side of Fig. 6 illustrates the hot and cold pool temperatures, which increase 141°C and 132°C, respectively. The temperature increases are necessary to generate the negative reactivity to compensate for the 47¢ of positive reactivity from the control rod withdrawal and return the core to 100% power. Despite an increase of 141°C in the hot pool, a margin of 53°C is maintained to the Service Level D limit.

The right side of Fig. 6 illustrates the peak fuel, cladding, and coolant temperatures. For the BOL UTOP, the fuel melting margin is 362°C and the sodium boiling margin is 122°C. The peak cladding temperature exceeds the slow eutectic threshold, but quickly decreases and remains below the threshold for most of the rest of the transient, resulting in a total eutectic penetration of 0.5% over the full thirty-minute simulation. There is no immediate risk of pin failures, other than from stochastic issues related to faulty fuel pins. All criteria previously defined were successfully met.

Because the critical position of the control rods is higher in the core at MOL conditions than at BOL conditions, a single rod withdrawal during a UTOP scenario results in a smaller reactivity insertion at MOL. This results in smaller temperature increases for the MOL UTOP scenario, where the fuel melting margin is 443°C and the sodium boiling margin is 231°C.

Table 3 presents the peak temperatures and minimum margins predicted for the unprotected transient overpower scenario at BOL and MOL conditions, respectively.

 

Fig. 5. BOL UTOP – (Left) Reactivity and (Right) Core Power.

 

Fig. 6. BOL UTOP – (Left) Hot and Cold Pool Temperatures and (Right) Peak In-Core Temperatures.

TABLE 3. UTOP PEAK TEMPERATURES AND MINIMUM MARGINS

|  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- |
|  | BOL | | MOL | | |
|  | Steady-State | Transient | Steady-State | Transient |
| Peak Fuel Temp. | 564°C | 861°C | 575°C | 772°C |
| Fuel Melting Margin | 659°C | 362°C | 640°C | 443°C |
| Peak Cladding Temp. | 548°C | 836°C | 544°C | 722°C |
| Max. Clad Penetration | 0% | 0.5% | 0% | 0% |
| Peak Sodium Temp. | 547°C | 833°C | 544°C | 720°C |
| Sodium Boiling Margin | 396°C | 122°C | 400°C | 231°C |
| Peak Hot Pool Temp. | 510°C | 651°C | 510°C | 604°C |
| Margin to Service Level D | 194°C | 53°C | 194°C | 100°C |

### Unprotected Loss of Heat Sink

The unprotected loss of heat sink transient is initiated by a simultaneous trip of the intermediate sodium pumps, which significantly reduces heat rejection through the IHXs. Additionally, all heat rejection through the balance of plant is assumed to be lost. The mechanism for this loss of the balance of plant heat rejection is not specifically defined; it is included as part of the transient definition in order to demonstrate the system’s response and progression when the normal heat rejection pathway is completely lost. It is also assumed that the DRACS air dampers open following the loss of heat rejection through the balance of plant. Long-term heat rejection responsibilities, therefore, transfer to both the DRACS and RVACS.

When the intermediate pumps trip at the start of the unprotected loss of heat sink, heat rejection through the IHXs significantly reduces. The ULOHS transient is driven by the resulting increase of the cold pool temperature, which causes the core inlet temperature to increase and results in negative reactivity feedbacks that reduce core power. The left side of Fig. 7 illustrates the core inlet and outlet temperatures during the beginning of the transient. Over the first ten minutes of the transient, the core inlet temperature increases to 492°C, while the core outlet temperature increases up to 519°C and then gradually decreases below its nominal 510°C due to the negative reactivity feedbacks decreasing power. Due to the slower progression of the transient, the magnitudes of the reactivity feedbacks are smaller than for the other transients. Although net reactivity only reaches a minimum of negative 6¢, core power reduces below 20% within seven minutes. Fission power reduces below decay heat within ninety minutes, leaving decay heat and pump heating from the still active primary sodium pumps as the main sources of heat generation in the plant, as illustrated by the right side of Fig. 7.

The BOL ULOHS transient was simulated for five hours as this is how long it takes for the RVACS and DRACS heat rejection to match total core power and primary pump heating. After five hours, temperatures throughout the system will stabilize and begin to decrease following the gradual decrease of decay heat. The hot and cold pool temperatures stabilize within a few degrees of the nominal hot pool temperature due to total power reducing below 2% and core flow remaining at full-flow conditions. Although the core inlet temperature increases significantly, peak temperatures in the core do not increase much from their initial pre-transient values. The peak fuel, cladding, and coolant temperatures only increase 4°C, 6°C, and 6°C, respectively. Because of these small temperature increases, safety margins are not significantly reduced for the ULOHS transient.

The ULOHS transient during both MOL and EOL conditions progresses similarly to the BOL ULOHS. In all three cases, peak temperatures in the core increase less than 20°C and large safety margins are maintained even as the cold pool temperature increases up to the hot pool temperature. The biggest difference between the three conditions is that during EOL conditions, the control rods are already fully withdrawn. Therefore, the positive reactivity feedback from the cold pool temperature increasing, which causes the reactor vessel to expand and pull the core down away from the control rod, is significantly smaller. Consequently, fission power decreases to negligible levels after only twenty minutes during the EOL ULOHS.

Table 4 presents the peak temperatures and minimum margins predicted for the unprotected loss of heat sink transient at BOL, MOL, and EOL conditions, respectively.

 

Fig. 7. BOL ULOHS – (Left) Core Inlet and Outlet Temperatures and (Right) Power

TABLE 4. ULOHS PEAK TEMPERATURES AND MINIMUM MARGINS

|  |  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- | --- |
|  | BOL | | MOL | | EOC | |
|  | Steady-State | Transient | Steady-State | Transient | Steady-State | Transient |
| Peak Fuel Temp. | 564°C | 568°C | 575°C | 580°C | 586°C | 598°C |
| Fuel Melting Margin | 659°C | 655°C | 640°C | 635°C | 615°C | 603°C |
| Peak Cladding Temp. | 548°C | 554°C | 544°C | 550°C | 551°C | 569°C |
| Max. Clad Penetration | 0% | 0% | 0% | 0% | 0% | 0% |
| Peak Sodium Temp. | 547°C | 553°C | 544°C | 550°C | 549°C | 568°C |
| Sodium Boiling Margin | 396°C | 388°C | 400°C | 391°C | 395°C | 371°C |
| Peak Hot Pool Temp. | 510°C | 516°C | 510°C | 516°C | 510°C | 530°C |
| Margin to Service Level D | 194°C | 188°C | 194°C | 188°C | 194°C | 174°C |

## CONCLUSIONS

A series of severe unprotected transient scenarios were analyzed with SAS4A/SASSYS-1 to assess the safety characteristics of the ARC-100 design. The postulated transients are very unlikely double-fault events with an assumed failure of the highly reliable reactor protection system. During these transients, the ARC-100 is able to maintain large safety margins due to the design features that utilize inherent passive responses to unanticipated conditions and equipment failures. The sodium coolant provides superior heat removal and transport characteristics at near atmospheric pressures with large margins to boiling, while the metallic fuel operates at relatively low temperatures due to its high thermal conductivity. The pool-type primary system provides a large thermal capacity while allowing for shutdown heat removal through the RVACS and DRACS systems via natural circulation.

Without a reactor protection system to scram the control rods, the inherent reactivity feedbacks are responsible for reducing power to match the total available heat rejection provided by the RVACS and DRACS, as well as the balance of plant if it is assumed to remain operational. The results of these accident simulations demonstrate the capability of the ARC-100 design to provide protection against reactor damage during low probability accident sequences resulting from multiple equipment failures.

ACKNOWLEDGEMENTS

The submitted manuscript has been created by UChicago Argonne, LLC, Operator of Argonne National Laboratory (“Argonne”). Argonne National Laboratory’s work was supported by Advanced Reactor Concepts, LLC under the Strategic Partnership Project Agreement No. 854V0: The ARC Reactor.

References

1. Advanced Reactor Concepts, LLC (2021), https://www.arcenergy.co
2. FANNING, T. H., et. al. eds., The SAS4A/SASSYS-1 Safety Analysis Code System, Version 5, ANL/NE-16/19, Nuclear Engineering Division, Argonne National Laboratory, March 31, 2017.
3. R. D. Lawrence, “The DIF3D Nodal Neutronics Option for Two- and Three- Dimensional Diffusion Theory Calculations in Hexagonal Geometry,” ANL-83-1, Argonne National Laboratory (1983).
4. BASEHORE, K. L. and TODREAS, N. E., 1980. SUPERENERGY-2: A Multiassembly, Steady- State Computer Code for LMFBR Core Thermal-Hydraulic Analysis. Technical Report. Pacific Northwest Laboratory, PNL-3379, COO-2245-57TR.
5. M. A. Smith, et al., “VARI3D & PERSENT: Perturbation and Sensitivity Analysis,” ANL/NE-13/8, Argonne National Laboratory, Argonne, IL (2013).
6. PRISM Preliminary Safety Information Document, GEFR-00793, Chapter 15, Volume IV.