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DOE-CEA Benchmark on SFR ASTRID Innovative Core: *Neutronic and Safety Transients Simulation*

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Abstract. ASTRID is a fast reactor being designed by CEA to achieve a level of safety that exceeds that of conventional fast reactors. In particular an axially heterogeneous core with an upper sodium plenum is employed to achieve a non-positive sodium void reactivity worth. In order to address the simulation challenges for this innovative concept, the USDOE Laboratories (ANL and INL) and CEA are performing neutronic and transient benchmark calculations for an ASTRID model based on design specifications provided by CEA. The blind comparison of the initial DOE and CEA results are found to be in good agreement, enhancing confidence in CEA predictions of key ASTRID safety-relevant parameters and transient behavior. For several parameters compared uncertainties in computed values are significant, and further studies are needed to reduce them.

1. Introduction

The Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) is a fast neutron reactor being designed by CEA and its industrial partners to achieve a high level of safety.[1] In particular, an axially heterogeneous core with an upper sodium plenum is employed to achieve a near zero global sodium void worth and promote inherently safe behavior of the core during unprotected transients. Under a framework agreement between U.S. and France for cooperation in low carbon energy technologies, Alternative Energies and Atomic Energy Commission (CEA) and U.S. Department of Energy (USDOE) are performing neutronics and transient benchmark calculations for a model of the current preliminary ASTRID design. The USDOE contributions to the neutronic and transient calculations are being performed by the technical experts at the Idaho and Argonne National Laboratories, respectively.

The primary objective of the collaboration is an assessment of key neutronics performance parameters and safety characteristics of the specified ASTRID configuration for a limited set of transients:

- Comparison of calculated k_{eff}, key reactivity feedback coefficients, and power distributions including the uncertainties and the effects of these uncertainties.
- Comparison of safety margins for two anticipated transients without scram.

After an intense team effort devoted to performing the benchmark calculations independently by both sides based on specifications provided by CEA, the first joint technicalmeeting was held in Saclay in December 2012. The results included in this paper reflect the blind comparisons of the calculations presented at the December meeting. Both the neutronics and transient analyses results obtained by the US and French teams were found to be in reasonably good agreement. For the integral sodium void worth, the discrepancy is found to be lower than \$0.5 with the calculated core power distribution, neutronics flux shape, and reactivity feedback coefficients being quite similar. The results of the selected transient benchmarks for the unprotected loss of heat sink and station blackout scenarios also compare favorably, confirming the applicability of both DOE and CEA codes for analysis of the key safety characteristics of this novel, axially heterogeneous ASTRID core design.

2. ASTRID Core Benchmark Model and Transient Specifications

ASTRID is 1500 MWth, sodium-cooled, oxide-fueled, pool-type fast reactor prototype.[1] A primary objective of the design is to obtain inherent safety with sufficiently large margins to sodium boiling and core melting during unprotected accidents, but also to achieve a negative internal void worth against even more severe hypothetical accidents. A simplified benchmark model of ASTRID has been created by CEA for comparative neutronics and transient analyses to confirm the key safety characteristics of the core and the plant design. Several aspects of the complete ASTRID design including the dedicated decay heat removal systems are disregarded for this benchmark model.

The axial core configuration is illustrated in Figure 1. This axially heterogeneous core configuration was designed to maximize the sodium leakage to achieve a negative overall sodium void worth. Two special zones are used to increase neutron leakage at the top of the core: First, a large sodium plenum is positioned immediately above the top of the core to increase the leakage probability. Second, placing a fertile blanket zone in the axial center of the inner core increases the flux at the top of the core.

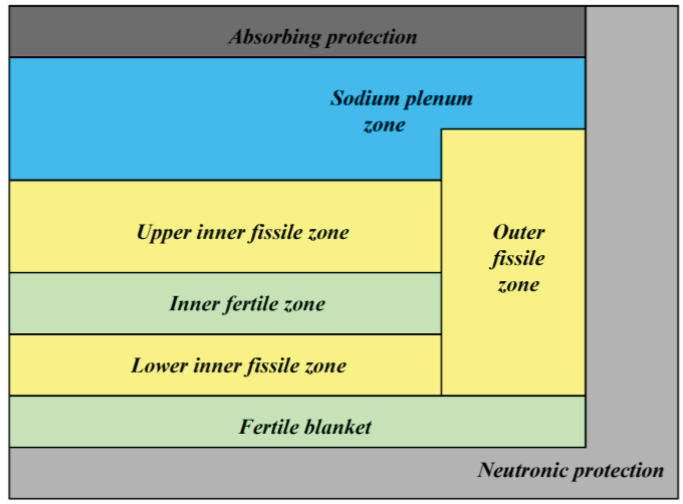

FIG. 1. Axial View of Core

The ASTRID core layout is illustrated in Figure 2. The inner core region has 177 fuel assemblies each with an active core height of 1.1 m in two fissile and two fertile regions. The outer fuel region has 114 fuel assemblies with an active core height of 1.2 m. To increase the flux at the edge of the core, the outer fuel assemblies are slightly taller and do not have an inner axial blanket.

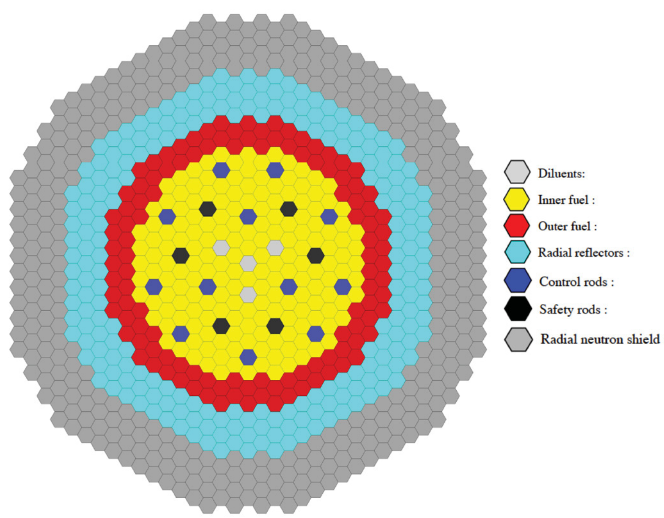

FIG. 2. Core Layout

The schematic of the ASTRID's primary sodium heat transport system is illustrated in Figure 3. The primary sodium system has three pumps and four intermediate heat exchangers (IHX). About 95% of the total flow from the pumps enters the core at 673 K and discharges at the top of the core at 823 K. The core outlet volume is part of the hot pool but, because it contains a large number of structural components, it is treated as a separate volume. Most of the sodium in the core outlet volume flows directly into the hot pool but 10% of the sodium flows up through the control plug volume and washes over the control rod drives.

In the diagrid, the 5% of total flow that does not enter the core flows upward along the reactor vessel wall through one of sixteen overflow pipes. This sodium then flows down and cools the outside of the hot pool shell, discharging back into the cold pool. For the benchmark model, the IHX intermediate side inlet temperature and sodium flow rate are specified as boundary conditions at 618 K and 1636 kg/s per IHX, respectively. The rest of the intermediate sodium loop is ignored in the benchmarks.

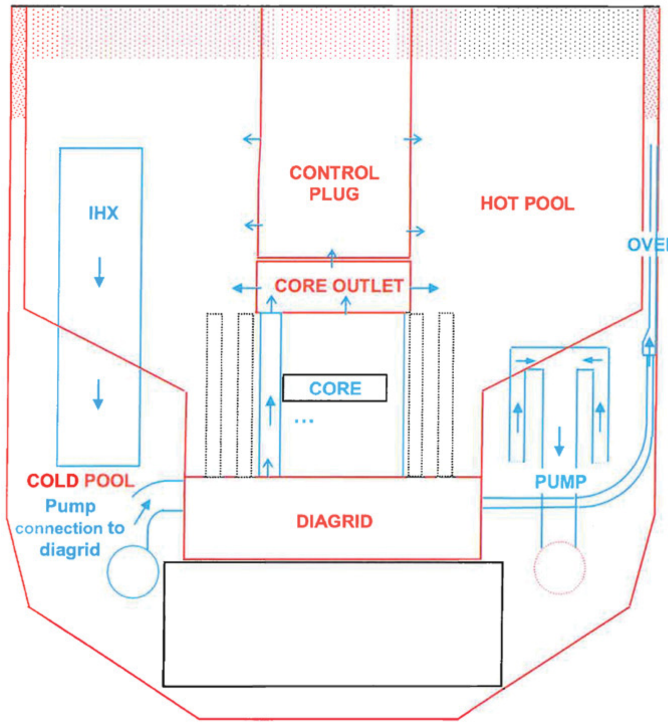

FIG. 3. Primary Sodium System

Two unprotected transients have been defined for the comparative analyses. The first transient is an unprotected loss of heat sink (ULOHS), which is an unscrammed intermediate pump trip defined by the mass flow rate vs. time values provided by CEA. The second transient is an unprotected loss of supply station power (ULOSSP), which is an unscrammed station blackout with pump trips in both the primary and intermediate loops, also defined by pump speed vs. time values provided by CEA.

3. Neutronic Benchmark

As part of the neutronics benchmark, the core multiplication factor, delayed neutron parameters, mass for main actinides, integral power per fuel assembly, maximal volumetric and linear power radial distributions, linear power axial profiles, maximal flux radial distribution, flux axial profiles, absorber rod bank worth, and Doppler constants were evaluated independently by the teams at CEA and INL. The sodium void effect was evaluated for the following three fuel assembly conditions: (1) draining of the sodium plenum, upper gas plenum, upper pin plugs, fissile, and axial inner blanket; (2) draining of the sodium plenum, upper gas plenum, upper pin plugs, and upper fissile; and (3) draining of the sodium plenum only. The reactivity feedback coefficients needed for the transients benchmarks were evaluated for the thermal expansion of sodium, fuel clad, wrapper, fuel/fertile pins, and the diagrid.

The codes used by CEA for the neutronics benchmarks include ERANOS[2], including SNATCH[3,4] solver, and TRIPOLI[5], both using JEFF3.1 nuclear data set.. At INL, the neutron transport calculations have also been carried out with both stochastic (MCNP5[6]) and deterministic codes (ECCO[7]/VARIANT[8]/BISTRO[9]/H3D-finite difference diffusion from the ERANOS code system). The INL team used the ENDF/B-VII.0 library as reference set for neutron cross-sections, but some calculations have been repeated using JEFF3.1 data and ENDF/B-VII.1. In both groups' calculations, the geometry has been represented with the maximum fidelity allowed (i.e., heterogeneous description of fuel assemblies) as shown in Figure 4.

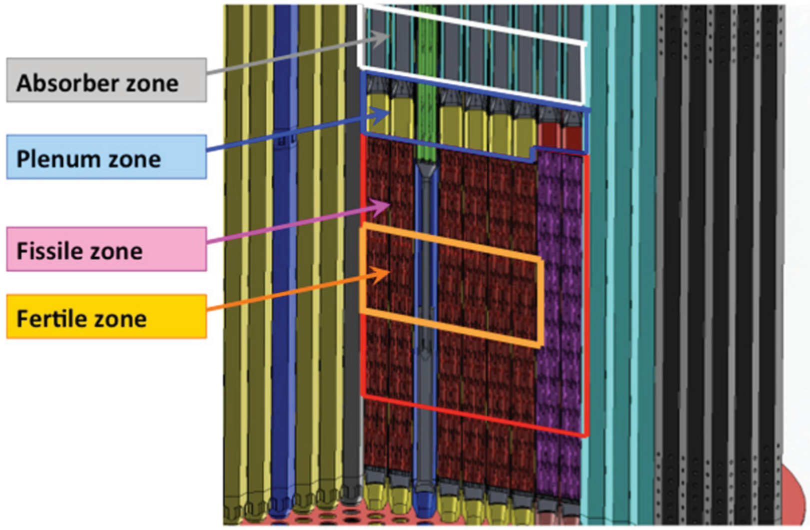

FIG. 4. ASTRID Core Layout Considered in the Neutronic Benchmark.

All calculations were performed using the ASTRID benchmark specifications for a BOL core configuration with all materials, geometries, and cross section data at (1) reference parameters for 20ºC, and (2) for nominal operations conditions. In the latter case, the geometry was expanded and material densities adjusted to conserve mass. Most regions of the core are assumed at a temperature of 750 K, the lower axial blanket, with associated pins, clad, fuel assembly wrapper, and sodium is at 900 K, the fuel and internal axial blanket, with associated pins, clad, S/A wrapper, and sodium, is at 1500 K.

The US and French values for neutronics parameters are found to be in reasonably good agreement. A comparison of the global parameters at 20°C and at nominal power for a BOL core is provided in Table I. The whole core sodium void worth calculated by both teams are also compared in Table II. And a comparison of the zone-dependent sodium void worth at nominal power for a BOL ASTRID core is provided in Table III for three separate voided configurations as shown in Figure 5: The whole core, the upper fissile zone + Na plenum, and Na plenum only.

Table I. A comparison of CEA and DOE (INL) results of BOL core reactivity (pcm)

	ERANOS – JEFF-3.1 CEA	MNCP – ENDF/BVII.0 DOE	DELTA CEA-DOE
20 °C	6764	6280	484
NP	4640	4205	435
20°C → NP	2124	2075	49

Table II. A comparison of CEA and DOE (INL) results of BOL Whole Core Sodium Void Worth

	ERANOS – JEFF-3.1 CEA	MNCP – ENDF/BVII.0 DOE	DELTA CEA-DOE
20 °C	-3.7 $	-4.1 $	0.4 $
NP	-2.9 $	-3.3 $	0.4 $
20°C → NP	-0.8 $	-0.8 $	0

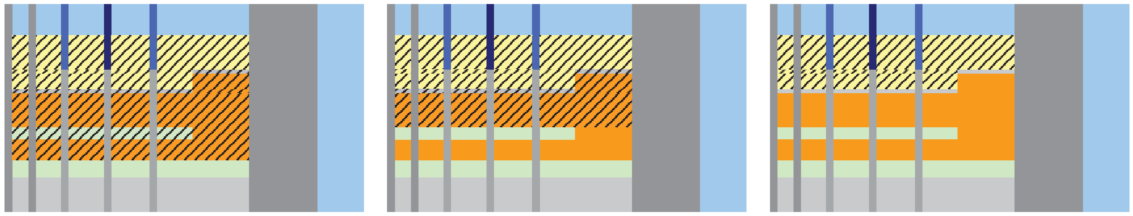

FIG. 5. Configurations for sodium void worth calculations (voided regions are shaded areas)

Table III. A comparison of CEA and DOE (INL) results of BOL Zone-Dependent Sodium Void Worth at Nominal Power.

Voided znoe	ERANOS – JEFF-3.1 CEA	MNCP – ENDF/BVII.0 DOE	DELTA CEA-DOE
Whole core	-2.9 $	-3.3 $	0.4 $
Upper fissile and plenum	-4.9 $	-5.1 $	0.2 $
Plenum only	-5.1 $	-5.4 $	0.3 $

The results indicate that the library bias from cross-section data set differences (ENDF vs. JEFF) as calculated with MCNP code is about 500 pcm and, for the sodium void worth, the maximum discrepancy is about $0.4. Doppler (temperature) reactivity effects were also compared (see Table IV) and differences were found to be relatively small, assuring good consistency for input data needed for the transient codes used in the subsequent safety analysis. The calculated power and neutronics flux shape, feedback coefficients are also quite similar, as further indication of the consistency between U.S. and French teams' results.

Table IV. A comparison of CEA and DOE (INL) results of BOL Zone-Doppler (temperature) reactivity effects at Nominal Power.

Doppler Effect (pcm)		
	ERANOS – JEFF-3.1 CEA	MNCP – ENDF/BVII.0 DOE
Everything + 1000K	-535	-494
Fuel @ 2500K	-338	-315
Blanket @ 1900K	-157	-149
Clad @ 2500 & 1900K	-19	-12
Wrapper @ 2500 & 1900K	-21	-18

In general, the observed differences stay within estimated uncertainties. A thorough, but still not exhaustive, uncertainty analysis was carried out by the two teams for the main integral parameters of interest, and nuclear data library effects were also studied in detail. In particular, effects of use of JEFF3.1 (the reference library for CEA) with respect to ENDF/V-VII.0 (the reference library for DOE) were broken down in terms of isotopes, reactions, and energy range using perturbation theory on a simplified R-Z model. A similar detailed analysis was performed by the two teams for the uncertainty evaluation where CEA used the COMAC[10] covariance matrix data and DOE used COMMARA 2.0[11]. The main differences and uncertainties were found to be associated to: ^{238}Pu (fission and capture), ^{239}Pu (fission), ^{23}Na (elastic and inelastic), and ^{238}U (inelastic). Table V summarizes data effects and uncertainties for k_{eff} and whole core sodium void worth.

Table V. Summary of data bias and uncertainty evaluation on k_{eff} and sodium void worth.

Parameter (1σ)	Library bias	Uncertainties (DOE)	Uncertainties (CEA)
Keff	500 pcm	1132 pcm	1434 pcm
$\Delta\rho_{Na}$	0.5 \$	0.5 \$	0.5 \$

The consistency of the results obtained with different nuclear data and codes is considered encouraging. However, the nuclear data uncertainties seem to dominate void worth estimates and need to be assessed particularly for local behaviors. The total uncertainties and bias in void worth is estimated to be about \$2 when considering three sigma uncertainty range and conservative (not statistical) uncertainty component combinations. Although expected to be small, consequences of these uncertainties on transient behaviors also need to be assessed.

4. Transient Analyses

CEA and Argonne teams independently developed their transient analysis models for the ASTRID benchmarks using their CATHARE-2[12] and SAS4A/SASSYS-1[13] systems analysis codes, respectively. With both codes, the geometry of ASTRID primary heat transport system is represented by a series of perfectly-mixed control volumes connected by liquid segments for modeling the sodium flow through the core, pumps, intermediate heat exchangers, control plug and overflow regions shown in Figure 3. Although the ASTRID design includes dedicated decay heat removal systems in the primary coolant hot and cold pools, they are excluded in the benchmark specifications and the ultimate heat sink is assumed to be constant ambient temperature outside the reactor vessel.

A single pin/channel model is assumed to characterize the fuel, coolant, and structure of an average pin in a fuel assembly. The fuel assemblies with similar reactor physics and thermal-hydraulics characteristics are grouped together to form a channel. The initial SAS4A/SASSYS-1 model of ASTRID core had two channels due to lack of details for the core flow distribution: Channel 1 represented the 177 inner fuel assemblies in an average sense, and Channel 2 represented the 114 outer fuel assemblies (i.e., the SAS4A/SASSYS-1 model currently calculates the peak temperature in two average fuel assemblies). The CATHARE-2 model of the ASTRID core has seven channels based on the estimated power density distributions in the core using the results of he neutronics benchmark. Because detailed design information for the other assembly types is not included in the benchmark, both the CATHARE-2 and SAS4A/SASSYS-1 core models are limited to the fuel assemblies.

The axial and radial power distributions as well as the reactivity feedback coefficients for the transient benchmark models come from the respective neutronics analyses in the France and U.S. Therefore, although estimated to be small, the transient analyses include the discrepancies in the neutronic benchmark. To obtain the flow distribution among the core channels, the mass flow in each channel is adjusted to give an outlet temperature of 823 K. Both the CATHARE-2 and SAS4A/SASSYS-1 reactivity feedback models include the Doppler, sodium density, fuel axial expansion, and core radial expansion. The control rod driveline expansion reactivity feedback was also included in the CATHARE-2 model but excluded in the initial SAS4A/SASSYS-1 calculations due to lack of design information. With CATHARE-2, the radial core expansion is assessed solely based on the diagrid expansion as a conservative approach, whereas with SAS4A/SASSYS-1, the expansion of the above-core load pads was also considered leading to a less conservative assumption.

SAS4A/SASSYS-1 currently has the capability to model upper and lower blankets but not internal blankets. For these simulations, the inner axial blanket region is modeled as MOX fuel for thermal-hydraulic purposes. The net fuel expansion feedback effect in the inner blanket is expected to be small during the transients; therefore, neglecting it is not assumed to lead to a significant discrepancy.

The results from the independently created Argonne and CEA models were compared for the ULOHS and ULOSSP transient scenarios for the first time at a meeting held in December 2012, and are presented here. The U.S. and French results compare reasonably well with similar trends for power and temperature histories and consistent timing of peak fuel and coolant temperatures, confirming the applicability of both USDOE and CEA codes for analysis of the ASTRID core. Estimated large margin to coolant boiling by both sides during both transients suggests that core damage or sodium voiding could be avoided for the ASTRID configuration studied in this benchmark.

During the ULOHS sequence, the early portion of the transient is driven by an increased inlet core temperature, which heats up the diagrid and causes a negative radial core expansion reactivity feedback. As power decreases, the negative radial expansion feedback is partially countered by a positive Doppler feedback as the core cools down. Around 100 seconds, the net reactivity reaches -6¢ for the Argonne model and -7¢ for the CEA model. Over the next 700 seconds, the net reactivity increases to -3¢ driven mostly by Doppler and also CRDL expansion in the CEA model. Around 800 seconds, net reactivity begins to decrease again as the diagrid continues to heat up, and at 3000 seconds it reaches -10¢ for the Argonne model and -9¢ for the CEA model.

As fission power decreases inherently during the ULOHS sequence, the core inlet and outlet temperatures converge as shown in Figure 6. Since the primary sodium coolant continues to remove the power at decay heat levels from the core at full flow, about 60 K difference between the asymptotic temperatures is attributed to differences in hydraulic resistances and rate of heat transfer to the ultimate heat sink (constant ambient temperature outside the reactor vessel). To better understand these differences, heat transfer between the hot and cold pools and the flow distribution between the core and overflow region are currently being evaluated and compared.

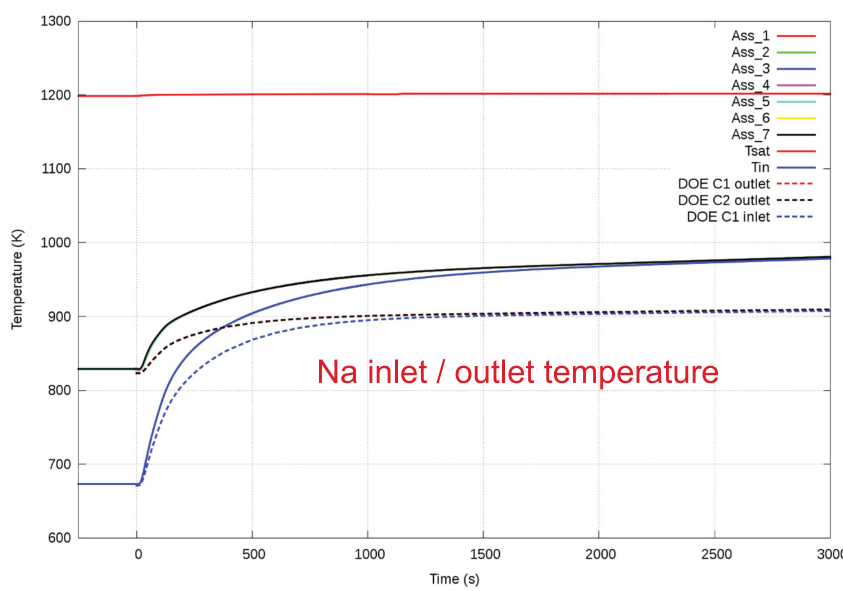

FIG. 6. ULOHS Inlet and Outlet Temperatures

The CATHARE-2 and SAS4A/SASSYS-1 models also predict similar progressions for the ULOSSP transient. While the beginning of the ULOHS transient is driven by increasing core inlet temperatures, the ULOSSP transient is driven by an increase in power to flow ratio that approaches a peak of 3. As the flow rate levels off, elevated temperatures in the core keep the net reactivity negative, even as the Doppler reactivity feedback from the cooling fuel inserts ~80¢ of reactivity. This leads to a sudden drop in power level as shown in Figure 7. While the SAS4A/SASSYS-1 model initially excluded the CRDL reactivity feedback, it predicts a more negative radial expansion reactivity feedback since it considers expansion of the above-core load pads. These compensation factors leads to similar trend for the net reactivity between the CATHARE-2 and SAS4A/SASSYS-1 models as illustrated in Figure 8 (the oscillatory appearance of the SAS4A/SASSYS-1 calculated net reactivity curve in Figure 8 is due to the table look-up method for modeling pump-coast-down).

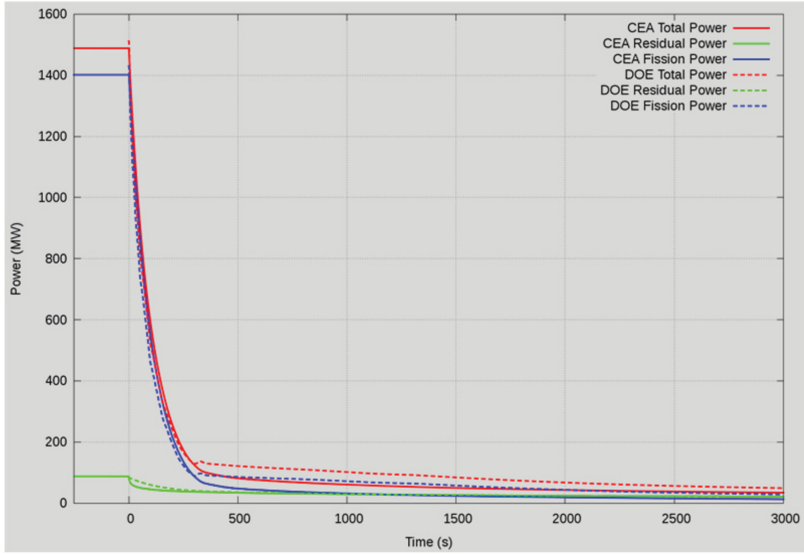

FIG. 7. Comparison of reactor power history for ULOSSP

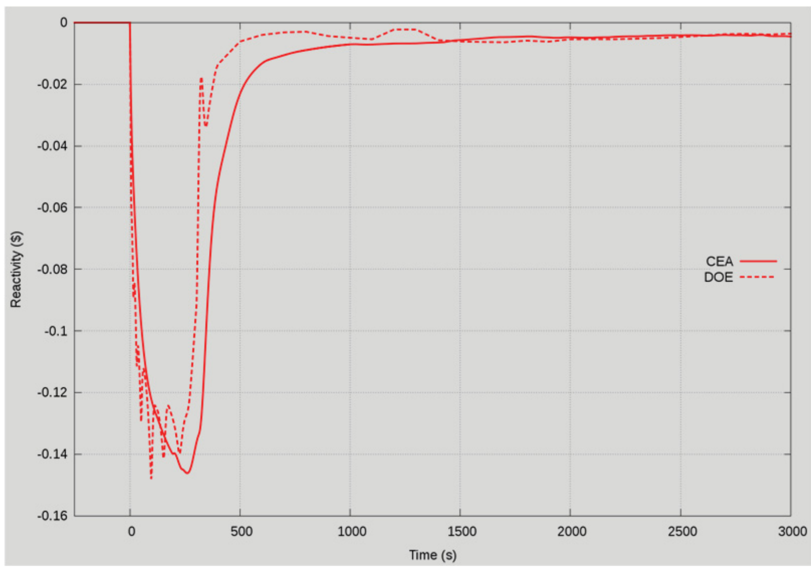

FIG. 8. Comparison of net reactivity history for ULOSSP.

Figure 9 illustrates the core inlet and outlet temperatures predicted by both models for the ULOSSP transient. Because the primary flow rate decays away, the inlet and outlet temperatures do not converge like in the ULOHS transient. As the cold pool heats up, both models predict the core inlet temperature would rise to ~800 K by 3000 seconds. Although the coolant outlet temperatures also follow a similar trend and reach to peak temperatures at about the same time (around 5 min mark), the predicted margin to coolant boiling differs by about 50 K between the two models. The minimum margin to sodium boiling is predicted as 50 K with the CATHARE-2 model whereas the margin is greater than 100 K with the SAS4A/SASSYS-1 model.

Three modeling differences have been identified as the main drivers of the observed discrepancies in the U.S. and French results:

- Because the transient benchmark specifications do not provide the fuel thermal conductivity and heat capacity, the default oxide-fuel thermo-physical properties in the CATHARE-2 and SAS4A/SASSYS-1 codes are assumed applicable. Potential differences in these default fuel properties will likely contribute to discrepancies in fuel temperature and Doppler reactivity feedback response during both transients analyzed in this effort.

- The CEA's CATHARE-2 model accounts for the differential expansion between the core, vessel, control rods, and control rod drives (each driven by a different time constant) to include the control rod driveline (CRDL) expansion reactivity feedback. Due to limited information in the benchmark specifications, the initial Argonne model with SAS4A/SASSYS-1 code did not include this feedback effect. During the second revision of the Argonne calculations, a simple CRDL expansion model that only accounts for expansion of the control rod drives and the vessel walls is also included.
- With CATHARE-2, the radial core expansion is assessed solely based on the diagrid expansion whereas with SAS4A/SASSYS-1, the expansion of the above-core load pads was also considered leading to a less conservative assumption. During the second revision of the Argonne calculations, only the diagrid expansion is considered to be consistent with the CATHARE-2 model.

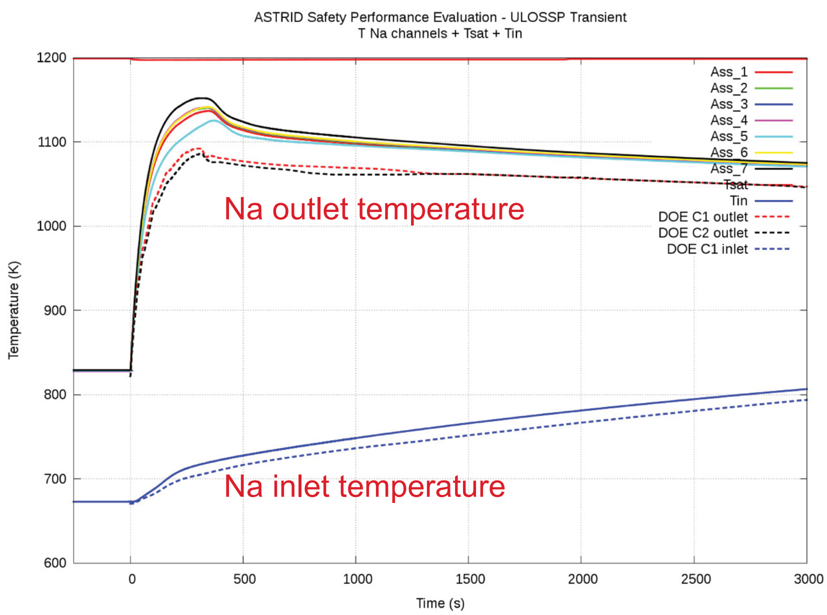

FIG. 9. ULOSSP Core Temperatures

5. Conclusions

ASTRID is a sodium-cooled fast reactor being designed by CEA with a unique axially heterogeneous core and upper sodium plenum as the main features aimed at achieving a near zero (or negative) global sodium void worth. According to a framework agreement between DOE and CEA, the teams at CEA, INL and ANL performed neutronics and transient benchmark calculations to verify the basic characteristics of the ASTRID design.

The CEA and INL teams evaluated both deterministic and Monte-Carlo methods for the neutronics benchmarks using ENDF/B as well as JEFF3.1 nuclear cross section data for cross comparisons. Generally good agreement is observed between the CEA and DOE results for core multiplication factor, sodium void worth, axial linear power, and flux distributions. However, based on the estimated uncertainties for separate components of void worth, the worst-case combination of uncertainties could still lead to a slightly positive integral sodium void worth for the configuration studied in this benchmark. Therefore, additional modeling is recommended to reduce the uncertainties.

As the comparison of blind calculations, the results of transient analyses are also found to be generally in good agreement, capturing the trends consistently and demonstrating the benign response of the ASTRID core for the both unprotected (without scram) transient sequences studied. The main difference in asymptotic coolant temperatures for the ULOHS case is attributed to differences in orifice coefficients, form losses, and other flow resistances. The heat-transfer from hot-pool to cold pool, to the annular vessel cooling bypass flow region, and eventually to the constant vessel outer temperature is also identified as a potential source of uncertainty. In addition, about 50K difference in U.S. and French results for the margin to boiling for the ULOSSP case is attributed to differences in implementation of radial core expansion feedback, control-rod driveline expansion effect (ignored in first round of USDOE calculations due to lack of data), and fuel Doppler feedback due to potential difference in fuel properties and gap conductance model. These uncertainties are being evaluated during the second round of calculations.

The consistency between the independent DOE and CEA results for both the neutronics calculations and the transient analyses represents significant progress toward the stated objectives of this collaboration. Despite some differences in the assumptions made by both sides using the state-of-the-art reference codes and methods based on best engineering judgment, the good agreement between the key core characteristics and transient behavior enhances confidence in the CEA predictions of key ASTRID safety-relevant parameters and behaviors. Proposed future activities are aimed at assessing (and reducing) the uncertainties arising from errors in computational models and data. A cost (or risk)/benefit analysis for very low sodium void worth cores is also considered important for both sides.

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