*Transactions of the Korean Nuclear Society Autumn Meeting*
*Gyeongju, Korea, October 26-28, 2016*

# Overview of Key Computer Codes for the *PGSFR* Safety Analysis

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## 1. Introduction

The engineering project for licensing and construction of a PGSFR (Proto-type Generation IV Sodium cooled Fast Reactor) was launched in 2012 [1]. Main interest of the PGSFR design has been focused on reducing an amount of the radioactive wastes discharged from the LWR (Light Water Reactor) spent fuel by transmutation. Its design could not only alleviate the toxicity of TRU (Transuranics) which is combination of Pu and MA (Minor Actinides; Np, Am, Cm), but also reduce half-life for long-lived nuclei.

The efficient electricity generation as well as the high level of safety is a requirement for the PGSFR design. In this context, the safety analysis is a key concern for the PGSFR specific design. In this regard, the present manuscript is aimed at sharing the knowledge on the PGSFR safety analysis with concerned individuals or organizations for a mutual understanding and collaboration. It introduces overall characteristics of the PGSFR design first, and then describes an accident classification with acceptance criteria, highlights of safety analysis computer codes, and discussion of covering ranges and availability of the codes.

## 2. Design characteristics of the PGSFR

### *2.1 Overall configuration*

The initial core of the PGSFR is loaded with a low enriched uranium metal fuel (U-10% Zr) for reactor performance demonstration and for fuel irradiation tests of TRU as a driver fuel. After the TRU fuel recycled from LWR (LWR-TRU) is qualified, the U-TRU-Zr fuel then will be replaced as a batch.

Figure 1 schematically represents the configuration of the PGSFR with the essential components. The primary heat transfer system (PHTS) is a pool type. All the structures and components of PHTS, 4 intermediate heat exchangers (IHXs) and 2 mechanical pumps are submerged into a large sodium pool confined by double vessels. A total of 217 fuel rods are arranged into a fuel subassembly with hexagonal configuration. The hexagonal core comprises a total of 112 fuel subassemblies. The active core height is about 90 cm. The cycle length of uranium equilibrium core is about 290 effective full power days (EFPDs). The core outlet sodium temperature is designed at 545$^{o}$C as a new fuel/cladding is developed, and a high temperature structure is available.

The decay heat is removed to the atmosphere by 2 active decay heat removal systems (ADHRs) and 2 passive decay heat removal systems (PDHRs). The active circuit has more than 50% of a passive heat removal capability even when the pump and blower are not operable.

The PGSFR has independent and diversified safety shutdown systems which consist of 6 primary control rods and 3 secondary shutdown rods. A passive shutdown mechanism is implemented into the secondary shutdown rods for an additional shutdown capability during beyond-design-basis accidents [1].

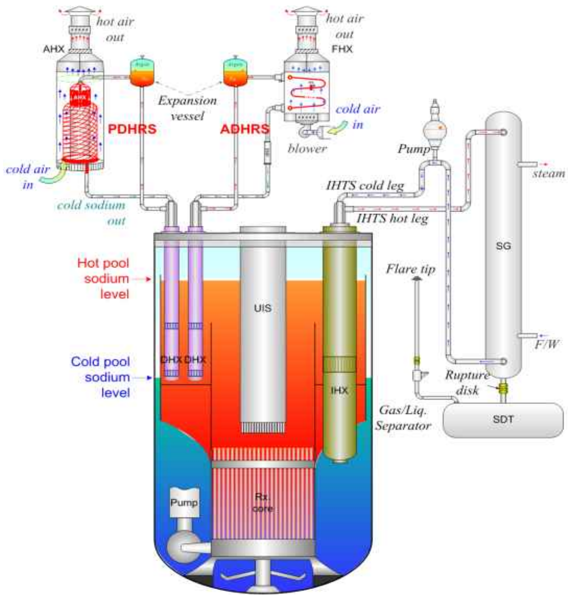


Figure 1. Schematic configuration of the PGSFR

### *2.2 Event classification and safety criteria*

The event classification and corresponding safety design acceptance criteria have been established in terms of Cumulative Damage Fraction (CDF) and a few temperatures. The safety criteria employed for the PGSFR safety analysis are classified into 4 categories, namely, anticipated operating occurrence(AOO), Design Basis Accident(DBA) case 1 and 2, and Design Extended Condition(DEC) as summarized in Table 1. The DEC consists of the unprotected accidents and severe accidents.

For events of AOO and DBA class-1, the conditions of both CDF<0.05 and thermal strain<1% can make an enough satisfaction with no reduction of fuel lifetime as well as a small fraction of cladding failure. For DBA

Class-2 events, some fuel pin failures are allowable as long as a pin coolable geometry is maintained and the fuel failures do not propagate. For DEC events, it was assumed that if coolant temperature is lower than the boiling temperature, the accident would not develop to a severe accident.

Table 1 summary of the event classification.

| **Event Category** | **Fuel/Cladding** | **Reactor Vessel/ Primary System** | **Containment** |
|---|---|---|---|
| **AOO** $10^{-1} > F \geq 10^{-2}$ | No reduction of fuel lifetime | ASME Service Level B limits | Design pressure and temperature not exceeded |
| | $CDF_{\Sigma AOO} < 0.05$ Strain < 1% | | |
| **DBA Class 1** $10^{-2} > F \geq 10^{-4}$ | A small fraction of fuel rod cladding failure | ASME Service Level C limits | |
| | $CDF_{event} < 0.05$ Strain < 1% | | |
| **DBA Class 2** $10^{-4} > F \geq 10^{-6}$ | Pin coolable geometry, with no pin failure propagation | ASME Service Level D limits | |
| | Fuel T< Solidus T Cladding T < 1075℃, Coolant T < Boiling T | | |
| **DEC** $10^{-6} > F \geq 10^{-8}$ | Core coolability with in-vessel retention | ASME Service Level D limits | |
| | Coolant T < Boiling T | | |

## 3. Computer codes for safety analysis of the PGSFR

For the safety analysis of the PGSFR, MARS-LMR is used for the analysis of AOO, and DBA class 1 and 2. MATRA-LMR/FB is a sub-channel analysis code to be applied to local phenomena such as an internal blockage in a subassembly or the fuel enrichment error that a higher enrichment fuel pin is loaded in a lower enriched subassembly. SAS4A/SASSYS-1 is used for analysis of the HCDA initiating phase, while the unprotected accidents are analyzed with MARS-LMR. No other code is available for the prediction of the HCDA (Hypothetical Core Disruptive Accident) progression beyond the initiating phase.

The Sodium-Water reaction in the Steam Generator (SG) is classified into AOO, DBA class 1 and 2 depending on a tube rupture size, and SWAAM-II is applied to the analysis. Meanwhile, the containment performance analysis is carried out with CONTAIN-LMR for sodium fire accidents in the containment. Source term generation is quantified with ORIGEN-2, and the source term transport inside the reactor vessel is tracked with ISFRA for events where source term is released in the vessel regardless of the classification. The MACCS-II code will be used for the atmospheric radioactive risk analysis. Following descriptions introduce highlights of individual code available.

### *3.1 MARS-LMR*

MARS-LMR [2] is a modified version of the MARS (Multi-dimensional Analysis for Reactor Safety) code [3] in order to analyze a SFR (Sodium cooled Fast Reactor) system. The neutron physics in MARS-LMR is based on a point kinetics model. To take into account the necessary reactivity feedbacks in the PGSFR core, the feedbacks due to the Doppler effect, sodium density/void, fuel axial expansion, core radial expansion, control rod drive-line, and reactor vessel expansion, were individually modeled and integrated into MARS-LMR [4].

The assessments of the MARS-LMR models newly implemented have been carried out with the shutdown heat removal test data in the EBR-II reactor [5] and the end-of-life natural circulation data in the Phenix reactor [2]. The predictions exhibited validity of the MARS-LMR applications to such integrated systems as EBR-II and Phenix reactors.

### *3.2 MATRA-LMR/FB*

In order to analyze local disturbances where a strong local cross-flow is anticipated within a subassembly, a technically sophisticated computer code must be applied for a reasonable prediction. The MATRA-LMR code [6] is such an option, because it is capable of representing the cross-flow within the fuel pin bundle through an axial and a lateral momentum equations by assuming axial flow is dominant over the transverse flow. The MATRA-LMR/FB is a revised version of the MATRA-LMR, and some of its models were modified to be eligible for the analysis of the SFR sub-channel blockage for wire-wrapped pins. It integrated the Distributed Resistance Model [7], which has generally been recognized as representing the wire-wrap effect more realistically.

The MATRA-LMR/FB had been qualified based on worldwide available experimental data such as Oak Ridge National Laboratory (ORNL) 19-pin tests [8,9], Karlsruhe 169-pin tests [8], and the EBR-II test [10]. Code-to-code comparative analyses with SABRE and the CFX code were also been performed [11,12]. All comparison results showed reasonable agreements.

### 3.3 *SAS4A/SASSYS-1*

SAS4A was developed at ANL (Argonne National Laboratory) to predict consequences of postulated accidents that could lead to fuel failures within a subassembly initiated by undercooling or overpower conditions. The SAS4A code is not usually used with a stand-alone version, but it is coupled with SASSYS-1 to provide the pre-conditions for SAS4A calculations. In the US, SAS4A severe accident models are validated with results from fuel tests run in the TREAT facility [13].

A combined version, SAS4A/SASSYS-1 computes fuel/cladding/coolant heating, coolant boiling, cladding failure, and fuel/cladding melting and relocation before an assembly duct is affected [14].

A renewed emphasis on the development and validation of the SAS4A metal fuel models is required because of two important phenomena that occur in metal fuel pins but are not present in the oxide fuel pins:

a) The radial migration of the U-Pu-Zr fuel components during irradiation, which leads to the formation of multi radial fuel regions with different composition as shown in Fig. 2, and

b) The formation of the fuel-cladding eutectic at the interface between the fuel and cladding, which leads to changes in the local composition of both fuel and cladding.

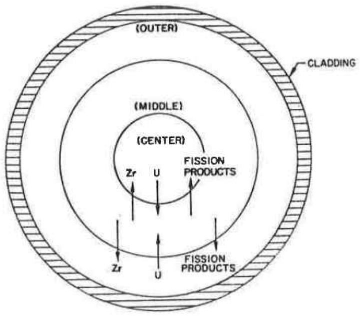


Fig. 2 Annular zones in U-Pu-Zr ternary fuel

The transient metal-fuel pin behavior such as fission gas formation and release, and pin axial expansion and fuel pin failure, is described by the new DEFORM-5 and FPIN2 models as illustrated in Fig. 3. The new PINACLE module describes the in-pin molten fuel relocation prior to cladding failure, *i.e.,* the molten cavity formation and pressurization, fuel relocation, fuel ejection above the original fuel column. Fuel ejection into the coolant channel and subsequent axial dispersal, including the post fuel failure with cladding relocation, and fuel freezing and channel geometry changes, are modeled by the PLUT02 and LEVITATE modules,

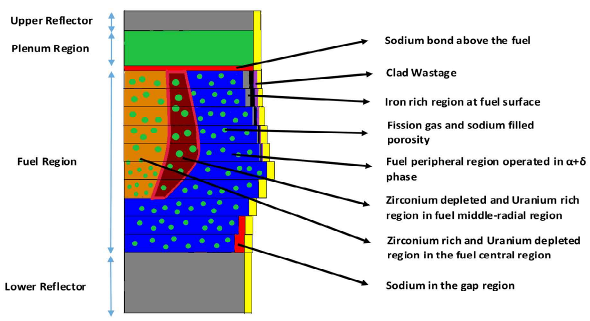


Figure 3. Metal Fuel Phenomena Modeled by DEFORM-5

*3.4. SWAAM-II*

A tube rupture in a SFR steam generator (SG) causes water/steam to contact with sodium (Sodium-water reaction), producing an exothermic chemical reaction with sudden generation of a large amount of hydrogen gas. The pressure pulses produced, thus, can exert a large forces on the structural materials and may threaten the structural integrity. In order to provide means for mitigating the pressure effect, a code which can be applied to a large class of events covering the entire spectrum of possible scenarios, must be provided.

The SWAAM-I1 (Sodium Water Advanced Analysis Method II) code [15] is designed to analyze a hypothetical sudden break of SG tubes in the SFR, called double-ended-guillotine (DEG). It basically calculates the pressure variations along the piping system by modeling the system with nodes and flow paths. It installs rigorous models on the leak flow blowdown, the fluid hammer effect in the sodium, the interactive dynamics of the sodium-water reaction and the hydrogen bubble growth, and the fluid-structure interaction in the IHTS piping and the steam generator shell side. Figure 4 represents a SWAAM-II calculation for pressure responses during 5 SG tubes rupture accident at various positions in IHTS.

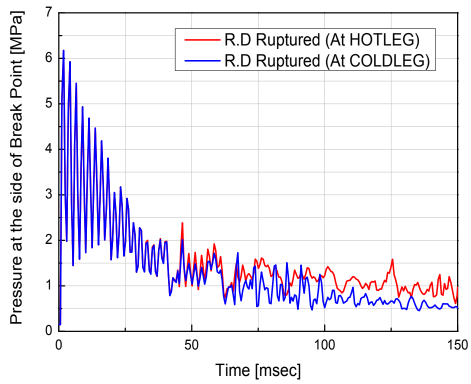


Figure 4. Pressure responses to SG tube rupture (5 tubes)

*3.5 CONTAIN-LMR*

The CONTAIN code is employed to evaluate internal threats to containment integrity and the radiological source term in the event of containment failure. CONTAIN-LMR [16] is an extended version of the CONTAIN code for SFR application. CONTAIN-LMR includes models for sodium-concrete interactions, debris bed phenomena, and other LMR-specific models such as sodium pool and spray fire models in an integrated manner.

*3.6 ISFRA*

The Integrated Sodium Fast Reactor Analysis (ISFRA) was developed for probabilistic safety assessment (PSA) in the Fauske & Associates [17]. In particular, the code is designed to represent the PGSFR pool design with metal fuel. The ISFRA code simulates the accident progression from a PSA perspective, which principally includes the release of fission products to the coolant, the reactor vessel, the containment, and to the environment. This will provide a necessary insight to assess the risk profile for the PGSFR design.

### *3.7 ORIGEN-2 and MACCS-II*

ORIGEN2.1 [18] is a one-group depletion and radioactive decay computer code developed at ORNL. The principal use of ORIGEN2.1 is to calculate the radionuclide composition and other related properties of nuclear materials.

Meanwhile, MACCS (Melcor Accident Consequence Computer System) was developed at Sandia National Laboratories (SNL) for the NRC. Its primary use is to evaluate the impact of accidental atmospheric releases of radiological materials on humans and on the surrounding environment. A recent version, MACCS-II [19], has been widely distributed and used by the NRC and its subcontractors, private industry, and the U.S. Department of Energy (DOE) complex.

## 4. Conclusion

So far, no problem has been found for the series of computer codes on application to the PGSFR safety analysis. Such a conclusion has been reached based on their validations, development backgrounds, availability, and practical uses for the PGSFR analysis. MARS-LMR has a wide range of applicability to accident analyses for an integrated system. SAS4A/SASSYS-1 also has a capability to model a system, but its models address more to the fuel failures during the initiating phase of HCDA. On the other hand, the codes such as MATRA-LMR/FB, SWAAM-II, and CONTAIN-LMR have their specific purposes and limited applications, while ORIGEN-2, ISFRA, and MACCS-II are used for the PSA purpose.

A code which can analyze the molten core progress post assembly duct failure is not available at present time. Since such capability is of importance in designing an adequate mitigation measure, there must be a plan how to cover the analysis range of the whole HCDA progress.

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