STUDSVIK CMS CAPABILITY FOR SPENT NUCLEAR FUEL

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ABSTRACT

A number of spent fuel pools around the world are close to reaching their filled capacity and thus alternative storage has to be found. To resolve this situation many utilities use dry storage containers. An accurate prediction of the spent fuel characteristics is mandatory for safe and economic solutions. So, it seems advantageous to couple the in-core and the back-end methods.

The purpose of this paper is to show the Studsvik Core Management System (CMS) capability to simulate spent nuclear fuel in dry storage starting from the In-Core Fuel Management (ICFM) calculations, followed by the evaluation of the residual heat sources using the SNF code, and finally, followed by the evaluation of safety related parameters (maximum creep strain, maximum clad stress, etc.) by ENIGMA.

1. Introduction

Knowing the isotopic composition of the fuel at the time of the discharge from the reactor is a prerequisite for back-end fuel cycle calculations with applications to storage, transportation and disposal of spent fuel. Solving the isotopic decay chains allows the calculation of the radiation as well as the decay heat sources at any time after the discharge from the reactor.

The CMS has the capability to simulate spent nuclear fuel in dry storage starting from the ICFM calculations using the CASMO5 and SIMULATE5 codes, followed by the evaluation of the residual heat sources using the SNF code, and finally, by the evaluation of the thermo-mechanical behaviour by ENIGMA.

The ENDF/B-VII data library used by CASMO5 includes a large number of actinides and fission products that are individually represented and tracked per fuel pin in ordinary ICFM depletion calculations. The operating history of a given fuel assembly is obtained from the ICFM 3-D simulations using SIMULATE5 core tracking calculations. The SNF program, developed by Studsvik, computes the residual heat and the radiation sources by the isotopic summation method. The isotopic concentrations required by the SNF code can be evaluated at the nodal level using the final burnup, spectrum history and power density history at each axial level for any given fuel assembly. Thus, the CMS state-of-the-art capabilities for ICFM are seamlessly integrated into a consistent back-end capability. The relatively high decay heat and low heat transfer coefficients during drying and dry storage may result in relatively high clad temperatures and high clad stresses that must be evaluated as part of the safety assessment. The ENIGMA code, developed by the UK National Nuclear Laboratory and recently added to the CMS code suite, is capable of predicting the thermo-mechanical behaviour of fuel rods during transient conditions as well as steady-state conditions of irradiation in Light Water Reactors (LWRs). Moreover, recent developments of the ENIGMA code extended its capabilities to pool cooling, drying and dry storage.

Section 2 provides a brief description of the CMS codes, namely: CASMO5, SIMULATE5, SNF and ENIGMA. Details are provided in the references at the end of this paper. Section 3 shows an example of the back-end calculations for a PWR rod irradiated close to 30 MWd/kgU and cooled for five years in the fuel pool.

2. Studsvik CMS code package

2.1 CASMO5

CASMO5 [1] is Studsvik's next generation LWR lattice physics code. It has many new features compared with its predecessors. Among them, it is important to mention in the context of back-end fuel calculations: (1) the generalized fuel storage racks simulation capability and (2) the generation of data for the code SNF. The data for SNF is comprised of isotopic number densities for important actinides, fission products and activation products. All data is provided by the ENDF/B-VII or JENDL-4 advanced data libraries [2]. These advanced libraries include additional isotopes (Th-230, Pu-236, Te-132, Te-127m, Te-129m, Tb-160, Ni-58, Ni-62, Te-127, Fe-54, and Fe-55) that are important for spent fuel characterization. Table 1 provides a complete list of the SNF isotopes available from CASMO5.

Actinides	Fission Products		Activation Products
U-234	Se-79	Te-132	Natural Fe
U-235	Kr-85	Xe-133	Co-60
U-236	Sr-89	Cs-134	Natural Ni
U-237	Sr-90	I-135	
U-238	Y-90	Xe-135	
U-239	Y-91	Cs-135	
Np-237	Zr-93	Cs-136	
Np-238	Zr-95	Cs-137	
Np-239	Nb-95	Ba-140	
Pu-236	Zr-97	La-140	
Pu-238	Nb-97	Ce-141	
Pu-239	Mo-99	Ce-143	
Pu-240	Tc-99	Pr-143	
Pu-241	Ru-103	Ce-144	
Pu-242	Ru-105	Pr-144	
Pu-243	Rh-105	Nd-147	
Am-241	Ru-106	Nd-148	
Am-242	Rh-106	Pm-147	
Am-242m	Ag-110M	Pm-148M	
Am-243	Ag-111	Pm-148	
Am-244	Sb-125	Pm-149	
Cm-242	Sn-126	Pm-151	
Cm-243	Sb-126	Sm-147	
Cm-244	Sb-127	Sm-151	
Cm-245	Te-127M	Sm-153	
Cm-246	Te-127	Eu-154	
Cm-248	Te-129M	Eu-155	
Cf-252	Te-129	Eu-156	
I-129	Tb-160		
I-131			

Table 1: CASMO5 SNF isotopes

2.2 SIMULATE5

SIMULATE5 [3] is Studsvik Scandpower's next generation nodal code developed to address deficiencies of existing reactor physics tools for today's aggressive core designs. Multi-group cross sections and other data for SIMULATE5 are generated from CASMO5. In the context of back-end fuel calculations, SIMULATE5 core follow calculations provide:

- operating data for fuel assemblies that will be analyzed with SNF
- thermal power densities for the rod(s) that will be analyzed with ENIGMA

The operating history from core follow calculations is transferred to SNF code through the SIMULATE5 restart file(s). A restart file written at the end of cycle depletion provides detailed assembly burnup and history distributions for each fuel assembly in the core. The final isotopic concentrations for some actinides and fission products with short half-lives are quite sensitive to the power density history. A user can provide detailed power history to SNF by requesting SIMULATE5 to write a restart file at various exposure points that reflects the operation history of the cycle. Cycle start-up and shutdown dates are available to the SNF program through SIMULATE5 restart files. The fuel rod(s) irradiation history from core follow calculations is transferred to ENIGMA through the SIMULATE5 pin file(s).

2.3 SNF

SNF [4] calculates the 3-D distributions of isotopic concentrations, decay heat and neutron and radiation source terms in a spent fuel assembly for cooling times up to 100,000 years. The SNF isotopic depletion method utilizes the accurate, local neutron spectrum based depletion calculations of the 2-D lattice physics code (CASMO5) as well as nodal operating history data from the 3-D nodal simulator (SIMULATE5).

The SNF power history model accounts for the influence of the actual power density history on the isotopic concentrations generated by the lattice code at a constant power density. The End-of-Life (EOL) concentration of a given isotope in a given node is obtained by integration of the relevant isotopic build-up/decay chains through the entire in-core lifetime. Short-lived fission product isotopes are obviously very sensitive to the local power density; however, isotopes also generated by neutron capture in fission products and many actinides depend quite strongly on the power density history.

The isotopic concentrations at discharge are used as initial conditions for solving the isotopic decay chains. A number of isotopes not present in the lattice physics codes are added in this process. The final isotopic concentrations provide the required basis for calculation of radioactivity, decay heat, gamma heat, spontaneous fission and (α,n) - reaction source neutrons as well as photon release rates and spectra.

SNF applications include analyses of radioactivity, decay heat and neutron emission rates of spent fuel assemblies to be loaded into transport/storage casks as well as full core or fuel pool decay heat calculations required for demonstration of compliance with cooling capacity limitations.

2.4 ENIGMA

The ENIGMA fuel performance code [5], developed by UK National Nuclear Laboratory, has recently been added to the CMS suite. ENIGMA calculates the thermo-mechanical behaviour of an LWR fuel rod in both steady-state and transient conditions. The active stack length region of the fuel rod is represented by a series of axial zones consistent with the axial discretization in the SIMULATE5 model. In each axial zone the fuel is divided into radial annuli. The free volumes associated with the fuel-clad gap, pellet dishes and chamfers, pellet cracks, the pellet bore and upper and lower plena are also modelled. Only radial, i.e. no axial or circumferential, heat flow is assumed and the fuel annuli are all considered to be subject to the same axial strain. The effects of shear stresses are approximated using models for axial extrusion and dish filling and for pellet hour-glassing which feed calculated strain increments back into the main solution scheme. Coupling between the axial zones is restricted to the coolant enthalpy, rod internal pressure and gas transport. Recently, ENIGMA has been extended for modelling dry storage scenarios including the analysis of pool cooling, and drying. This involves: (1) incorporating an out-of-pile clad creep model; (2) including the ability to simulate annealing of the clad irradiation damage; (3) suppression of the clad corrosion modelling during out-of-pile conditions. Rossiter [5] provides a detailed account of these modifications.

ENIGMA is validated against a large database of LWR fuel rod irradiations in both commercial and test reactors. In total, over 500 rod irradiations with burnups up to 90 MWd/kgHM are included in the validation database. ENIGMA's out-of-pile clad creep model has been independently validated against experimental data and so that the integral clad strain predictions are in good agreement with measurements for the creep testing of dry stored rod segments performed as part of the Dry Cask Storage Characterisation (DCSC) Project [7].

3 Example

An SNF / ENIGMA dry storage assessment has been performed for a UO_2 fuel rod irradiated in a PWR core to approximately 30 MWd/kgU and then cooled in a fuel pool for five years. In the example that follows, the analysis was performed assuming one assembly per cask. Furthermore, the spent fuel was dried for a period of 24 hours using the vacuum drying method before it is transferred to the cask. The spent fuel is assumed to be covered with helium during its cask storage.

The in-pile irradiation history was calculated using CASMO5/SIMULATE5. Fig. 1 shows the rod average rating as a function of rod average burnup for the selected rod.

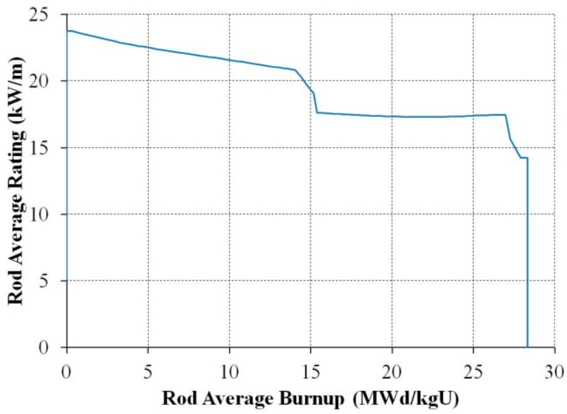


Fig. 1: In-pile irradiation. Rod average linear heat generation rate vs. rod average burnup.

Clad surface temperatures during in-pile irradiation and pool cooling were computed by ENIGMA and set to 50°C respectively. The calculation of the decay heat output, performed with SNF, was used to compute the clad surface temperatures during drying and cask storage. Clad surface temperatures were evaluated using the correlations provided by Manteufel and Todreas [6]. Fig. 2 shows the maximum axial rod surface temperature during and after the in-pile irradiation.

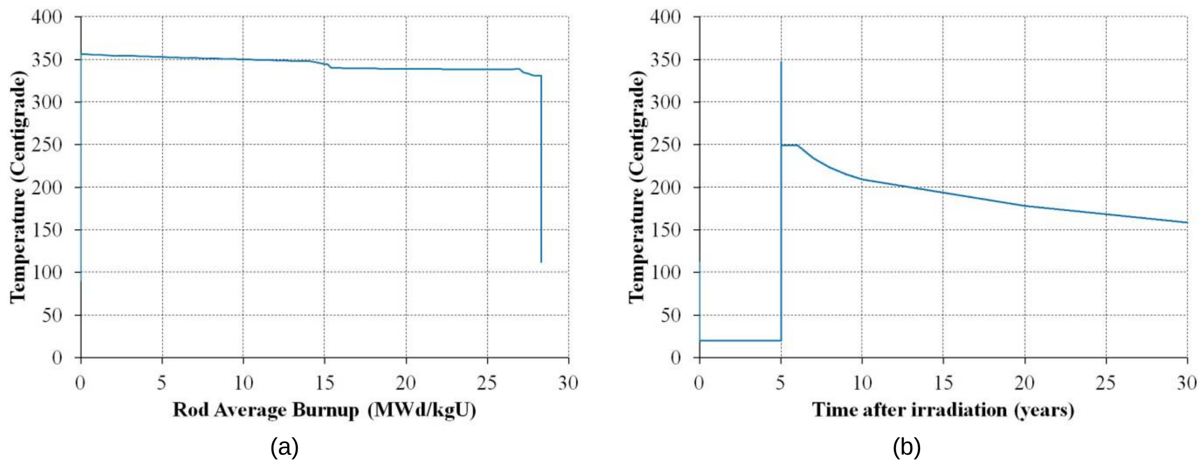
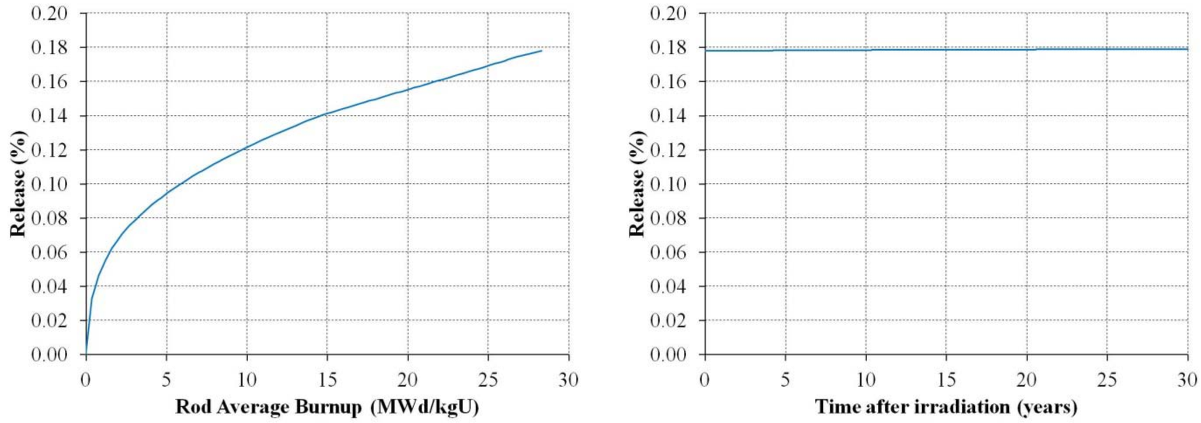


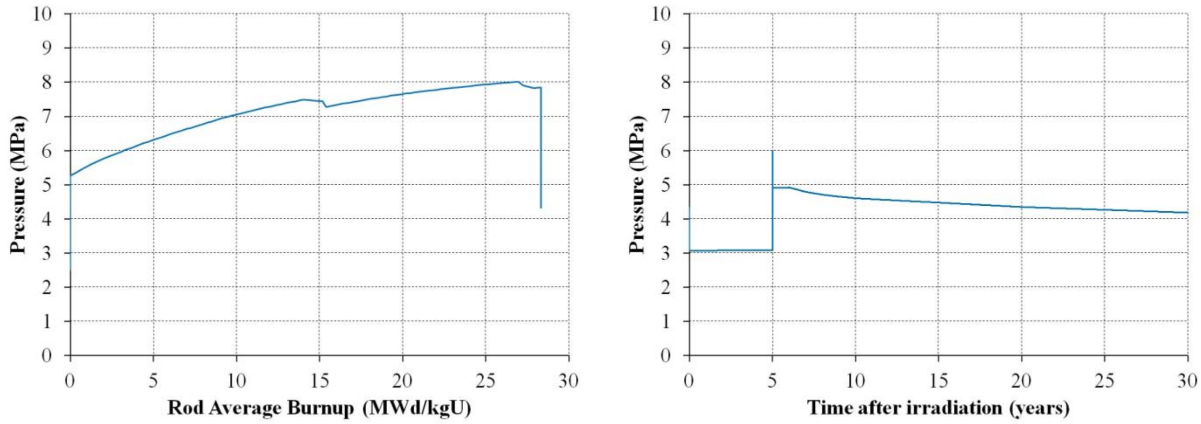
Fig. 2: Maximum axial rod surface temperature: (a) during irradiation, (b) after irradiation.

During in-pile irradiation, fission gas release may lead to the over-pressurisation of the rod. The waterside corrosion may reduce the cladding wall thickness. Both phenomena reduce the performance of the cladding. Therefore, fission gas release, rod internal pressure and the maximum axial clad oxide thickness were evaluated with ENIGMA. They are shown in Fig. 3-a, Fig. 4-a, and Fig. 5-a respectively.



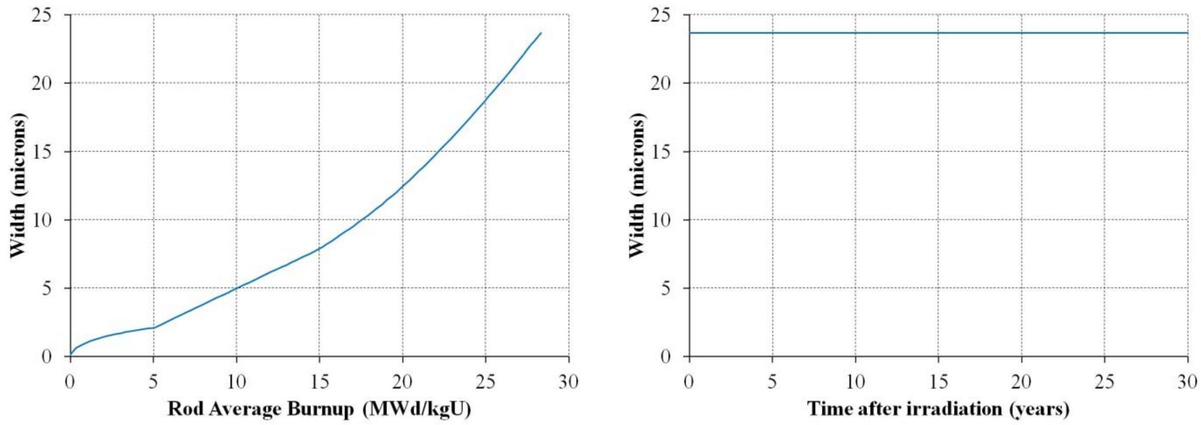
(a) (b)

Fig. 3: Fission gas release: (a) during irradiation, (b) after irradiation.



(a) (b)

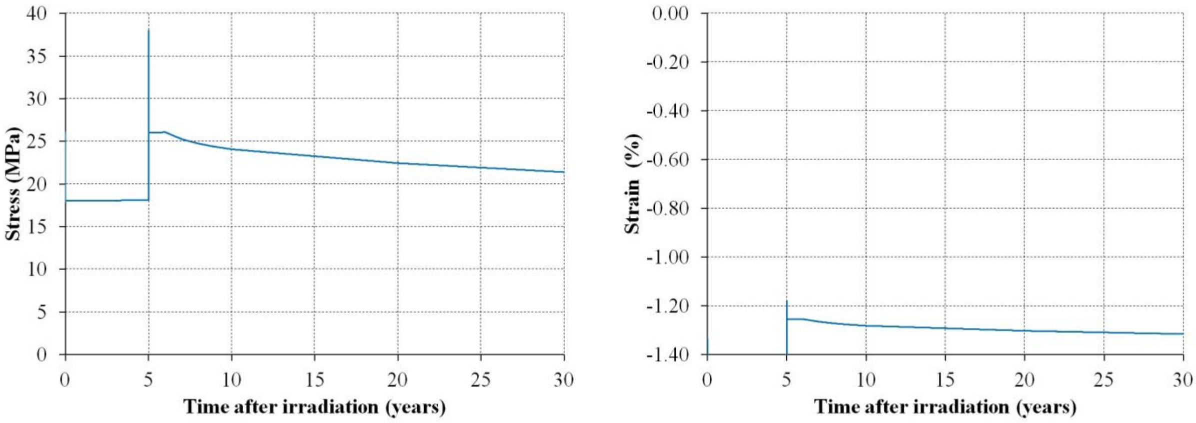
Fig. 4: Rod internal pressure: (a) during irradiation, (b) after irradiation.



(a) (b)

Fig. 5: Maximum axial oxide thickness: (a) during irradiation, (b) after irradiation.

It is clear from the previous figures that the mild irradiation of this rod did not result in either rod over-pressurisation or excessive (>10%) thinning of the clad wall. Therefore, it is to be expected that during cask storage the maximum clad hoop stress and strain are within the safety limits as clearly shown in Fig. 6. For completeness, Fig. 3-b, Fig. 4-b, and Fig. 5-b show the fission gas release, the rod internal pressure and the oxide thickness after the in-pile irradiation.



(a) (b)

Fig. 6: (a) Clad hoop stress, (b) Clad hoop strain.

Summary

The Studsvik CMS capability to simulate spent nuclear fuel in dry storage starting from ICFM calculations, followed by SNF and ENIGMA evaluations of safety related parameters has been described and exemplified for a fuel rod irradiated in a PWR core.

Acknowledgments

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