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Fuel Performance Code for Light Water Reactor, GIFT: Current development status and path-forward

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1. Introduction

While designing a nuclear reactor, fuel performance modeling is crucial because it shows changes in the overall state of the nuclear fuel rod during the operation of the nuclear power plant and determines whether the rod will remain stable throughout operation without failure. In addition, the code simulation results should confirm that the number of regulatory restrictions on the rod is satisfied.

For this purpose, Korea has generally used the FRAPCON code, a computer code for the calculation of steady-state, thermal-mechanical behavior of oxide fuel rods for high burnup, so far. FRAPCON was created for the United States Nuclear Regulatory Commission (US NRC) by the Pacific Northwest National Laboratory (PNNL) [1]. Korea's nuclear regulatory agencies have utilized this nuclear fuel code by using the executable file as it is, or with adding a new model to the source code.

However, more extended simulation capabilities have begun to be required for nuclear fuel codes recently. There have been various changes in the nuclear power academia, such as introduction of several new fuel concepts like accident tolerant fuel (ATF), a movement to use fuel until it reaches a higher burnup level over 60MWd/kgU, and interest in spent nuclear fuel, etc.

Amid this trend, the development of the GIFT code, Generalizable Integral Fuel-Life Tracker, has been carried out from 2021 to the present. By mounting a newly made high-performance model based on previously developed physical and mathematical models, GIFT-1.0 enhances the accuracy and fidelity of the light water reactor simulation. Also, it is designed to facilitate the establishment of regulatory standards for new types of nuclear fuel by developing and adding independent models suitable for ATF simulation.

In this paper, in addition to an overview of GIFT code, how to use GIFT code and some test results will be discussed.

2. Overview of GIFT

GIFT can recognize fuel rods by dividing them into pellets, gaps, and cladding, and can also briefly simulate reactor environments such as coolants. The overall module structure of the GIFT is shown in Fig. 1.

Fig. 2 shows the calculation flow chart of GIFT. After initializing the code with received user input, the code is executed around the convergence loop for pressure and the other loop for temperature difference in the gap. In this process, the power and burnup, the temperature and deformation at each point of the fuel, the amount of fission gas produced and released, and the volume and internal pressure in the fuel rod are calculated.

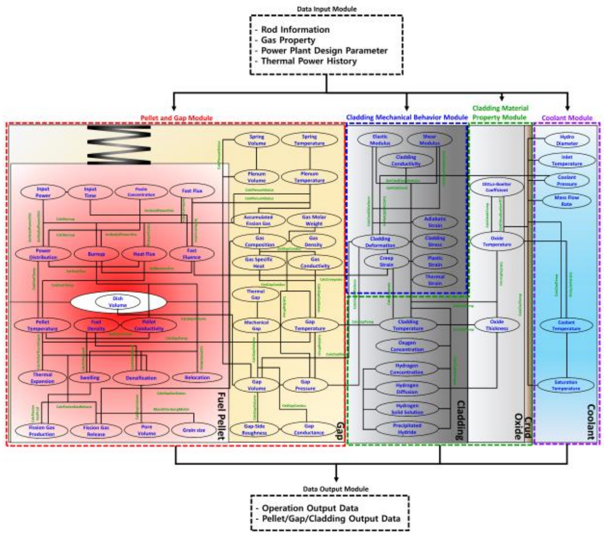

Fig. 1. Overview of GIFT code structure

As a nuclear fuel performance code for LWR, GIFT adopts the following basic configurations:

- *Geometry*. In the code, a fuel rod consists of several axial segments, which are divided into radial ring segments while assuming azimuthal symmetry. Pellet and cladding have different numbers of ring segments.

- *Thermal Analysis*. Dittus-Boelter equation was used for heat transfer from cooling water to cladding and Ross-Stoute gap conductance model was used for cladding to pellet heat transfer. Inside the cladding, heat conduction equation with one-dimensional approximation is used, and inside the pellet, the temperature is calculated through a two-dimensional FDM assuming azimuthal symmetry.

- *Stress and Strain Analysis*. Thermal expansion, swelling, densification, and relocation causing deformation of a pellet are calculated by the empirical model. An isotropic assumption applies to other components except relocation. In the case of cladding, the 2D axisymmetric structural analysis function is used to calculate the stress and strain at each mesh. This function receives pressure, temperature, creep, and plastic strain as its input.

- *Fission Gas Release*. Forsberg-Massih model is used for the fission gas release. This model performs the calculation considering the diffusion of fission gases produced by irradiation and their release from the fuel surface.

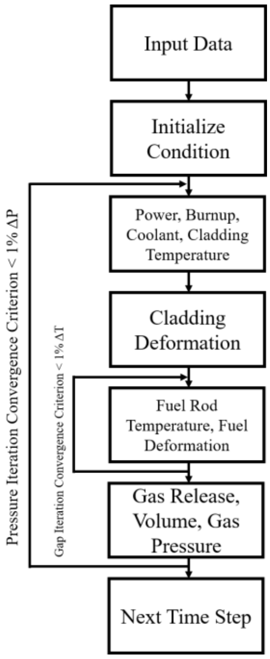

Fig. 2. Flowchart of GIFT code

In addition to the basic light-water reactor simulation capabilities of existing nuclear fuel performance codes, this GIFT code contains several notable functions and specialized models. Table 1 summarizes those features; cladding structural analysis function, chromium-coated cladding model, SiC cladding model, spent fuel model, and hydrogen diffusion model. The presence of these distinct features demonstrates that GIFT code is capable of performing equivalent tasks to other existing codes, as well as achieving better performance and responding to the development of new nuclear fuel concepts.

These characteristics differentiate GIFT code from other traditional nuclear fuel codes. For example, the FRAPCON code uses a 1D model for individual axial meshes when calculating the cladding deformation. In the case of chromium-coated ATF or SiC duplex cladding, however, it is difficult to calculate using this model, and the effect of the new cladding concepts on nuclear fuel performance cannot be properly evaluated.

Table 1. Description of major features of GIFT code

Features	Description
Cladding Structural Analysis Function	Perform calculations for multiple layers of cladding using a two-dimensional axisymmetric model
Chromium-Coated Cladding Model	Simulate the individual behavior and interaction of two layers, each with the properties of chromium and zircaloy
SiC Cladding Model	Simulate two types of SiC cladding, monolith and composite, up to three layers, including pseudo ductility of composite
Spent Fuel Model	Based on experimental data, creep deformation and swelling of spent nuclear fuel are simulated
Hydrogen Diffusion Model	Simulates hydrogen diffusion in vertical direction within cladding, considering precipitation and hydrides

3. Input and Output Interfaces of GIFT

GIFT operates through text-based input and output interfaces. The input file consists of individual block units that contain information about fuel rod and reactor separately. An example of an input file is shown in Fig. 3. The brief descriptions of the basic information to be included in individual blocks is shown in Table 2.

```
------------------------------<CLADDING>------------------------------
CladThickness[m]: 570e-6
CladRoughness[m]: 5.08e-7
CladColdwork[na]: 0
CladType[-]: Zr4
------------------------------<GAP>------------------------------
GapWidth[m]: 82.5e-6
GapPressure[Pa]: 2.41e+6
GasType[-]: He
------------------------------<PLENUM>------------------------------
PlenumLen[m]: 0.525
SpringTurnNum[#]: 60
SpringDiam[m]: 8e-3
SpringWireDiam[m]: 1.2e-3
------------------------------<COOLANT>------------------------------
CoolantPressure[Pa]: 15.51e+6
CoolantInletTemp[K]: 565
CoolantMassFlowRate[kg/m^2-s]: 3200
------------------------------<ROD>------------------------------
RodDiam[m]: 9.5e-3
RodPitch[m]: 0.0125
ActiveFuelLen[m]: 3.66
------------------------------------------------------------
```

Fig. 3. Input example of GIFT code

In addition to the basic information given above, GIFT code has an additional list of information that can be entered according to the user's purpose. These optional inputs are required for MOX, ATF, spent fuel functions, etc. A description of the blocks used to enter the optional information is as follows:

- MOX: an additional block used when fuel type is set to MOX. Contains information on plutonium concentration in fuel.
- SPENT_FUEL: a block added when you want to use spent fuel simulation. Instructs the time step when wet storage and dry storage periods begin.

- CR_COATING: a block added when you want to use the Chromium-coated ATF option. You can enter the thickness of the coating and the material properties of the chromium used in the coating.
- SIC: a block required when using the SiC cladding option, which can be used to specify the structure and material type of the SiC cladding.

Table 2. The default input list of GIFT code categorized by blocks.

Block Name	Components
NUMBERS	the number of time steps
	the number of axial meshes
	the number of fuel radial meshes
	the number of cladding radial meshes
TIME	calendar days
POWER	linear heat generation rate
	positions for the shape function
	values of the shape function
	index space of the shape function
	the number of time steps for each shape function index
PELLET	pellet dish depth
	pellet dish shoulder width
	pellet chamfer width
	pellet chamfer height
	pellet roughness
	pellet void radius
	pellet length
	fuel theoretical density
	enrichment
GAP	initial gap width
	internal pressure
	gas type
CLADDING	initial cladding thickness
	cladding roughness
	cold work
	cladding material type
PLENUM	plenum length
	spring number of turns
	spring diameter
	spring wire diameter
COOLANT	coolant pressure
	coolant inlet temperature
	coolant mass flow rate
ROD	reactor type
	rod diameter
	pitch
	active fuel length

GIFT's text output has its own format, but it is not suitable for users to read immediately. For this reason, the GIFT code package contains a script that converts the output file into graphs for individual physical quantities. Fig. 4 is an example of graphs made by the script, using GNUplot software.

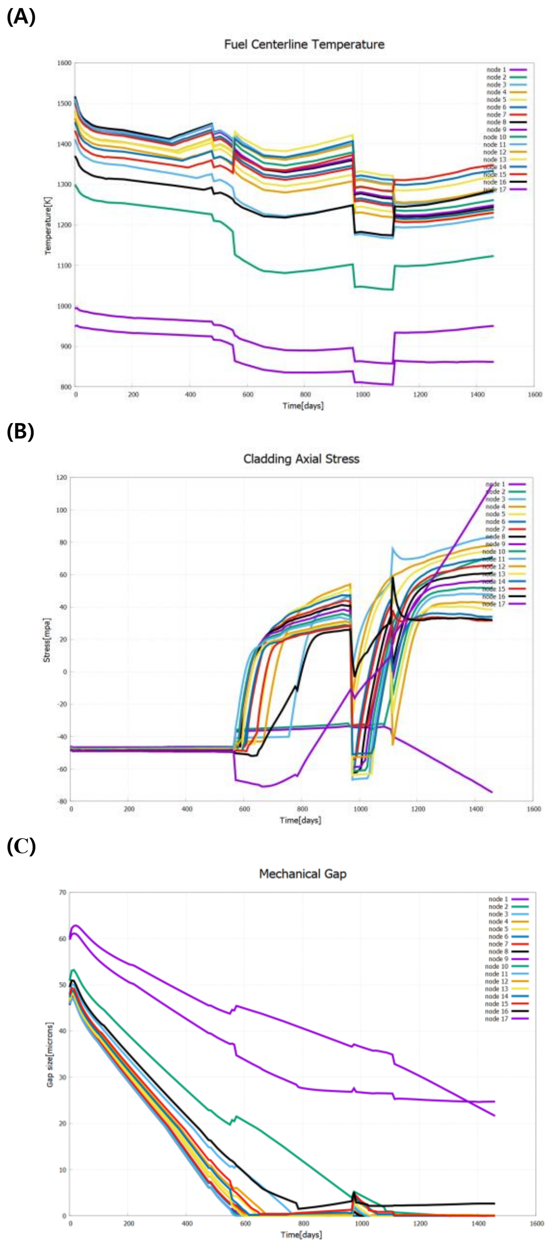

Fig. 4. Graphs post-processed by the script of GIFT code package, (A) fuel centerline temperature, (B) cladding axial stress, (C) mechanical gap width

4. Validation with OECD/NEA fuel experiment data bank

GIFT code was verified by comparing the experimental results of the OECD/NEA IFPE database with the code execution results. Table 3 lists the six cases used to verify GIFT code to date, and Fig. 5 shows all the calculated burnup results of the entire case, where the calculated value and experimental data closely match. Some cases containing additional information have individually described with extra paragraphs following.

Table 3. List of the cases used for the verification of GIFT code

	Case Name	Number of Rods
1	CONTACT [2]	3
2	Br-3 [3]	4
3	HBEP [4]	63
4	USPWR 16x16 [5]	4
5	Regate [6]	1
6	IFA-534 [7]	2

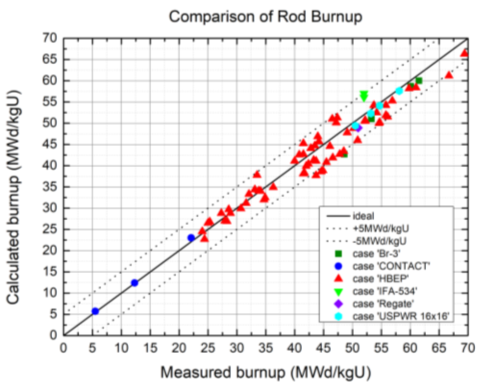

Fig. 5. Comparison between calculated and measured burnup of all cases

4.1. case 'CONTACT'

The data for the case 'CONTACT' comes from an in-pile test program carried out in the Siloe reactor. Rods of Zr-4 clad UO2 pellets of standard PWR 17x17 assembly were used [2]. A comparison between calculated and measured fuel centerline temperature of rods of 'CONTACT' case are presented in Fig. 6. It can be seen that the two values match well, over the entire period.

4.2. case 'Br-3'

The 'Br-3' case is a program about exposing 5 PWR-type test rods to high burn-up in the Br-3 reactor in Mol [3]. Table 4 shows the results of comparative verification for the four rods from the case 'Br-3'. In the results of the fission gas release, except for an error of about 20% in the rod '36-I-8', it can be seen that only a small error of less than 10% was observed overall.

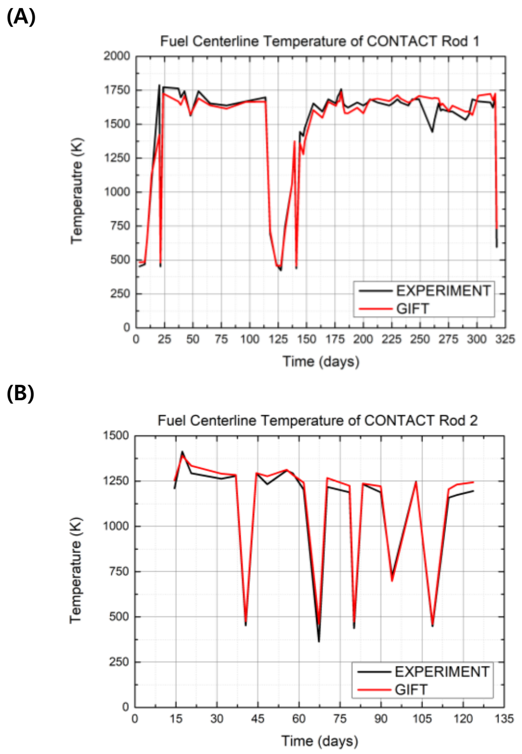

Fig. 6. Comparison between calculated and measured fuel centerline temperature of the case 'CONTACT' (A) rod 1 and (B) rod 2

Table 4. Comparison of burnup and fission gas release between GIFT execution results and BR-3 experiment results

CASE	FGR[%]		
	EXPERIMENT	GIFT	error rate[%]
24-I-6	21.8	23.6	7.69
28-I-6	13.2	13.3	0.70
36-I-8	33.8	42.6	20.6
111-I-5	14.4	13.8	3.98

4.3. case 'HBEP'

From the case 'HBEP,' high burnup effects program carried out at the Battelle North-west Laboratories, data of 63 rods are used for verification [4]. Fig. 7 shows the comparison of the internal pressure at the end of operation among the experimental data of the case 'HBEP' and the calculation result of the GIFT code. Most of the code execution results showed a difference of less than 1 MPa from the experimental data.

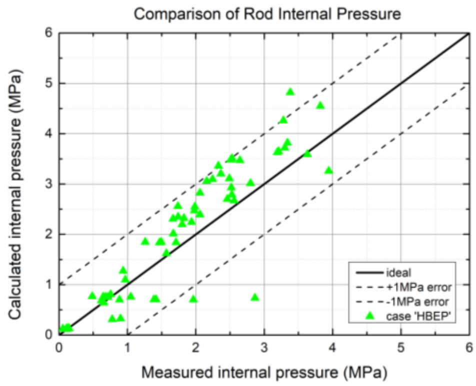

Fig. 7. Comparison between calculated and measured rod internal pressure of the case 'HBEP'

4.4. case 'USPWR 16x16'

The data in the case 'USPWR 16x16' is from the 1980s US-PWR 16x16 lead test assembly extended burnup demonstration program [5]. In a commercial PWR in the United States, two 16x16 assemblies were irradiated and their data is included in this case.

Table 5 includes comparison results between 'USPWR 16x16' experimental data and GIFT results for the void volume inside the rod and the total amount of collected gas. An error of no more than 2mL was shown in both rod data, demonstrating that the experimental data and GIFT's calculations were consistent with high accuracy.

Table 5. Comparison of void volume at end of life and total collected gas volume in STP between GIFT execution results and USPWR 16x16 experiment results

CASE	Void Volume [mL]			Collected Gas Volume in STP [mL]		
	EXPERI-MENT	GIFT	error [mL]	EXPERI-MENT	GIFT	error [mL]
TSQ 002	17.8	16.0	1.8	21.8	23.6	1.8
TSQ 022	31.0	32.2	1.2	13.2	13.3	0.1

5. Conclusion

The domestic nuclear fuel performance code GIFT has been developed to respond to the emergence of new nuclear fuel concepts. GIFT achieved high accuracy and fidelity by embedding both existing developed models and new models with higher performance, which were then verified by comparing them to actual experimental data. In the future, the code and its individual model will be updated as verification with more data proceed.

ACKNOWLEDGEMENT

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