REACTIVITY INITIATED ACCIDENT ANALYSIS METHOD USING MULTI-PHYSICS COUPLED CODE SYSTEM BASED ON RAST-K v2.0

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ABSTRACT

A Reactivity Initiated Accident (RIA) analysis method is demonstrated using the multi-physics coupled code system of RAST-K. Four computational codes based on three different physics models are coupled in one direction without feedback. RAST-K, CTF, and FRAPCON/FRAPTRAN represent neutronics, sub-channel thermal hydraulics (T/H), and fuel performance, respectively. The neutronics code calculates fuel pin power distribution, the sub-channel T/H code calculates coolant properties, and the fuel performance codes calculate fuel behaviours. Steady-state calculation is performed by RAST-K, CTF, and FRAPCON to simulate the long-term behaviour of the reactor during burnup. Considering normal operating conditions as an initial point, a transient calculation is performed with RAST-K and FRAPTRAN to simulate a Rod Ejection Accident (REA) for the peak pin. Cycle 1 of a typical OPR1000 LWR reactor is analysed by the coupled code system at the beginning of a cycle. A relative core power level of 270%, a fuel centreline temperature of 860 °C, and a 25 cal/g fuel enthalpy increase were achieved for the peak pin during the REA simulation.

1. Introduction

Ulsan National Institute of Science and Technology (UNIST) has developed a nuclear reactor core analysis code system called STREAM/RAST-K [1], [2], sponsored by Korea Hydro & Nuclear Power Central Research Institute (KHNP-CRI). The neutron transport code (STREAM) performs a lattice calculation to provide nuclear data to the nodal diffusion code RAST-K, adopting a two-step approach for nuclear reactor analysis. To construct a high fidelity, multi-physics, and multi-scale reactor core analysis code system, RAST-K has implemented multi-physics capability by employing the sub-channel thermal hydraulics (TH) code CTF [3] and fuel performance codes FRAPCON [4] and FRAPTRAN [5]. The coupled calculations are performed in one direction.

A Reactivity Initiated Accident (RIA) is a nuclear reactor accident which involves an unwanted increase in reactor power [6]. The reactor power increase could damage the reactor core, and could lead to disruption of the normal performance of the reactor. In Pressurised Water Reactors (PWR), control Rod Ejection Accidents (REAs) can occur due to mechanical failure of the control rod drive mechanism or its housing, such that the reactor coolant system pressure would cause the ejection of a partially or fully inserted control rod, and drive the shaft to its fully withdrawn position. If the reactor is operating at or close to the critical position, the consequences of this mechanical failure include rapid reactivity insertion and core power increase, together with an asymmetric core power distribution. This may lead to localised fuel rod damage. The fuel temperature rapidly increases resulting in fuel pellet thermal expansion, and in very severe cases, failure in the cladding. For this reason, resistibility to RIA accident is an important parameter for nuclear reactor safety and licensing.

In a recent trend, conventional conservative analysis for nuclear reactor safety is being replaced by best-estimate analysis. Concurrent with this trend, an RIA analysis method using a multi-physics coupled code system was devised, and applied to the UNIST nodal diffusion code, RAST-K. The calculation is performed in a once-through way without feedback because the establishment of the multi-physics coupled code system is the starting point. In addition, the T/H properties as determined by steady-state calculations are used for the fuel performance calculation. With respect to the initial condition achieved based on the steady-state calculation, RAST-K simulated REAs to calculate pin power distribution and FRAPTRAN was used to analyse the fuel behaviour of the peak pin during the simulation. The RIA analysis method and its application to REA simulation of a typical OPR1000 type reactor is presented in this report.

In Chapter 2, the computational codes involved in the multi-physics coupled calculations are examined and Chapter 3 is a description of the once-through coupling strategy for both the steady and transient state, to analyse RIA in a PWR. In Chapter 4, the test results for a typical OPR1000 type reactor is presented.

2. Overview of the coupled calculation scheme

2.1. Computational codes

STREAM [1] is a lattice physics code that is being developed at UNIST, and it uses the Method of Characteristics (MOC) for the transport calculation. STREAM generates nuclear data, such as 2-group cross of a fuel assembly and the reflector models used in RAST-K. By adopting a Pin based Slowing down Method (PSM) as the resonance treatment approach, STREAM can achieve numerical results with a higher accuracy. STREAM provides 2-group cross-section and group constant data to the codes for reactor analysis.
RAST-K [2] is a reactor core analysis code developed at UNIST for steady and transient analysis of the reactor core, based on diffusion theory. The process involves solving a nodal diffusion equation using a 3-dimensional 2-group UNM (unified nodal method), and it adopts CRAM (Chebyshev Rational Approximation Method) and uses a micro depletion method for the depletion calculation and the theta time differencing method for transient calculations.
Linkage of STREAM and RAST-K is performed by the STORA code which gathers STN files containing cross-section and group constant data calculated by STREAM, which is then reformatting for use in RAST-K. Fig 1 presents a flowchart of the STREAM/RAST-K code system.

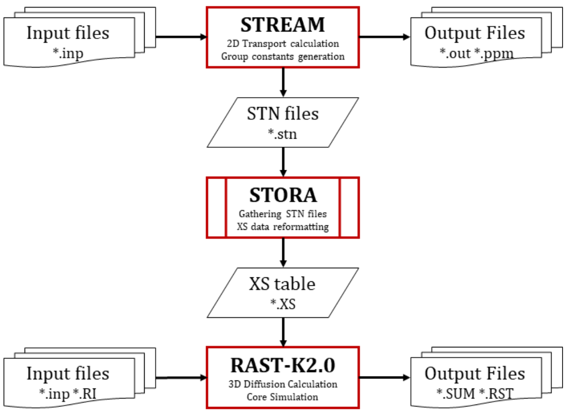


Fig 1. Two-step flowchart of STREAM/RAST-K code system.

CTF [3] is a sub-channel T/H code designed for LWR vessel analysis. It solves sub-channel forms of 9 conservation equations using a two-fluids, three-field (fluid film, fluid drops, and vapour) modelling approach. Because CTF models multi-rod arrays in the reactor core, channel to channel flow (cross-flow) can be considered in the simulation, which results in more accurate coolant properties than other T/H models that simulate single fuel rods. CTF can be easily coupled with other physics codes (neutronics and fuel performance codes) because it provides rod-centred output quantities (although it performs channel-centred calculations) and a coupling interface module,

FRAPCON [4] calculates the steady-state thermal-mechanical response of oxide fuel rods in LWRs, during long-term burnup conditions of normal power reactor operation. For each time step, coolant heat conduction in the axial direction, heat transfer of the fuel rod in the radial direction, deformation of fuel and cladding, and fission product generation and release are sequentially calculated. Accurate fuel rod property behaviours can be determined because the oxide fuel property changes and fission gas release are considered. Moreover, other parameters, such as fission gas release, cladding corrosion, cladding hoop strain, and gap thickness are available for quantification. The steady-state fuel behaviour calculated by FRAPCON can be used as the initial input condition for FRAPTRAN.

FRAPTRAN [5] is a computational code for analysing the thermomechanical behaviour of LWR fuel rods under transient conditions and accidents such as Loss of Coolant Accident (LOCA) and RIA. It uses an axisymmetric representation of the fuel rod geometry, but a model which accounts for local non-axisymmetric cladding deformation can be used. Cladding failure under RIA is predicted using a strain-based failure criterion. The calculated plastic deformation in the hoop direction for each time step is compared with a threshold value, which is correlated to the temperature and hydride content of the cladding material. The failure correlation is based on uniform elongation data from burst tests and uniaxial ring tensile tests on irradiated Zircaloy-2 and Zircaloy-4, which makes the failure criterion conservative.

2.2. Steady-state calculation

To simulate the normal operating condition of the reactor core based on a steady-state calculation, RAST-K, CTF, and FRAPCON are coupled in a once-through way. The static library feature is used for the coupling of the computational codes. At the first call of CTF, input files are generated, and these input files are used for further T/H calculation with the assumption of a constant reactor core geometry. The main programme of the FRAPCON is converted to a subroutine and input parameters are entered when calling the FRAPCON subroutine without generating an input file for this subroutine. Since FRAPCON performs analysis on a single pin, the subroutine is called for every fuel rod in the model. In current coupling, FRAPCON simulates each of the burnup steps from the fresh fuel. i.e. when RAST-K performs a calculation at the n^{th} burnup step, FRAPCON simulates from the 1^{st} burnup step to determine the fuel behaviour at the n^{th} burnup step.

For each burnup step, the 3-dimensional node power distribution is computed by solving the nodal diffusion equation. Using the form function generated by STREAM, RAST-K reconstructs the pin-by-pin power distribution from the node power distribution and provides it to CTF and FRAPCON. Using the pin power distribution, the sub-channel T/H calculation is performed. Coolant T/H properties are computed and the coolant temperature and pressure for each node are transferred to FRAPCON. The coolant temperature and pressure from CTF are used as the boundary conditions for the heat transfer calculation in the fuel rod. Steady states of each physics are reached in order of RAST-K, CTF, and FRAPCON because the coupling is in a once-through manner. As a result of the steady-state calculation, nuclear safety parameters of normal operation such as the coolant and fuel temperature distribution, gap thickness, ZrO_2 oxide thickness, hoop stress, and strain can be calculated, while it is impossible to determine these parameters based on a standalone neutronics calculation.

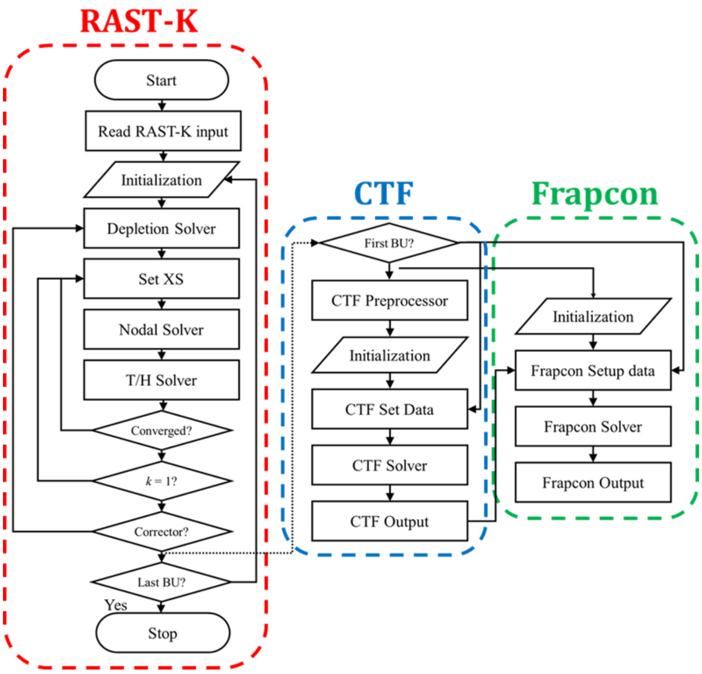


Fig 2. Steady-state coupled calculation flowchart

2.2. Transient calculation

The steady-state calculation results from RAST-K and FRAPCON are used as the initial conditions for the transient calculations of RAST-K and FRAPTRAN, respectively. Since REA simulation with short time period is performed in this study, it is assumed that the fuel rod is adiabatic such that the coolant bulk temperature is constant. i.e. the T/H properties determined based on the steady-state CTF calculations are used as boundary conditions for the thermomechanical calculations of the fuel rod by FRAPTRAN.

During the initiation of the REA simulation, the coupled code system performs calculations of normal operating conditions. As the steady state of the selected burnup step is achieved, RAST-K simulates a rod ejection accident for the same reactor core to determine the core power level and pin power distribution during the process. Using the restart file generated by FRAPCON of the steady state calculation as an initial condition of the fuel rod's pin power distribution of the peak pin, FRAPTRAN calculates fuel behaviour during the REA simulation. Fig 3 represents the transient calculation scheme for the RIA analysis.

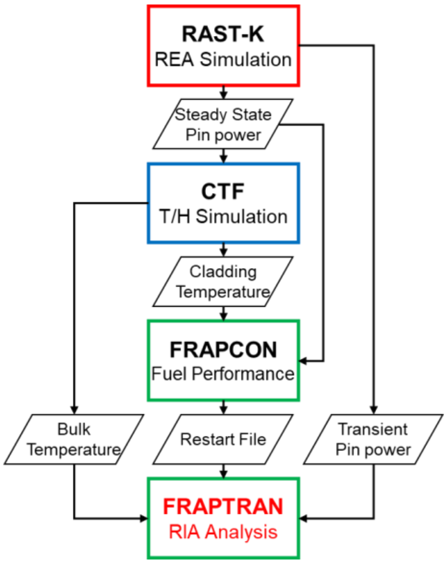


Fig 3. RIA analysis flowchart of RAST-K/CTF/FRAPCON/FRAPTRAN

3. Coupled calculation model

3.1. OPR1000 model

Cycle 1 of a typical OPR1000 reactor is selected for the numerical test of the coupled calculation. Fig 4 shows the loading pattern of the core that is composed of 177 fuel assemblies (FA) with a 16 by 16 array of 236 fuel pins and 5 guide tubes. The fuel enrichment varies according to assembly types A, B, and C, from 1.42 wt% to 3.42 wt%. A quarter core model was used with 48 axial nodes: 2 in the top and bottom reflector regions each, and 46 in the active fuel region for the neutronics calculation. For the T/H and fuel performance calculations, 10 axial nodes were used for the fuel region. Each length of 6 T/H and fuel performance nodes at the middle corresponds to 5 neutronics nodes, and the nodes at the top and bottom correspond to 4 neutronics nodes. From the beginning of cycle (BOC), 17 burnup steps are analyzed up to 13.8 GWd/MTU. The total core power is 2,815 MWth, the coolant total flow rate is 16,315 kg/s, and the inlet temperature is 296.1 °C.

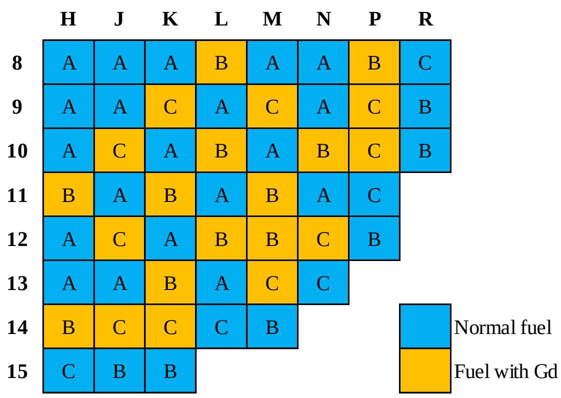


Fig 4. Loading pattern of OPR1000 Cycle 1

3.2. Control rod ejection scenario

Usually, the increase in enthalpy under a REA accident at BOC condition with a given prompt reactivity insertion is higher than at EOC because fresh fuel assemblies exist in the core. For a more conservative evaluation, the BOC condition is selected for the RIA analysis. At the zero-power condition of BOC, the control rods of the R4 group are ejected in 0.1 s. The core model includes the regulating bank (R1-R5), shutdown bank (SA, SB), and the part strength (PS). The regulating and shut down banks use B_4C as the absorber material, and the part strength uses Inconel to control the axial shape only. Fig 5 shows the locations of the control rod in the model.

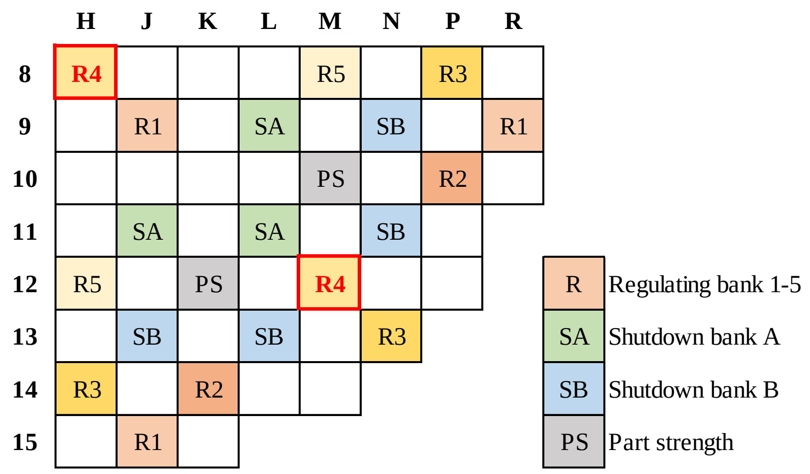


Fig 5. Control rod locations in the model

3.3. Numerical results

Firstly, RAST-K calculates the fuel pin power distribution during the REA simulation. Using the power distribution, FRAPTRAN calculates fuel mechanical properties, such as fuel enthalpy increase, fuel centreline temperature, cladding temperature, and hoop strain, and these parameters are quantified for RIA safety analysis.

Fig 6 shows the relative pin power distribution of the power peak node at 0.28 s. Fig 7 shows the reactor core power level relative to the normal operation power (2,815 MWth) and peak pin power during the simulation. The control rod is completely ejected in 0.1 s and a maximum core power level of 270% is reached at 0.28 s.

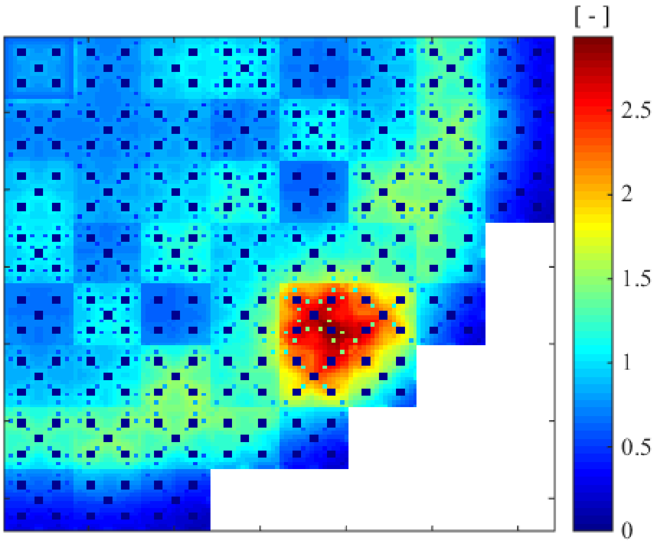
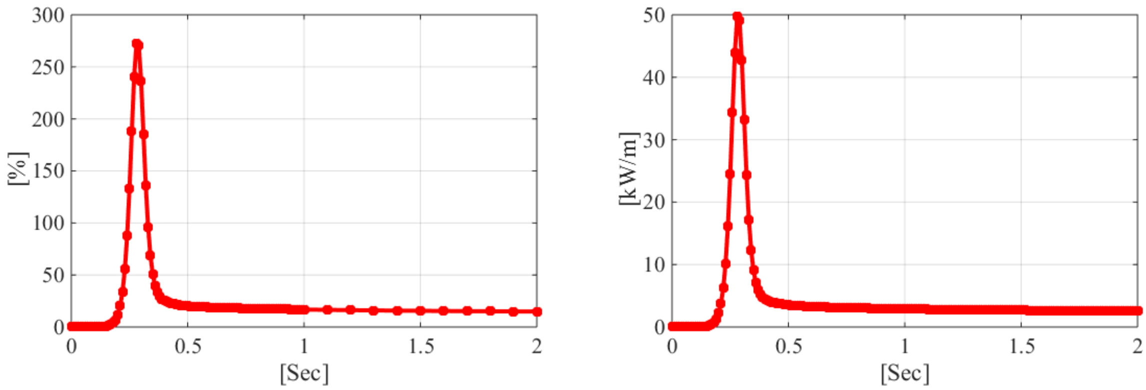


Fig 6. Relative fuel pin power distribution at peak position at 0.28 s

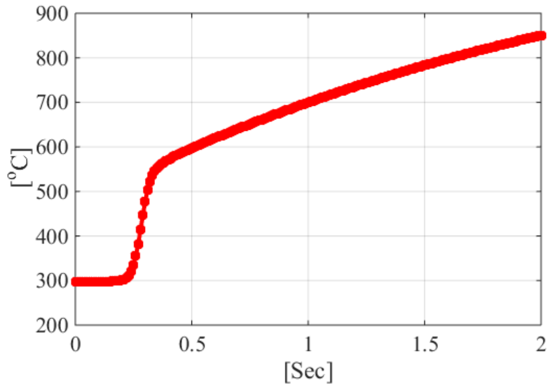


(a) Core power level

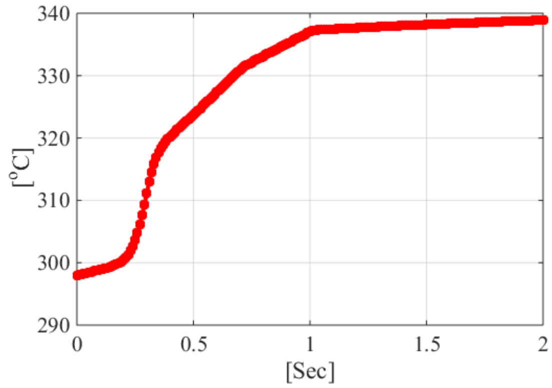
(b) Peak pin power

Fig 7. Reactor core power level during the RIA simulation by RAST-K

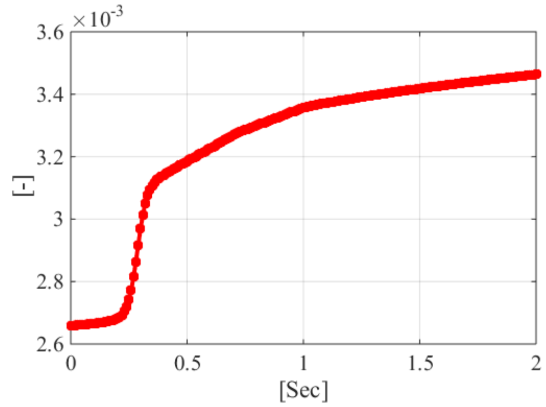
Fig 8 shows the behaviour of the RIA safety parameters including the fuel centreline temperature, cladding temperature, hoop strain, and fuel enthalpy increase from steady-state, of the 7th axial node from the bottom with the highest power and temperature. At steady-state condition, the fuel temperature is 298 °C which is the same as the coolant temperature, and the fuel enthalpy is 17.6 cal/g. During the RIA analysis, the fuel centreline temperature reaches 860 °C and fuel enthalpy increase approaches 25 cal/g. Since the input coolant temperature is fixed, only increases of the safety parameters are observed, which is a limitation of the RIA analysis.



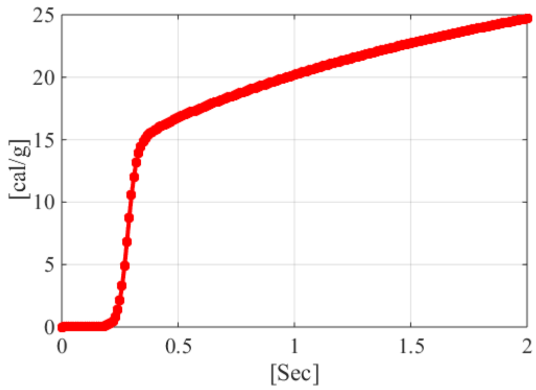
(a) Fuel Centreline temperature



(b) Cladding temperature



(c) Relative hoop strain



(d) Fuel enthalpy increase

Fig 8. Behaviour of RIA safety parameters for the peak pin

4. Conclusion

In this investigation, an RIA analysis method was demonstrated using a high-fidelity multi-physics coupled code system. A control rod ejection accident was simulated by the code system. The simulation was performed at BOC of Cycle 1 of a typical OPR1000 type reactor, where fresh fuel rods are loaded under Hot Zero Power (HZP) condition. A core power level of 270%, a fuel centreline temperature of 860 °C, and a 25 cal/g fuel enthalpy increase for the peak pin are achieved during the REA simulation. The assumption of a constant coolant temperature results in a limitation in the observation of the long-term behaviour of the RIA safety parameters.

In future work, a transient T/H calculation will be included to observe long-term changes in the mechanical property of the fuel rod during REA simulation. In addition, our investigation will include a fully coupled calculation of RAST-K, CTF, and FRAPCON to generate improved best-estimate initial reactor core conditions for the transient calculation. This calculation will also be performed with a fully coupled calculation by RAST-K, CTF, and FRAPTRAN.

5. Acknowledgements

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6. Reference

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