Development of INCTAC Code for Analyzing Criticality Accident Phenomena

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Aiming at understanding nuclear transients and thermal- and hydraulic-phenomena of the criticality accident, a code named INCTAC has been newly developed at the Institute of Nuclear Safety. The code is applicable to the analysis of criticality accident transients of aqueous homogeneous fuel solution system. Neutronic transient model is composed of equations for the kinetics and for the spatial distributions, which are deduced from the time dependent multi-group transport equations with the quasi steady state assumption. Thermo-hydraulic transient model is composed of a complete set of the mass, momentum and energy equations together with the two-phase flow assumptions. Validation tests of INCTAC were made using the data obtained at TRACY, a transient experiment criticality facility of JAERI. The calculated results with INCTAC showed a very good agreement with the experiment data, except a slight discrepancy of the time when the peak of reactor power was attained. But, the discrepancy was resolved with the use of an adequate model for movement and transfer of the void in the fuel solution mostly generated by radiolysis. With a simulation model for the transport of radioactive materials through ventilation systems to the environment, INCTAC will be used as an overall safety evaluation code of the criticality accident.

KEYWORDS: INCTAC, criticality accident, safety evaluation, aqueous fuel solution, multi-group transport equations, validation test, radiolysis void, TRACY

1. Introduction

With the JCO criticality accident in 1999 a need for studying criticality accident was greatly increased. Particularly, from the view point of safety evaluation of the fuel reprocessing plant, its commercial plant is now under construction in Japan, the analysis of the nuclear transients and thermal- and hydraulic-phenomena of the criticality accident postulated for fuel solution system has become one of important items to be studied.

A study of the criticality and thermo-hydraulic phenomena of the JCO accident was made with a simplified model of the point reactor kinetics including temperature and void reactivity coefficients and models for void dissipation and heat removal.[1] A couple of noteworthy study items were obtained from the study: behavior of radiolysis void and heat transfer have important influences in determining the neutronic kinetics in short and long ranges of the transient respectively, use of reactivity coefficients is not preferable because of complicated thermo-hydraulic phenomena during the transient and difficulty of coefficient evaluations, and nuclear cross sections and their changes in the system should be directly treated in the analysis.

2. Development of INCTAC

2.1 Basic Requirements

The code should be applied for analyzing the nuclear transients and thermal- and hydraulic-phenomena of the criticality accident postulated for the fuel solution system. The transients will be very rapid and short in those cases generally postulated, but will continue for a couple of minutes or hours under some uncommon conditions. For completing the safety assessment of the accident, the analysis of behavior and transport of radioactive materials including fission products generated by the chain reactions should be included too. But, analysis of mechanical failure of components, which might be caused by the accident, will be excluded, and an extended change of the boundary conditions, such as deformation of fuel solution vessel, relocation of fuel solution or additional heating up by any other cause, will not be considered.

2.2 Model and Equations

2.2.1 Neutronic Model

Two basic equations, the time-dependent transport equation for the neutron flux (the Boltzmann transport

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equation) and the time-dependent equations for the delayed neutron precursor densities, are used.

With the assumption that the time-dependent neutron flux distribution is described as a product of the amplitude and the shape functions, the numerical solution method based on the quasi-steady state approximation is applied. The amplitude function is determined with the equation for the point reactor kinetics with appropriate coefficients, which are deduced for each term in the basic transport equation. The time-dependent equation for the normalized value of the delayed neutron precursor density is also obtained. The shape function is determined with the equation similar to the basic time-dependent transport equation under the normalization constraint.

2.2.2 Thermo-Hydraulic Model

The two-phase flow model with two components, liquid and gaseous phases, is applied, and a set of the six differential equations for mass, momentum and energy is prepared. An additional set of the equations to determine the momentum and energy transfer between the phases, and the laws to describe the friction loss and the heat transfer at wall surfaces is also used. Because a very wide change of two-phase conditions, from liquid to gaseous, is expected during the transient, many different modes of the friction loss and the heat transfer for the two-phase flow should be considered in the thermo-hydraulic models.

2.3 Structure of INCTAC

2.3.1 Overall Structure

The INCTAC code is composed of a couple of modules corresponding to the analytical models, and these modules are connected and operated on a Parallel Virtual Machine (PVM) under the regulation of the master control module as shown in Figure 1.

The PVM is developed at ORNL [2] and other institutes for realizing the parallel processing on a non-parallel computer, and is used in the INCTAC code for linkage of the analytical modules.

The master control module makes overall control, but the data necessary for calculations in each module are directly transferred between the analytical modules. The calculated results are stored as database for later output and retrieval.

2.3.2 Reactor Dynamics Module

The reactor dynamics module is composed of two sub-modules: amplitude function module and shape function module.

In the amplitude function, or reactor kinetics, module, the reactivity for whole reactor is evaluated with the angular fluxes and the adjoint angular fluxes and then the power of reactor is calculated. This module is developed based on the Kaganove's method used in the TRAC-PF1 code.[3]

In the shape function module, the angular fluxes, the reactor power distributions and the fission rates for whole reactor are calculated with the analytical models based on the quasi-steady state approximation. This module is based on two-dimensional discrete-ordinate method code TWODANT developed at LANL.[4]

2.3.3 Thermal- and Hydro-Dynamics Module

Numerical analysis of the thermo-hydraulic model is embodied with TRAC code, which has been developed at LANL and used widely in the liquid-gaseous two-phase flow analyses. The semi-implicit method is applied for numerical integrations: the pressure propagation and the liquid-gaseous phase interactions which have a relatively short time scale in their changes, are treated by the implicit method, whereas the propagations of mass, momentum and energy, or the convection terms, which have a relatively long time scale in their changes are treated by the explicit method.

In the TRAC code, in addition to the evaporated vapor, incondensable gases can be treated in the gaseous phase, and this feature is very useful in the analysis of criticality accident, where a large amount of the radiolysis gases will be generated. Models for describing the radiolysis gas generation are additionally prepared.

2.3.4 Cross-section Preparation Module

All neutron cross-sections to be used in the reactor dynamics module are prepared in the cross-section preparation module with fitting functions. Fuel solution is the major component or material to be considered in the criticality accident analysis, and its neutronic characteristics are generally dependent on the temperature and the density, and additionally on the fuel or fissionable nuclides concentration when intense evaporation is expected.

The neutron cross-sections of structure materials and control rods are also prepared in the cross-section preparation module, but their dependencies on the thermal and hydraulic conditions are not considered.

Preparation of the original cross-section data using the SCALE-4.4 systems, [5] and generation of the fitting functions using the least square method is also integrated into the module.

2.3.5 Fission Product Analysis Module

To evaluate the radiation source term to the environment, the fission product analysis module is prepared to calculate the amounts of the fission products during the criticality accident. Widely used point depletion code ORIGEN [6] is used.

2.3.6 Overall Control

As shown in Figure 1, the operation of the INCTAC code is regulated with the master control module. The module has two key functions of controlling the analytical steps: control of the subordinate process modules and control of the event time. The former control function includes the selection of the analytical modules and the direction of their start-and-stops, and the latter monitor the data generated by each analytical module and controls the time progression.

2.3.7 Database

The results calculated in the reactor dynamics module and in the thermal- and hydro-dynamics module and the applied calculation conditions are stored into the database. Member of the database corresponding to each calculation case is composed of the header section to store the calculation conditions and the trend section to store the calculated results during the transient.

The database can be handled similar to ordinary used one, and the functions of 'add,' 'delete,' or 'retrieval' of the data is provided. Printing or plotting of the data is also possible

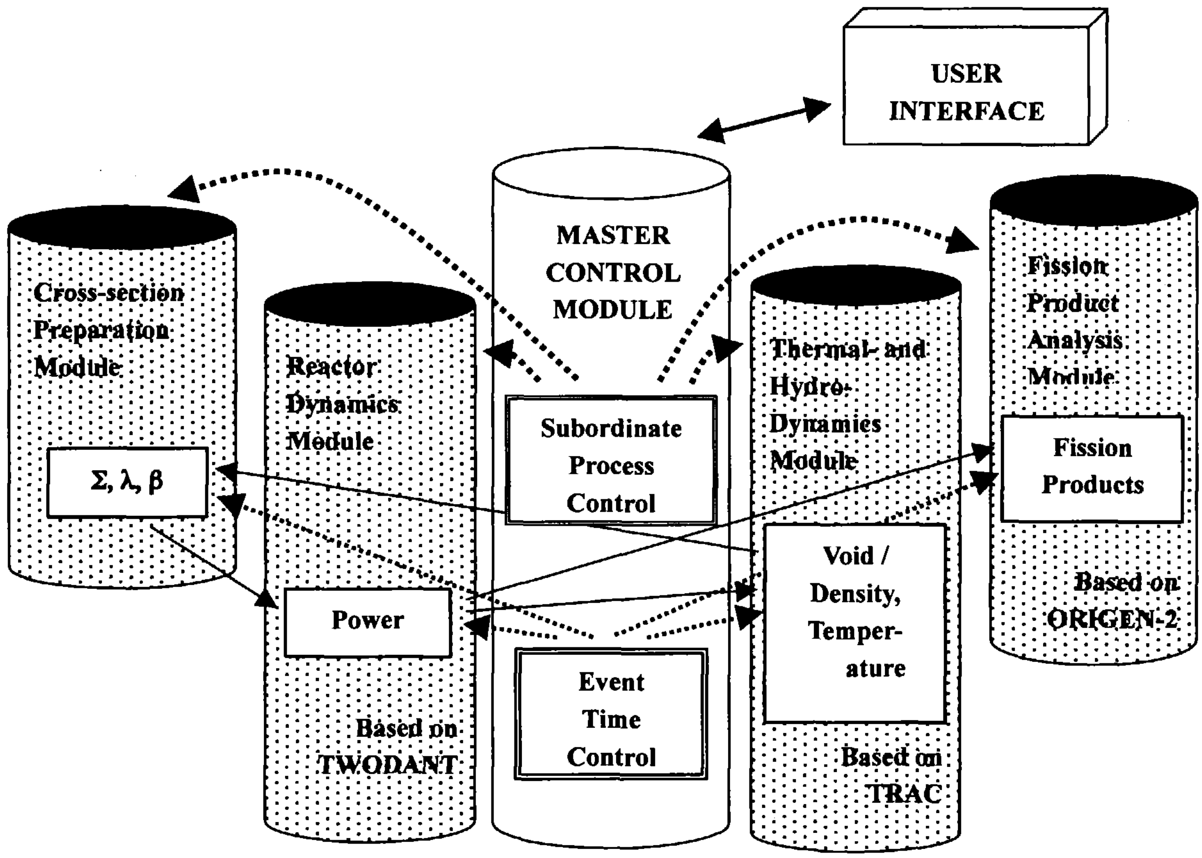

Fig. 1 Module structures and calculation flows of INCTAC code.

3. Validation Tests

3.1 TRACY Experiments

3.1.1 Outline of Facility

TRACY is a JAERI's transient experiment critical facility, and is used for the study of transient phenomena under the supercritical condition of uranyl nitrate solution: to confirm the safety margin included in the safety evaluation of the critical accidents postulated in a fuel reprocessing plant. TRACY attained its first criticality in December 1995.The reactor core of TRACY is an annular tank of 50 cm in diameter and the aqueous solution of uranyl nitrate of 10 % enriched. At the central hole of 7 cm in diameter, a transient-rod is equipped.

Three different methods of reactivity inserting are provided: ramp or pulse withdrawal of the transient-rod and fuel solution feeding.

Details of the facility and the experiment conditions and observed results were available in the databook published recently, [7] and the validation tests of INCTAC were made for a couple of experiments of different experiment conditions.

3.1.2 Data Preparation

The analytical model of TRACY is schematically shown in Figure 2. The reactor-tank is made of stainless steel (SS) and the absorption material of borated carbon (B_4C) is used in the transient-rod. Cover gas of the fuel solution in the reactor-tank is air, but is replaced with aluminum (Al) in the model for preventing too slow convergence of the neutron fluxes.

To complete the transient calculation within an acceptable computation time, as small as nine neutron energy groups were used, and the atomic number densities of the fuel solution were calculated using the formula of Moeken.[8] Determination of the fitting functions and regeneration of the cross-section data with the fitted functions were precisely examined and their applicability were confirmed.

The parameters for reactor kinetics, such as delayed neutron fractions, decay constants of the precursor nuclides and energy spectra of the delayed neutrons, were also prepared with appropriate calculation models and data including fuel conditions.

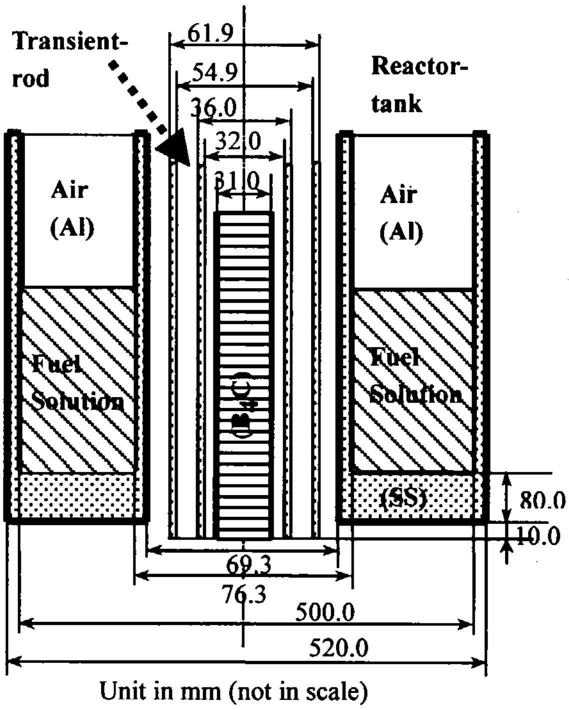

Fig. 2 Analytical model for the TRACY experiments: annular reactor-tank and transient-rod at the center.

3.2 Transient-rod Ramp Withdrawal

TRACY Run 109, one of the transient-rod ramp withdrawal experiments, was conducted at November 10th, 1998. The rod was withdrawn at the speed of 875 cm/min, and the integrated reactivity of 2.97 \$ was inserted in 7 seconds.

The calculated results of the INCTAC code were shown and compared with the observed one in Figure 3. Using the INCTAC code at the development stage of 2000, a relatively good agreement of the calculation and the observation was confirmed for the overall transient, except a slight discrepancy of the time when the reactor power reached its peak and the shape by which the power was declining after the peak. An inadequate model for the hydraulics or the behavior of radiolysis voids was thought as a cause of the discrepancy. And, the more precise model for the thermo-hydraulics described in sections 2.2.2 and 2.3.3 was programmed. As sown in the figure, the improved result with the INCTAC (2001) code showed a very good agreement with the observed results for the timing of peak power and the shape of transient.

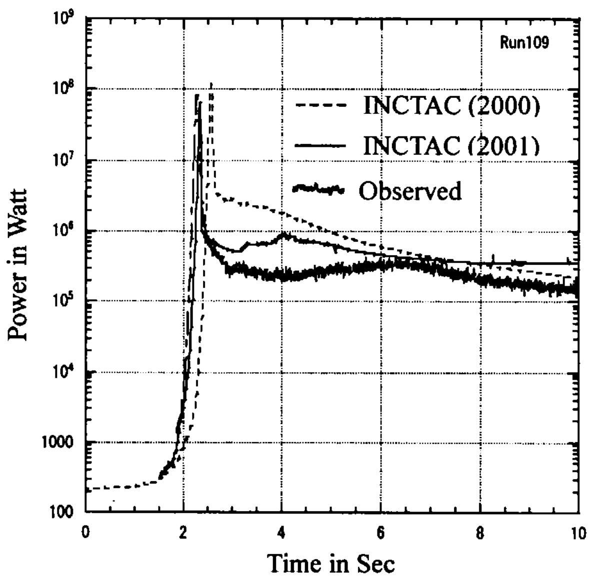

Fig. 3 Power transient of Run No. 109.

3.3 Fuel Solution Feeding

TRACY Run 163, one of the fuel solution feeding experiments, was made at July 4th, 2000. The fuel solution was fed at the rate of 60 liter/min, or the solution level rising at about 5 mm/sec, for the duration of about 34 seconds, and the inserted reactivity was integrated to 1.25 \$.

The calculated result of the INCTAC (2001) code was shown and compared with the observed result in Figure 4. A very good agreement with the observed transient for the shape and level of the power after the initial short transient was obviously indicated.

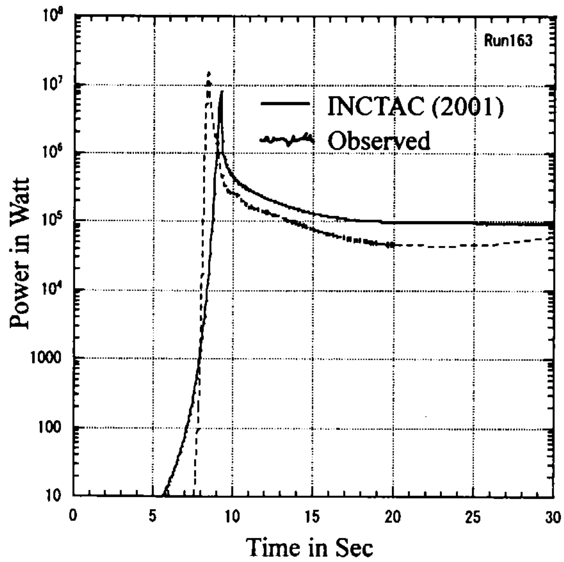

Fig. 4 Power transient of Run No. 163.

4. Further Development

4.1 Long-sustained Transients

Long-sustained transient experiments, up to five hours, were conducted for the fuel solution feeding cases.

Simulation of theses experiments will be useful for validation study of the thermo-hydraulic models including heat removal mechanisms and void transfer behaviors under higher temperature and evaporation conditions. And also to be studied in the simulation of the long-sustained experiments is the generation and accumulation of fission products in fuel solution, which might be a source of the radioactivity hazard with criticality accidents.

4.2 Overall Safety Evaluation

Safety evaluation of the criticality accidents is one of the important study items where the INCTAC code will be applied. For this purpose, physical and chemical models of analyzing the radioactive material and fission product transfer from fuel solution to upper covering space, and simulation models of predicting radioactivity release to the environment through cover gas treatment and ventilation systems should be integrated to the code.

5. Conclusion

Aiming at understanding the nuclear transients and thermal- and hydraulic-phenomena of the criticality accident, a numerical analysis code named INCTAC was newly developed at the Institute of Nuclear Safety.

The code is applicable to the analysis of criticality accident transients of the aqueous homogeneous fuel solution system. The neutronic transient model is composed of the equations for the kinetics and for the spatial distributions, which are deduced from the time dependent multi-group transport equations with the quasi steady state assumption. The thermo-hydraulic transient model is composed of a complete set of the mass, momentum and energy equations together with the two-phase flow assumptions.

The validation tests of INCTAC were made using the data obtained at TRACY. Both the transient-rod ramp withdrawal at its nearly maximum speed and the fuel solution feeding at a fixed rate were simulated and studied. The calculated results with INCTAC showed a very good agreement with the experiment data. A slight discrepancy calculated with an earlier version of the code for the time when the peak of reactor power was attained was resolved with application of the adequate model for movement and transfer of the void in the fuel solution mostly generated by the radiolysis process.

With simulation models for the transport of radioactive materials through ventilation systems to the environment, INCTAC will be used as an overall safety evaluation code of the criticality accident in the near future.

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