JNES / SA&E Div - Japan
October 2004

CRITICALITY ACCIDENT CODE

IDENTIFICATION SHEETS

INCTAC

DRAFT

GENERAL INFORMATIONS
(1)

Designation of the code	INCTAC	
Summary (General purpose)	INCTAC is applicable to the analysis of criticality accident 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 complete set of the mass, momentum and energy equations together with the two-phase flow assumptions.	
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Status of code	First version released (date and reference number)	Not yet.
	Current version released (date and reference number)	Not yet.
	Language program / Modularity	FORTRAN77 + C
	Operating system (windows, linux, unix, …)	Sun Solaris, IBM-UIX, HP-UX
	Software requirements (fortran compiler, …)	FORTRAN-77 compiler, pvm 3.4.1
	Portability (PC / Workstation / Supercomputer)	Sun Blade 2000, IBM RS-6000, HP J6700
	Avaibility / web site (executable, source files, data files, …)	N/A
	Typical running time (for one calculation)	About 10 hours
Comments		

GENERAL INFORMATIONS
(2)

User Interface	Users access Workstation from PC for executing calculations, and post processing of calculated results on PC.
Calculated Standard Outputs / and Units Time step output: power, energy, pressure, temperature, … Main characteristics: first peak power, total energy release, maximum pressure, temperature, time of boiling, …	Time step output: total power (watt, fission/s), energy release (Joule, fissions), reactivity (total, void, temperature, inserted), liquid level R-Z distributions: temperature, pressure, density, void ratio, velocity, mass flow Main characteristics: first peak power, total energy release, maximum pressure, temperature
Graphic editor	TRAC-PF1 Graphic tool (X TRAC VIEW [ver 2.3c])
Quality Assurance (data and code package)	
Contact Person (name of the contact for the code)	Hideo KONISHI Safety Analysis and Evaluation Division, JNES 3-20 Toranomon 4-chome, Minatoku, Tokyo 105-0001 Japan
Comments	

GENERAL DESCRIPTION
(3)

Fissile Materials	**Physical Forms**				
	Solution (nitrate,fluorure, sulfate, …)	Powder (dry, wetted, …)	Metal (dry, wetted, …)	Fuel rods	…
Uranium (isotopic content %)	Nitrate solution				
Plutonium (isotopic content %)	Nitrate solution				
Mixed Plutonium / Uranium (isotopic content %)	Nitrate solution				
Geometry description Cylindrical, spherical, … Space Dimension (1D, 2D, …), … Meshing / Region, … Finite Element Method, …	Cylindrical, 2D (R-Z), Meshing				
Comments					

<table>
<tr><th colspan="3">DESCRIPTION OF MODELS USED</th></tr>
<tr><td>Neutronic Power / Kinetics</td><td>Point kinetic equation, transport or diffusion theory, …</td><td>Quasi-steady state approximation, or transport theory</td></tr>
<tr><td rowspan="2">Reactivity and Reactivity feedback</td><td colspan="2">Transport or diffusion theory, mathematical formulas, input or calculated data
(reactivity insertion, temperature coefficients, Doppler, dilatation, …)</td></tr>
<tr><td colspan="2">Transport perturbation theory
$$\begin{aligned}&\rho(t)\\ &= 1+\frac{1}{F(t)}\left\langle w(E,\vec{\Omega},\vec{r})\left[-\vec{\Omega}\nabla\,\psi(E,\vec{\Omega},\vec{r},t)-\Sigma_t(E,\vec{r},t)\psi(E,\vec{\Omega},\vec{r},t)\right]\right\rangle\\ &+\frac{1}{F(t)}\left\langle w(E,\vec{\Omega},\vec{r})\int dE'\,d\vec{\Omega}'\,\psi(v',\vec{\Omega}',\vec{r},t)\Sigma_s(E',\vec{\Omega}'\rightarrow E,\vec{\Omega},\vec{r},t)\right\rangle\end{aligned}$$
$F(t)$: perturbation denominator
$\psi(E,\vec{\Omega},\vec{r},t)$: directional flux
$w(E,\vec{\Omega},\vec{r})$: initial adjoint directional flux
Input:
rod withdraw velocity
cross sections Σ (T_f, ρ)
Calculated data:
total reactivity, feedback reactivities
effective delayed neutron fraction
prompt neutron generation time</td></tr>
<tr><td rowspan="3">Thermal – hydraulics
Hydrodynamics</td><td>Thermal
(heat conduction, convection, boiling, …) / Meshing and region</td><td>Pseudo-three-dimensional (R-Z-Θ) calculation</td></tr>
<tr><td>Multi-phase flow</td><td>Two-phase flow model with liquid and gaseous phases</td></tr>
<tr><td>Fluid motion
/ Meshing and region</td><td>Set of the six differential equations for mass, momentum and energy
Equations to determine the momentum and energy transfer between the phases
Laws to describe the friction loss and the heat transfer at wall surfaces</td></tr>
</table>

	Pressure modeling	Momentum equations with given boundary conditions for inlet velocity and outlet pressure
Radiolysis (for solutions)	Radiolytic formation and migration models	
	Gas (void) generation rate $\Gamma = M_a GN$: after the saturation ($C \geq C_0$) Γ : radiolysis gas (void) generation rate [(kg/m^3)/s] M_a : molecular weight of radiolysis gas [kg/mol] G : energy to generate radiolysis gas [mol/J] N : power density [watt/m^3] C : radiolysis gas concentration [mol/m^3] C_0 : radiolysis gas saturation (threshold) concentration [mol/m^3] Gas concentration $\frac{\partial C}{\partial t} + \nabla \cdot (Cv_l) = \Gamma_{mol} = GN$: before the saturation ($C < C_0$) v_l : liquid flow velocity [m/s] $\Gamma_{mol} = GN$: radiolysis gas generation rate [(mol/m^3)/s] $C = C_0$: after the saturation ($C \geq C_0$) Migration process: mass conservation of non-condensable gas (void) $\frac{\partial \alpha \rho_a}{\partial t} + \nabla \cdot (\alpha \rho_a v_g) = 0$ α : void fraction [-] ρ_g : density of void [kg/m^3] v_g : void flow velocity [m/s]	
Data libraries External or/and Internal (constants, calculated / tabulated, experimental, bibliography, …) …	Neutronics – kinetics (cross sections libraries, Kinfinity, neutron lifetime, delayed neutrons, …)	Cross sections: SCALE-4.4 44 group library, CSASN-XSDRN-ICE 9 groups collapsed Delayed neutrons: SRAC95-JENDL3.2 library
	Thermal and hydrodynamics (heat capacity, conductivity, …)	TRAC-PF1 models Jay W. Spore, et al., “TRAC-PF1/MOD2 VOLUME 1. THEORY MANUAL,” LA-12031-M, Vol.1, NUREG/CR-5673 (1993). L. A. Guffee, et al., “TRAC-M Fortran77 Version 5.5 PROGRAMMER’S GUIDE VOLUME III,” (1998). R. G. Steinke, et al., “TRAC-P USER’S GUIDE VOLUME II,” (1993).

	Radiolysis (yield, threshold formation, velocity, …)	Yield and threshold: CRAC experimental data Forehand, H.M., Jr., “Effect of Radiolytic Gas on Nuclear Excursions in Aqueous Solutions,” NUREG/CR-2517 (1982).
Comments		

VALIDATION BASE OF THE CODE

Summary of the main assumptions in the code	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.	
Limitations to the use of the code	Aqueous solution of fissile media Cylindrical 2D (R-Z) geometry	
Experimental benchmarks (reactor : fissile media, geometry, reactivity insertion, duration,…)	Reactor: TRACY / JAERI's transient experiment critical facility Fissile media: uranyl nitrate solution Geometry: annular tank of 50 cm in diameter and a transient–rod at the center hole of 7 cm in diameter Reactivity insertion: ramp or pulse withdrawal of transient-rod and fuel solution feeding Benchmarks: Run Nos. 109, 110, 160, 161, 163, 164	
Past accidents (simulations)		
Codes comparison	Validation of the modelling with standard codes (neutronics, thermal, …)	
	Accidents code	
Domain of validation and level of confidence	Comparisons with experimental benchmarks from TRACY show good agreement for both their first rapid transients and longer slow oscillations.	
References (reports, communications, publications,…)	S. Mitake, et al., "Development of INCTAC Code for Analyzing Criticality Accident Phenomena," Proc. of ICNC2003, 2003.	