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# Recent Improvements to the RELAP5-3D Code

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**Abstract –** *The RELAP5-3D computer program has been recently improved. Changes were made as follows: (1) heat structures are allowed to be decoupled from hydrodynamic components, (2) built-in material properties for heat structures have been made consistent with those in MATPRO and the Nuclear Systems Materials Handbook (they are now documented in the RELAP5-3D manual), (3) Schrock's flow quality correlation is now used for a downward oriented junction from a horizontal volume for the stratification entrainment/pullthrough model.*

## I. INTRODUCTION

The RELAP5 series of codes has been developed at the Idaho National Laboratory for over 25 years under sponsorship of the U. S. Department of Energy, the U. S. Nuclear Regulatory Commission, members of the International Code Assessment and Applications Program, members of the Code Applications and Maintenance Program, and members of the International RELAP5 Users Group. Specific world-wide applications of the code have included simulations of transients of light water reactor systems such as loss of coolant, anticipated transients without scram, and operational transients such as loss of feedwater, loss of offsite power, station blackout, and turbine trip. RELAP5-3D[1], the latest in the series of RELAP5 codes, extends the applicability of earlier versions to include an integrated multidimensional thermal-hydraulic/neutronic capability. In addition to calculating the behavior of a reactor coolant system during a transient, it can be used for simulation of a wide variety of hydraulic and thermal transients in both nuclear and nonnuclear systems involving mixtures of vapor, liquid, noncondensable gases, and nonvolatile solute. The 3D capability in RELAP5-3D includes 3D hydrodynamics and 3D neutron kinetics (the 3D neutronics is based on the NESTLE[2] code). RELAP5-3D has recently been modified to include all the ATHENA[3] features and models that were previously only available in the ATHENA configuration. The ATHENA features and models are currently used primarily in Generation IV reactor applications, space reactor applications, and nuclear fusion applications.

RELAP5-3D is also used in a SCDAP/RELAP5-3D[4] configuration that is designed to calculate for severe accident situations the overall reactor coolant system thermal-hydraulic response, core damage progression, and reactor vessel heatup and damage. RELAP5-3D (also true of SCDAP/RELAP5-3D) is also used in an integrated code system configuration consisting of RELAP5-3D and other codes such as FLUENT, CFX, and CONTAIN. The coupling of the codes in this configuration is coordinated using an executive program[5] in concert with the Parallel Virtual Machine (PVM) message passing software. The coupling can be done explicitly or semi-implicitly. For example, the FLUENT/RELAP5-3D[6] coupling configuration is designed to perform detailed 3D analyses using FLUENT's capability while the boundary conditions required by the FLUENT calculation are provided by the balance-of-system model created using RELAP5-3D. The FLUENT/RELAP5-3D coupling configuration is currently used primarily in Generation IV reactor applications. RELAP5-3D is also used for the thermal-hydraulic module in the real-time nuclear plant simulation code RELAP5-R/T[7,8,9], which is used in training simulators at nuclear power plants around the world. There is also a visualization system for the various RELAP5-3D configurations, which is called the RELAP5-3D Graphical User Interface (RGUI)[10].

This paper discusses Version 2.3 of the RELAP5-3D computer program. The version includes the optional decoupling of heat structures from hydrodynamic components; built-in heat structure material properties consistent with MATPRO and the Nuclear Systems Materials Handbook; and Schrock's flow quality correlation for a downward oriented junction.

## II. HEAT STRUCTURE DECOUPLING

An input option has been added to RELAP5-3D. It allows the user to decouple a heat structure from the hydrodynamic components. Decoupling a heat structure means that the heat structure responds to the hydraulic conditions in the hydrodynamic volumes to which it is connected, but the energy removed from (or added to) the surface of the heat structure by convection/boiling/condensation is not added to (or removed from) the hydrodynamic volumes. This option is allowed for all three heat structure geometry types: rectangular, cylindrical, and spherical. The hydrodynamic quantities affected are the heat transfer from the wall to the liquid and vapor/gas as well as mass transfer at the vapor/liquid interface in the boundary layer near the wall for vapor generation (boiling) and liquid condensation.

This decoupling option is useful for the Code Scaling, Applicability, and Uncertainty (CSAU) methodology[11] and for debugging.

## III. BUILT-IN MATERIAL PROPERTIES

Thermal properties of five materials are stored internally (built-in) in RELAP5-3D: fuel rod gap gas, carbon steel, stainless steel, uranium dioxide, and zircaloy. Previously, the built-in material properties were present only to demonstrate capability, and they were not documented in the RELAP5-3D manual. The built-in data in the RELAP5-3D code are now from referenceable sources. The sources are MATPRO and the Nuclear Systems Materials Handbook[12]. MATPRO is documented in Volume IV of the SCDAP/RELAP5-3D code manual[4]. The user is also allowed to input one's own data.

Constant room-temperature densities are multiplied by temperature-dependent specific heat capacities to generate volumetric capacities used by the code. Arbitrary low and high temperature values of 5 K and 5,000 K are included to prevent code failures with out-of-range material property data. Constant room-temperature densities are used to develop the volumetric heat capacity input because the RELAP5-3D heat conduction solution does not include the effects of geometric changes to the structures. Specifically, the cladding deformation model will calculate the change in the radii of the fuel rod cladding. This change is accounted for in the adjacent fluid volume (its volume and flow area change), but this information is not accounted for in the conduction calculation; the geometry is the same as was input. Since the geometry does not change, using a constant density will maintain a constant structural mass, which is correct. It is generally appropriate, therefore, to use a room temperature reference density in calculating the volumetric heat capacity, since the dimensions of the structure being modeled are normally known (and taken) at room temperature.

For fuel rod gap gas, representative gap gas properties were developed for a combination of fill and fission product gases. A 0.1066/0.1340/0.7594 mole fraction He/Kr/Xe mixture is modeled. The gas mixture thermal conductivity is taken from standard textbook formulas and is documented in both the RELAP5-3D manual and the MATPRO manual. A perfect gas relation is used to determine the specific heat capacity. A representative fuel rod internal pressure of 4.1 MPa at room temperature of 300 K is assumed to determine the gap gas density using the perfect gas relation.

For carbon steel, the Nuclear Systems Materials Handbook[12] was used since it contains data for three medium carbon steels. Data for C-Mn steel (>1.0% Mn, <0.10% Si) was used as its thermal conductivity was in between the values for the two other alloys presented in the handbook. The equations used for the thermal conductivity, specific heat capacity, and density are documented in the RELAP5-3D manual.

For stainless steel, type 304 stainless steel is used as a representative stainless steel. The thermal conductivity and specific heat capacity equations are documented in the MATPRO and RELAP5-3D manuals. The density used is also documented in the RELAP5-3D manual.

For uranium dioxide, the thermal conductivity is taken from MATPRO. Both the equation for the thermal conductivity of solid uranium dioxide fuel and liquid uranium dioxide are documented in the MATPRO and RELAP5-3D manuals. The transition from solid to liquid uranium dioxide is modeled to occur between 3,113.15 K and 3,114 K. Both the equation for the specific heat capacity of solid uranium dioxide fuel and liquid uranium dioxide are documented in the MATPRO and RELAP5-3D manuals. The reference density used is also documented in the RELAP5-3D manual.

For zircaloy, the thermal conductivity is taken from MATPRO. Both the equation for the thermal conductivity of solid zircaloy and liquid zircaloy are documented in the MATPRO and RELAP5-3D manuals. The transition from solid zircaloy to liquid zircaloy is modeled to occur at 2,098 K. The specific heat capacity for zircaloy is calculated using a table lookup from MATPRO and is documented in the MATPRO manual. The reference density used is documented in the RELAP5-3D manual.

## IV. SCHROCK'S FLOW QUALITY CORRELATION

Recently, a RELAP5-3D calculation of a VVER 440/213 plant calculation was carried out for a small break loss-off-coolant accident (SBLOCA). The break was assumed to be at the bottom of the cold leg pipe. The calculation was run with the stratification entrainment/pullthrough input option invoked for a downward-oriented junction. The calculation showed that the break junction void fraction was at times greater than that the cold leg void fraction. This is an incorrect result for downward vapor/gas pullthrough; the break junction void fraction should be less than or equal to the cold leg void fraction. Analysis of the correlations being used showed that Schrock et al.'s correlation[13] was used for the inception height for a downward-oriented junction, and that Smoglie et al.'s correlation[14] was used for the offtake flow quality for a downward-oriented junction. This is the methodology used by Ardron and Bryce[15,16], and this methodology was the basis for the stratification entrainment/pullthrough model put into RELAP5. When it was put in the code, Schrock[17] questioned the methodology because it mixed correlations from different research groups.

The RELAP5-3D code was changed to use Schrock et al.'s correlation[13] for the offtake flow quality for a downward-oriented junction. With this change, the VVER plant calculation break junction void fraction is now always less than or equal to the cold leg void fraction (correct result).

## V. CONCLUSIONS

This paper has discussed recent improvements to the RELAP5-3D code. The improvements are in Version 2.3. The improvements are an optional decoupling of heat structures from hydrodynamic components; built-in heat structure material properties consistent with MATPRO and the Nuclear Systems Materials Handbook; and Schrock's flow quality correlation for a downward oriented junction.

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## REFERENCES

1. THE RELAP5-3D CODE DEVELOPMENT TEAM, "RELAP5-3D Code Manual," Idaho National Laboratory, INEEL-EXT-98-00834, Revision 2.3, April (2005). http://www.inel.gov/relap5/r5manuals.htm.

2. P. J. TURINSKY et al., "NESTLE: A Few-Group Neutron Diffusion Equation Solver Utilizing the Nodal Expansion Method for Eigenvalue, Adjoint, Fixed-Source Steady State and Transient Problems," Idaho National Engineering Laboratory, EGG-NRE-11406, June (1994).

3. G. W. JOHNSEN, C. B. Davis, and P. D. Bayless, "ATHENA-3D for Generation IV Reactor Analysis," BE-2004, American Nuclear Society Winter Meeting, Washington, DC, USA, November 14-18 (2004). http://www.inel.gov/relap5/athena/athena.htm.

4. THE SCDAP/RELAP5-3D CODE DEVELOPMENT TEAM, "SCDAP/RELAP5-3D Code Manual," Idaho National Engineering and Environmental Laboratory, INEEL-EXT- 02-00589, Revision 2.2, October (2003). http://www.inel.gov/relap5/scdap/scdap.htm.

5. W. L. WEAVER, E. T. Tomlinson, and D. L. Aumiller, "A PVM Executive Program for Use with RELAP5-3D," 2001 RELAP5 International Users Seminar, Sun Valley, ID, USA, September 5-7 (2001). http://www.inel.gov/relap5/rius/sunvalley/weaver.pdf.

6. R. R. SCHULTZ and W. L. Weaver, "Using the RELAP5-3D Advanced Systems Analysis Code with Commercial and Advanced CFD Software," ICONE-11, 11th International Conference on Nuclear Engineering, Tokyo, Japan, April 20-23 (2003).

7. K. A. WILLIAMS, "Full-Scope Simulators Running Real-time RELAP5-R/T," 2001 RELAP5 International Users Seminar, Sun Valley, ID, USA, September 5-7 (2001). http://www.inel.gov/relap5/rius/sunvalley/williams.pdf

8. D. W. HILTBRAND and K. A. Williams, "Utilization of a RELAP5 RCS and Secondary Plant Model in a Nuclear Power Plant Training Simulator," 2001 RELAP5 International Users Seminar, Sun Valley, ID, USA, September 5-7 (2001). http://www.inel.gov/relap5/rius/sunvalley/hiltbrand-williams.pdf

9. J. L. JUDD et al., "High-Fidelity, Real-Time Simulation with RELAP5/NESTLE," American Nuclear Society Winter Meeting, Washington, DC, USA, November 10-14 (1996).

10. G. L. MESINA, Visualization of RELAP5-3D Best Estimate Code, BE-2004, ANS Winter Meeting, Washington, D.C., November 14-18 (2004). http://www.inel.gov/relap5/products.htm.

11. TECHNICAL PROGRAM GROUP, "Quantifying Reactor Safety Margins: Application of CSAU to a LBLOCA," *Nuclear Engineering and Design,* **119** (1990).

12. ORNL, "Nuclear Systems Materials Handbook – Volume 1, Design Data," TID-26666, Revision 37, Oak Ridge National Laboratory, September (1998).

13. V. E. SCHROCK, S. T. Revankar, R. Mannheimer, and C. H. Wang, "Small Break Critical Discharge – The Role of Vapor and Liquid Entrainment in a Stratified Two-Phase Region Upstream of the Break, NUREG/CR-4761, LBL-22024, Lawrence Berkeley Laboratory, December (1986).

14. C. SMOGLIE, "Two-Phase Flow Through Small Branches in a Horizontal Pipe with Stratified Flow," KfK 3861, Kernforschungszentrum Karlsruhe GMBH (KfK), Karlsruhe, FRG, December (1984).

15. K. ARDRON and W. Bryce, "Assessment of Horizontal Entrainment Model in RELAP5/MOD2," AEEW-R 2345, Atomic Energy Establishment Winfrith, April (1988).

16. W. BRYCE, "Numerics and Implementation of the UK Horizontal Stratification Entrainment Off-Take Model into RELAP5/MOD3," AEA-TRS-1050, AEEW-R 2501, Atomic Energy Establishment Winfrith, March (1991).

17. V. E. SCHROCK, NRC Advisory Committee on Reactor Safeguards Thermal-Hydraulic Phenomena Subcommittee Meeting, Idaho Falls, Idaho, USA, August 28-29 (1989).