RE-ASSESSMENT OF RELAP/SCDAPSIM/MOD3.X USING HISTORICAL INTEGRAL EXPERIMENTS

CHRIS M. ALLISON, JUDITH K. HOHORST

Innovative Systems Software

iss@srv.net jkh@srv.net

The RELAP/SCDAPSIM code, designed to predict the behavior of reactor systems during normal and accident conditions, is being developed by Innovative Systems Software (ISS) as part of the international SCDAP Development and Training Program (STDP). The ISS developed RELAP/SCDAPSIM uses the publicly available SCDAP/RELAP5 models and correlations developed by the US Nuclear Regulatory Commission in combination with proprietary features developed by ISS and STDP members including (a) advanced programming and numerical methods, (b) advanced user options such as integrated graphical user environments, coupled 3D reactor kinetics options, and uncertainty analysis methods, and (c) advanced models and correlations. MOD 3.4 is the current production version. Experimental versions developed to support specific RELAP/SCDAPSIM users' design and analysis activities include MOD3.5 and MOD3.6. MOD 3.5 contains improved models for LWRs including improved SCDAP models and correlations for fuel rods, B_4C control rods, electrically heated fuel simulator rods such as those used in the CORA, QUENCH, and PARAMETER experimental facilities in Germany and Russia. It also includes new modeling options for generalized structures as well as the influence of air ingression. MOD3.6 includes the MOD3.5 improvements plus PHWR specific models and correlations developed for CANDU and Atucha-2 channel type reactor designs. These new models are currently being actively assessed by selected users, ISS, and university students and faculty members participating in the SDTP University Support program. New versions of MOD3.5 and MOD3.6 will be released as needed based on the results of the on-going assessment activities.

This paper will summarize and describe the new models that were incorporated into the experimental versions of RELAP/SCDAPSIM along with the extensive assessment and verification activities that are currently underway at ISS and various universities and institutes around the world along with examples of thee assessment results. These assessment activities include the verification and validation of the new models by comparing the RELAP/SCDPSIM/MOD3/x calculated results to the measured experimental data as well as to the predicted results from earlier versions of SCDAP and SCDAP/RELAP5 (USNRC codes released in the 1980s and 1990s). The experiments used for this assessment and discussed in this paper include integral thermal hydraulic and severe accident experiments conducted in the LOFT and PBF facilities at the Idaho National Laboratory (formerly INEL), the French PHEBUS facility at Cadarache, and the CORA and QUENCH facilities at KIT in Karlsruhe, Germany (formerly kfk, and fzk).

1. INTRODUCTION

The RELAP/SCDAPSIM code being developed at Innovative Systems Software (ISS) as part of the international SCDAP development and training program [1.2], is designed to predict the behavior of reactor systems during normal and accident conditions. RELAP/SCDAPSIM uses the publicly available RELAP/MOD3.3 [3] and SCDAP/RELAP5/MOD3.2 and MOD3.3 [4,5] models developed by the US Nuclear Regulatory Commission in combination with proprietary (a) advanced programming and numerical methods, (b) user options, (c) models developed by ISS and other STDP members. [6] During the development activities, new models, new capabilities, and error corrections have been implemented in the code. Section 2 of this paper will give a short description of the RELAP/SCDAPSIM code, Section 2 will give a description of the new models and improvements to the code, Section3 will describe the assessment activities and show representative results from the ongoing assessment activities and Section 4 will give the conclusions.

2. BRIEF DESCRIPTION OF RELAP/SCDAPSIM/MOD3.X

RELAP/SCDAPSIM is designed to describe the overall reactor coolant system (RCS) thermal hydraulic response and the behavior of the core under normal operating, design basis, and severe accidents conditions. The RELAP5 models calculate the overall thermal hydraulic response of the RCS, control system behavior, reactor kinetics and the behavior of special reactor system components such as valves and pumps. The SCDAP models calculate the behavior of the core and vessel structures under normal and accident conditions. The SCDAP portion of the code includes user-selectable reactor component models for LWR fuel rods, Ag-In-Cd and B_4C control rods, BWR control blade. Channel boxes, electrically heated fuel rod simulators, and general core and vessel structures. The SCDAP portion of the code also includes models to treat the later stages of a severe accident including debris and molten pool formation, debris/vessel interactions and structural failure (creep rupture) of the vessel structures. The later models are automatically invoked by the code as the damage in the core and vessel progresses during an accident.

MOD3.4 is the current production version of the code. Two additional experimental versions are currently being assessed. MOD3.5 is being used to analyze LWR's, experimental facilities such as QUENCH and CORA, BWR's for Fukushima type accidents and air ingress accidents. MOD3.6 is being used specifically for CANDU and ATUCHA 2 type reactors.

3. THE NEW MODELS IMPLEMENTED

To better analyze experimental programs and Fukushima like behavior, improvements to models currently in the code and new models, and extensions to the output were implemented. [7,8,9] The changes to the code included:

1. Improved heated simulator fuel rod model
2. The correlations used to model the B_4C control rod were improved
3. The correlations for gap conductance and thermal conductivity for the fuel rods were improved
4. Major improvements to the shroud model were implemented
    a. The ability to model diverse designs of a shroud
    b. Radiation heat transfer across the gap in the shroud
    c. The ability to better simulate thermal-mechanical failure of the shroud at high temperature and during quenching
    d. Double sided oxidation of the shroud
5. Oxidation models improved to include Zircaloy oxidation in air and the uptake of Nitrogen
6. Improved Zr-Nb correlations (CANDU and VVER)
7. Channel to Channel radiation heat transfer
8. The COUPLE models were improved to include
    a. Improve the accuracy for core plate options
    b. Core-concrete models and correlations
9. Bounding options for liquefied {U-Zr}-O_2 relocation and freezing
10. The addition of more plot variables
11. Error corrections

The above listed changes to the code will improve a) the fuel and cladding predicted temperature for steady state and design basis transients where stored energy is important; b) the Zircaloy embrittlement and oxidation rates in the presence of air for spent fuel storage and mid-loop operation in LWRs: c) the thermal and oxidation behavior of the Zr-Nb cladding used in CANDU and VVER reactors; d) the behavior of channel type reactors such as ATUCHA2 and CANDUs during normal and severe accident conditions; e) bundle behavior including the heat-up, oxidation, and melting in experiments using electrically heated fuel rod simulators (CORA, QUENCH, and PARAMETER); f) the behavior of B_4C control rods used some experimental programs and reactors where the control rods are B_4C rather than a B_4C control blade in a channel box; g) the ability to model core plate reactions and MCCI.

3.1 IMPROVEMENTS TO THE FUEL ROD AND CORE MODELS

The extensions to the fuel rod and general core models to better represent the out-of-pile fuel rod experiments and fuel rod behavior during an accident included improvements to the simulator model, fuel rod, and shroud. The fuel rod simulator model was extended to include tantalum, molybdenum, and brass as well as tungsten to better represent experimental programs such as PARAMETER [7] and QUENCH. The QUENCH experiments performed by Karlsruhe Institute of Technology (KIT) in Germany used tungsten heaters in their simulator rods where the Russians used tantalum. The Russian experiments also used different end fittings from the molybdenum used in QUENCH. These fitting consisted of brass and a special alloy of molybdenum. To accommodate differences in the designs of simulator rods the code was extended to allow the user to define the material composition and radial position of each material as a function of elevation. The materials property data was also extended to define as a function of temperature the required properties of the new materials (electrical resistivity, thermal conductivity, heat capacity, density, thermal expansion and emissivity).

In addition to the generalization of the simulator model, a model was added to calculate the conductance of heat across the gap in the fuel rod and simulator model. The original model in the code assumed the fuel or simulator material to be in soft contact with the cladding, independent of cladding ballooning. The new gap conductance model which is applied at each axial node of the modeled fuel rod or simulator, uses the calculated extent of ballooning to calculate the effect of ballooning on the fuel-cladding gap. This model considers the size of the gap and the composition and pressure of the gas in the gap. The most important applications of the model will be during periods of rapid quench, rapid oxidation, and when the fuel rod cladding balloons.

For analysis of experimental programs using a shroud on the active bundle, improvements to the shroud model were incorporated into the shroud model. Since the performance of the flow shrouds surrounding the bundle is a function of the radiation heat transfer across any gap present in the shroud. The rate of heat transfer across a gap increases as the temperature increases. A new model was developed that is applied at each axial node of the shroud for any sector of the radial meshes modeling the shroud that contains a gas. The gas sector in the shroud is identified by comparing the user defined density for each radial mesh with a density threshold value of 10 kg/m^3. If the defined density in at radial mesh point of the shroud is less than 10 kg/m^3, that mesh point is consider to be part of a gap region in the shroud. The thermal conductivity of the gas in that region is increased to a value above of that of the gas in the gap to account for the heat transfer across to the gap due to radiation heat transfer.

3.2 ZIRCALOY OXIDATION IMPROVEMENTS

To better model the ingress of air into a reactor core or a spent fuel storage pool the Zircaloy oxidation models in the presence of air were improved. Air when it comes I n contact with Zircaloy can lead to accelerated fuel degradation and enhanced release of fission products especially ruthenium [ ]. Due to a higher heat of reaction, the rate of oxidation Zircaloy in air is accelerated from that in steam due to accelerated kinetics caused by oxide breakaway. Improvements have been made to better represent the initial parabolic kinetics, the transition to linear (breakaway) kinetics as the protective effect of the initially formed oxide is progressively lost, and the post breakaway kinetics. These improvements include the rate of formation of the zirconium nitride and the consideration of the stresses applied to the protective oxide layer which are a function of temperature, composition of the gas surrounding the cladding, and the type of cladding.

3.3 IMPROVEMENTS TO MODEL CHANNEL TYPE REACTORS

Channel type reactions such as ATUCHA2 and the Canadian CANDU required models that consider channel to channel radiation heat transfer.

3.4 IMPROVEMENYS TO THE COUPLE MODEL

Modifications to the COUPLE module in RELAP/SCDAPSIM included the introduction of a convective heat transfer area calculation for planar geometry. Also the model to calculate natural circulation in the molten pool for planar geometry was activated. The equivalent radius definition from spherical geometry was changed to an equivalent radius which corresponds to a half section of a cylinder with a length 1.

4.O Assessment Activities

To assess the new models incorporated into RELPASCDAPSIM an extensive program has been initiated using historical severe accident experiments. The desired calculations are being performed by interns at the ISS office, students and faculty at various universities around the world, licensed users and the ISS staff. The predicted behavior from the new models is being compared to the experimental data and the predicted results from older versions of the code and standalone SCDAP. The historical experiments being revisited are the 1) PBF-SFD (Power Burst Facility Severe Fuel Damage) experiments and LOFT conducted at the Idaho National Laboratory; 2) CORA and QUENCH conducted at Karlsruhe Institute of Technology formerly Kernforzentrun Karlsruhe and Forzentrum Karlsruhe; 3) PHEBUS conducted at Cadarache in France; 4) TMI2. Examples of some of the assessment results are given in the following sections.

4.1 SFD-ST SCOPING TEST

SFD-ST [9,10]1was the first in a series of severe fuel damage test conducted in the Power Burst Facility at the INEL (Idaho National Engineering Laboratory, now Idaho National Laboratory). The objective of Scoping Test was 1) to gain experience in performing an in-pile test; 2) to understand fuel bundle dynamics at high pressure, 6.9 MPa, with slow bundle heating; 3) to understand hydrogen production and fission product behavior during a high temperature transient; and 4) to observed the coolability of the damaged fuel under reflood. The experimental bundle was comprised of 32 0.91 m in length fresh fuel rods in a 6 x 6 array with the corner rods removed surrounded by a Zircaloy-4 lined insulating shroud to simulate the cladding of another row of fuel rods. The test bundle was preconditioned through several power cycles in the PBF driver core develop an intercalibration between the driver core and the experimental bundle power. During the transient part of the experiment the bundle power was slowly increased to 93 KW with the coolant flow maintained at 0.02 L/s. When the desired temperature was attained the reach was scrammed and the test bundle quenched.To assess the new models against this experiment comparisons were made to the published results from Version 18 of standalone SCDAP [ ]. Examples of these comparison are shown in Figures 1 and 2.

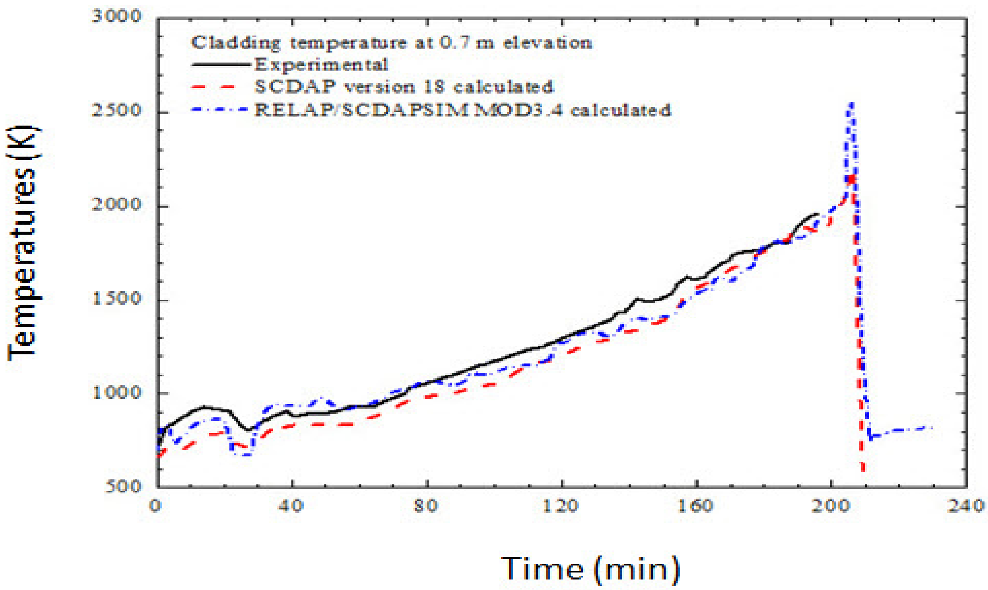

Figure 1. Comparison of Cladding Temperature at the 0.7 mm Elevation

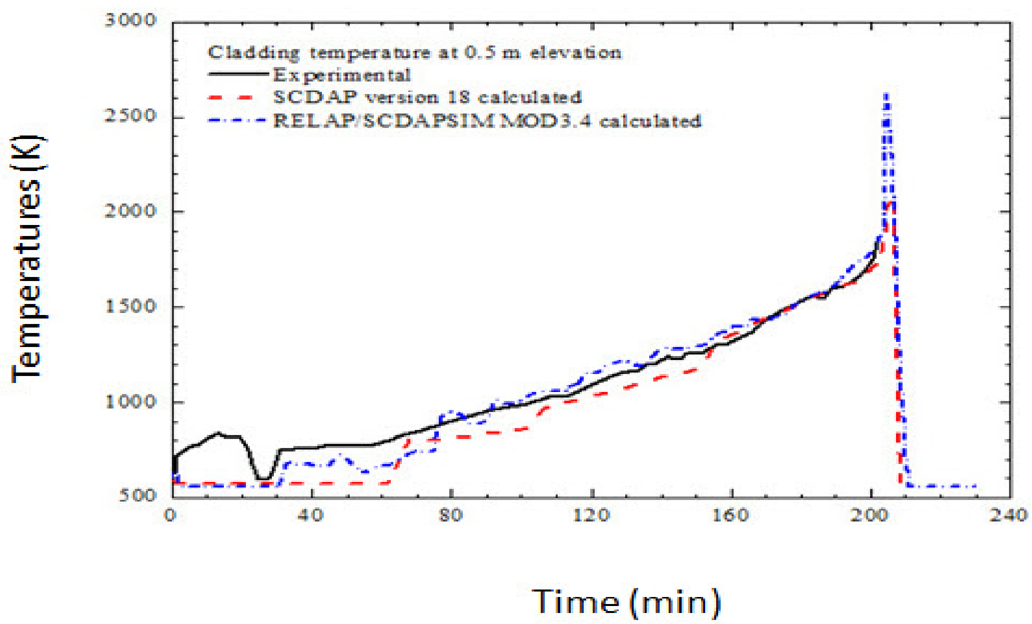

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Figure 2. Comparison of the Cladding Temperatures at the 0.5 mm Elevation

As shown in Figures 1 and 2 could match the experimental measurement and the predicted results from SCDAP v18 as well as the rise in temperature attributed to quenching of the test bundle.

4.2 CORA EXPERIMENTS

The CORA experiments performed at Kernforzentrum Karlsruhe, KfK, during the 1980's and early 90's are being used to assess the new models in RELAP/SCDAPSIM as they cover a wide range of thermal hydraulic conditions. These electrically heated experiments that used Zircaloy clad fresh fuel rods, studied 1) variations in the initial heating rates; 2) variations in the pre-oxidation period of the bundle; 3) effect of oxide thickness; 4) different bundle sizes; 5) PWR and BWR configurations; and 6) different methods of cooling (slow cooldown and reflood).

CORA-17 was selected to assess the improvements to the heater rod model, the insulating shroud and boron carbide properties. Though other experiments have been performed with a boron carbide control rod, the CORA BWR experiments simulated the control blade and channel box in a BWR. This feature is important in assessing the code for Fukushima Daiichi like accidents. The CORA-17 test bundle consisted of 12 Zircaloy-4 clad heater rods containing UO_2 pellets and a central 6mm in diameter tungsten heating element. 6 unheated fresh fuel rods, two Zircaloy channel box walls and an absorber blade representing a portion of a BWR absorber blade. Figure 3 shows the CORA 17 test bundle.

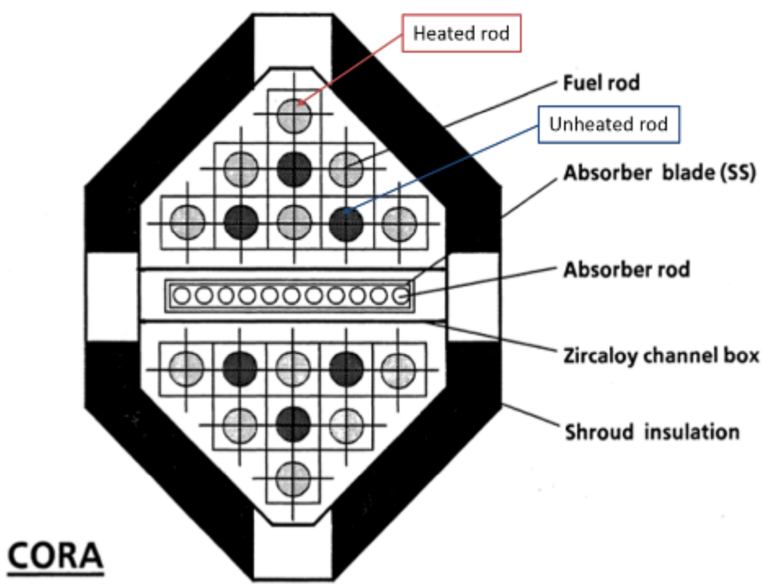

Figure 3. CORA-17 test Bundle

To assess the new models comparison were made using two versions of RELAP/SCDAPSIM, the production version MOD3.4 and an experimental version MOD3.5 which contained the new models. Two input decks were also used for the assessment, one the original deck developed by Oak Ridge National Laboratory (ORNL) for the ORNL developed control blade/channel box model and the other a deck modified for the new models. Figures 4 and 5 show some representative results from these calculations.

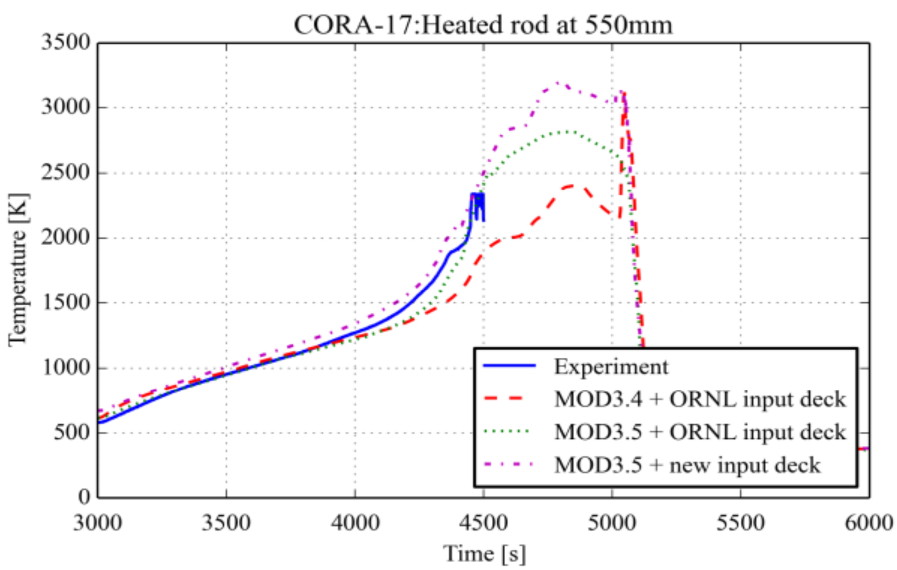

Figure 4. Temperature History of the heated rod at the 550 mm elevation

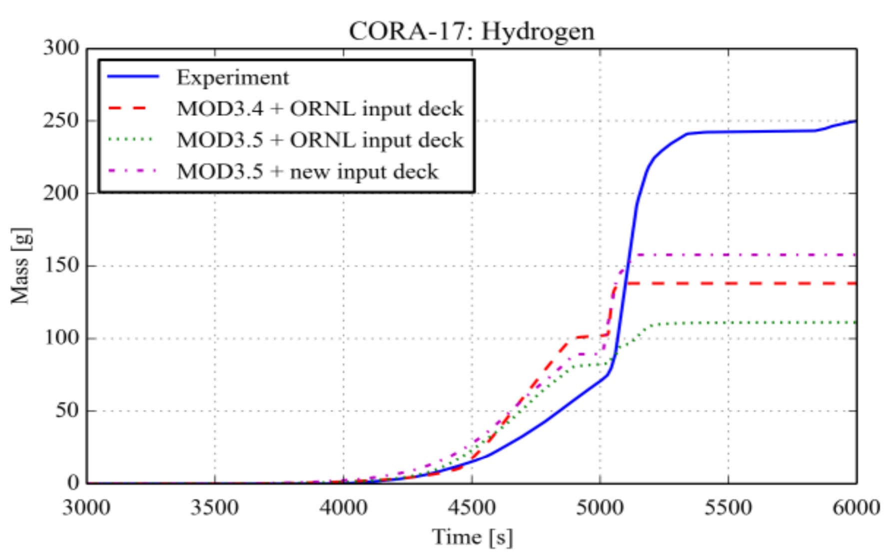

Figure 5. Hydrogen production comparison

From the above two figures, the new models which better represent the CORA facility more accurately predict the temperature history during the experiment and hydrogen production. With the new models hydrogen production increased, but was still lower than the original measured value. A re-evaluation of the hydrogen production was performed and the value calculated by RELAP/SCDAPSIM is to the new value. [Yuri Struckert]

5.0 Additional Assessment Activities

Assessment activities are being performed by various institutions around the world and results of these calculations are being presented. Currently, various QUENCH experiments are being analyzed using the new models. These experiments include QUENCH-6, an international standard problem, and QUENCH 14 with M5 cladding. QUENCH-6 is routinely used as a training exercise for new users prior to analyzing a different QUENCH experiment. Du ring the assessment process most of the QUENCH experiments will be analyzed using the new models and the predicted results compared to the measured data.

Ongoing assessment activities include CORA, PHEBUS, LOFT, TMI-2 and other PBF-SFD experiments, SFD 1-1 and SFD 1-3.

6.0 Conclusion

The new models which allow the user to accurately model the experimental facility. The ability to model tantalum and tungsten heater rods has made the code more versatile, since the Russian electrically heated parameter experiments use tantalum heating elements. The addition of a gap conductance model for the fuel rod has improved temperature predictions. The ability to model different material at each axial location in the shroud as well as gap conductance in the shroud and double-sided oxidation of the shroud material has improved the predicted temperatures of the shroud at each axial elevation. The improvements to the control rod mod to include B_4C has allow the user to analyze with accurate results experiments using a central B_4C control rod as well as reactors using B_4C control rods.

From the calculations performed to date, the new models accurately predict the thermal hydraulic behavior of the experimental test bundles and the predicted melt relocation better matches the Post Irradiation Examination (PIE) results.

7.0 References

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