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OVERVIEW OF COMPUTATIONAL CHALLENGES IN THE DEVELOPMENT OF EVALUATION MODELS FOR SAFETY ANALYSES OF THE IRIS REACTOR

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ABSTRACT

The pressurized light water cooled, medium power (1000 MWt) IRIS (International Reactor Innovative and Secure) reactor plant has been under development for three years by an international consortium of over 20 organizations from nine countries. The plant conceptual design was completed in 2001 and the preliminary design is currently underway. The pre-application licensing process with NRC started in October 2002 and IRIS is one of the designs considered by US utilities as part of the ESP (Early Site Permit) process. Major characteristics of the IRIS design and supporting analyses have been previously reported.

This paper focuses on the computational challenges that need to be addressed in the development of appropriate Evaluation Models for IRIS safety analyses.

In the analyses of the IRIS non-LOCA transients and accidents, most of the computational and computer modeling challenges are related to the adoption of an integral reactor coolant system with novel components (steam generators, pumps and pressurizer) for which appropriate constitutive models need to be developed. These challenges in the description and analysis of IRIS integral reactor coolant systems in normal and abnormal conditions are introduced here and discussed in more detail in companion papers.

The IRIS novel approach to safety, discussed in a companion paper, introduces a strong coupling between the integral reactor coolant system and the containment following small break LOCAs. An appropriate evaluation model for the analyses of these events will therefore require the development of an evaluation tool that can analyze both the reactor systems and the containment and its related systems. The coupling of a system thermal-hydraulic code (RELAP5) and a containment code (GOTHIC) has been considered as the most promising approach. The code coupling is briefly presented in this paper.

Key Words: IRIS, Evaluation Models, RELAP, GOTHIC, CFD

1. INTRODUCTION

Section 15 of the NRC Standard Review Plan (NUREG0800) [1] describes transients and accidents that the NRC reviews as part of a license application, and the acceptance criteria from 10CFR50 Appendix A that apply to each transient and accident. It also indicates that "acceptable Evaluation Models should be used to analyze these transients and accidents."

The concept of "Evaluation Models" is further detailed in other NRC documents, and in particular DG-1120 "Transient and Accident Analysis Methods" [2]. In DG-1120, an evaluation model is defined as the calculation framework for evaluating the behavior of the reactor system during a postulated transient or design basis accident. It may include (a) one or more computer programs (including general purpose computer codes); (b) one or more special models; and (c) all the information needed for application of the calculation framework to a specific event (procedures for treating input and output and in particular the code input arising from plant geometry, plant assumed state, and transient initiation).

DG-1120 also describes the Evaluation Models Development and Assessment Procedure (EMDAP) that can be used to develop evaluation models for a new application on the basis of a 5 step approach: (1) determine requirements for the evaluation model (phenomena identification and ranking tables); (2) develop an assessment base consistent with the determined requirement (test data); (3) Develop the Evaluation Model; (4) assess the adequacy of the Evaluation Model against the assessment base; (4) follow an appropriate quality assurance protocol during EMDAP; (5) provide comprehensive, accurate, up-to-date documentation. The principles of the EMDAP were developed and applied in the Code Scalability, Assessment and Uncertainty (CSAU) program [3].

IRIS is an innovative design, but is firmly grounded on the extensive technical and analytical experience matured in the past 50 years of design and operation of pressurized water reactors, and as such does not require a complete application of the EMDAP procedure. As described in DG1120, a limited application of the EMDAP can be performed to integrate existing models in those areas where novel characteristics are identified. The specific areas where the application of the EMDAP, or at least of part of it, will be required to develop a complete set of Evaluation Models for IRIS are currently being assessed.

This paper will focus on some of the computational challenges and proposed solutions in the development of Evaluation Models for IRIS, especially in the thermal-hydraulic area.

2. COMPUTATIONAL ISSUES IN THE ANALYSIS OF NON-LOCA TRANSIENTS AND ACCIDENTS

The final selection of codes for transient and accident analyses can be completed only at the end of the EMDAP procedure, but a preliminary selection is necessary to assist and support the development of the EMDAP, in particular to assist in the identification and ranking of important phenomena.

From a phenomenological point of view, IRIS response to several non-LOCA transients and design basis accidents is not significantly different from other PWRs, so that current Evaluation Models developed by Westinghouse for the analyses of this class of events have been used as a reference in developing preliminary Evaluation Models for IRIS and assessing the areas where challenges might surface. Also, several of the plant systems are based on Westinghouse experience with passive plants and do not pose any new challenge.

For these reasons, a general purpose code developed for the analyses of light water reactors can be used, at least in principle, for analyses of the IRIS reactor. For IRIS system analyses, the RELAP5 Mod3.3 [4] code has thus been selected. This code has been favored over Westinghouse in-house codes due to the international nature of the IRIS project and due to the vast experience that several members of the IRIS consortium have in the development and use of the RELAP code.

Although the RELAP code has been extensively used in the analyses of light water reactors, and has also been used in the transient analyses of the advanced Westinghouse passive plants [5], the introduction of a new reactor and supporting systems poses great challenges to the development of an appropriate plant nodalization. In particular, the IRIS integral reactor coolant system layout is sufficiently different from the typical loop PWR to require a completely new approach to develop the coolant system model. Based on the best experience acquired by the University of Zagreb in the use of the RELAP code for safety analyses and on Westinghouse experience in PWR analysis, a state-of-the-art plant nodalization has been developed for the plant. This RELAP5 plant model and steady state qualification is discussed in a companion paper [6] in this conference, and an overview of the approach used in developing the model is shown in Figure 1. This activity has required an extensive effort and has been completed during the second quarter of 2002.

This nodalization is being used with a modified version of the RELAP5 code that is being developed by the University of Zagreb to perform an initial safety assessment of all the main events indicated by the NRC in Section 15 of the SRP. This assessment will create a sufficient knowledge of the plant response to transients and accidents to assist in the completion of the first step of the EMDAP procedure, identifying and ranking the important physical phenomena that the Evaluation Model has to address. The important phenomena will then be addressed to identify necessary testing to complete a sufficient assessment base for the code selection and nodalization validation.

During the development of the plant model and the initial safety assessment of various accident sequences, several areas where limits in the proposed methodology could have an impact on the analyses have been identified. Most of these limits are due to the lack of appropriate constitutive models to represent IRIS integral components, since there is a lack of appropriate experimental data. Also, some computational issues in the used methodology have been identified, especially in the analyses of small break LOCAs, as discussed in the following sections.

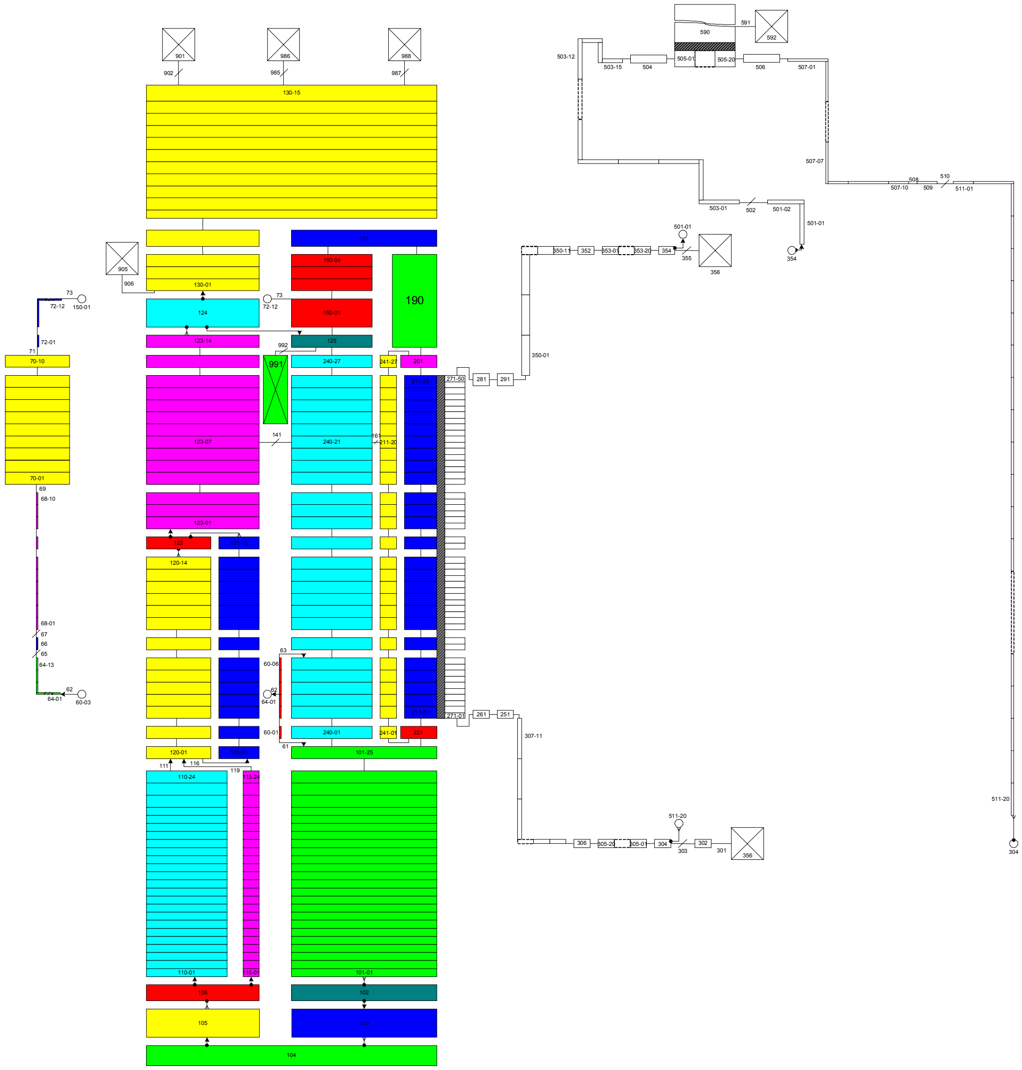

Figure 1. Overview of IRIS nodalization for RELAP5 Mod3.3

2.1 Integral Reactor Coolant System

The RELAP5 hydrodynamic model is a one-dimensional, transient, two-fluid model for flow of a two-phase steam-water mixture. The code has been developed and used for the analysis of light water reactors (and also for CANDU analyses) with a loop design. The IRIS integral reactor coolant system presents a completely different system layout and, as discussed in the previous section, this has required the development of a completely new nodalization that will need extensive validations before its qualification can be considered complete.

It is interesting to point out is that an integral reactor such as IRIS actually presents, from a computational point of view, a structure where each of the reactor coolant pumps and steam generators are coupled and, in normal operation, constitute a separate flow path. The nodalization logic is therefore not as different from the one used in PWRs as it could be expected at first sight.

Aside from the development of an appropriate overall nodalization, the main challenges in the development of safety analyses for the integral reactor coolant system are due to the new integral components and to the analyses of mixing effects in the downcomer and pressurizer regions of the system.

2.1.1. Integral Components

The IRIS integral reactor coolant system features three components that present significant differences from other PWRs: the reactor coolant pumps, the steam generators and the pressurizer.

IRIS adopts integral reactor coolant pumps that are completely enclosed inside the pressure vessel [7]. While these pumps are significantly different form both shaft seal pumps (used in current plants) and canned motor pumps (featured in the AP600/AP1000 plants), from the point of view of modeling and analysis using the RELAP code, they don't present significant challenges. This is because pumps are defined in the code simply through their hydraulic performance (homologous curves) and coastdown characteristic. Therefore, to correctly model the IRIS spool pumps, it has been sufficient to properly define a sufficient database to represent the pump behavior.

More challenging is the modeling of the helical coil steam generators [8]. Preliminary studies on this component and the modeling approach utilized are discussed in reference [9] and are therefore not discussed here, but an extensive testing campaign will be required to properly characterize the SG thermal-hydraulic characteristics (mainly pressure losses and heat transfer models) over the whole range of conditions over which these components will be operating in normal and abnormal conditions. This is the only way to provide to the RELAP code an appropriate set of constitutive models validated for safety related analyses.

Finally, the integral pressurizer does not pose significant challenges from a safety analyses point of view, but will require a careful design to guarantee appropriate mixing and response to insurge and outsurge events. If the design confirms the preliminary choice of eliminating the spray system, appropriate mixing of the pressurizer water will have to be provided by other means to ensure uniformity in boron concentration with the flowing coolant and detailed CFD analyses will be required. Also, heat transfer and eventually insulation at the boundary between the coolant system and the pressurizer will be studied using CFD analyses. Since the focus of this paper is on computational challenges in the development of evaluation models for safety analyses, no discussion on the pressurizer behavior are included.

2.1.2. Mixing phenomena in the downcomer and lower plenum region

Mixing phenomena in the downcomer and lower plenum are important in the safety analyses of current PWRs for some asymmetrical events, such as the locked rotor or the steam system piping failures. These asymmetrical conditions are typically studied by defining conservative mixing coefficients that empirically account for mixing and segregation phenomena. Several efforts are devoted to use CFD codes to reduce the conservativness in the analyses of these asymmetrical events.

At first sight, the IRIS integral reactor coolant system layout would appear to increase the importance of 3D and mixing effects, while the proposed nodalization for RELAP safety analyses currently assumes perfect mixing in these regions. This apparent contradiction can be explained on the basis of the very low velocities and long residence times (more than 20 seconds) of the coolant in the downcomer and lower plenum. In fact, in loop-type PWRs mixing effects are typically important for full flow transients, while a more uniform condition tends to exist in low flow, natural circulation conditions. However, this simplification needs to be verified, and eventually appropriate models to represent the fluid mixing in the downcomer region need to be developed. A research program has already been defined. Figure 2 shows a half section of the downcomer and lower plenum that will be used for this analysis. The SG and core conditions calculated with RELAP will be used as boundary conditions for the CFD model, and the core inlet distribution of temperature and flow will be assessed for different operational and abnormal conditions. Temperature, flow and boron mixing phenomena will be evaluated, and the effect of different lower plenum and core support geometries will be assessed.

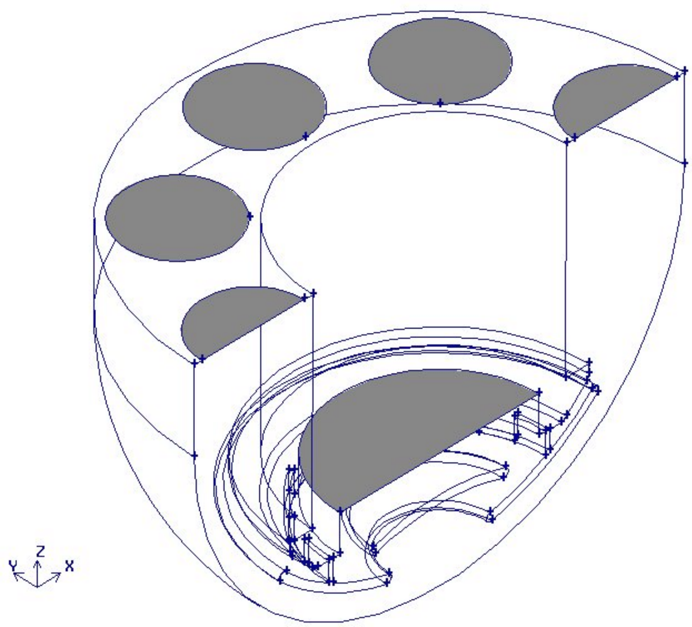

Figure 2. Downcomer 3-D modeling for the CFD code (in dark gray the SG modules outlet and the core inlet, used as input boundary and output boundary for the CFD code, respectively).

Figure 3 shows the mesh and velocity profiles for one of the lower plenum designs that have been considered. The FLUENT CFD code has been used for these analyses.

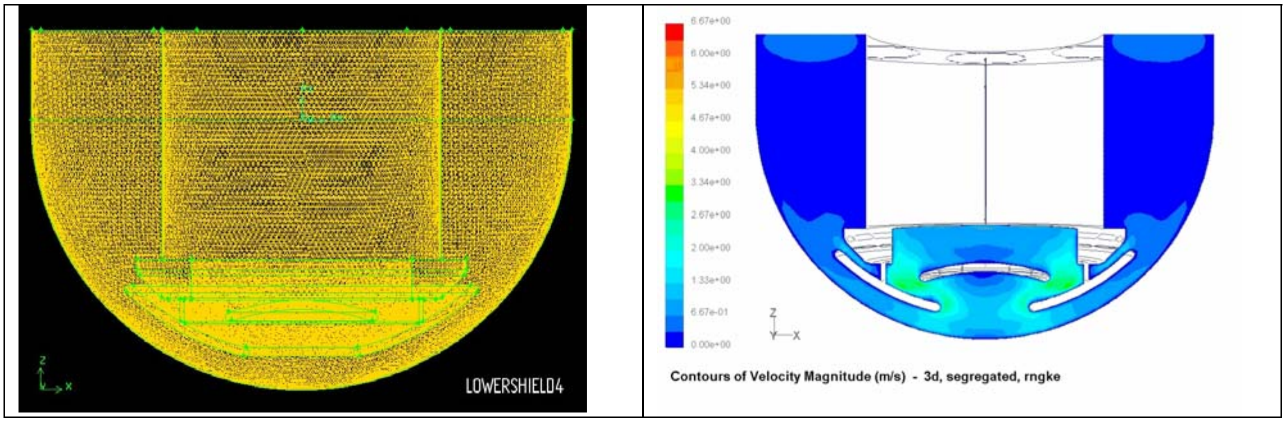

Figure 3. Downcomer 3-D mesh (hexahedral and tetrahedral, unstructured) for the CFD code and CFD results: velocity profiles in the downcomer with modified lower shielding.

Based on the results of this activity, the importance of mixing phenomena and the effect of the lower plenum design in different transients will be assessed. For those events where the perfect mixing assumption is not justified and not conservative, appropriate models will have to be implemented in the RELAP plant analysis model. One approach would be to simply follow procedures similar to that used in the analyses of current PWRs, defining an appropriate set of mixing coefficients (e.g. representing a perfect mixing, a null mixing, and the so-called design mixing), with an appropriate nodalization for each case in the system code. Another approach that will be considered is to use the CFD code as an intermediate step between the system code and the core analysis code. Typically, core inlet conditions are determined with a mono-dimensional analysis code (be it RELAP or other similar code) and then provided to a core subchannel analysis code, where they are used as boundary conditions in an appropriate model for the evaluation of the thermal margin (i.e., departure to boiling ratio evaluation).

The simpler way of using CFD in this analysis loop is as an intermediate step where the system parameters for the SG outlet conditions are provided to the CFD model, which in turn provide the core with 3D inlet boundary conditions. An alternative to this segregate approach would be an effective coupling of the CFD code with the system code, as is being currently proposed by several different researchers (see for example [10])

2.2 Core Thermal-Hydraulic analysis

The IRIS core is not different from a conventional PWR core, except for a slightly enhanced moderation ratio to improve fuel utilization. As such, NRC approved methodologies [11] and codes will be used to quantify the thermal margin in the core during normal operation and transients. The system code (RELAP) will be used only to provide core boundary conditions (neutronic power, inlet flow rate, inlet temperature, and inlet pressure) to a 3D core model developed for the VIPRE-01 code [12]. The VIPRE core model developed for IRIS is based on a one-pass modeling approach. In one-pass modeling, hot channels and their adjacent channels are modeled in detail, while the rest of the core is modeled simultaneously on a relatively coarse mesh. Figure 4 shows a detail of the hot assembly for the core model currently used in safety analyses.

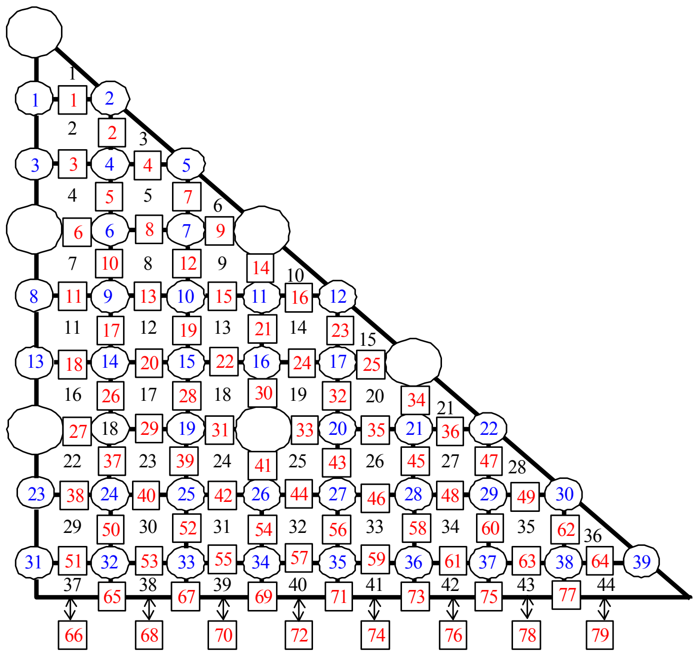

Figure 4. Hot Assembly model for IRIS transient analyses (rods number in blue, channel numbers in black and gap numbers in red)

The only IRIS feature that will require some development are appropriate correlations for departure from nucleate boiling, that will have to be specific for the IRIS fuel assembly design (enhanced moderation ratio and innovative grid design) and thermal hydraulic (low core flow velocity) conditions.

2.3 Secondary Side and Safety Systems

IRIS balance of plant and safety systems do not present significant computational challenges. IRIS relies on passive safety systems [13] that are similar in phenomenological behavior to AP600/AP1000 systems, for which Westinghouse has extensive analytical experience. Limited testing will be required to provide a sufficient assessment base for some of the systems.

3. COMPUTATIONAL CHALLENGES IN THE ANALYSIS OF IRIS SMALL BREAK LOSS OF COOLANT ACCIDENTS

An innovative safety approach has been developed to mitigate the IRIS response to small break LOCA. The IRIS response to a small break LOCA event is presented in detail in a companion paper [13], and only some general considerations are provided here. The IRIS strategy is based on the interaction of the IRIS compact containment with the reactor vessel to limit initial blowdown, and on depressurization of both the reactor and containment through the use of a passive Emergency Heat Removal System (EHRS), which removes heat (condenses steam) using the SGs inside the reactor vessel. A small Automatic Depressurization System (ADS) provides supplementary depressurization capability. A pressure suppression system is provided to limit the containment pressure peak following the initial blowdown to well below the containment design limit. The ultimate result is that during a small-to-medium LOCA, the core remains covered for an extended period of time, even without credit for emergency water injection or external core makeup. The IRIS LOCA response is in fact based on "maintaining water inventory" in the vessel rather than on the principle of safety injection.

This novel safety approach poses significant issues for computational and analysis methods since the IRIS integral reactor coolant system and containment are strongly coupled, and the system response is based on this interaction. A preliminary assessment has led to the conclusion that in order to develop an appropriate Evaluation Model for the IRIS SBLOCA, the containment/vessel coupling had to be correctly described.

For this reason a coupled RELAP5/GOTHIC model was developed [14] by the University of Zagreb. A simple direct coupling of RELAP5/mod3.3 and GOTHIC 3.4e [15] was used with connections at the points of hydraulic contact (the break, ADS, and gravity makeup flow paths). The connections are comprised of a time dependent volume component on the RELAP5 side and a flow boundary condition on GOTHIC side. The existing detailed RELAP5 model of the reactor coolant system and of the engineered safety features is used for these analyses, together with a simplified GOTHIC model of the containment.

The coupling is direct and doesn't require the use of any additional software tool or protocol. Also, the coupling is explicit in time and RELAP5 is the leading part of the coupled code. Containment conditions from the previous time step are used in the RELAP5 new time step calculation of the integral reactor coolant system and connected systems. At the end of the converged RELAP5 calculation time step, the interface subroutines transfer boundary condition data to GOTHIC. GOTHIC performs one or more time step calculations and then the interface subroutines forward the conditions for next time step to the RELAP5 code. The structure of the coupled codes is shown in Figure 5.

Each code can perform independent time step calculations with smaller time steps, but during this process interface information is kept constant.

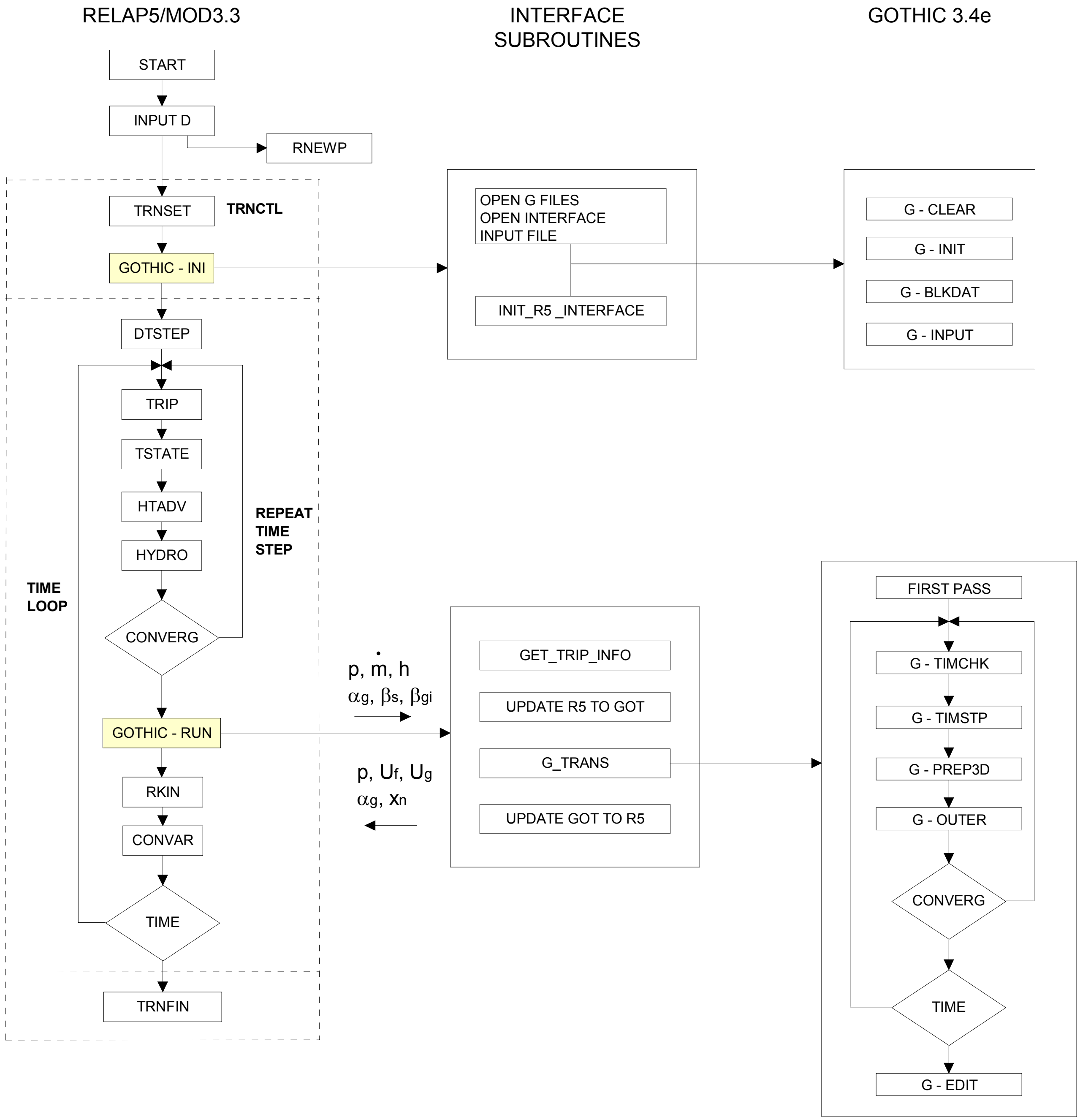

Figure 5. Structure of RELAP5/GOTHIC coupling

The two codes use different integration variables and therefore the coupling interface has to perform the necessary conversions. The interface subroutines responsible for providing GOTHIC data to the RELAP5 code use GOTHIC liquid and droplet data to produce RELAP5 liquid phase data. Optional conversion of the RELAP5 liquid flow to droplets is automatically handled by the GOTHIC flow boundary condition depending on input data.

The details of the code coupling have been presented in [14], and as a result of the coupling both RELAP5 and GOTHIC are applied to the portions of the plant where they can best perform.

3.1 Simplified Containment Model and Preliminary Verification of RELAP/GOTHIC Coupling

The use of RELAP for the study of the integral reactor coolant system and the engineered safeguards features allows the use of the detailed plant model developed for IRIS in the analyses of small break LOCA. For preliminary analyses and assessment of coupling the RELAP and GOTHIC codes, a Simplified Containment Model (SCM) has been developed and is shown in Figure 6.

The coupling was initially checked by comparison with RELAP5 stand-alone results in a situation where it is assumed that both codes can give similar predictions. The base case for containment modeling (label *R5 only* in Figure 7) is one containment node modeled as a RELAP5 *branch* component (initial conditions at 101.325 kPa, 40 ^{o}C, nitrogen filled). The corresponding coupled code case (label *R5+G*) uses one control volume and one flow boundary condition on the GOTHIC side, and the *branch* component is replaced with time dependent volume component on the RELAP5 side.

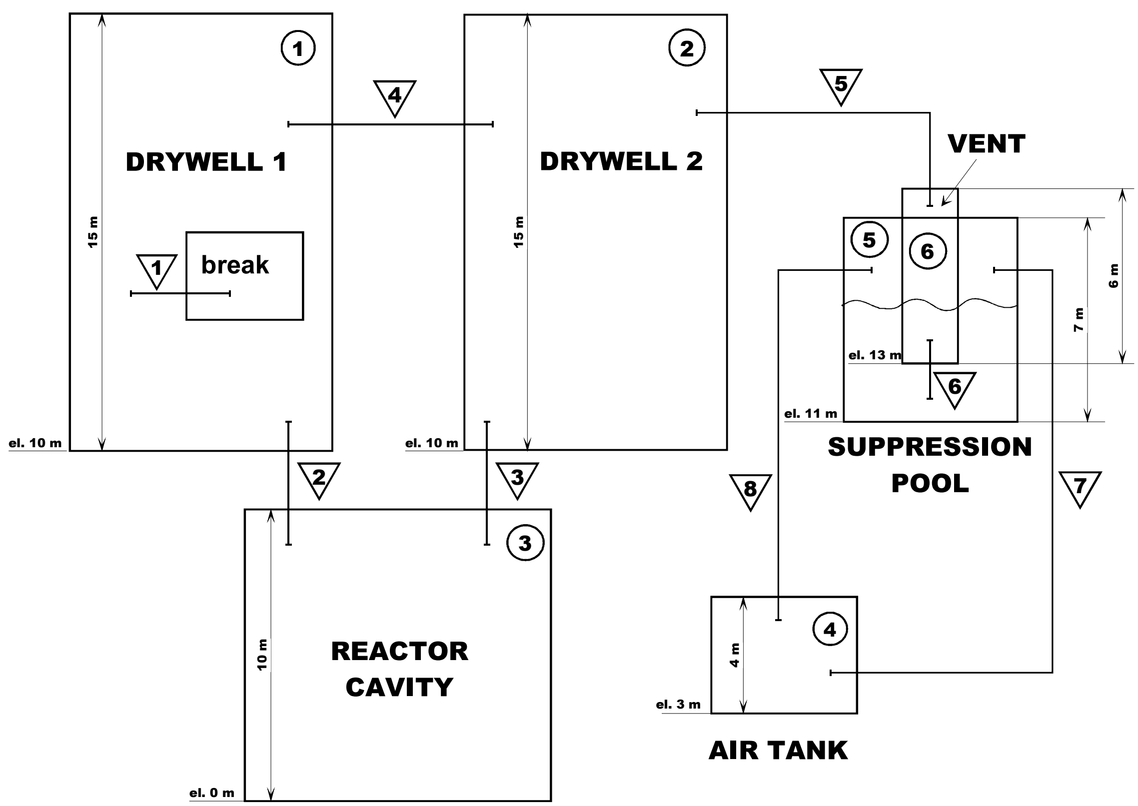

Figure 6. IRIS Simplified Containment Model for GOTHIC

An additional functional test of the coupled code (label *R5+G SMC*) was performed for the simplified containment model with six GOTHIC volumes simulating the pressure suppression function of IRIS containment. This case represents a preliminary, more physically true analysis of the SBLOCA in IRIS. The RELAP5 nodalizations used (and the steady state restart file) are exactly the same as those used in the one-node case.

Results of the initial assessment are presented in [14] and show a good performance of the coupled code, as shown also by the reactor vessel and containment pressures for the three cases in Figure 7.

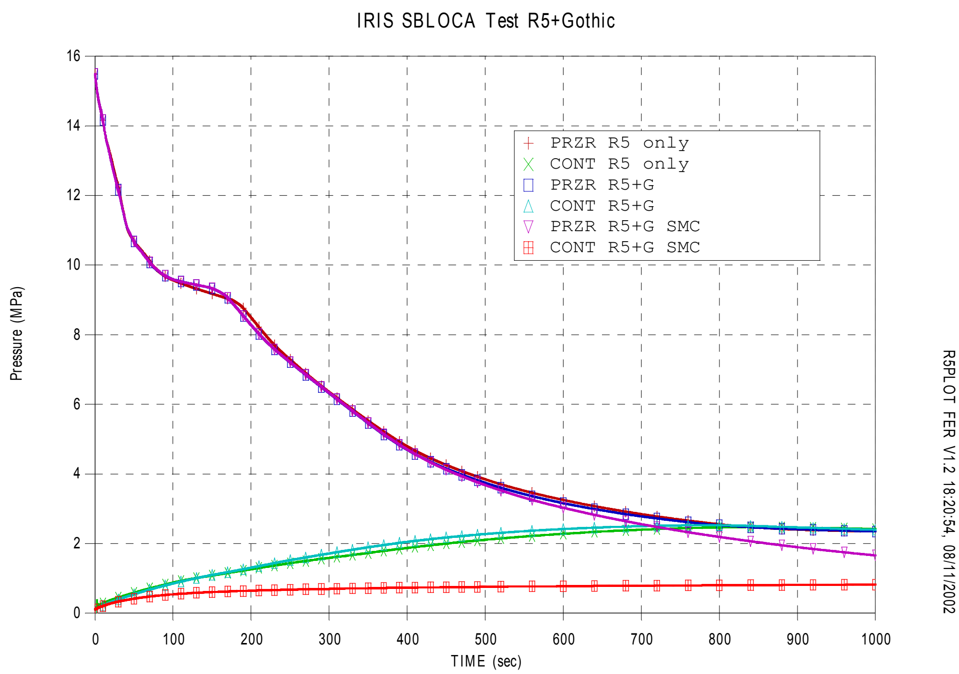

Figure 7. IRIS SBLOCA system and containment pressures calculated by RELAP5/mod3.3 and coupled RELAP5-GOTHIC code for single node and SMC containment

3.2 Future activities in the analyses of Small Break LOCAs for IRIS

The preliminary results of the GOTHIC/RELAP coupling have been promising, and the coupled code is currently considered the tool of choice for the analysis of SBLOCA events on IRIS.

While the initial assessment featured the break as the only connection between containment and vessel, multiple connections between system and containment models are possible. Also, the connections are not limited to atmospheric regions only (water level effect on boundary condition pressure is taken into account in GOTHIC). In addition to the coupling of fluid systems it is possible to exchange trip information between the two codes and heat structures in one code can be connected to control volumes in another code. The GOTHIC capability to allow subdivision of the containment lumped volumes can be used in a coupled version to perform multidimensional calculations. This will allow proper definition of the other safety systems that need to be modeled in the SBLOCA analyses of IRIS and will also allow the trip logic for the sequence to be properly defined.

The coupled code will therefore be used to perform the IRIS SBLOCA analysis, and an appropriate testing campaign will be defined to validate this approach. Preliminary validation will be performed on available test data, for example from the PANDA facility. Each code will be applied to the areas where they can perform best, but new testing will be required and will include separate effect tests for new components or operating conditions, and integral effect tests to validate to the use of the coupled codes.

4. CONCLUSIONS

The IRIS reactor is an innovative PWR with an integral reactor coolant system and a new approach to safety based on the “safety by design” approach. While IRIS is firmly grounded on the extensive technical and analytical experience matured in the past 50 years on pressurized water reactors, new Evaluation Models for the safety analyses and licensing of the reactor will need to be developed. This paper presents an overview of the main challenges, from a computational and analysis point of view, that have been or will have to be addressed as appropriate Evaluation Models are developed.

In particular, the development of a coupled system/containment code for the analysis of the IRIS small break LOCA is introduced.

ACKNOWLEDGMENTS

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