Pressurized Water Small Modular Reactor (SMR),
Design Basis Accident Analysis using the ASTEC code

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ABSTRACT

According to classifications adopted by the IAEA, small reactors are characterized by
an equivalent electric output of less than 300 MW while medium-sized reactors by an
equivalent electric power between 300 and 700 MW. Pressurized water small and medium
sized reactors (SMR) generally adopt an integral layout of the primary circuit with in-vessel
location of steam generators and control rod drives; one compact modular loop-type design
features reduced length piping, an integral reactor cooling system accommodating all main
and auxiliary systems within a leak-tight pressure boundary, and leak restriction devices.

In this paper, a description is given of the development of the modelling and noding of
the primary loop, secondary loop passive core cooling system and containment for a SMR,
based on the available data of the SPES3-IRIS integral test facility.

SPES3-IRIS is under construction at SIET laboratories in Piacenza (Italy), simulating
with 1:100 volume scale and 1:1 height scale, the primary, secondary, containment and safety
systems typical of the IRIS small modular reactor.

Three ASTEC code modules were adopted: the ICARE module to predict the in-vessel
phenomena, the CESAR module to compute two-phase thermal–hydraulics in the Reactor
Cooling System (RCS) and for the control and safety systems, and the CPA module to
evaluate thermal–hydraulic and aerosol behaviour in the reactor containment.

The SMRs as well as the advanced nuclear water-cooled reactors rely on containment
behaviour to achieve some of their passive safety functions. Steam condensation on
containment walls, where non-condensable gas effects are significant, is an important feature
of the new passive containment concepts. Thus, to simulate correctly the main phenomena
involved during an accident scenario, the coupling between primary circuit and containment
has to be reproduced accurately. Furthermore, given that the containment plays a fundamental
role during every accident scenario, it has to be taken into account just as a real safety system.

The worst design basis event for the SMR was analysed, and the calculated results were
compared with those obtained by the University of Zagreb in collaboration with
Westinghouse using the coupled codes RELAP-GOTHIC. The aim of this work is to evaluate
the applicability of ASTEC coupled modules in the safety analyses of the new reactor systems
with strong interaction between primary system and containment.
INTRODUCTION

Today, there is a renewed interest in the development and near term deployment of advanced/innovative Small and Medium sized Reactors (SMRs). This technology is the most suitable option for deployment in countries with small electrical grid capacity or electricity demand, as well as for non-electrical nuclear energy applications, i.e. seawater desalination, district heating, hydrogen production and other process [1]. From the technological point of view, the SMRs are similar to large reactors, but they differ for the higher degree of innovation implemented in their designs to achieve competitiveness and reliable performances. The SMR designs (integral type, PWR based concepts), such as NuScale, mPower and W-SMR, share a common set of design principles adopted also for the IRIS reactor to enhance plant safety and robustness [2]. The SMR common design consists of: incorporation of primary system components into a single vessel, increased relative inventory in the primary reactor vessel, more effective heat removal, increased relative pressurizer volume, vessel and component layout that facilitate natural convection cooling for core and vessel [3]. The integral layout permits to fulfill the so named “safety-by-design” approach, that means trying to eliminate accident initiators, or, if not possible, trying to limit their consequence and/or their probability of occurring. The lack of large primary penetrations in reactor vessel or large loop piping eliminates by-design the possibility of large releases of primary coolant (Large Break LOCA accidents). The strategy to mitigate LOCA consequences is based on “maintaining water inventory” rather than on the principle of safety injection, [4]. This new safety approach poses significant issues for computational and analysis methods since the IRIS vessel and containment are strongly coupled, and the system response is based on the interaction between the two. In order to explore the reactor vessel and containment interaction in more detail, an ASTEC [5] model was developed and applied for the analysis of a postulated 2-inch Double Ended Guillotine (DEG) break on the Direct Vessel Injection (DVI) line. The ASTEC calculated results will be compared with those obtained by the University of Zagreb (FER) in cooperation with Westinghouse using the RELAP5 and GOTHIC coupled codes, and available in Refs. [6] and [7].

1. SMALL MODULAR REACTOR NODALIZATION

Figures 1 and 2 show a basic view of the Reactor Vessel (RV) and the containment of the IRIS reactor, whilst a graphical representation of the ASTEC primary and secondary circuits is given in Figure 3. The primary circuit model is a coarse node representation of the hydraulic system and structures comprising 89 hydraulic control volumes, and includes the upper part of the integral reactor vessel above the core level, reproducing 8 internal loops (1+1+3+3), the Emergency Boration System (EBS) with 2 tanks and 2 Refueling Water Storage Tanks (RWSTs). The model respects the specifics of the SPES3 facility described in [8] and [9]. The reactor vessel is simulated by mean of five radial rings and twenty-one rows, for 105 meshes, using the data available in [9] and [10]. The secondary system is nodalised in 74 fluid meshes, which include eight Once Through Steam Generators (2+2x3 OTSGs) and 4 Emergency Heat Removal System (EHRS) trains (1+1x3). The model also includes a preliminary version of the basic reactor protection system [10] that will be improved and extended after preliminary accident sequences analysis. Control variables are provided for calculation of: transferred power (core, SG, EHRS), fluid mass in all-important parts of the nodalisation. The containment model was subdivided in 10 control volumes and 13 flow paths Figure 4, only the drywell walls are taken into account, the other heat structures are not part of the model. The drywell space is further split into four parts (volumes D1, D2, D3 and D4); the lower parts are connected to the reactor cavity (volume...
CAV). The horizontal connections among the volumes are represented with two flow paths to simulate mixing between volumes.

Figure 1: Layout of the Reactor Vessel (RV)  
Figure 2: Reactor Containment (RC)  
Figure 3: Integral vessel nodalisation  
Figure 4: Containment nodalisation
The pressure suppression pool (volume PSS) is connected to the volume VENT; this volume simulates the vent pipes that connect the suppression pool to the containment atmosphere (D3, D4). The red arrows in Figure 4 represent the seven connections between the CESAR and CPA module, one for the break, two for the ADS valve connection, two for the Long Term Gravity Make-up System (LGMS) lines and two lines going from the reactor cavity to the DVI. Because of the small containment volume, a high level of interaction between containment and primary system is expected during the LOCA, moreover the model will have to be able to reproduce the pressure suppression capability of the contained suppression pools, which also provide the long-term supply source to cooling the reactor. In this calculation, the SP was simulated using the ‘insertion junction’ model of CPA.

2. IRIS RESPONSE TO TRANSIENTS AND POSTULATED ACCIDENTS

As stated before, IRIS is designed to limit the loss of coolant from the vessel rather than relying on active or passive systems to inject water into the RV, Figure 5, moreover the integral design eliminates the possibility of large break LOCAs, since no large primary system piping is present in the reactor coolant system. This is made possible by the advantage of the following three features of the design:

- The initial large coolant inventory in the reactor vessel;
- The EHRS which removes heat directly from inside the RV thus depressurizing the RV by condensing steam, rather than depressurizing by discharging mass;
- The compact, small diameter, high design pressure containment that assists in limiting the blowdown from the RV by providing a higher backpressure in the initial stages of the accident and thus rapidly equalizing the vessel and containment pressures.

After the LOCA initiation, the RV depressurizes and loses mass to the containment causing the containment pressure to rise (Blowdown Phase). The mitigation sequence is initiated with the reactor trip (P_{cont} > 1.7 bar), pump trip; the EBTs are actuated to provide borated water; the EHRS is actuated to depressurize the primary system by condensing steam on the steam generators (depressurization without loss of mass); and finally the ADS is actuated to assist the EHRS in depressurizing the RV. The Pressure Suppression System (PSS) and the reduced break flow due to the EHRS heat removal from the RV, limit the containment pressure. At the end of the blowdown phase the RV and CV pressure become equal (Pressure Equalization) with a containment pressure peak of approximately 8-9 bar. The break flow stops and the gravity makeup of borated water from the LGMS tank becomes available (Depressurization Phase). The PSS and LGMS pressures tend to decrease slower than the Drywell pressure, generating a pressure gradient among the zones inside the containment. This gradient favours water injection from the LGMS tanks to the RV and causes the ejection of a portion of suppression pool water through the vents and assists, which floods the vessel cavity. The Long Term Cooling Phase follows the depressurization phase, where the RV and containment pressures are slowly reduced as the core decay heat decreases. Since decay, heat is directly removed from within the vessel and the vessel and containment are thermodynamically coupled, the long-term break flow does not depend on the core decay heat, but it is in fact limited to only the containment heat loss.
Before any transient calculation, it is necessary to run a steady-state calculation. The aim of this calculation is to bring the initial state of the reactor, defined in the input deck, to the physical state at which the reactor is operating in normal conditions. This is performed using regulations, which act on the pressurizer heater, pumps, and so on, in order to reach the desired physical values.

Table 1: Thermal–hydraulic parameters of the IRIS reactor at nominal power

<table>
<thead>
<tr>
<th>Parameters</th>
<th>Unit</th>
<th>RELAP5 –GOTHIC[5][6]</th>
<th>ASTEC v2r2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pressurizer pressure</td>
<td>MPa</td>
<td>15.5</td>
<td>15.5</td>
</tr>
<tr>
<td>Vessel flow</td>
<td>kg/s</td>
<td>4524</td>
<td>4504</td>
</tr>
<tr>
<td>Core inlet temperature</td>
<td>°C</td>
<td>291</td>
<td>293</td>
</tr>
<tr>
<td>Core outlet temperature</td>
<td>°C</td>
<td>329</td>
<td>329</td>
</tr>
<tr>
<td>Steam inlet temperature</td>
<td>°C</td>
<td>224</td>
<td>224</td>
</tr>
<tr>
<td>Steam exit temperature</td>
<td>°C</td>
<td>318</td>
<td>317</td>
</tr>
<tr>
<td>Total steam flow</td>
<td>kg/s</td>
<td>503</td>
<td>513</td>
</tr>
<tr>
<td>Water mass (RV)</td>
<td>kg</td>
<td>325400</td>
<td>316330</td>
</tr>
</tbody>
</table>

Table 1 shows some characteristic values of the primary and secondary circuits at the end of the steady state calculation at full power. One can see that the overall agreement between ASTEC and RELAP-GOTHIC is very good.
4. TRANSIENT ANALYSIS

A 2 inch Double Ended Guillotine Break (DEGB) on the DVI line was chosen for this analysis. The transient starts after 30000 s of steady state. The break opening is initiated at t=0 s transient time. The first 10000 s of accident are calculated and the main parameters are compared. One EBT tank (intact loop) and four EHRS sub-systems are actuated following the SCRAM signal on high containment pressure at 30 s. The powers removed by the EHRS and the EBT injection flow rate calculated by the two simulations are in agreement (Figure 6 and Figure 7). During the fluid discharge phase, the ASTEC case predicts a higher RV depressurization rate until 400 s (Figure 8). The pressure equalization occurs around 2000 s for both calculations, Figure 9. However the two simulations show some differences; when the blowdown phase runs out, the RELAP-GOTHIC case calculates a decrease of the RV pressure well below the drywell pressure, while in the ASTEC case the RV pressure assumes the same value of the containment pressure until 7000s (Figure 9). After the pressure equalization, the flow at the break position and at the ADS valve will change direction. Figure 10 shows the reverse steam flow coming from the break and the ADS 1 valve calculated by ASTEC. All the steam condensed in the suppression pool causes an increase of the liquid level into the pool zone and consequently a backpressure rise. Moreover, after the peak, the drywell pressure decreases more quickly than PSS pressure, thanks to steam condensation on the containment wall, and after the RV and drywell pressures are coupled, by means of the EHRS heat removal from the primary circuit. Figure 11 illustrates the pressure gradient among the different zones inside the containment during the transient. The PSS pressures calculated by ASTEC tend to decrease more rapidly than those computed by GOTHIC. This difference is due to the model used to simulate the suppression pool, which is not able to reproduce a liquid volume inside the vent pipe volume and the two zones remain connected by mean an atmospheric junction.
When the differential pressure between PSS and DW is sufficient to overcome the hydrostatic head of the PSS vent pipes, a reverse flow starts from the PSS to the drywell through the vent lines; the ejected water floods the cavity of the reactor ensuring external cooling of the integral vessel. Figure 12 shows the water level in the cavity zone. During this phase, the LGMS system on the intact loop starts to discharge into the RV as illustrated in Figure 13. The calculated flow shows that a great difference exists between the ASTEC and RELAP-GOTHIC injection trend. The ASTEC code calculates a mass flow greater than the RELAP-GOTHIC code throughout its operation time of at least 5 times more. In this way, the LGMS are empty around 6000s for ASTEC and 15000 s for RELAP-GOTHIC. The faster LGMS tanks emptying, also affects the PSS depressurization observed. The reason of this discrepancy could be the different pressurization of LGMS with respect to the DVI, but the results do not show pressure differences to justify the discrepancy. Overall, most of the main parameters calculated during the transient showed the same trend, and show the capability of ASTEC code to reproduce these kind of phenomena.

5. CONCLUSION

The analysis of the DVI DEG break transient by the comparison between the ASTEC code and RELAP-GOTHIC coupled codes has shown general agreement in the occurring phenomena from the qualitative point of view. The interaction of the IRIS primary system with the containment following the accident scenario and the behaviour of the passive safety systems are well reproduced by the ASTEC code. This preliminary testing of the model has showed that the discretization approach is acceptable and that the model produces reasonable
steady-state and transient results. Further improvements and corrections are expected after the next phase of model testing that will be performed by means of sensitivity analyses. After this phase of testing, the model could be used to simulate a hypothetical severe accident scenario.

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REFERENCES


