Applicability of Coupled Thermalhydraulic Codes for Safety Analysis of Nuclear Reactors

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1. Introduction

Performance assessment of LWR safety systems during Design-Basis-Accidents (for instance LOCA, Reactivity Initiated Accidents, etc) and Beyond-Design-Basis Accidents should be done to ensure safe and stable operation of the nuclear power plant facilities. To this end computational codes like RELAP and TRACE are used to model thermal-hydraulic response of nuclear power plant during an accident. By careful modeling and significant user experience these system codes are able to simulate the behavior of primary system and the containment to a reasonable extent. Comparatively decoupled simulation is simple but might not produce reality and the physics involved in an accurate manner. Thus simulation using two different system codes is interesting as the whole system is coupled through the pressure in the containment and flow through the break. Using this methodology it might be possible to get new insight about the primary and containment behavior by the precise simulation of the accident both in the current reactors and future Gen-III/III+ reactors.

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2. Couple Codes

Couple codes have been in use since quite a time in the Nuclear Engineering domain. Coupled neutronic-thermalhydraulic codes together with structural mechanics input have set a benchmark in the safety analysis of nuclear reactors. Tightly coupled reactor system with strong thermalhydraulic feedback is the rationale behind using such approach. The idea of using coupled thermalhydraulic code is new and is being studied extensively in the case of an ESBWR reactor where there is a strong interaction between the performance of PCCS pool and the pressure in the containment. Further it could be interesting to use it in the place of traditional two step safety analysis methodology where first the response of the primary system is calculated using a standalone system code and then mass and energy release is given to the containment code for further calculations (Keco et al 2005).

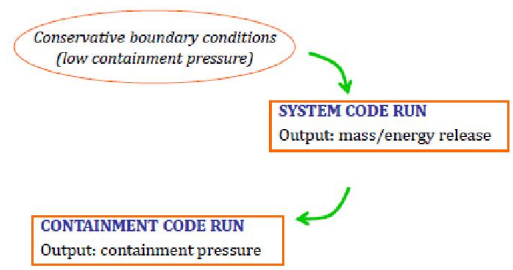

Fig 1. Two step decoupled calculation approach

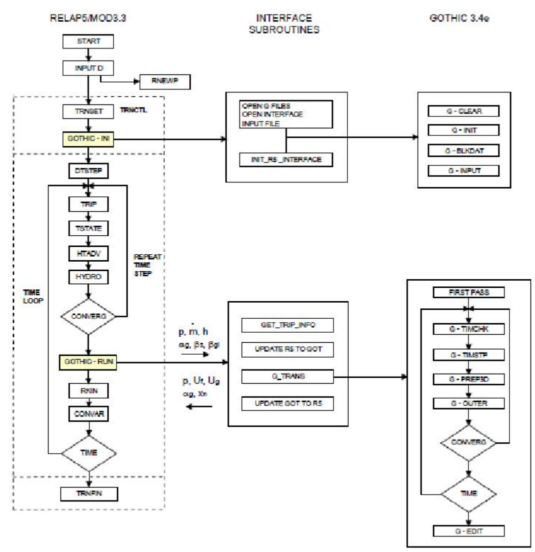

Fig 2. Couple R5G code (RELAP-GOTHIC) Keco et al 2005

2.1 Couple thermalhydraulic Codes applied to PWR systems

This approach gives the analyst a chance to see whole system in a multiphysics based simulation where the containment environment, break mass flow, cladding temperature etc are metamorphosed in a dynamic fashion.

However the applicability of these codes to the GEN-II PWR systems has found limited success where there is almost low to negligible coupling is found between the primary and the containment system, rendering them to be essentially decoupled system. It is further shown in Fig 3 that the pressure transient calculated using both standalone GOTHIC and couple R5G code produce the identical results.

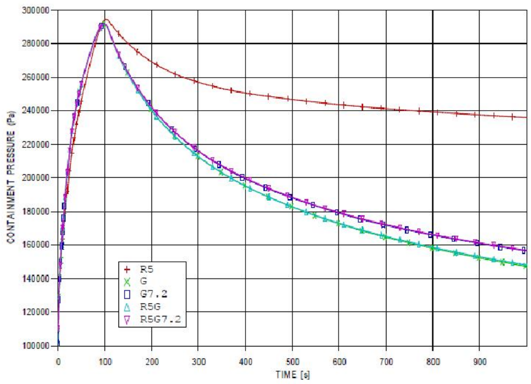

Fig 3 Containment pressure calculated using R5G and GOTHIC, Keco et al, 2005

Nevertheless some interesting studies can still be performed using the idea of coupled thermalhydraulic code methodology. One instance of which is the study of sump stratification issue and its effect on the long term cooling capability of the ECCS system. The rationale behind this is the dynamic response of the sump to the containment back pressure, in here the onset of stratification phenomenon in the sump may pose a considerable challenge to the ECCS system and its ability to cool down the plant.

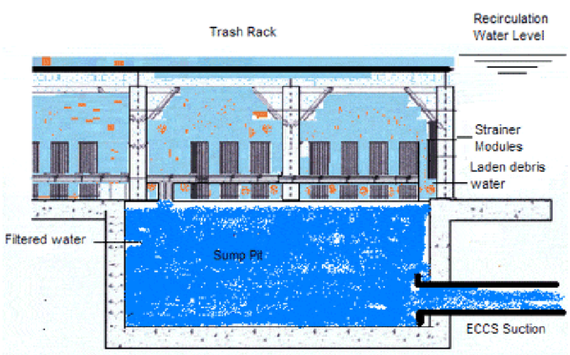

Fig 4. PWR sump

2.2 Couple thermalhydraulic Codes applied to BWR systems

Couple code calculations are also know for BWR systems (ATHLET-COCSYS) The dynamics of the accident can be a little different, nevertheless there are minor differences during the accident progression and both the standalone and coupled version produced the identical results. However when there is a small difference of pressure between the drywell and the RPV the situation started to change a little and the couple code started to give a different result than the standalone version. It can be seen in Fig.5 that at the end water level started to rise faster in the coupled code run than in the.

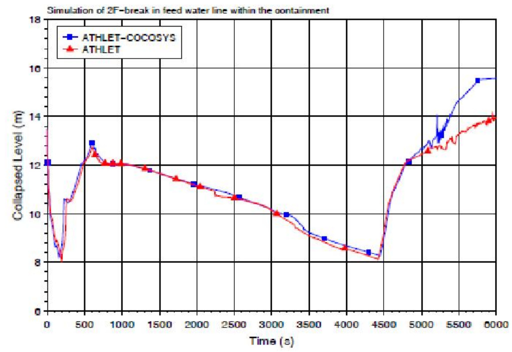

Fig 5 Collapsed water level in BWR RPV (ATHLET-COCOSYS),Hoffmann et al,2011

standalone run, pointing towards the fact that containment back pressure is indeed affecting the break mass flow.

Papini et al have studied and validated the TRACE-GOTHIC coupled code using the ISP-42 experiment for ESBWR reactor. The results proposed by them were quite encouraging for the usage of the couple code methodology in an ESBWR reactor system. However the applicability of the couple code for the GEN-II systems is found to be limited; nevertheless some of the specificities are still not explored fully and thus provides a room for improvement and further investigations.

3. Conclusions

Couple thermalhydraulic code methodology is still new and require further investigations. Applicability of such methodology to the GEN-II plants have met with limited success, however a number of situations in which this methodology could be applied are still unexplored and thus provides a room for improvement and modifications. One such possible situation in the context of PWR is discussed in this paper which could be explored.

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