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IMPLEMENTATION OF AN EXTERNAL CODE FOR THE CONTROL AND PROTECTION SYSTEM OF ATUCHA I NUCLEAR POWER PLANT

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Abstract. One of the fields of study of paramount importance when modeling the behavior of a nuclear power plant is the control and protection systems. Depending on the scope of the implementation, the degree of accuracy of the control and protection actions may vary from a full spectrum coverage up to an extremely simplified model with respect to the real system installed in the plant. In the light of this consideration, certain tools and approaches might be more suitable for programming the control and protection logics depending on what the ultimate subject of the engineering study is (i. e. operational or safety transients). For instance, RELAP (which is a widely used code in the nuclear industry for performing transient calculations for nuclear installations) offers the possibility of modeling a control system through numerical cards that define generic control components such as arithmetic operations, integrators, lead-lag operators, mathematical functions, etc. Nevertheless, this tool is quite complex to be implemented if a high degree of accuracy is desired. On the other hand, this task could be far more simple to fulfill using a high-level programming language such as C or Fortran. If the protection and control actions are programed in an external code, both RELAP and the external program should be able to exchange information with each other (and eventually others) in order to perform coupled calculations successfully. Taking this last concept into account, an extension of RELAP, developed by TECNA S. A., capable of exchanging information through shared memory and synchronizing the different processes involved with semaphores has been used to perform a coupled calculation between the RELAP extension (modeling the plant) and an ad-hoc program written in Fortran (modeling the control and protection systems). In addition to this, a radiation model between the fuel rods and the coolant channel as well as the computation of the net reactivity were implemented. In this paper a general description of these models are presented as well as a brief description of the exchanged data flow. To conclude, the advantages of implementing this approach versus the one available with RELAP standalone are highlighted.

1 INTRODUCTION

A nuclear power plant is an extremely complex installation which aims to produce electricity from the energy released when heavy nuclei are split into lighter elements, in the case of a fission nuclear power reactor. This process, among others, has to be carefully controlled in order to guarantee power supply as well as safe operation points. During the normal operation of a nuclear reactor, some disturbances might occur that could generate a disequilibrium between the different systems causing some magnitudes to shift from their setpoint. If no action is taken, these deviations could lead to other operational points (which could be undesirable) or to uncontrolled scenarios.

For instance, let's consider the situation of reducing the nuclear power from 100% to 80% of full power. In order to achieve this goal, control rods should be inserted in order to turn the reactor subcritical and, consequently, to produce a decrease in the nuclear power. Sooner or later, the control rods must be extracted in order to return to the critical condition but the power control system should also deal with the appearance of more xenon, which will cause the control rods to be more extracted at the end than what they were at the beginning of the analysis to compensate for the negative reactivity introduced by the increase of Xe-135. Without a power control system, this operational transient could not have been executed as the appearance of xenon would have shut the reactor down in the short-term.

Another case deem of attention is when the control system actions cannot handle the magnitude of the transient occurring. In these situations the protection system must guarantee the safety functions, namely the reactor shutdown, core cooling and prevention of radiation release. It is a compulsory requirement for licensing a nuclear power plant to evaluate the performance of this systems in case of certain initiating events (chapter 15 of the Final Safety Analysis Report). This evaluation needs to be performed using computational codes, as the real test will imply damaging the plant and the possibility of radiation release. To achieve this task, RELAP (Information Systems Laboratories Inc., 2001a) is a widely used program for calculating both, operational and accidental transients.

Atucha I is a pressure-vessel heavy-water reactor with online full-power refueling. It is both cooled and moderated by heavy water. The reactor has 250 vertical cooling channels, each one containing one fuel element. The complete fuel column has a height of approximately 6 meters, whereas the active length is 5.30m long, consisting of 37 rods with a Zy-4 cladding. Originally, the reactor was designed to operate with natural uranium in the form of UO_2 pellets. Currently, because of economic reasons, it operates using Slightly Enriched Uranium (0.85% of ^{235}U).

Inside the Reactor Pressure Vessel, the moderator and the coolant are separated. The coolant channels are immersed inside the moderator tank but they are not directly connected in the core. For reactivity reasons, in average, the moderator is maintained approximately 100°C colder than the coolant. The moderator tank has some openings that communicate with the upper plenum in such a way that both the moderator and the coolant are kept at the same pressure of 115 bar.

The original gross electric design power was 330 MWe with a thermal reactor power of 1100 MW. Two changes were implemented afterward: in 1977 the gross power was increased up to 357 MWe (8%), generating a net power of 335 MWe and a thermal power of 1179 MW. From 1995 up to 2000, a progressive conversion from natural uranium to slightly enriched uranium was performed.

In the next sections, the main control systems as well as the main protection signals will be described for the Atucha I nuclear power plant. In addition to this, the implementation of these systems in RELAP will be specified as well as their implementation in an external program written in Fortran. Furthermore, the data flow exchange between RELAP-CPL (modeling the plant) and the external code (modeling the control and protection system) will be depicted. Finally the advantages of implementing the control and protection actions in an external code will be highlighted with respect to the RELAP standalone case.

2 CONTROL SYSTEM

The control system in a nuclear power plant is in charge of keeping plant magnitudes within certain margins with respect to the established setpoint. The principal regulation models are presented below.

2.1 Power regulation

The power control system is in charge of the control rods position. Only two groups of control rods are modeled, namely the gray and the black banks. The control rods movements are continuous, with an insertion/extraction velocity depending on two power error limits which distinguish between "high/low" power and "very high/low" power situations. It is also modeled a limitation for the control rods that takes into account the relative position of the two banks. A continuous movement without a hysteresis band is not representative of the actual movements of the control rods in the reactor because the reactivity inserted/extracted depends on the time step used (for small time steps the reactivity change will be small and for big time steps it will be greater; with a hysteresis band a certain reactivity is inserted/extracted regardless of the time step used). Nevertheless, for accidental cases the continuous model is accurate enough as the reactor shutdown is always demanded. For future analysis considering operational transients, this issue shall be improved.

In this regulation it is also included the rod # 16, which is employed to rapidly decrease the reactor power when, for example, a load rejection occurs. The movement of this rod is modeled as discrete, i.e., it falls during a period of time and remains still for another period of time.

2.2 Primary coolant system pressure regulation

The primary system pressure is regulated through the action of a spray (which injects water from the cold leg into the top of the pressurizer) and an electric heater. Both components have different stages depending on the pressure error. In the case of the spray, there are four stages which are represented by four values of mass flow into the upper volume of the pressurizer. There exists a fifth spray stage modeled, which is the permanent one.

With respect to the heater, its action is quite similar to the spray: depending on the pressure error up to three stages can be demanded. Each stage is represented with a certain amount of power delivered to the water in the bottom of the pressurizer.

2.3 Regulation of the pressurizer level

To control the pressurizer level, the volume control system plays a crucial roll. This system will command the extractions from the moderator tank in order to achieve the pressurizer level setpoint. During normal operation, the volume control system injects a constant mass flow into the moderator tank through the action of high pressure positive displacement pumps. As a consequence, controlling the extracted mass flow, the pressurizer level can be modified.

2.4 Regulation of the steam generators level

The steam generator level is controlled through the action of the feed-water valves. There are two regulating valves (which operate in a parallel arrangement) for each steam generator, namely one for high power and another one for low power. The commutation from high-power regulation to low-power regulation (and conversely) is performed manually in the plant, so the actions described in the operation manual have been programed to model this switch.

2.5 Regulation of the live steam admission valve to the turbine

Currently, Atucha I presents a load following program by turbine, i. e., a reactor thermal power reference level is set and the power delivered by the generator will be regulated in order to keep a constant live steam pressure. The live steam admission valve is in charge of this regulation.

3 PROTECTION SYSTEM

The protection system is in charge of monitoring some plant magnitudes (such as power, temperature, pressure, etc) and take the corresponding actions when these magnitudes go beyond certain limits. The ultimate purpose of the protection system is to guarantee the reactor shutdown, core cooling and to avoid uncontrolled radiation releases to the environment.

Once an abnormal situation is identified, this system will demand the action of several components (valves, shutdown rods and pumps just to name a few) in order to accomplish the task mentioned in the last paragraph. Some of the signals generated by this system are described below.

3.1 Reactor shutdown signal

Atucha I, as all PHWRs, has two shutdown systems, one based on neutron absorbent rods and another one based on fast boron injection into the moderator tank. This last system is demanded when the depressurization rate of the primary coolant system is greater than a certain value or when the insertion of rods fails.

3.2 Small break signal

This signal is generated when the following conditions are met:

1. low primary system pressure and,
2. high containment pressure or low pressurizer level.

3.3 Large break signal

Similar to the SB signal, this one involves:

1. low primary system pressure and,
2. primary system depressurization rate grater than a limit.

3.4 General interlocks

The above mentioned signals are just a few of the many signals generated by the protection system. This signals, among others, play a vital role in the different components interlocking.

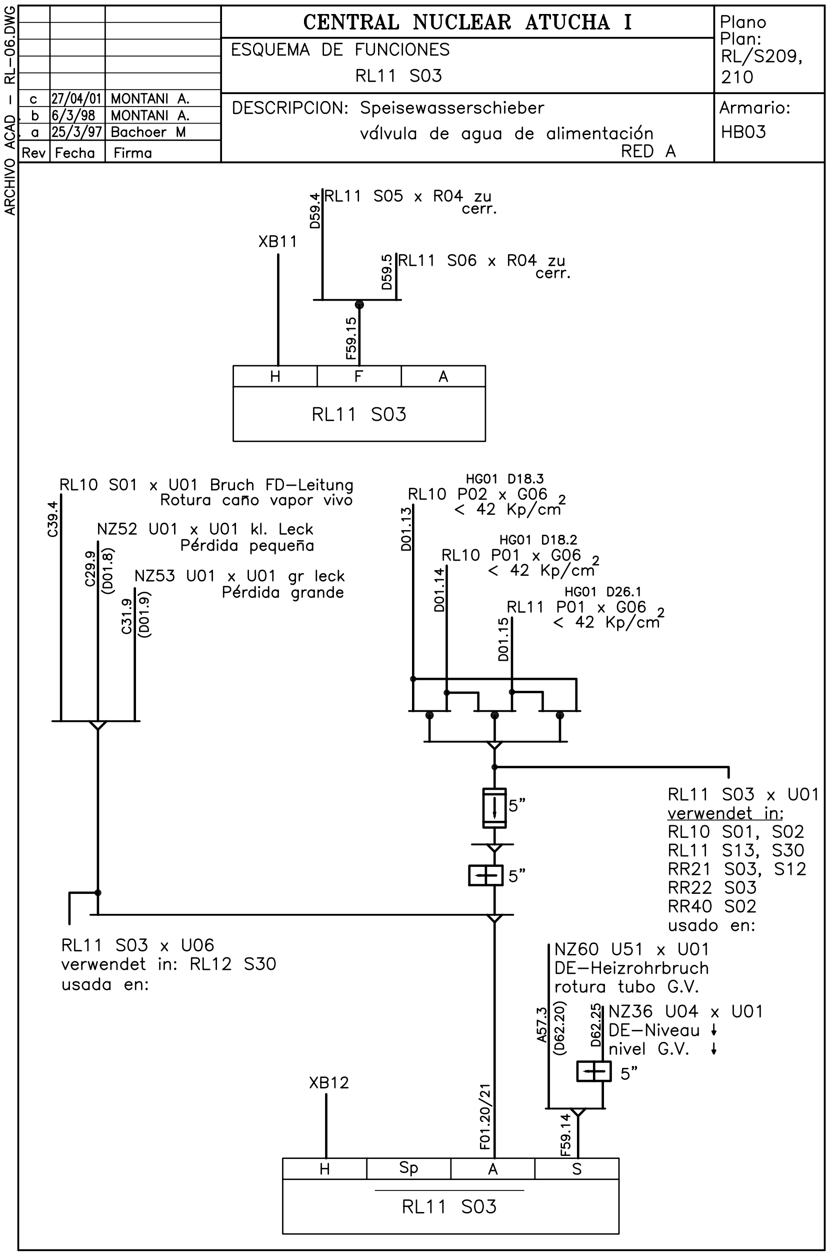

Figure 1: Interloking sheet for a valve that belongs to the live steam system. The upper block corresponds to the opening condition and the lower to the closing one. The gates correspond to the automatic (A), manual (H), opening/closure action enabled (F), opening/closure action blocked (Sp) and protection (S) signals.

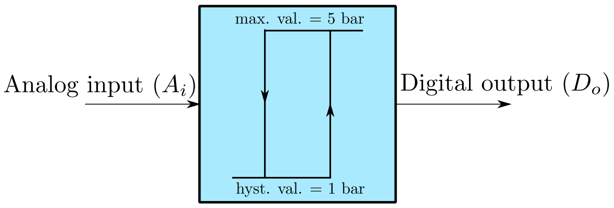

Figure 2: Representation of a threshold value comparator with hysteresis in a block diagram. Both, the maximum value and the hysteresis value is shown, which in this case are 5 bars and 1 bar respectively. The output value is defined in equation 1.

In figure 1 a typical valve interlocking sheet that states under which conditions the valve should close or open is shown. There are different types of gates for each action, namely manual (H), automatic (A), opening/closure action enabled (F), opening/closure action blocked (Sp) and protection (S).

4 ORIGINAL IMPLEMENTATION

All the regulations and interlocking logic above mentioned were first implemented through control variables and trips in a RELAP representation of the whole plant, which included the thermohydraulic, neutron-kinetics and control models. This representation was developed ten years ago to perform safety calculations and partially updated to incorporate some logic related to the installation of a second heat sink.

As it was mentioned earlier, depending on the level of detail required for the control and protection logic the RELAP implementation could result into a complex goal to achieve. Just to give a few examples let's consider the case of a simple threshold value comparator with hysteresis which should be activated when the pressure is greater than 5 bars and should be deactivated when the pressure drops below 4 bars. The conventional block diagram for representing a threshold value comparator with hysteresis is shown in figure 2, where the output value D_o depends on the input value A_i as follows

$$D_o = \begin{cases} \text{true} & \text{if } A_i > \text{max. value} \\ \text{false} & \text{if } A_i < (\text{max. value - hyst. value}) \\ \text{last value} & \text{if (max. value - hyst. value)} < A_i < \text{max. value} \end{cases} \tag{1}$$

The RELAP threshold value comparator with hysteresis implementation is depicted in figure 3 where the resultant trip is E, initialized as “false”. Figure 4 shows a postulated pressure evolution as well as the different trip values in every time step.

Another example is shown in figure 5 where the reactor power scram value calculation is represented in a block diagram. If these actions should be repeated to a n number of calculations, the same diagram should be explicitly repeated n times, due to the lack of statements such as iterative loops.

It should be noticed how complicated programing this logic could be, especially if the control and protection logic are meant to be as similar as possible to the actual systems installed in the plant. Additionally, there exists a restriction in the number of control variables available as well as in the number of trips.

5 CURRENT IMPLEMENTATION

Based on the previous experience with Atucha II where the protection, limitation and control logic are complex and they had to be programmed in an external code to avoid running

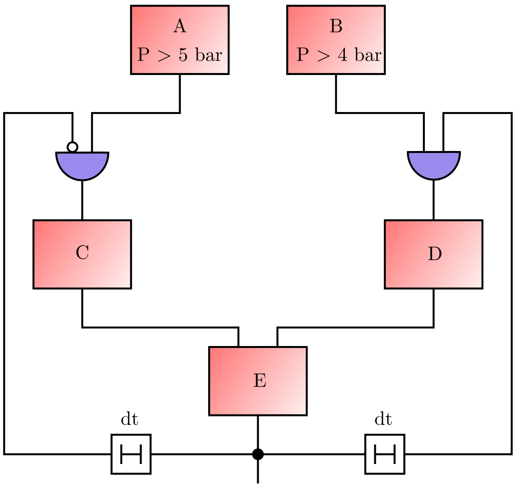

Figure 3: Trip operations in order to program a threshold value comparator with hysteresis which should be activated when the pressure is greater than 5 bars and deactivated when the pressure is lower than 4 bars. Trips A and B are variable trips, whereas trips C, D and E are logical trips. Trip E is calculated as an "OR" gate and trips C and D are calculated with the value of E in the previous time step.

out of control variables (ten thousands for the extended notation) for the desired accuracy extent, Atucha I control and protection logic were also written in an external code. In addition to this, the protection system updates were incorporated as well as the different components interlocking modifications.

So, the RELAP representation of the plant will be in charge of modeling the thermohydraulic and neutron-kinetic aspects of the calculation, whereas the protection and control actions, reactivity and power distribution calculations will be performed by an external code.

The aforementioned task has been coded in a set of Fortran subroutines which are implemented as a plug-in of the wasora framework, in charge of performing the data exchange and synchronizing the different processes, among other functions. Some of the subroutines were already written as they belonged to the ad-hoc program used by the Atucha I designer to perform transient calculations many years ago.

An extension of RELAP called RELAP-CPL (TECNA S. A., 2010) capable of exchanging information and synchronizing with other codes has been employed. Consequently, apart from coding the Fortran subroutines, the RELAP input file had to be modified in order to adequate the way in which the external variables will be received. Moreover, the former RELAP implementation of what has been programed externally was removed from the input file and a corresponding coupling file was developed.

5.1 Radiation model and reactivities calculation

The new RELAP nodalization incorporates 49 coolant channels, i. e., 34 coolant channels more with respect to the last nodalization available. The RELAP in-built radiation model only allows up to 99 heat structures being affected by this model according to Information Systems

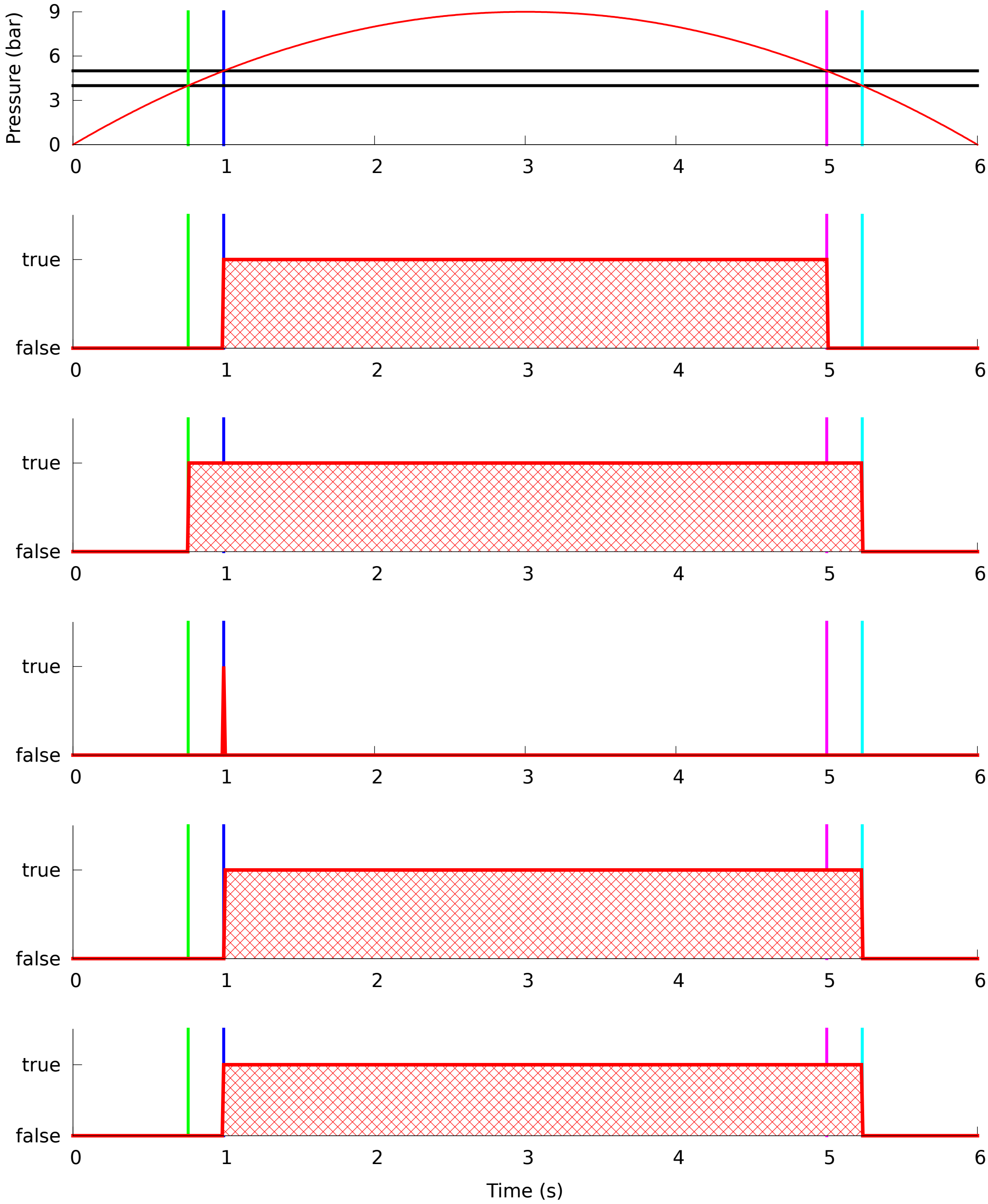

Figure 4: Postulated pressure versus time and the different trips involved in figure 3.

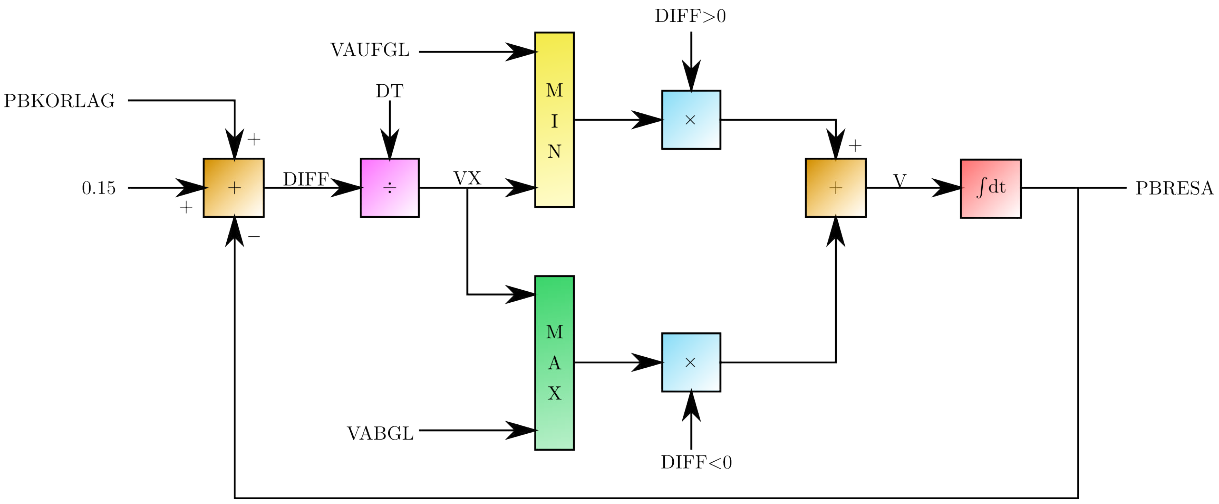

Figure 5: Block diagram representation for the calculation of the reactor power scram value.

Laboratories Inc. (2001b). Each channel consists of 20 axial cells, so the in-built RELAP radiation model cannot be implemented for the entire core. There still exists the possibility of programming this model using control variables, which will imply approximately 5000 control variables and 1000 variable trips (the maximum number of this kind of trips available) for the activation of the radiation model. For these reason all these calculations were performed externally taking advantage of the versatility of high level programming languages such as C or Fortran and the coupling capabilities provided by RELAP-CPL.

Regarding the feedback reactivities, they are averaged using kinetics weight factors which depend upon the magnitude P_i^2/V_i, being P_i the nuclear power generated within the cell i and V_i the volume of the ith-cell. Again, this calculations are far more simple to implement with a DO loop rather than using control variables.

5.2 Coupled scheme and information exchanged

In order to be able to perform the calculations, the external code needs to be fed with some plant magnitudes such as pressures, temperatures, levels, densities, etc., which are computed by RELAP-CPL. Similarly, RELAP components such as valves, pumps, etc., and reactivity and power distribution depend on the external code calculations.

Consequently, both codes should interact exchanging certain information in an organized manner. For that end, RELAP needs (apart from the input file describing the problem) a coupling file with information about which variables must be exported, which must be imported and when that processes should take place.

In this context, RELAP-CPL should export plant magnitudes for the external code to read them. This last program should perform the corresponding calculations and write the required information back in order to RELAP-CPL to be able to use it in the next time step. RELAP-CPL writes to shared memory the reactor total power, densities, void fractions, pressures, collapsed levels, temperatures, mass flows, velocities, internal energies, quality values and some control variables values and read from shared memory valve stem positions, pump trips, valve opening/closing trips, mass flows, reactivity values and the power dissipated in the fuel elements, the moderator tank and the coolant channel wall for the 20 axial cells.

6 ADVANTAGES OF THE COUPLED SCHEME VERSUS THE STANDALONE ONE

Some of the advantages of splitting the task have been mentioned so far, mainly related with the ease of implementing some calculation in one way in comparison with the other. In order to be more explicit, the next two lines are the Fortran implementation of the hysteresis depicted in figure 3:

```
      IF (P .GT. 5.e+5       ) TRIP = .TRUE.
      IF (P .GT. 5.e+5 - 1.e5) TRIP = .FALSE.
```

With respect to the block diagram representation of figure 5, here is the RELAP code for programming it:

```
* pbresa calculation
20536300    diff     sum      1.0     0.0   1 * pbkorlag + .15 - pbresa
20536301             0.15
20536303            -1.0   cntrlvar   3640    * pbresa
20536304             1.0   cntrlvar   3624    * pbkorlag

20536310     vx      div     1.0      0.0         0
20536311           dt          0     cntrlvar  3630

20536320    xx10   stdfnctn  1.0      0.0   0 * limited_to_v_min
20536321             min
20536322           cntrlvar  9788   * vaufgl
20536323           cntrlvar  3631   * vx

20602020  cntrlvar  3630  gt   null   0   0.0 n -1.0 * diff > 0.0

20536330  tpux_gt0 tripunit  1.0      0.0         0 * tripunit diff > 0
20536331             0202

20536340    xx11     mult    1.0      0.0         0
20536341           cntrlvar  3632    cntrlvar   3633 * xx10 X tpux_gt0

20536350    xx12   stdfnctn  1.0      0.0         0  * limited_to_v_max
20536351             max
20536352           cntrlvar  9787   * vabgl
20536353           cntrlvar  3631   * vx

20602030  cntrlvar  3630  lt   null     0     0.0 n -1.0 * diff < 0.0

20536360  tpux_lt0 tripunit  1.0      0.0         0 * tripunit diff < 0
20536361             0203

20536370    xx13     mult    1.0      0.0         0
20536371           cntrlvar  3635    cntrlvar   3636 * xx12 X tpux_lt0

20536380     v       sum      1.0     0.0        1   * correction vel.
20536381             0.0
20536383             1.0   cntrlvar   3634    * xx11
20536384             1.0   cntrlvar   3637    * xx13

20536400  pbresa   integral  1.0   1.15  0  2  1.15   * pbresa
20536401           cntrlvar  3638
```

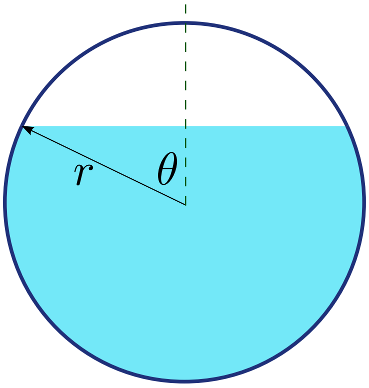

Figure 6: Section of a cylindrical tank set horizontally, partially filled with liquid. The liquid fraction of the RELAP single volume which models the tank is equal to the ratio of the colored area and the complete circle area. Knowing the tank geometry (namely the radio r), the θ angle can be obtained through an iterative process.

whereas in Fortran, the implementation is as follows:

```
      DIFF=PBKORLAG-PBRESA + 0.15
      V=VAUFGL
      IF(DIFF.LT.0.) V=-VABGL
      IF(ABS(DIFF).LE.ABS(V*DT)) PBRESA=PBRESA+DIFF
      IF(ABS(DIFF).GT.ABS(V*DT)) PBRESA=PBRESA+V*DT
      IF(PBRESA.GT.1.15) PBRESA=1.15
```

Another aspect to highlight is that with the inbuilt RELAP radiation model, only the high power hydraulic zones (namely 7 and 8) where affected by the model, whereas with this new implementation the radiation model is applied to the whole core.

Additionally, no auxiliary calculation should be performed in order to obtain the bias reactivity for each reactivity feedback term. This is automatically done in the external code, as the reactivity change is reported as $\Delta\rho = \rho_{\text{tran}} - \rho_{\text{stdy-st}}$ where $\rho_{\text{stdy-st}}$ is the reactivity computed during the steady state, and ρ_{tran} corresponds to the reactivity computed during the transient. During the steady state, ρ_{tran} is set equal to $\rho_{\text{stdy-st}}$ so when the transient begins the net reactivity is zero.

The fact of having all these calculations in an external code written in Fortran running under the wasora framework also allows the user to debug the routines in search of any problem or for a better understanding of the algorithms implemented under certain situations. This action is quite complicated to perform with RELAP, and if the user finally decides to suffer for a while and give it a chance, it could only be done if the user owns the source code.

Finally, with the external developed code it is possible to perform some iterations that need to converge before the problem time advances. Such is the case of the calculation of the collapsed level in a cylindrical horizontal tank (represented as a RELAP single volume) where only the liquid fraction is the input data. Every time step a certain liquid fraction is calculated in the interior of the tank and with a iteration process it could be known the angle named θ in figure 6. With this iteration process a more accurate collapsed water level (involved in some components interlocking logic for example) can be obtained instead of the mere product of the liquid fraction times the volume length, which is the value reported by RELAP.

7 CONCLUSIONS

A powerful tool for the calculation of the control and protection logic using Fortran subroutines and the wasora framework with the corresponding plug-in for performing coupled calculations between RELAP-CPL and the external code has been developed. This implementation presents several advantages when it comes to performing transient calculations versus the RELAP standalone mode.

First of all, not only is the new approach far easier to be developed and understood in comparison with the RELAP way, but also minimizes the typos when the same calculation has to be performed several times.

With respect to some RELAP limitations regarding restrictions in the number of control variable or trips or number of heat structures to be affected by a certain model, the external code presents a simple as well as powerful solution to deal with them. With the original implementation, only the hydraulic zones 7 and 8 were affected by the radiation model, whereas in the current implementation the model is applied to the entire core.

Furthermore, it was shown that some iterations are possible to be done with the new approach in order to obtain more accurate results in comparison with the RELAP standalone case.

For future calculations, the RELAP-CPL input file has been also adapted to implement another external program that solves spatial kinetic equations for examples. Although in the current approach every axial node of the fuel element heat structures receives the power dissipated in that node as a fraction of the total nuclear power calculated by RELAP-CPL with a point kinetic model, in a future implementation this power can be calculated by a 3D neutronic code, which could also increase the calculation accuracy.

By the time this article is being written, the tool is being tested (with some satisfactory partial test results) in order to start the calculations for the update of the Atucha I Final Safety Assessment Report.

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