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METHODOLOGY OF A PWR CONTAINMENT ANALYSIS DURING A THERMAL-HYDRAULIC ACCIDENT

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ABSTRACT

The aim of this work is to present the methodology of calculation to Angra 2 reactor containment during accidents of the type Loss of Coolant Accident (LOCA). This study will be possible to ensure the safety of the population of the surroundings upon the occurrence of accidents. One of the programs used to analyze containment of a nuclear plant is the CONTAIN. This computer code is an analysis tool used for predicting the physical conditions and distributions of radionuclides inside a containment building following the release of material from the primary system in a light-water reactor during an accident. The containment of the type PWR plant is a concrete building covered internally by metallic material and has limits of design pressure. The methodology of containment analysis must estimate the limits of pressure during a LOCA. The boundary conditions for the simulation are obtained from RELAP5 code.

1. INTRODUTION

With the occurrence of the first nuclear accidents, regulators agency in the world included in its safety analysis reports some theoretical studies of accidents considered as project base (loss of primary coolant accident), and are being studied accidents beyond project base (meltdown) , as occurred at Chernobyl and Fukushima. For this type of study, have been developed some computer programs that make the analysis of nuclear plant behavior when subjected to some kind of accident.

The analysis of transients and accidents in nuclear reactors are made with the use of some sophisticated software, i.e., computer codes. Most of these programs have a realistic philosophy (best estimate) and all were developed to simulate accidents and transients in light water cooled reactors, Pressurized Water Reactor (PWR), and associated systems. However, in order to meet the demand of works in the licensing area of nuclear facilities in the country, the tool selected by the licensing authority (CNEN - National Commission of Nuclear Energy) is RELAP5 code [1], which has a very important role in this study.

With mass adding and energy generated by RELAP5 during a LOCA , its possible study the integrity of the containment. The CONTAIN code [2] is the most widely used for analyzing the containment of a nuclear power plant, which is a concrete building internally coated by metalic material and have limits of design pressure. The simulation should be performed to ensure that all radionuclides originating from accidents in the plant are not be released to the environment.

The objective of this work is to present the methodology for analyzing the Angra 2 containment behavior with the CONTAIN code for the basic design of accidents , which are the cases of Large Breaks Loss of Coolant Accident (LBLOCA) on the pipes of primary circuit that will be simulated with RELAP5 code .

2. METHODOLOGY

The methodology that will be used in this work is presented following:

2.1 Accidents description

For this study will be considered the nuclear power plant Angra 2, wich is a PWR reactor located from 150 km away from the city of Rio de Janeiro , at the city of Angra dos Reis. This reactor was designed by Siemens / KWU generating 3,765 MW of thermal power at rated pressure of 15.8 MPa. It had its initial criticality authorized in July 2000 and currently finds itself in operation [3]. It consists of the reactor vessel, four coolant pumps, four cooling circuits, four steam generators and one pressurizer, this latter connected only to one of the cooling circuits.

It also has an Emergency Core Cooling System (ECCS) consisting of eight accumulators that inject water in the hot and cold legs. Four pumps of high-pressure inject water in the hot legs, and four pumps of low-pressure inject water to remove residual heat.

The reactor also contains the Reactor Protection System (RPS) that has a logic with all possible shutdown signals provided for each of the basic design accidents. This means that for each event postulated in the Final Safety Analysis Report (FSAR) [3] many important safety parameters are constantly faced with limits for operation. Depending on the simulated transient, the reactor does not need necessarily be turned off and can only be demanded for performance of control systems and / or limitation. The protection system can identifies the possible need for reactor shutdown. If the first shutdown signal is not effective, other signs will also indicate this same need, getting guaranteed total loss of the core power generation except of the decay of fission products.

In the event of a LBLOCA in a PWR, coolant mass and energy are released from the reactor coolant system to the containment through the break. This kind of accident is characterized by a rapid depressurization of the primary circuit to values which occurs the water injection by accumulators and, soon after, the safety injection of low pressure. The maximum value of the break area consists from the complete rupture of the coolant line (double-ended break). If the accident is not successfully mitigated by the action of safety systems can occur the core meltdown, The containment response during a LBLOCA will be simulated with the CONTAIN computer code.

The breaks that will be investigated in the loop, where is located the pressurizer, are following:

- Between the reactor coolant pump and reactor pressure vessel of (cold leg);
- Between the reactor pressure vessel and steam generator (hot leg);
- Between the steam generator and reactor coolant pump (pump section).

The LOCA is typically divided into four phases:

- Blowdown - the period from accident initiation (when the reactor is at steady state operation) to the time that the reactor coolant system reaches initial equilibrium with containment;
- Refill - the period when the lower plenum is being filled by accumulator and safety injection water;
- Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched;
- Post-Reflood - describes the period following the reflood phase.

2.2. Codes Description

In the present work, the RELAP5 and the CONTAIN 2.0 will be used to simulate the considered experiment. The first one is used for the thermal-hydraulic part, and the second one for the containment analysis. Both codes are described below.

2.2.1. RELAP5 code

The RELAP5 was developed by the Idaho National Laboratory. This code was originally designed for the analysis of thermal-hydraulic transients in PWR. The RELAP5 can model the primary and secondary cooling systems of experimental facilities and of Nuclear Reactors with geometric details. The program uses the non-homogeneous non-equilibrium two-fluid model, and considers the conservation equations of mass, momentum and energy for the liquid and gas phases. One-dimensional model is used to treat the fluid flow and the heat conduction in the structures; however, in some special cases such as the cross flow in the reactor core and the rewetting region in flooding model, the two-dimensional model is used. The RELAP5 code simulates a primary LOCA, small or large break. In addition, also simulates transient: loss of electrical power, water loss, flow loss, etc.

The hydrodynamic model is based on the control volume model for fluid. Scalar properties as pressure, energy density and void fraction are represented by average in volume control and are located in central point of this. By other side, vector properties, such as velocity, are located on the joints. The model of flow of RELAP5 uses two-phase model, not homogeneous and not balanced (one-dimensional). The heat transfer model is also based on a one-dimensional approach to calculate the temperatures and heat flows.

2.2.2. CONTAIN 2.0 code

The CONTAIN - developed by Sandia National Laboratories, under sponsorship of the US Nuclear Regulatory Commission (USNRC) - is a tool that analyzes the physical and chemical conditions and the distribution of radioactive material within the containment of a nuclear reactor in an accident. CONTAIN 2.0 is intended to replace the earlier version, CONTAIN 1.12, released in 1991.

This program provides the response of the thermo-hydraulic behavior in the containment and the release of radionuclides to the environment in case of rupture of the containment. The code considers also the aerosol behavior, behavior of fission products and the interactions between the thermo-hydraulic phenomena.

The code includes several models, to perform problems from simple to more complex, as intercell flows, condensation and evaporation in structures and aerosols, aerosol behavior, gas combustion, heat conduction into structures, decay and transport of fission products, radioactive decay heating and decontamination effects of fission products and terms-hydraulic designed. It also includes models for the phenomena of the reactor cavity, such as concrete core interactions and coolant pool boiling.

Even considering the significant uncertainty about many phenomena when dealing with accident analysis, the CONTAIN enables the user-specified input parameters. Still it has many models available that deal with serious accidents in a containment.

A postprocessor program called Aptplot will be used for extract plot files from the CONTAIN outputs. The Fig.1 shows the simulation flow diagram used in this work.

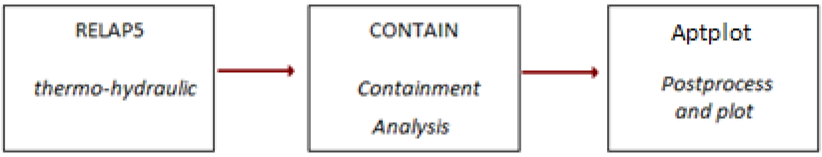

Figure 1: simulation flow diagram

2.3. CONTAIN Code Input

The structure of input to CONTAIN is quite flexible, designed with several key features for physical models and parameters. It is divided on cell level input and global input (for process common to more than once cell). The information required in global input specifies control information, material and fission products. The input also required information about:

- Containment Compartments: the geometry of the containment input model is based on the reference [3], and subdivided in compartments (Fig. 2 and 3).
- Engineered Safety System: Safety Injection Pumps (Low and High Pressure), Accumulators and Residual Heat Removal Pumps are the available safety systems of Angra 2.

- Boundary Condition: once CONTAIN code do not model phenomena in the reactor coolant system, a time-dependent source of coolant will be taken from the RELAP5 code output.
- Initial Conditions: The initial pressure inside the containment is prescribed to be 1.0 bar and 50% of relative air humidity. The initial temperature of the containment atmosphere and heat structures was 40°C, except for the rooms in the region of reactor cavity, whose initial surface temperature was 90°C.

These data are being properly collected of the FSAR -A2 [3] and of the simulation of each LBLOCA considered in this work. Later, it will be added as input data of the CONTAIN. Following are described the phases that were realized so far.

3. RESULTS

In first step of this work the student has contacted with the Contain code input. Some sensitivity studies have been realized to determine which parameters are significant, and which physic models of CONTAIN that should be applied in the simulation of Angra 2 containment.

The worst condition of break in the LOCA is the total rupture of the reactor coolant line, known as double-ended break. It is assumed that the end of the pipes is displaced so that the exit flow from each end is free. The proposed model to the Angra 2 containment is described following:

1) In case of a break in the cold leg - LBLOCA between the reactor pump and the pressure vessel, that is the worst case of a LOCA - the coolant flows from the pressurizer, through the hot leg, and the pressure vessel until the break, and then to the containment (Fig. 2).

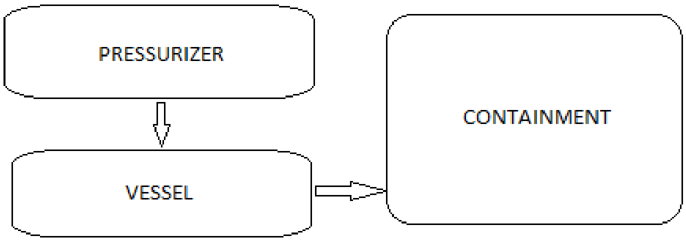

Figure 2: LBLOCA in a cold leg flow diagram

2) In case o f a break in the hot leg, the coolant flows to containment from the both sides, from the pressurizer and from the vessel (Fig. 3).

3) In case of a break in pump section, between the steam generator and the reactor coolant pump, the coolant flows to containment from the both sides, from the pressurizer and from the vessel (Fig. 3).

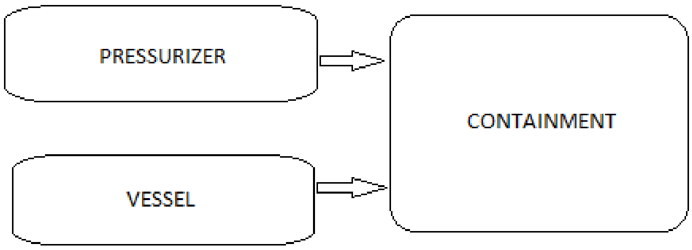

Figure 3: LBLOCA in a hot leg or pump section flow diagram

All the three cases describe above will be simulated by RELAP5 and CONTAIN codes and compared with FSAR-A2 [3], that uses the COCO code [4] for the containment analyze.

4. CONCLUSION

The objective of this work was achieved, i.e., it was presented the methodology of analysis of the Angra 2 containment during some cases of LBLOCA for this nuclear power plant. In this cases will be used the RELAP5 to obtain the boundary condition used in the CONTAIN, that will describe containment behavior.

All results this analysis will be presented in the final work of the Master degree of the Dayane Faria Silva student.

ACKNOWLEDGMENTS

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