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TRANSIENT SIMULATION OF ALWR PASSIVE SAFETY SYSTEMS USING RELAP5/MOD2

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ABSTRACT

Numerical simulation is presented of some passive safety systems currently incorporated in the design of the next generation advanced light water reactors (ALWRs). The performance and effectiveness of ex-core natural convection cooling and the concept of gravity driven water injection at high pressure are investigated using the RELAP5/MOD2 thermal-hydraulic code. The study identifies areas that should be investigated more fully in future experimental programs related to hypothetical large and small LOCA in ALWRs.

INTRODUCTION

Worldwide research and design activities are currently directed to develop a passively safe, commercially attractive and licensable nuclear power plant. Some of the concepts have reached the stage of detailed engineering design. For instance, in the Westinghouse AP600 reactor, higher safety is achieved by simplifying the traditional PWR design with respect to the number of systems and equipments, operations, inspections and maintenance requirements while employing proven light water reactor (LWR) component technology. The safety systems are predominantly passive, relying on the natural forces of gravity, convection, evaporation and natural circulation rather than on active components.[1,2] Thus, evaluation and safety analysis concentrate mainly on demonstrating the feasibility of the passive safety concepts and on predicting the reactor behavior during operation and abnormal transients. This paper presents a numerical model for accident analysis in reactors conceptually similar to the AP600 reactor. The model utilizes the IBM version of the computer code RELAP5/MOD2/CYCLE 36.[3]

The passive safety concept incorporates several components such as, high and low pressure gravity driven injection systems, natural convection heat exchangers, and automatic depressurization systems (ADS). The performance of these new safety features are investigated parametrically in this work during large and small hypothetical LOCA events. In particular, the paper concentrates on the thermal and hydraulic processes controlling the operation of the core make-up tanks (CMT), the passive residual heat removal heat exchanger (PRHR-HX) and the ADS. It is realized that an advanced pressurized water reactor, such as the AP600, is a complex and highly interactive system and that changes in operational procedures, logic or design parameters may affect the transient behavior under accident conditions. Nevertheless, it is believed that the results of this study demonstrate the basic characteristics of a design based transient in advanced pressurized water reactors and can point out safety related areas where additional theoretical and experimental verifications are needed.

THE SIMULATION MODEL

The reactor system nodalization for a large break LOCA analysis is shown in Figure 1. The physical model consists of about 140 RELAP5 "components", each containing up to 10 "volumes", which are necessary for the joint modeling of both the reactor coolant system (RCS) and the passive safety systems (PSS). Design and operational data required to construct the numerical model were based primarily on published papers and reports on AP600. Missing information items were complemented by applying engineering judgment and scaling of available information on commercial PWRs. This work complements earlier studies performed for other types of presently operating light water reactors.[4,5]. Modeling efforts have concentrated mainly on a realistic representation of the RCS and its PSS.

The model is a two-loop representation of the power plant consisting of hydrodynamic volumes (fluid control volumes), hydrodynamic junctions (momentum control volumes), and heat structures to represent heat transfer surfaces which store, transfer and generate energy such as fuel pins and steam generators' tubes. The RCS model includes the reactor vessel with the core and two parallel flow loops. Each loop consists of a U-tube steam generator with a pump (equivalent of two parallel main recirculation pumps) attached to it, a hot leg and a cold leg. Two parallel pipes are used to represent the cold leg in the broken loop. One of the loops contains a pressurizer. The model also accounts for the PSS which consists of a PRHR, an in-refueling water storage tank (IRWST), the containment, the core make-up tanks (CMT) and accumulators. A point kinetics model, including reactivity feedback effects, is implemented for core power calculations.

The secondary system of the plant model is composed of several hydrodynamic components which represent the SG boiling region (around the U-tubes), steam generator upper and lower downcomer, steam separators steam dome, feedwater preheater section and feedwater supply line. The feedwater supply rate and temperature are modeled by time-dependent volumes and junctions. The steam flow rate is defined by a pressure regulated valve. This ensure a proper stead-state in which the steam flow rate equals the feedwater rate while the SG pressure is maintained at a predetermined level.

The numerical model was validated in comparison to steady-state data related to the conceptual design of the AP600 reactor under normal operating conditions. Steady-state conditions were calculated by performing a *null-transient*, i.e., running the transient option of the RELAP5 code for steady boundary conditions, using a simplified model without neutron kinetics and with reduced thermal inertia. Steady-state was achieved normally within less then 20 sec of *null-transient* computations. Typical steady-state results are listed in Table 1 along with reported AP600 reference plant data.

The calculated total core power in Table 1 is in good agreement with the reference plant data indicating that adequate input data are used to calculate the reactivity coefficients and the SG heat transfer area. Generally good agreement is indicated for all flow and pressure parameters. This is probably because small discrepancies in the medium geometries and heat transfer coefficients are well compensated in the model by proper choice of the heat transfer areas. The predicted exit steam condition is saturation with respect to the steam dome pressure. The exit steam flow rate equals the feedwater flow within better than 1% indicating that suitable control parameters have been used in the code to define the steam valve position. The feedwater pressure and temperature are given as input tables in the model.

Table 1: Steady-State Operating Values

Parameter	Reference Plant	RELAP5
Reactor Power (Total Rated) [MWth]	1800	1802
Core Flow Rate [kg/s]	8795	8648
Core Bypass Flow [kg/s]	512	460
Core Inlet Temperature [K]	561	566
Core Exit Temperature [K]	596	597
Core Pressure [bar]	158	159
Pressurizer Pressure [bar]	157	156
Cold-Leg Temperature [K]	561	564
Feedwater Temperature [K]	508	508
Feedwater Flow Rate [kg/s]	517	514.4
Steam Flow Rate [kg/s]	517	519.2
Steam Drum Temperature [K]	552	552
Steam Drum Pressure [bar]	63	63

TRANSIENT RESULTS AND DISCUSSION

In order to investigate the performance of the passive safety features of the AP600 reactor, the numerical model described in the previous section, has been applied to study the capability of the plant and its PSS to mitigate the consequences of a spectrum of postulated LOCA events. Special consideration is directed to the performance of the CMT and the PRHR systems during hypothetical small and large break LOCA. Both transient were initiated by a sudden breach of the primary system. A small break LOCA is simulated by a break of the 0.075 m (3 in) line leading from the CMT to the reactor vessel. A large LOCA consists of a 200% break of one of the four cold leg pipes. In both events the reactor scram signal is assumes to actuate by low pressure signal. In the small LOCA event the scram occur at about 2 seconds following the transient initiation. At that time a large negative reactivity is inserted into the core simulating the insertion of the reactor's control and safety rods. After the scram, the core power during the transient is practically determined by the decay heat only as shown in fig. 2. It drops below 200 MWth after 3 sec reaching less than 100 MWth at 30 sec. The core power curve is the same in the small and large LOCA events.

Several safety systems are actuated following the

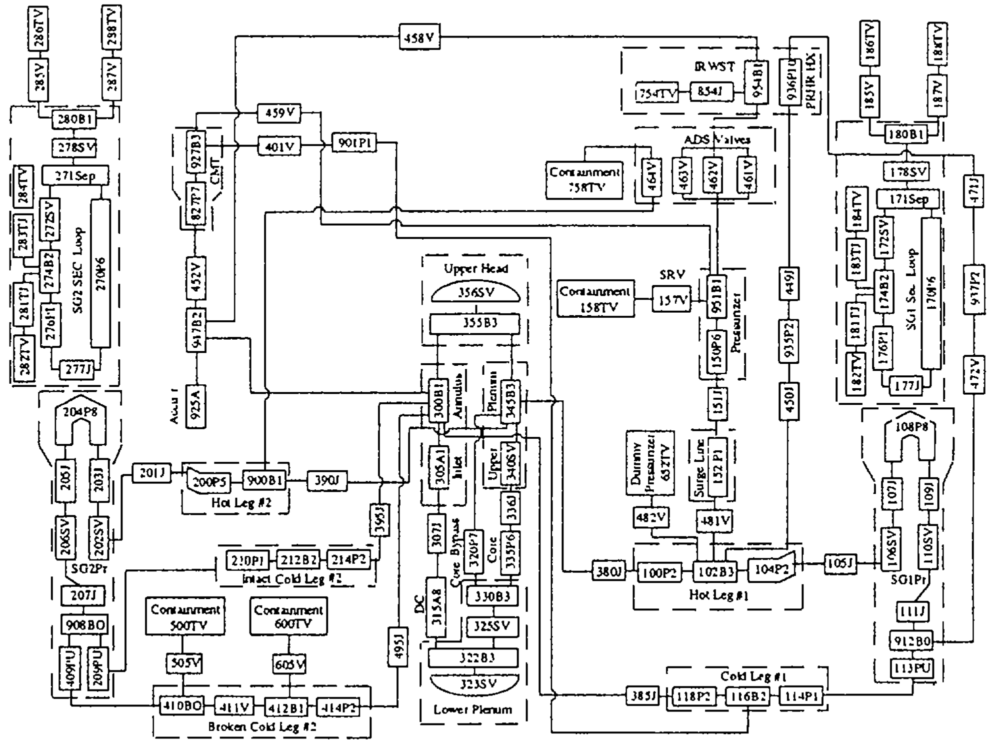

Figure 1: RELAP5 nodalization scheme for large break LOCA simulation

break. The CMT (valve 452) opens by a low pressure signal at about 144.8 bar. The CMTs are located above the reactor vessel and maintained at the reactor pressure by two pressure equalizing lines and check valves (459 and 451) connecting the top of the CMTs to the vapor space of the pressurizer and to the cold leg. They are designed to inject coolant by gravity directly into the reactor downcomer, independent of the reactor pressure. The two CMTs contain about 113 m^3 of borated water which can be injected into the core. In the small LOCA simulation it is assumed that only one CMT is available.

The ADS is design to reduce the reactor pressure in a controlled manner in order to allow for the low pressure injection systems to come into operation. The order at which the ADS is operated is of considerable importance to the performance of the system. The ADS performance was studied here parametrically by simulating several modes of operation. The results in this section were all obtained using the following operational logic which is similar to the one used in the AP600 reactor:

- The first depressurization stage (valve 461) opens when the CMT level reaches 80%.
- The second depressurization stage (valve 462) is initiated when the CMT level is below 80% and the pressure in the pressurizer reaches 75.7 bar.
- The third stage (valve 463) starts at a pressurizer pressure of 13.8 bar and a CMT level below 80%.
- The last set of two 0.2 m dia. valves in parallel (valve 464) opens to the containment atmosphere at the "low low" core make-up tank level of 20%.

Figures 3 and 4 show the leakage flow and the safety injection rates in the two accident, respectively. After a short period of subcooled choked flow, the break flow transits into saturated critical. The leakage flow is then influenced by the thermodynamic conditions in the primary system. In the small LOCA case, the leakage rate drops sharply when the ADS valves are opened at $t = 350$ sec. At that time the reactor pressure is reduced causing the CMT flow to increase and actuating the accumulators. This processes are not as distinct in the large LOCA case where the leakage rate decreases

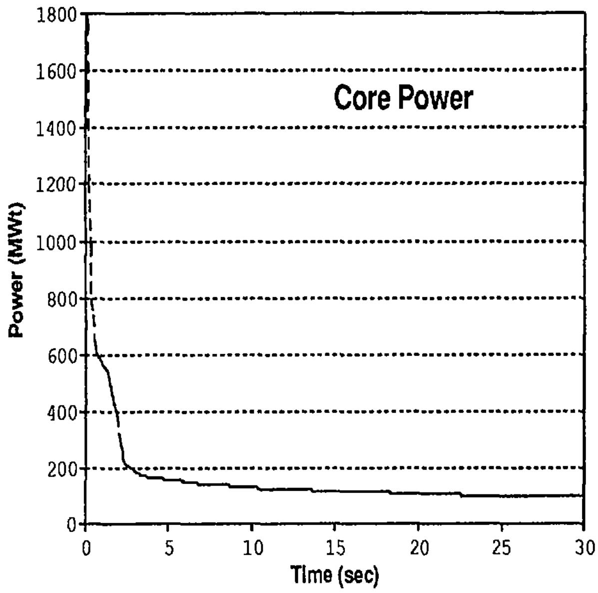

Figure 2: Core power history during large LOCA

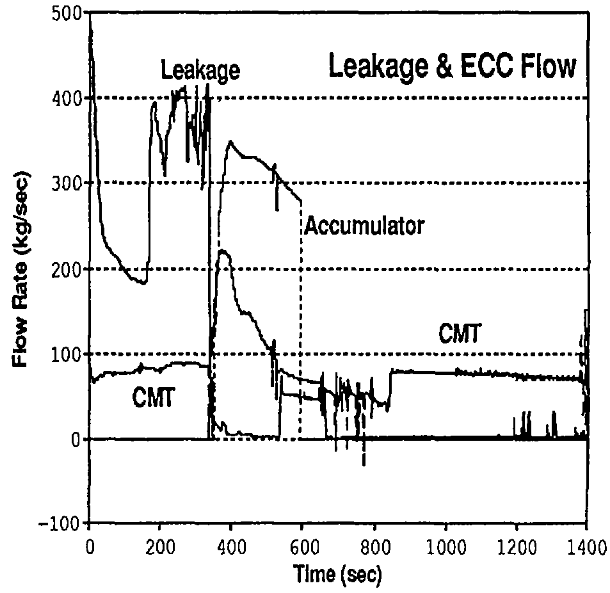

Figure 3: CMT, accumulator and leakage flows vs time during small LOCA

monotonically and the CMT flow is almost constant during the first 30 sec of the transient.

It is noted that in both small and large LOCA events, the CMT flow constitutes only a small fraction of the leakage flow. During the initial high pressure phase of the accident the CMTs injection rate is less than 100 kg/sec compared to a leakage flow through the break of about 400 kg/sec in the small LOCA event and over 10^4 kg/sec (average) during the first 10 sec of the large LOCA event. The study has also shown that the CMT flow is largely influenced by the pressure distribution in the primary system. For instance, shortly after opening the ADS valves the core pressure could exceed momentarily the pressurizer or the cold leg pressures. At that period the CMT flow was observed to stop completely due to closure of the exit check valve (valve 452). In spite the limitations of this study it is nevertheless believed that the CMT concept must be further investigated, theoretically and experimentally, to assess its utility and performance during high pressure events.

The most important data for assessing the performance of reactor safety systems and the containment response, is the total energy released into the containment during hypothetical LOCA events. In particular, such data is required as input for determining the performance of the passive containment cooling systems. Figure 5 shows the total power released through the break during large LOCA. This is basically defined by

$$E = W_b h_e \qquad (1)$$

where W_b is the break mass flow rate and h_e is the enthalpy of the leakage flow.

The leakage power through the break reaches a maximum of more than 300 MWth at about 13 sec. At that time the rate of discharge of the stored energy exceeds the total core decay power. The energy is carried into the containment by the steam and water flow. This curve may be used to set the requirements for the containment cooling system which must dissipate the heat without substantially raising the containment pressure.

As the reactor pressure decreases, the primary recirculation pumps are normally tripped-off and a PRHR flow is initiated by opening valve number 472. The PRHR is a natural circulation loop connecting the bottom of the pressurizer to the suction side of the recirculation pump. It requires no active components for its operation. The PRHR flow direction is defined by buoyancy forces and by the static pressures in the hot and cold legs (branches 102 and 912 in Fig. 1), respec-

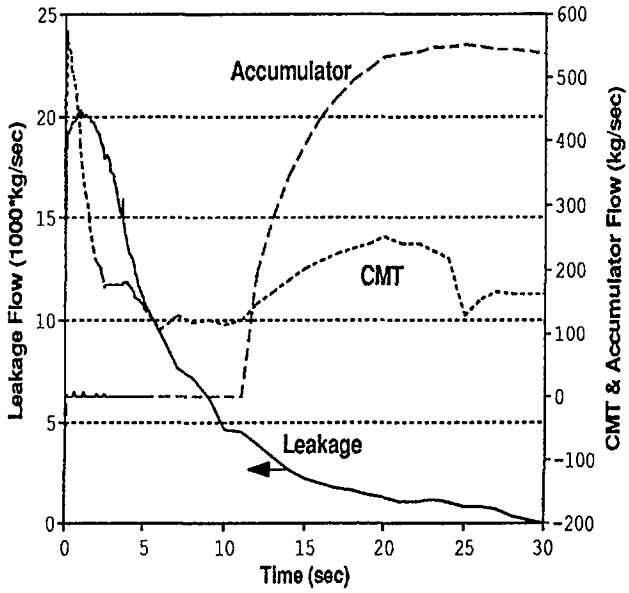

Figure 4: CMT, accumulator and leakage flows during large LOCA

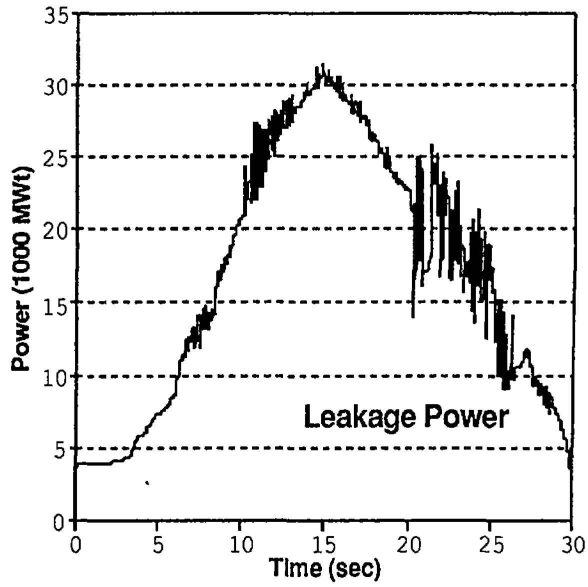

Figure 5: Total power released through the break during large LOCA

tively. Regardless of the flow direction, heat is always transferred at the PRHR-HX from the primary system to the IRWST. Figure 6 depicts the total power dissipated by the PRHR during a large LOCA. The PRHR power is defined by

$$E_{PRHR} = W_p(h_{in} - h_{out}) \tag{2}$$

where W_p is the PRHR mass flow rate and h_{in} and h_{out} are the inlet and outlet enthalpy to the PRHR heat exchanger, respectively.

It is noted that although the PRHR is designed for long term cooling, it also constitutes an effective heat removal system during the early phases of LOCA transients. In the first few seconds of the transient the rate of heat transfer by the PRHR heat exchanger reaches a maximum of about 220 MWth. The heat dissipation in the PRHR is comparable to the total decay heat throughout the transient.

CONCLUSIONS AND RECOMMENDATIONS

A numerical model is presented to simulate operational and abnormal events in the AP600 reactor. The results may serve as a basis for evaluating the plant compliance with regulatory requirements in general, and for design and sizing support of its PSS, in particular.

Two transients were discussed corresponding to small and large LOCA events. The evolution of the thermal and hydraulic processes in the two accidents is quite different. The progression of the small LOCA event can be divided into three phases. The first is a rapid depressurization of the reactor system down to the saturation pressure according to the core liquid temperature. This is followed by a core voiding phase at constant pressure until the ADS valves are opened. The opening of the ADS valves marks the beginning of the third and last phase of the accident which brings the pressure down to almost atmospheric and allows the long range cooling by the vast amount of water available in the IRWST. The transient phases are not as distinct in the case of large LOCA. However, in both cases, the rate of high pressure coolant injection from the CMTs is shown to be limited relative to the initial break flow. On the other hand, heat transfer in the PRHR provides an effective heat removal mechanism.

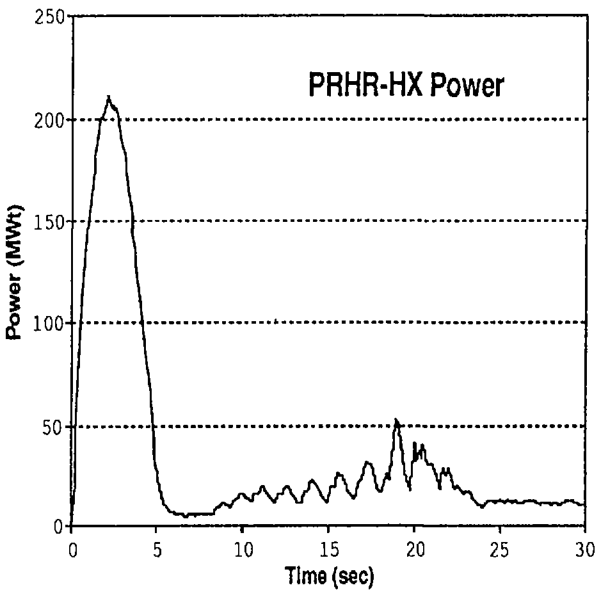

Figure 6: PRHR power dissipation during large LOCA

ACKNOWLEDGMENT

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REFERENCES

1. R. VIJUK and H. BRUSCHI, " AP600 Offers a Simpler Way to Greater Safety, Operability and Maintainability," *Nucl. Eng. International*, 22–28 (1988).

2. H. BRUSCHI and T. ANDERSEN, "Turning the Key," *Nucl. Eng. International*, 15–22 (1991)

3. V. H. RANSOM et. al., "RELAP5/MOD2 Code Manual," NUREG/CR-4312, EGG-2396, Rev. 1 (1987).

4. D. HASAN, E. ELIAS and E. WACHOLDER, "RELAP4/MOD6 Model for the AP600 RCS," *Trans. Israel Nucl. Soc.*, 16, 155 (1990).

5. D. HASAN, E. ELIAS and E. WACHOLDER, "AP600 PSS Modeling Using RELAP5/MOD1," *Trans. Israel Nucl. Soc.*, 16, 152 (1990).

6. R. M. KEMPER, C. M. VERTES, "LOCA Performance of the Advanced 600 MWe PWR," Proc. ANC Topical Meeting, Seattle, p. 750 (1988).