A RELAP5 MODEL FOR THE THERMAL-HYDRAULIC ANALYSIS OF A TYPICAL PRESSURIZED WATER REACTOR

by

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This study describes a RELAP5 computer code for thermal-hydraulic analysis of a typical pressurized water reactor. RELAP5 is used to calculate the thermal hydraulic characteristics of the reactor core and the primary loop under steady-state and hypothetical accidents conditions.
New designs of nuclear power plants are directed to increase safety by many methods like reducing the dependence on active parts (such as safety pumps, fans, and diesel generators) and replacing them with passive features (such as gravity draining of cooling water from tanks, and natural circulation of water and air). In this work, high and medium pressure injection pumps are replaced by passive injection components. Different break sizes in cold leg pipe are simulated to analyze to what degree the plant is safe (without any operator action) by using only these passive components. Also station blackout accident is simulated and the time response of operator action has been discussed.

Key words: *simulation, PWR, ECCS, passive plant, RELAP5*

Introduction

A major safety advantage of passive plants is that long-term accident mitigation is maintained without operator action or reliance on off-site or on-site AC power. New passive plants use extensively analyzed and tested passive systems to improve the safety of the plant. The passive safety systems are significantly simpler than traditional PWR safety systems and do not require the large network of safety support systems needed by typical nuclear plants. That includes AC power, heating, ventilation, air conditioning, cooling water systems, and the seismic buildings needed to house these components [1].

Passive systems use gravity, natural circulation, and compressed gas. No pumps, fans, diesels, chillers, or other rotating machines are used in the safety sub-systems. New passive

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plants designs have fewer valves, less piping, less control cable, fewer pumps and less seismic building volume than a similarly sized conventional plant [2]. Figure 1 shows a comparison between current typical PWR designs and passive designs (AP1000 as an example).

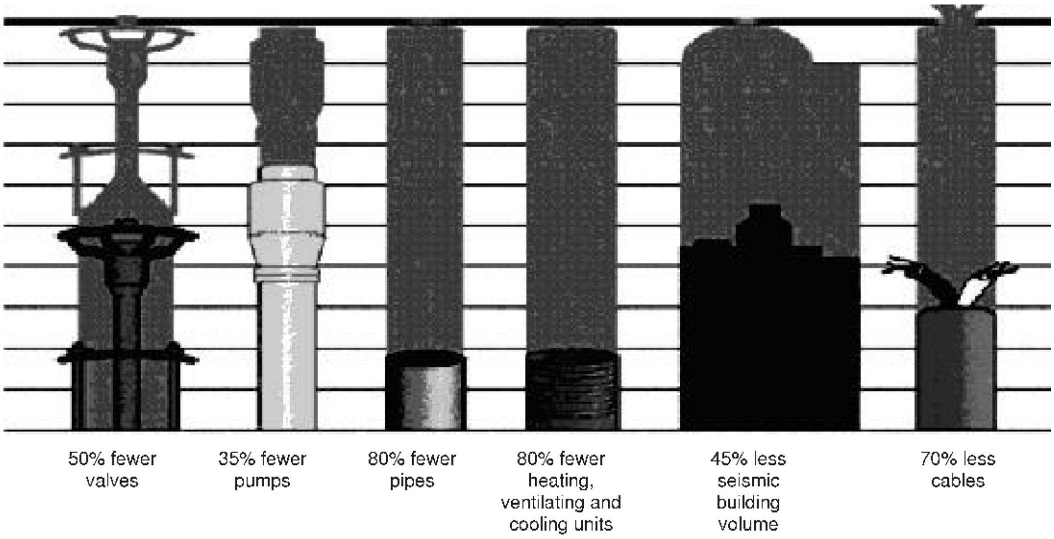

Figure 1. Comparison between active current designs and passive designs (AP1000) of PWR

The simplified construction will also reduce operator actions. The passive design means that the operators would not need to take immediate actions after an accident, with the reactor, instead, safely shutting down on its own. Also, with passive safety features and extensive plant simplifications that enhance the construction, operation, maintenance and safety.

RELAP5 Code

The RELAP5 hydrodynamic model is a one-dimensional, transient, two-fluid model for flow of a two-phase steam-water mixture. It was developed at the Idaho National Engineering Laboratory (INEL) for the U. S. Nuclear Regulatory Commission (NRC). Code uses include analyses required to support rulemaking, licensing audit calculations, evaluation of accident mitigation strategies, evaluation of operator guidelines, and experiment planning analysis. The code has been developed and used for the analysis of light water reactors (and also for CANDU analyses) with a loop design. Although the RELAP code has been extensively used in the analyses of light water reactors, and has also been used in the transient analyses of advanced Westinghouse passive plants, the introduction of a new reactor and supporting systems poses great challenges to the development of an appropriate plant representation in RELAP5 [3].

Description of the model

The reference plant chosen for the present study is a three-loop typical PWR design (modeling of two- and four-loop designs are similar). The major components of the plant are:

– pressurizer (1 for the plant),
– steam generator (SG) (1 for each loop),
– reactor pressure vessel (1 for the plant),
– reactor coolant pump (1 for each loop),
– connecting pipes, and
– passive safety injection system (2 systems).

The nominal plant operating parameters are given in tab. 1.

Table 1. Normal plant operating values

Parameter [Units]	Value
Reactor power [MW_t]	2300
Coolant presure [MPa]	15.51
SG presure [MPa]	5.5
Active core height [m]	4.1
Core flow rate [kgs^{-1}]	12,725
Inlet core temperature [K]	559
Outlet core temperature [K]	592

The nodalization of the reference PWR plant model used in this work represents the standard nodalization scheme used at INL. A schematic of the nodalization of the typical PWR system used in this work is shown in fig. 2.

Figure 3 shows the two passive safety injection systems used in this work. These two passive injection systems are:
– high pressure safety injection tank, and
– medium pressure safety injection system (accumulator).

High pressure safety injection systems are used when the normal makeup system is inadequate or is unavailable. They are filled with cold borated water and designed to function at any reactor coolant system (RCS) pressure using only gravity, and the temperature and height differences from the RCS cold leg as the motivating forces [4]. These tanks are located above the RCS loop piping. If the water level or pressure in the pressurizer reaches a set low level, the reactor, as well as the reactor coolant pumps, is tripped and the tanks discharge isolation valves open automatically. The water from the high pressure injection (HPI) tanks recirculates then flows by gravity through the reactor vessel. It is always that the primary pressure and natural circulation is established when valves are open and cold borated water enters reactor and hot primary water flows to HPI thank head. The detailed governing equations of the HPI systems are founded in references [5, 6].

Medium pressure safety injection, accumulators, are required for loss of coolant accidents (LOCAs) to meet the immediate need for higher initial makeup flows to refill the reactor vessel lower plenum and downcomer following the RCS blowdown. The accumulators are pressurized to about 5 MPa with nitrogen gas [7]. The pressure differential between the pressurized accumulators and the dropping RCS pressure ultimately forces open check valves that normally isolate the accumulators from the RCS. The accumulators continue delivery to supplement the HPI systems in maintaining water coverage of the core.

Model validation

Model validation is important to determine whether a physical model can properly describe the phenomena they are designed to simulate. Validation is based on theoretical analysis (typical behavior), experimental, or numerical assessment.

As the results issued from numerical simulations of transients and accidents in real reactors cannot be compared with experimental results, the detection of a mistake in the numerical

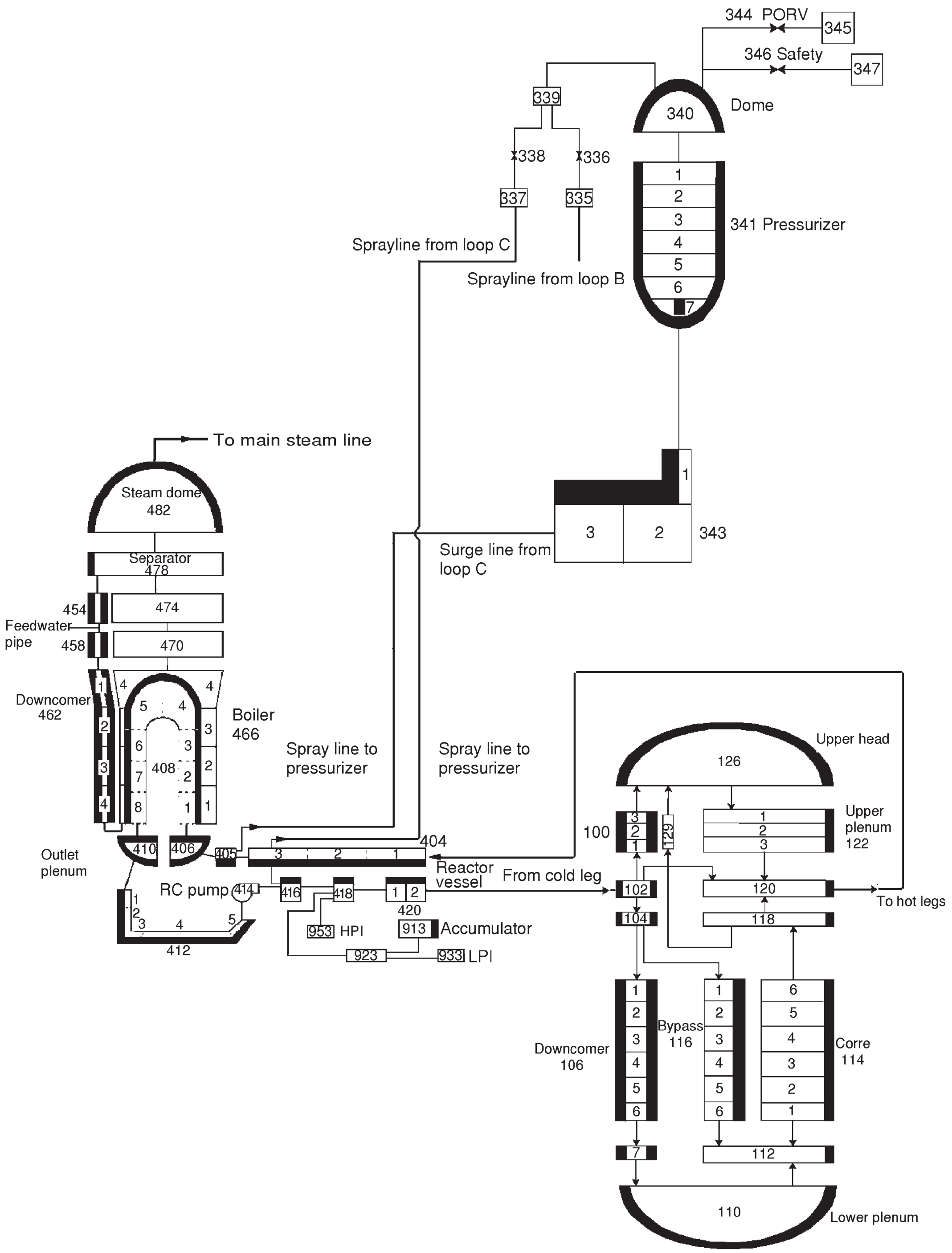

Figure 2. A schematic of the nodalization of the typical PWR system

model is impossible. Thus a preliminary work of validation of the model is compulsory [8]. The validation of the model used in this work is done by two different methods: numerical and theoretical validation methods.

Numerical validation (Convergence of the results around initial conditions)

The first validation method of the used model is obtained by performing a computation under normal operating conditions. The transients extremity remains clogged. As computations are initialized approximately in operating conditions, the convergence of the results around the initial conditions is sufficient to prove that the model is correct and fig. 4 shows the convergence of the results of the pressure of the pressurizer when we use different initial conditions.

Theoretical validation (Typical behavior of a reference accident)

Another method of validation is a comparison of accidental behavior of the current used model with the typical behavior of the accidents which is well understood as explained in refs. [9-12]. Station blackout accident (SBO) is taken as an example for validation. SBO accident is an important sequence that induced core damage. It is initiated by a loss of alternating current (AC) power and also loss of off-site power. Decay heat removal cannot be maintained for a long time because there is no AC power for the motor driven pumps, and the turbine-driven auxiliary feed water (AFW) pumps are also assumed fail to supply water.

In the current case for validation, it is assumed that the accident and the reactor trip occur at 0.0 second and the RCS trip

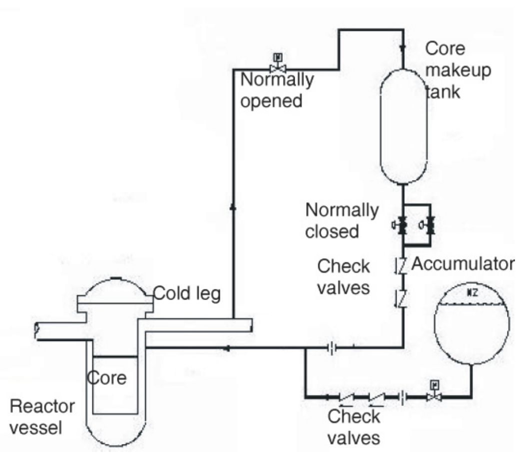

Figure 3. Two passive injection systems used in the model

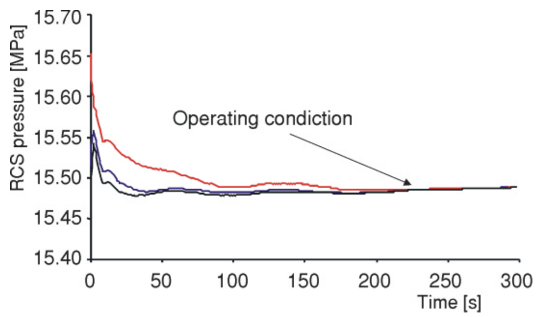

Figure 4. Numerical validation: convergence of RCS pressure around operating condition (15.5 MPa) for different initial conditions

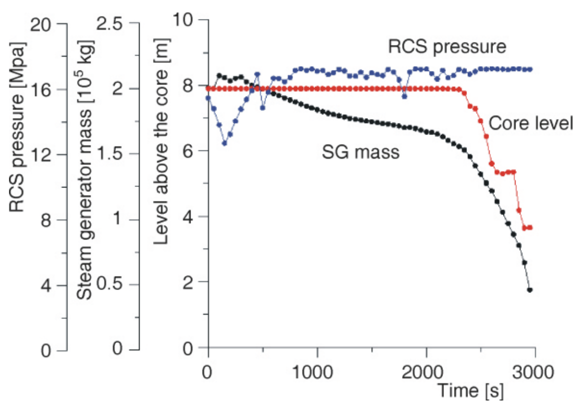

Figure 5. Theoretical validation: time responses in the case of SBO of RCS pressure, SG mass, and the level of water above the core

occurs at 150 seconds. Also it is assumed that no recovery action is taken. Following a reactor trip, the RCS pressure must drop due to a sudden decrease in heat generation from the core. When the SG dry out, the RCS volume expansion and pressurization occurs. Steam generators lose their heat removal capability because of dry out of the secondary side if power cannot be restored. Heat up of the primary coolant by decay heat pressurizes the RCS to the set point of the pressurizer power-operated relief valves (PORVs). Coolant inventory in the RCS decreases because of steam boil off through cycling of the PORVs. The core is uncovered and eventually damaged if the power cannot be recovered in time. This typical behavior is shown in fig. 5.

Results and discussion

After running the model under normal operation conditions for suitable time to reach stability and after checking of the model consistency by changing the initial conditions (fig. 4) as shown in previous section, the transients and accidents simulation may be now carried out.

Two accident scenarios are taken as base cases in this work. These two scenarios are:
- small break loss of coolant accident (SBLOCA), and
- station blackout (SBO).

SBLOCA is modeled in RELAP5 by simulating the rupture by using an imaginary valve which modeled between the place of the break and a sink volume. SBO is modeled by stopping RCS pumps and closing feedwater valves.

The base case of SBLOCA sequence assumes that the small breaks with different diameters (starting from 5 inches break) occur in the cold leg of the plant. The recovery action of the two used passive emergency core cooling system is examined in all the cases. The main attention is "Can we say the plant is safe by using these two passive emergency core cooling systems (ECCS) in the case of SBLOCA for 1500 seconds after the accident?" and if the answer is yes, "To what extent we can say that?"

The base case of SBO sequence assumes that the SBO occurs at time 0.0 second and different operator actions are modeled to mitigate this accident. The main attention is "Can we decrease the dependence on the operator action by using automatic depressurization system in the case of SBO for 4000 seconds after the accident?"

Tables 2 and 3 show the analytical results of both cases SBLOCA and SBO scenarios, respectively.

The cases which are modeled are the following:

SBLOCA:
- base case (without action) for 5, 6, 7, and 8 inches cold leg rupture,
- 5 inches clod leg rupture with passive ECCS,
- 6 inches cold leg rupture with passive ECCS,
- 7 inches cold leg rupture with passive ECCS, and
- 8 inches cold leg rupture with passive ECCS.

SBO:
- base case (without action),
- 10 min. operator action,
- 30 min. operator action,and
- 45 min. operator action.

Table 2. SBLOCA analytical results in cold leg and mitigation measure (sequences per second) for different break sizes

Progression	5 inch break		6 inch break		7 inch break		8 inch break	
	Base case	Mitigation measure	Base case	Mitigation measure	Base case	Mitigation measure	Base case	Mitigation measure
Simulation starts	0	0	0	0	0	0	0	0
Accident begins	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Reactor trip	15	15	15	15	15	15	15	15
Core begins to uncover	180	220	95	95	67	75	26	28
HPI starts	–	38	–	25	–	20	–	17
Accumulator starts	–	620	–	300	–	200	–	100
HPI empty	–	1500	–	950	–	350	–	200
Accumulator empty	–	–	–	1350	–	600	–	350
Core completely uncover	1200	–	600	–	500	–	100	1400
Maximum clad temperature >1472 K	1200	–	720	–	630	–	400	1500
Simulation ends	1500	1500	1500	1500	1500	1500	1500	1500
Consequence of accident and mitigation measure	**Core melt**	**Core melt is prevented**	**Core melt**	**Core melt is prevented**	**Core melt**	**Core melt is prevented**	**Core melt**	**Core melt is prevented**

Figure 6 through fig. 9 shows (for all the cases) the time response of the following important parameters: normalized level of both passive ECCS used, level of water above the bottom of the core, and maximum clad temperature (which must not exceeds 1472 K [13]).

Conclusion and future work

We have presented models that increase the effectiveness and the ability to predict the response of nuclear power plants in different cases: normal operation, operational transients, and hypothetical accidents.

We have modeled a new PWR power plant with two passive ECCS: HPI passive tanks and MPI accumulators. Figure 7 clearly shows that the modeled PWR is safe in the case of SBLOCA until size of 7 inches break 1500 seconds after the accident by using only two short term passive ECCS without any operator action. The core melts after 1500 seconds if the break size is equal to 8 inches as shown in fig. 8(b).

Passive safety designs have succeeded in minimizing the dependence on the operator actions as shown in the modeling of SBO scenarios with different time response of the operator actions. If an automatic depressurizing concept is used with passive ECCS injection, the operator duty will be eliminated totally.

Table 3. Analytical results for 3 inch SBO accident and mitigation measure in sequences per second

Progresion	Base case	10 min. operator response	30 min. opeator response	45 min. operator response
Simulation starts	0	0	0	0
Accident begins	1	1	1	1
Reactor trip	15	15	15	15
Pressurizer full	1800	1600	1800	1800
Core begins to uncovered	2820	–	2275	2820
SG empty	3000	–	–	3000
Pressurizer PORVs opened manually	–	600	1800	2400
HPI system initiation	–	700	1900	2500
AFW initiation	–	–	–	–.
Core completely uncovered	4000	–	–	–
Core reflooded	–	No need	3550	–
Maximum clad temperature >1472 K	3760	–	–	Maximum of 1200 K at 3700
Simulation ends	4000	4000	4000	4000
Consequence of accident and effectiveness of mitigation measure	**Core melt**	**Core melt is prevented**	**Core melt is prevented**	**Core melt is prevented but dangers case**

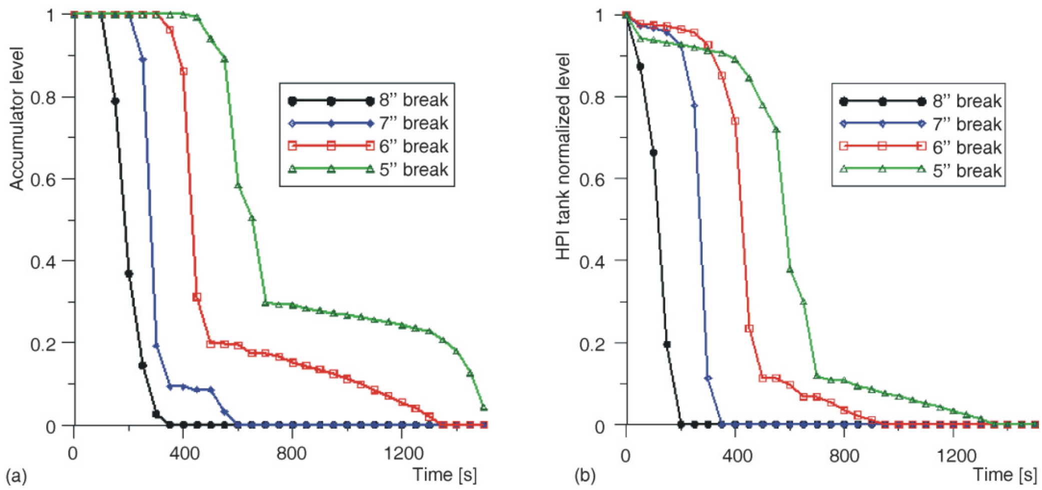

Figure 6. Normalized level of the two passive ECCS used:
(a) accumulators, (b) high pressure injection tank,
1500 seconds after the accident for different cold leg pipe break sizes

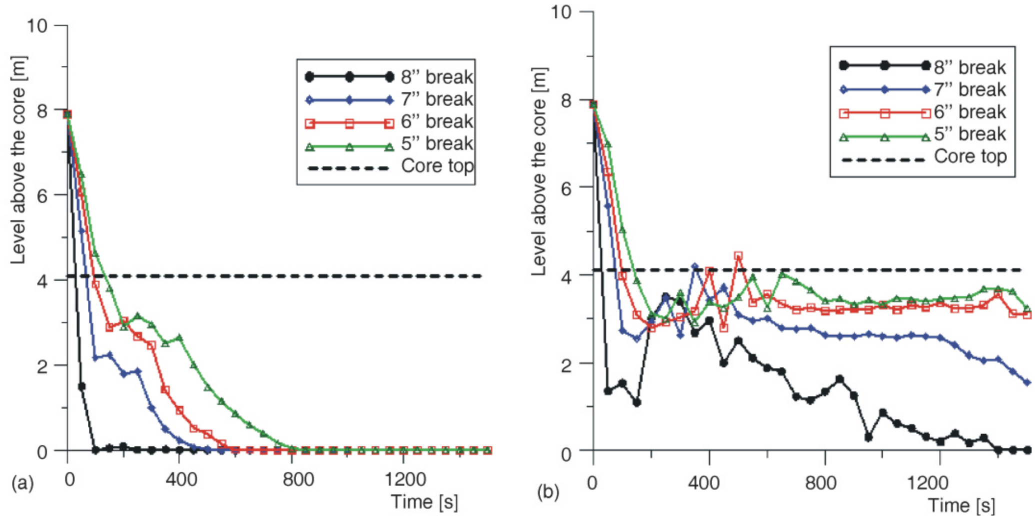

Figure 7. Time response (1500 seconds after the SBLOCA occurs) for different cold leg pipe break sizes
(a) level above the core (without action), (b) level above the core (with passive ECCS)

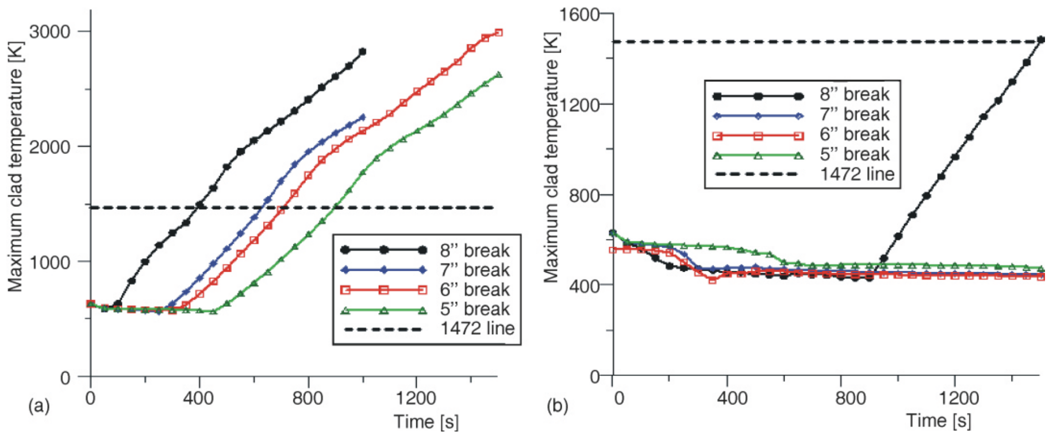

Figure 8. Time response (1500 seconds after the SBLOCA occurs) for different cold leg pipe break sizes
(a) maximum clad temperature (without action), (b) maximum clad temperature (with passive ECCS)

Using more passive safety systems enhance "safety margins" of nuclear power plants because the very quick response of the passive systems which do not depend on operator action. That is because the very important conclusion that the new designs succeeded in minimizing the dependency on the operator action in some kinds of accidents as SBLOCA and SBO.

Future work can focus on: modeling of more different accident scenarios to implement passive safety features, modeling complete passive power plant, studying safety of passive safety plants for longer times, and studying to know which place in the RCS cycle is more suitable to be the driving force to the HPI system.

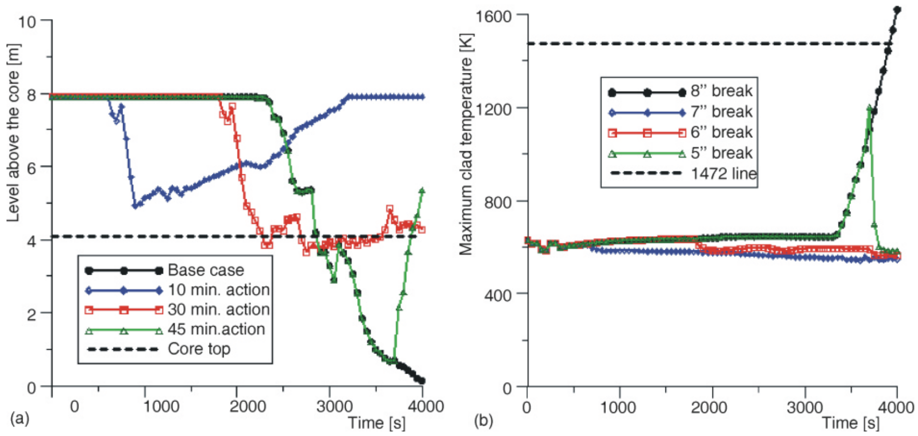

Figure 9. Mitigated case of SBO accident with different operator time response actions
(a) Level above the core, (b) maximum clad temperature

The future work must be concentrated on the analysis of the other different types of accidents to see to which degree the using of passive safety systems increase the safety margins and also modeling other passive safety systems to cover larger time scales after the accidents.

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