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Analysis for Break- Size Effects on SBLOCA Scenario in a 4-Loop PWR Using RELAP5/MOD 3.3

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---***ABSTRACT***---
Small-Break loss of coolant issue was highlighted in 1979, after the Three Mile Island accident. The reactor system response to a small break is characterized by a small rate of coolant discharges and slow pressure variations with time. The depressurization may be slow enough to delay the accumulator's intervention for some time. The accident scenarios may change drastically due to many factors; the break size is one of these factors.
In this study, RELAP5/MOD 3.3 thermal hydraulic computer code is used to simulate the effect of break size on the consequences of SBLOCA in a 4-loop PWR Westinghouse design. Plant nodalization consisting of two loops, the first one represents the broken loop and the second one represents the other three intact loops, is considered. All the plant main components in addition to the emergency core cooling system (ECCS) trains are modeled. To investigate the worst break size in the cold leg, a spectrum of five different break sizes with diameters 1 inch, 2 inch, 4 inch, 6 inch and 8 inch are considered.
Results show that for break sizes 1 and 2 inches, the charging system and the high pressure safety injection overcome and limit the consequences. The worst consequences occur at break size 6-inch where most of the core uncovered for a period of time accompanied with a sharp increase in fuel rod cladding temperature. A maximum cladding temperature of approximately 1037 ^{o}F occurs before the accumulator's intervention.
KEYWORDS: *Small-break loss-of-coolant accident breaks size Thermal hydraulic phenomena Safety injection 4-loop PWR Core uncovery.*

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I. INTRODUCTION

The purpose of accident analysis is to demonstrate compliance of plant performance against applicable regulatory requirements and acceptance criteria, thus assuring nuclear safety under postulated initiating events. One of the postulated initiating events is the small break loss of coolant accident (SBLOCA). The attention of reactor safety research was shifted to SBLOCA behavior after the March 1979 accident at the Three Mile Island Unit 2 reactor because its consequences could be sufficiently severed to warrant safety concerns. Many test facilities were constructed and many research projects were prepared for detail investigation of thermal hydraulic related phenomena involved in SBLOCA [1]. Also, counterpart experimental tests have been performed on PWR test facilities available in European community such as LOBI, SPES, BETHSY, and LSTF, on small break LOCA in PWR to understand the involved phenomena, the key parameters affecting it's scenario, and to collect experimental data to be used in the validation process of computer codes such as RELAP5 thermal hydraulic system codes [2]. In addition, many other researchers and organizations all over the world handled the effects of SBLOCAs on the reactor safety and the capabilities of the applicable computer codes in predicting their related phenomena [3-15].

The main characteristic of the SBLOCA is the slow depressurization of the primary loop, therefore the late intervention of the emergency core cooling system (ECCS). Previous studies show that, SBLOCA scenarios are depend on many factors such as reactor design, break location, Safety injection set points, break size, boron concentration or non-condensable gases concentration, core coolant bypasses, and the reactor operator actions [1]. Also, SBLOCA is characterized by five periods: blow-down, natural circulation, loop seal clearance, boil-off, and core recovery, while the duration of each period is break-size-dependent [16]

To confirm the previous results regarding the main characteristics of SBLOCA and to determine the worst SBLOCA size in a 4-loop PWR the present work is proposed. The tool used in the analysis is the RELAP5/MOD3.3 Thermal Hydraulic (TH) system code. A spectrum of SBLOCA sizes in the cold leg of a loop not containing the pressurizer is considered. Five break sizes in the range which may affect the intervention of the accumulators are considered. These break sizes are 1 inch, 2 inch, 4 inch, 6 inch and 8 inch in diameter. All the postulated accidents occur while the plant operates at nominal power and all the engineering safety measures are available without consideration of single failure. Complete Nodalization for the main plant components and emergency core cooling was done. The nodalization consists of two loops; broken loop represents one of the plant loops without pressurizer and intact loop represents the three other plant loops.

RELAP5/MOD3.3

RELAP5/MOD3.3 TH system code has been developed for best-estimate transient simulation of light water reactor coolant systems during postulated accidents. The code models the coupled behavior of the reactor coolant system and the core for large and small loss-of-coolant accidents and operational transients such as anticipated transient without scram, loss of offsite power, loss of feed water, and loss of flow. A generic modeling approach is used that permits simulating a variety of thermal hydraulic systems. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers, and secondary feed water systems. RELAP5 system code passes a rigorous process of validation and accuracy quantification through comparison with experimental date and benchmarking with other TH codes [2, 9, and 13]

II. REFERENCE PLANT

The plant considered is a Westinghouse 4-loop PWR Nuclear Power Plant (NPP) with thermal power 3411 MW_{th}. The reactor core consists of 193 fuel assemblies. Each fuel assembly is arranged in a 17x17 arrays and includes 264 fuel rods. Each loop consists of a hot leg, U-tube steam generator, intermediate leg, reactor cooling pump, and cold leg. A Pressurizer connected to the hot leg of one of cooling loops. An emergency core cooling system connected to the cold leg of the four loops and consists of four accumulators, four branches from the charging system, and four branches from the safety injection system. Some reference plant related data is shown in Table 1.

Table 1 Reference plant data [17]

Parameter	Value
Primary flow rate/Loop (Ib/s)	10154
Primary pressure (psia)	2250
Core inlet/outlet temperature (^oF)	560/620
SG main feed water temperature (^oF)	440.5
SG auxiliary feed water temperature (^oF)	100
Charging system coolant temperature (^oF)	100
Safety injection system coolant temperature (^oF)	100
Accumulators coolant temperature (^oF)	120
Auxiliary feed water design flow rate /loop (Ib/s)	62
Charging system maximum flow rate/loop (Ib/s)	29

PLANT NODALIZATION

The plant nodalization is shown in Figure 1. The nodalization consists of two loops; broken loop simulates one of the plant loops other than that containing the pressurizer and intact loop simulates the other plant loops. The nodalization simulates all the main components of the reactor, such as the reactor vessel internals, main coolant pumps, steam generators, pressurizer, feed water systems…etc. For each loop, the ECCs is simulated as two time dependent junctions (represent the charging system and the safety injection system) and accumulator. The ECCs capacity for the intact loop is three folds that of the broken loop. The charging system injects water at primary pressures less than the nominal pressure based on a low pressurizer water level signal. The safety Injection system serves in the pressure range from 1500 psia and up to the atmospheric pressure. The accumulators cover the pressure range less than 600 psia. The core is simulated as one average channel divided to six axial volumes and connected to the lower and upper plenums. Table 2 presents the main components and their equivalent code number in the nodalization.

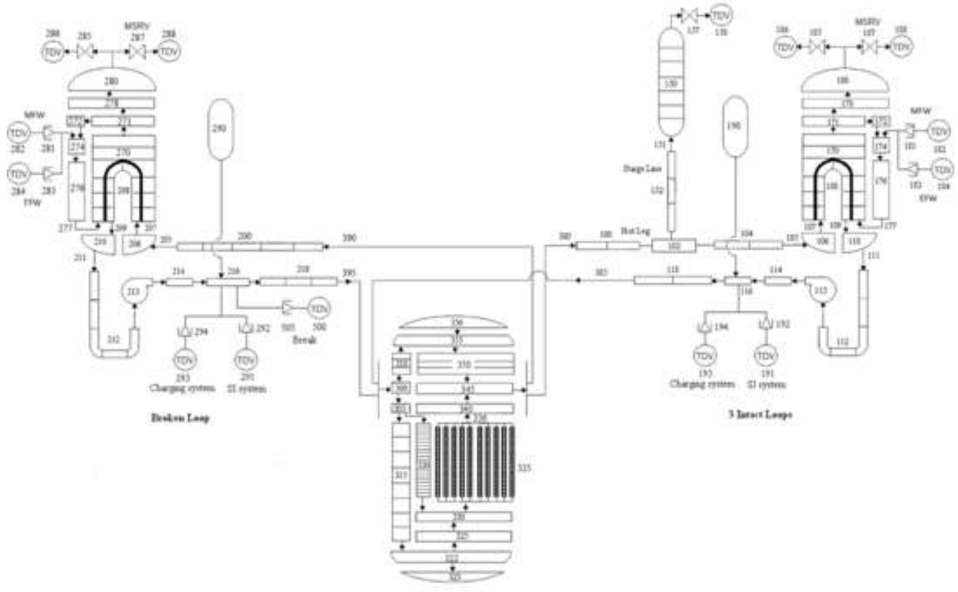

Figure 1 NPP Nodalization

Table 2 Main Plant Components and the Corresponding Nodalization Numbers

Component	**Equivalent Code**
Hot Leg	100, 200
Cold Leg	116,118, 216, 218
Steam Generator Primary Side	108, 208
Steam Generator Secondary Side	170-180, 270-280
Reactor Primary Pumps	113, 213
Pressurizer/ Accumulators	150 / 190, 290
Main Feed Water System (Main/Auxiliary)	182, 282 / 184, 284
Safety Injection System	191-192, 291-292
Charging System	193-194, 293-294
Reactor Core channel /heat structure	335/336
Break Valve	505

ACCIDENT DESCRIPTION AND ASSUMPTIONS

The transient analyzed is a Small Break Loss of Coolant Accident (SBLOCA) in the cold leg of one of the loops other than that contains the pressurizer. The break occurs where the reactor operates at full power. All the ECCS trains and the auxiliary feed water pumps are available. The simulation of accident was performed by incorporating the operational logic of the reactor protection system. The imposed events involved in this transient with their set points are outlined in Table 3. Due to a lack of data, the set point for stop/start of the charging system is assumed at ± 10 % of the pressurizer level. All the passive and active safety injection components of the ECCS are connected to the cold legs. In the SBLOCA category, break sizes of 1, 2, 4, 6, 8 inches in diameter are considered. The break is simulated by a horizontally oriented valve component connected to the cold leg side.

Table 3 Safety system actuation set-points during the accident

System/Function	**Time/Set point**
Steady-state operation at normal conditions	0 – 100 s
Break initiation	at 100 s
Reactor protection system trip signal	Pressurizer pressure 1860 psi (12.82 MPa)
Reactor coolant pump stop/Main feed water stop	Reactor trip signal
Main steam valve closure	Reactor trip signal
Charging system (very high safety injection) start/stop	Reactor trip time + low pressurizer water level Signal (± 10 % of nominal level)
Auxiliary feed water system in the intact or broken loops start/stop	14 sec. delay after reactor trip + high/low setting of void fraction at SGs volume 172 and 272 (0.39578/0.30838)
High Safety injection (HPSI) start	Pressurizer pressure 1500 psi (10.34 MPa)
Accumulator injection start	Pressurizer pressure 600 psi (4.14 MPa)
End of transient	2000 sec

III. RESULT ANALYSIS AND DISCUSION

The first step before the transient simulation is the nodalization qualification through comparing the RELAP5 results from a steady state run with the plant nominal parameters. This step was performed in a previous paper for the authors and good results were obtained [14]. For the transient simulation, the main thermal hydraulic parameters which demonstrate the plant behavior under SBLOCA such as pressure, clad temperature, core void fraction, core collapsed water level, break flow, and total safety injection flow for each break size are discussed in the following paragraphs. These parameters for each break size are shown in Figures 2-5. Also the timing of the key events is illustrated in Table 4.

Table 4 Timing of the main events for the different break sizes

Event Phenomena	Time (s)				
	1 inch break	2 inch	4 inch	6 inch	8 inch
Break initiation	100.0	100.0	100.0	100.0 s	100.0 s
Reactor trip	319.8	156	114	107	105
Charging system intervention	319.8	156	114	107	105
Safety Injection intervention	715.8	192	126	115	112
Accumulators intervention	-------	-----	928	320.6	306
Emergency/Auxiliary feed water actuation (on/off)	336/1844	170/1722	128/1723	121/1786	119/1820

As shown in the Figures for the different break sizes, the plant behavior during the SBLOCA passes through different stages. The first stage starts immediately after the initiation of break and extended for a short period. This stage is characterized by sharp decrease in primary pressure, step increase in the secondary pressure, large break flow rate, and no formation of voids where the primary coolant is sub-cooled liquid. The second stage is characterized by primary loop void formation, the primary pressure slowly decreased, the primary pumps stopped and the natural convection in the primary loop started. During this stage the break and the steam generators are required to dissipate the stored energy in the primary side, therefore the primary pressure remains higher than the secondary pressure. The Period of this stage is controlled by the break size; increasing the break size shortens this stage. The third stage characterized by highly voiding of the core and the primary loops including the SG tubes and the loop's seal part, the break discharge becomes mainly steam, the primary pressure decreases below the secondary pressure, and the SG becomes inefficient heat sink for the primary side.

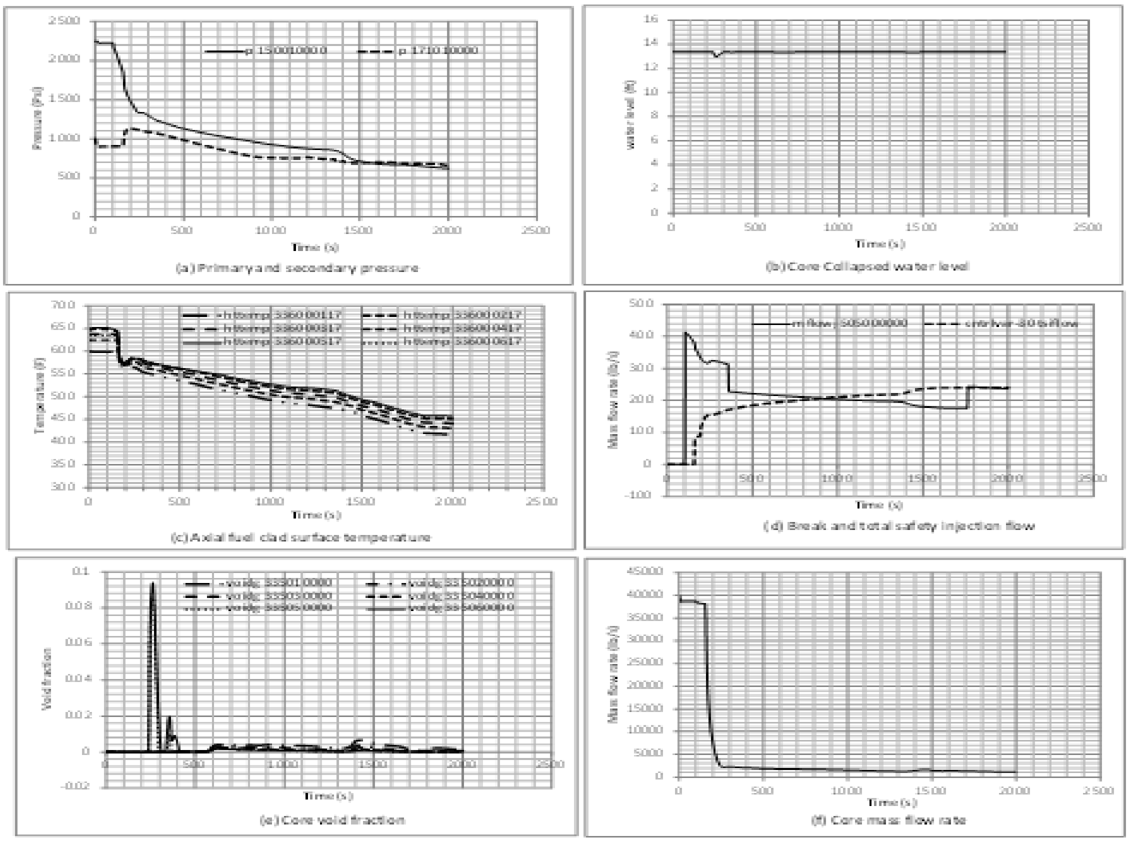

Figure 2 Key parameters during 2 inch SBLOCA

Depending on the break size, the core collapsed water level reaches its minimum and the upper part of the core may be uncovered. The pressure drop is faster than in stage two and therefore the accumulator's intervention occurs in this stage. Stage four characterized by nearly stabilization in pressure, the break flow may be totally vapor, liquid, or intermittent. The specific results for each break size are discussed in the following paragraphs.

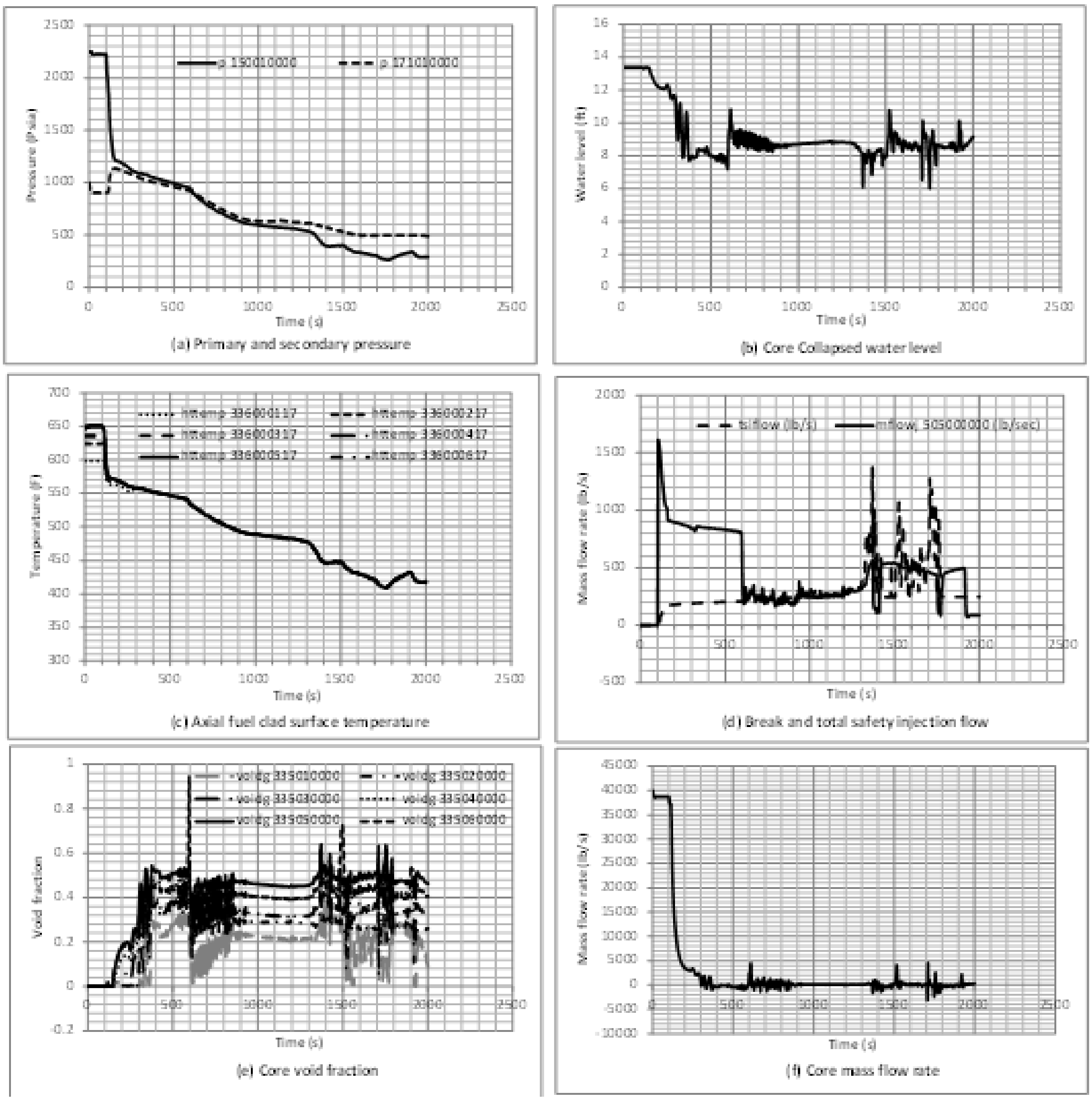

Figure 3 Key parameters during 4 inch SBLOCA

As shown in Figure 2 for 2 inch SBLOCA, the core totally covered with water with the formation of very small fraction of voids. Except the latest time period of the transient; the primary pressure remains higher than the secondary pressure. Therefore the effectiveness of the SGs as a heat sink and the establishment natural circulation in the primary loop are continued. The primary pressure remains higher than the accumulator's set point where the charging system and the safety injection in its high range are sufficient to overcome the consequences of the accident.

In 4 inch SBLOCA, shown in Figure 3, the core totally covered with a two phase coolant with average void fraction nearly 0.4. After nearly 600s, the primary loop seals cleared and the steam confined in the primary loops discharges through the break. Therefore the primary pressure decreases gradually and becomes lower than the secondary pressure. The accumulators start a weak intervention at nearly 928 s and efficient intervention after 1600 s where a vast drop in the primary pressure occurs. After 1600 s, a huge amount of cooled water injected in the primary loops which temporary reduces the average void fraction and a repeatable loop seal formation and clearance occurs. During the transient, the core does not overheat and the clade temperature remains below the nominal operating temperature.

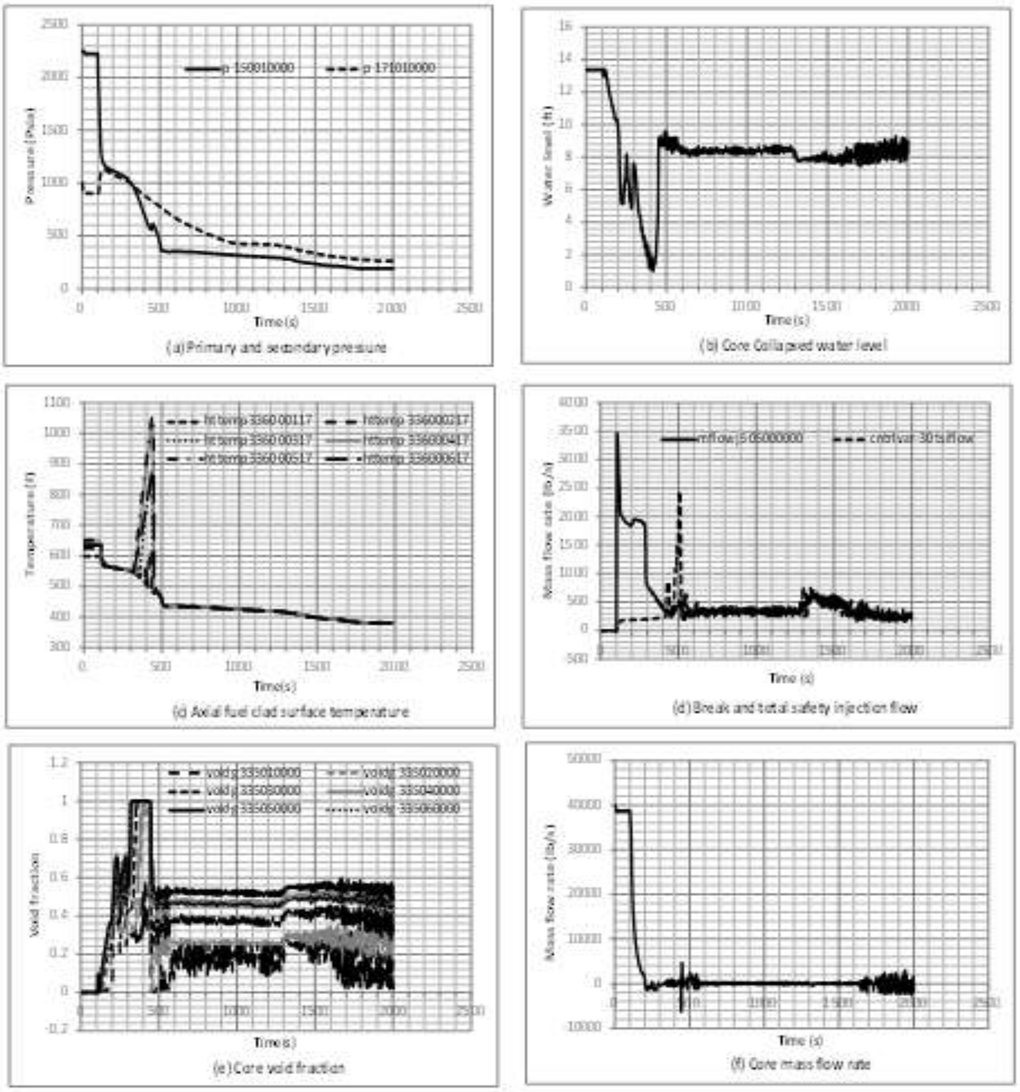

Figure 4 Key parameters during 6 inch SBLOCA

The key parameters during the 6 inch SBLOCA are shown in Figure 4. Due to the break large discharged flow, the primary pressure and core collapsed water level decreases rapidly. An early loop seal clearing occurs which increases the drop rate in pressure. The core voided extensively and its upper parts are uncovered for a period of time sufficient for heating up the core fuel elements before the intervention of accumulators. A Peak Cladding Temperature (PCT) of 1037 ^{o}F occurs at nearly 430 sec. At nearly 424 s the accumulators start injection of cooled water, where core void fraction and the clad temperature decrease.

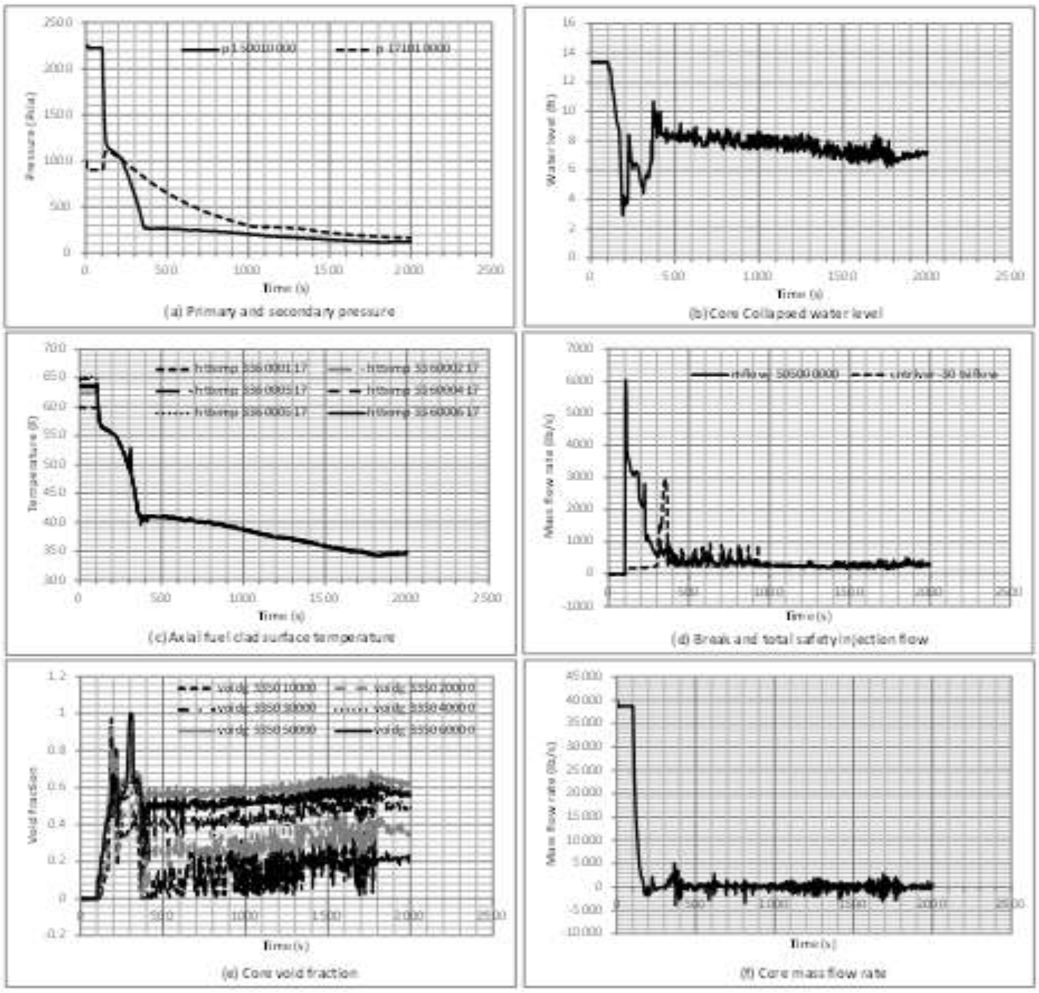

Figure 5 Key parameters during 8 inch SBLOCA

During 8 inch SBLOCA, shown in Figure 5, the decrease in the primary pressure occurs rapidly enough for early intervention of accumulators. The core uncovered and heating up occurs for a very short period of time. The PCT is 527 ^{o}F at nearly 312 s. The starting time of accumulator's intervention can early limiting the consequences of the accident.

IV. CONCLUSIONS

RELAP5/MOD 3.3 is used to simulate a Small Break Loss of Coolant Accident (SBLOCA) in the cold leg of 4-loop PWR NPPs. A spectrum of five different break sizes with breaks of 1 inch, 2 inch, 4 inch, 6 inch and 8 inch in diameter were selected to determine the main characteristics and the worst break's size within the considered rang. The results showed that the charging system and the safety injection system (high range) are sufficient to overcome the consequences during break sizes from 1- 2 inch where there is no violent vapor generation, no core uncovering, and no overheating. A repeatable loop seal clearing occurs during the later phase of 4 inch SBLOCA size. The worst consequences occur at 6 inch break size due to a late in the accumulator's intervention and PCT of 1037 oF occurs at nearly 430 s (330 s from the start of transient). During 8 inch break size, the core uncovering and heating up occurs for a very short period of time and a PCT of 527 oF is attained. Therefore, even at 6 inch break size; there is no violation for the acceptance criteria of the ECCS. These results are best estimate without consideration for uncertainties or conservative assumptions.

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