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Main Steam Line Break Analysis for Lungmen ABWR

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ABSTRACT

The object of this paper is to develop methodologies for analyzing the behaviors of fuel rod, vessel, and containment during main steamline break (MSLB) transient. The broken area of the RPV side was assumed to be $0.0984\mathrm{m}^2$ (flow limiter). And the broken area of the main steam header side was assumed to $0.319\mathrm{m}^2$ (main steam line area). According to FSAR, for conservative assumption, MSIVs started to close at 0.5sec and fully closed at 5.0sec after the transient started. The results of TRACE/PARCS coupling calculation were compared with those of both FSAR and GOTHIC data, indicating that the TRACE/PARCS coupling model has the ability to predict the MSLB transient, and both RPV integrity and containment integrity criteria are met. After that, the output data from TRACE/PARCS calculation was put into FRAPTRAN code as boundary conditions to analyze the thermo-mechanical behavior and calculate the stress, strain, oxide thickness, etc. The values of these factors were compared with the criteria. And the final results show that the fuel rod integrity criteria are met.

FOREWORD

The US NRC is developing an advanced thermal hydraulic code named TRACE for nuclear power plant safety analysis. The development of TRACE is based on TRAC, integrating RELAP5 and other programs. NRC has determined that in the future, TRACE will be the main code used in thermal hydraulic safety analysis, and no further development of other thermal hydraulic codes such as RELAP5 and TRAC will be continued. A graphic user interface program, SNAP which processes inputs and outputs for TRACE is also under development. One of the features of TRACE is its capacity to model the reactor vessel with 3-D geometry. It can support a more accurate and detailed safety analysis of nuclear power plants. TRACE has a greater simulation capability than the other old codes, especially for events like LOCA.

Taiwan and the United States have signed an agreement on CAMP (Code Applications and Maintenance Program) which includes the development and maintenance of TRACE. To meet this responsibility, the TRACE/PARCS/FRAPTRAN model of Lungmen NPP has been built. In this report, the TRACE/PARCS/FRAPTRAN model of Lungmen NPP was used to evaluate the Lungmen main steamline break transient.

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EXECUTIVE SUMMARY

An agreement in 2004 which includes the development and maintenance of TRACE has been signed between Taiwan and USA on CAMP. NTHU is the organization in Taiwan responsible for applying TRACE to thermal hydraulic safety analysis in order to provide users' experiences and development suggestions. To fulfill this responsibility, the TRACE/PARCS model of Lungmen NPP is developed.

According to the user manual, TRACE is the product of a long term effort to combine the capabilities of the NRC's four main systems codes (TRAC-P, TRAC-B, RELAP5 and RAMONA) into one modernized computational tool. The 3-D geometry model of reactor vessel, which is one of the representative features of TRACE, can support a more accurate and detailed safety analysis of NPPs. On the whole TRACE provides greater simulation capability than the previous codes, especially for events like LOCA.

PARCS is a multi-dimensional reactor core simulator which involves a 3-D calculation model for the realistic representation of the physical reactor while 1-D modeling features are also available. PARCS is capable of coupling the thermal-hydraulics system codes such as TRACE directly, which provide the temperature and flow field data for PARCS during the calculations.

Lungmen NPP is the fourth NPP in Taiwan. It has two identical units of ABWRs with 3,926 MWt rated thermal power each, consisted of 872 GE14 assemblies with 205 control rods. The steam flow is 7.64×106Kg/h at rated power condition. The designed rated core flow is 52.2×106 Kg/h. Compared with BWRs, ABWR replaced the recirculation loop by 10 RIPs (reactor internal pumps), eliminating the probability of large break LOCA. 10 RIPs could provide 111% rated core flow at the nominal operating speed of 151.84 rad/sec.

The object of this paper is to develop a complete flow chart for analyzing the nuclear system transient, such as behaviors of fuel rod, vessel, and containment.

The double-ended MSLB transient in Lungmen ABWR was chosen to be a subject of case study in this paper. The MSLB is the design-basis accident analysis of containment, presenting in FSAR section 6.2 [1]. According to FSAR 6.2, double-ended MSLB transient is the limiting case for DW pressure. Lungmen NPP, the fourth NPP in Taiwan, has two identical units of ABWRs with 3,926 MWt each, consisted of 872 GE14 assemblies (10×10 with two water rods) with 205 control rods. Compared with BWR containment, there are two main differences: a) drywell (DW) is divided into upper-drywell (UDW) and low-drywell (LDW), which are connected by 10 drywell-connecting-vents (DCVs); b) wetwell (WW) is isolated from reactor building, which is connected with DW by 10 vertical vents with 3 horizontal vents each.

The codes, TRACE, PARCS, and FRAPTRAN are all developed and provided by US NRC. The Lungmen TRACE/PARCS coupling model with only nuclear steam supply system (NSSS) had been established and verified that it has respectable accuracy shown in previous papers of our laboratory [2][3][4]. In order to develop a complete flow chart for analyzing the nuclear system transient, the Lungmen containment model and FRAPTRAN model were established in this research. The results of TRACE/PARCS coupling calculation, with containment model, were compared with those of both FSAR and GOTHIC [1][5], indicating that the TRACE/PARCS coupling model has the ability to predict the MSLB transient, and both RPV(Reactor Pressure Vessel) integrity and containment integrity criteria are met. After that, the output data from TRACE/PARCS calculation was putted into FRAPTRAN code as boundary conditions to analyze the thermo-mechanical behavior and calculate the stress, strain, oxide thickness, etc. The values of these factors were compared with the criteria. And the final results show that the fuel rod integrity criteria are met.

ABBREVIATIONS

CAMP	Code Applications and Maintenance Program
DCVs	Drywell-Connecting-Vents
DW	Drywell
LDW	Low-Drywell
LOCA	Loss Of Coolant Accidents
MSLB	Main SteamLine Break
NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
NTHU	National Tsing Hua University
RIP	Reactor Internal Pump
RPV	Reactor Pressure Vessel
SP	Suppression Pool
SRV	Safety Relief Valve
TCV	Turbine Control Valve
TBV	Turbine Bypass Valve
UDW	Upper-Drywell
WW	Wetwell

1 INTRODUCTION

The object of this paper is to develop a complete flow chart for analyzing the nuclear system transient, such as behaviors of fuel rod, vessel, and containment, as shown in Figure 1.

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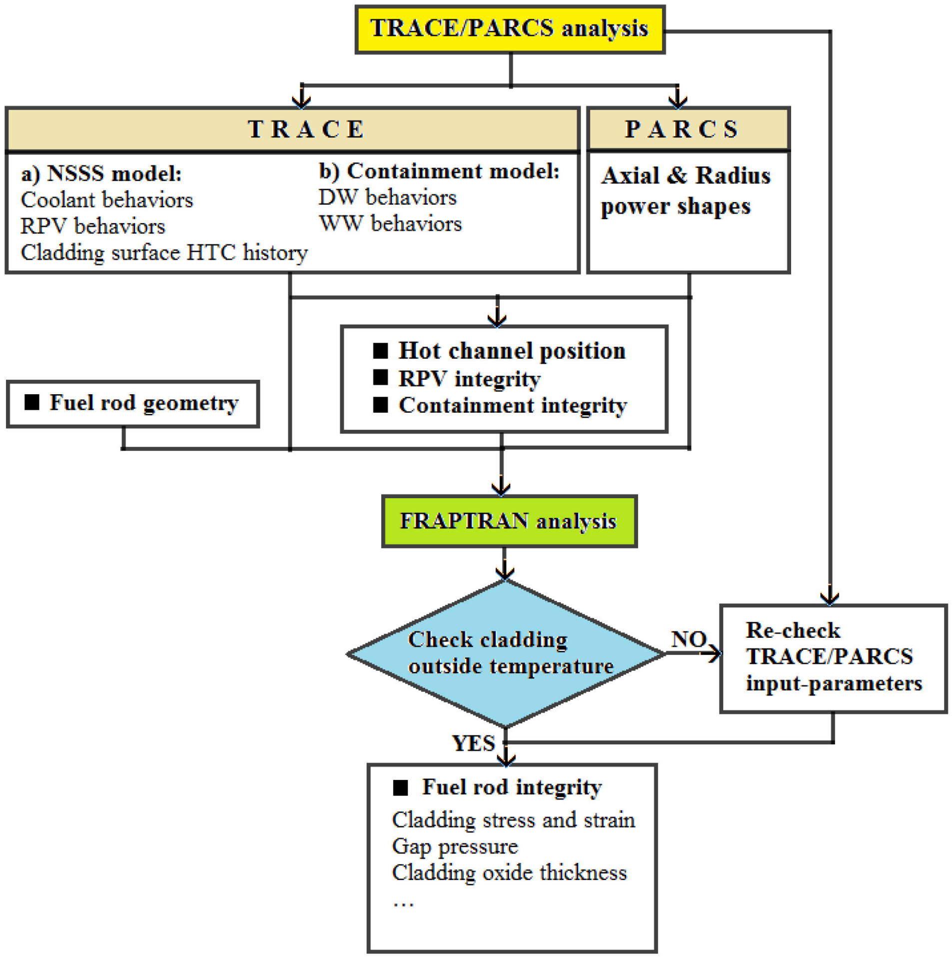


Figure 1 Flowchart of combining TRACE/PARCS and FRAPTRAN codes

2 MODELS OF LUNGMEN ABWR

2.1 Lungmen TRACE Model

The preliminary Lungmen TRACE model is established based on the relevant documents, as shown in Figure 2 [6]~[9]. There are three major control systems implemented in Lungmen TRACE model: feedwater control system, pressure control system, and RIP control system. The core region was modeled by 22 thermal-hydraulic channels to simulate the T-H behavior of 872 fuel assemblies. In the region around the dropped rod, each channel represented a single assembly in order to reflect accurately the T-H reactivity feedback effects following a control rod drop. In other region, each channel represented several fuel assemblies. The number of axial nodes in each channel is 11. According to the assemblies in the real reactor, the vessel was divided into eleven axial levels, four radial rings, and six azimuthal sectors. The six azimuthal sectors are orientated in 36°, 36°, 108°, 36°, 36°, 108°, 36°apart, and each azimuthal sector is connected with the feed water line inlet (six feedwater lines). There are four main steam lines connected to the 36°azimuthal sector of vessel and ten RIPs connected to six azimuthal sectors, one for every 36°. The ten RIPs were separated into three groups, four RIPs not connect to M/G sets (RIP3) and six RIPs connect to M/G sets (RIP1 and RIP2, thee for each). There are four sets of valves included in this model. The MSIVs and Turbine control valves (TCVs) are normally opened. The turbine bypass valve (TBV) and six groups of safety relief valves (SRVs), simulating eighteen SRVs distributed at the four main steam lines with different setpoints, are normally closed. In addition, the Moody choke flow model was adopted for limiting the maximum SRVs' flow.

In addition, the steady state plant parameters from Lungmen TRACE model had been successfully verified with those from FSAR and RETRAN02. The verified results reveal that there is respectable accuracy in the Lungmen TRACE model [10][11].

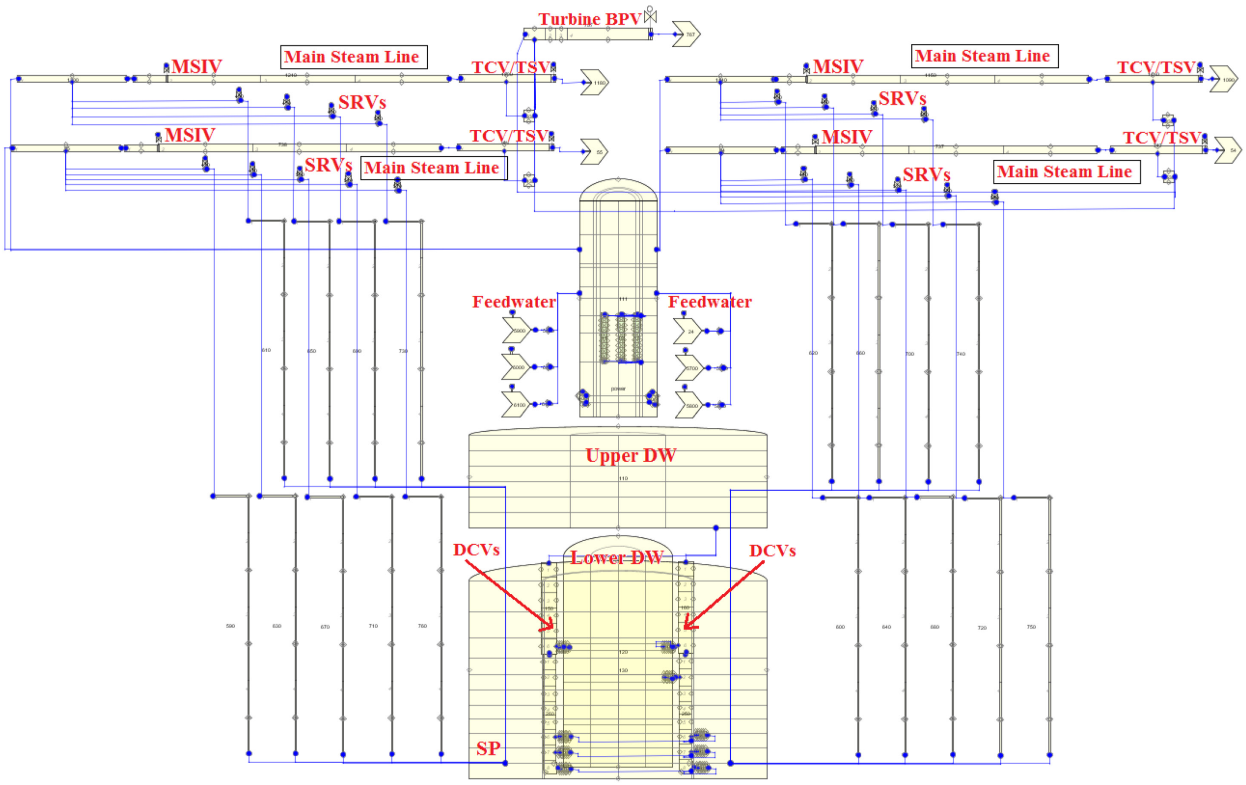


Figure 2 Lungmen TRACE model

2.2 Lungmen PARCS Model

PARCS involves 3D reactor core simulator for the realistic representation of physical reactor, and it can solve steady-state and time-dependent, multi-group neutron diffusion and SP3 transport equations in orthogonal and hexagonal core geometries. Figure 3 shows the core pattern for Lungmen PARCS model. For radial mesh, there are 1012 nodes in Lungmen PARCS model: 872 nodes model 872 fuel assemblies (yellow square); 140 nodes model the reflector outside the core (blue square). And the number of axial planes is 25 in the effective fuel region. The cross-section data used in PARCS calculation is provided by PMAXS file which is generated by GenPMAXS program from the macroscopic cross-section libraries and the results of lattice code, CASMO [12].

The preliminary Lungmen PARCS model is established by our laboratory colleagues, Chen [13] and Chang [14]. The k_{inf} calculated from PARCS had been verified by that from SIMULATE. The result shows the respectable accuracy in Lungmen PARCS model that the error bar is smaller than 10^{-5}.

Figure 3 is the code pattern of Lungmen PARCS model. The marked positions, (11,9) and (11,28), are the fuel assemblies which were chosen for FRAPTRAN analyses.

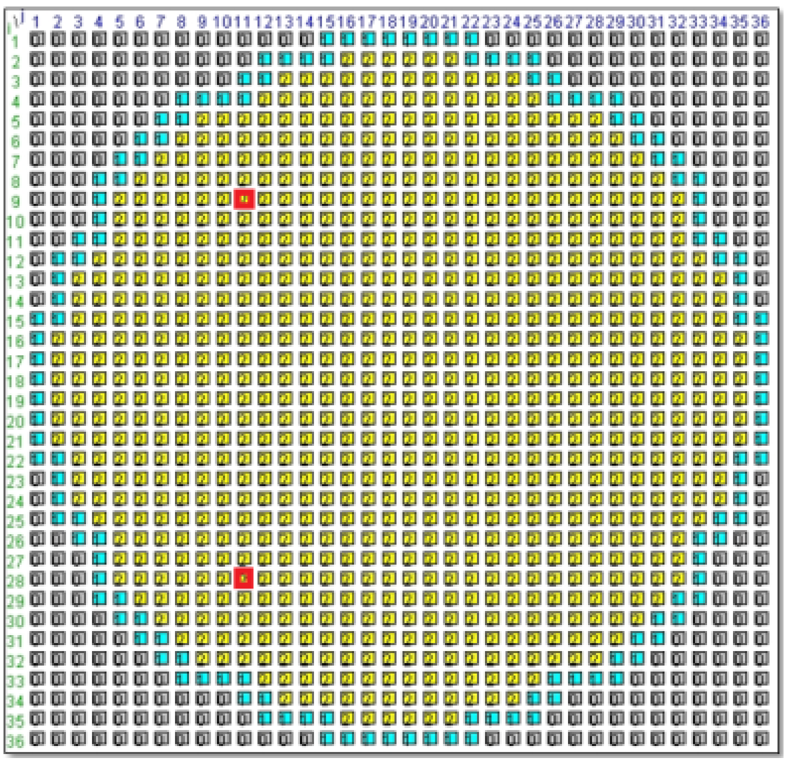


Figure 3 Core pattern for Lungmen PARCS model

2.3 Lungmen TRACE/PARCS Coupling Model

Figure 4 displays the flowchart of TRACE/PARCS coupling model. During the transient calculation, PARCS determines the core power distribution by using T-H conditions provided by TRACE. The power information is then transferred back to TRACE to calculate the new T-H conditions for PARCS. Thus the TRACE/PARCS coupling model gives the actual core power and T-H distribution at any time point.

Based on this preliminary Lungmen TRACE/PARCS coupling model, Feng et al.[15] analyzed the loss feed water heater transient and compared the results with plant vender data. It shows that the Lungmen TRACE/PARCS coupling model has an ability of transient simulation of Lungmen NPP.

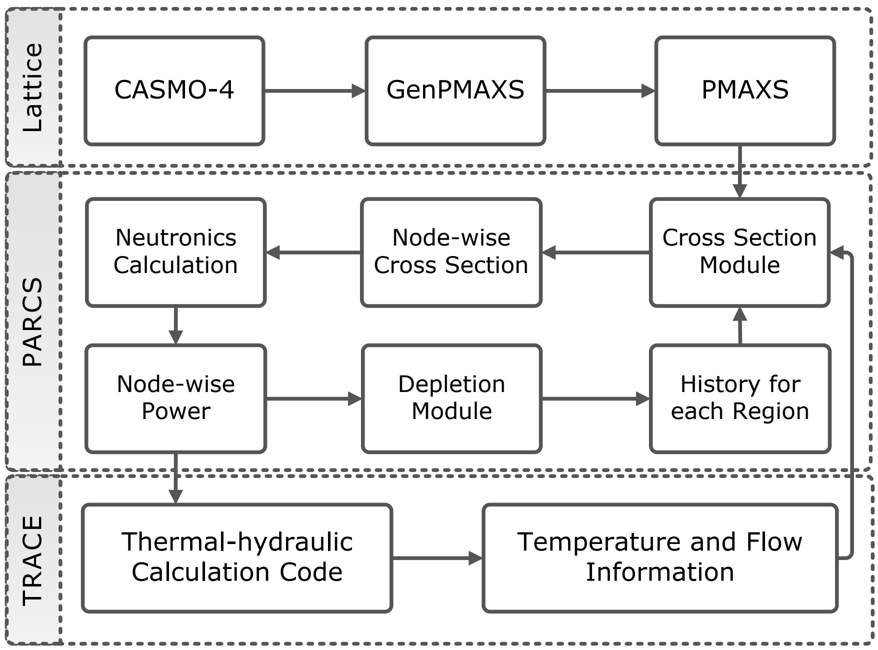


Figure 4 The procedure of TRACE/PARCS coupling calculation [16]

2.4 Lungmen TRACE/PARCS/FRAPTRAN Model

FRAPTRAN is a computer code for analyzing the thermo-mechanical behavior of light water reactor fuel rod under transients and accidents, such as LOCAs and RIAs [17]. Figure 5illustrates the schematic of fuel rod in FRAPTRAN model. The axial fuel length from bottom to top was divided into 12 nodes, and the fuel radial direction was divided into 17 nodes, including 15 nodes in the pellet and 2 nodes in the cladding. Although different numbers of axial node were used in these codes, important physical parameters could be obtained by simple linear interpolation.

Figure 1 shows the flowchart of combining FRAPTRAN and TRACE/PARCS. The input file of FRAPTRAN mainly composes of three parts to define the transient problems: a) Fuel rod geometry; b) Power history, including axial pin power shape and pin power history; c) Coolant boundary conditions, including coolant temperature, coolant pressure, and cladding-coolant heat transfer coefficient. In FRAPTRAN code, there are two modes we can choose to input the coolant boundary condition: COOLANT mode and HEAT mode. In this report, HEAT mode was chosen because the coolant boundary condition can be defined certainly from TRACE/PARCS output data. In addition, the reference temperature used in the calculation of fuel and clad enthalpy was defined at 298.15K.

The mechanical model used in FRAPTRAN for calculating the mechanical response of the fuel and cladding is the FRACAS-I model. This model does not account for stress-induced deformation of the fuel and therefore is called the rigid pellet model. This model includes the effects of thermal expansion of the fuel pellet; rod internal gas pressure; and thermal expansion, plasticity, and high-temperature creep of the cladding. After the cladding strain has been calculated by the mechanical model, the strain is compared with the value of an instability strain obtained from MATPRO. If the cladding effective plastic strain is greater than the cladding instability strain, then the cladding cannot maintain a cylindrical shape and local ballooning occurs. And the ballooning model, BALON2, is used to calculate the localized, nonuniform straining of the cladding. For the local region at which instability is predicted, a large deformation ballooning analysis is performed. No further strain is calculated for non-ballooning nodes. Modification of local heat transfer coefficients is calculated as the cladding ballooning progresses and additional surface area is presented to the coolant.

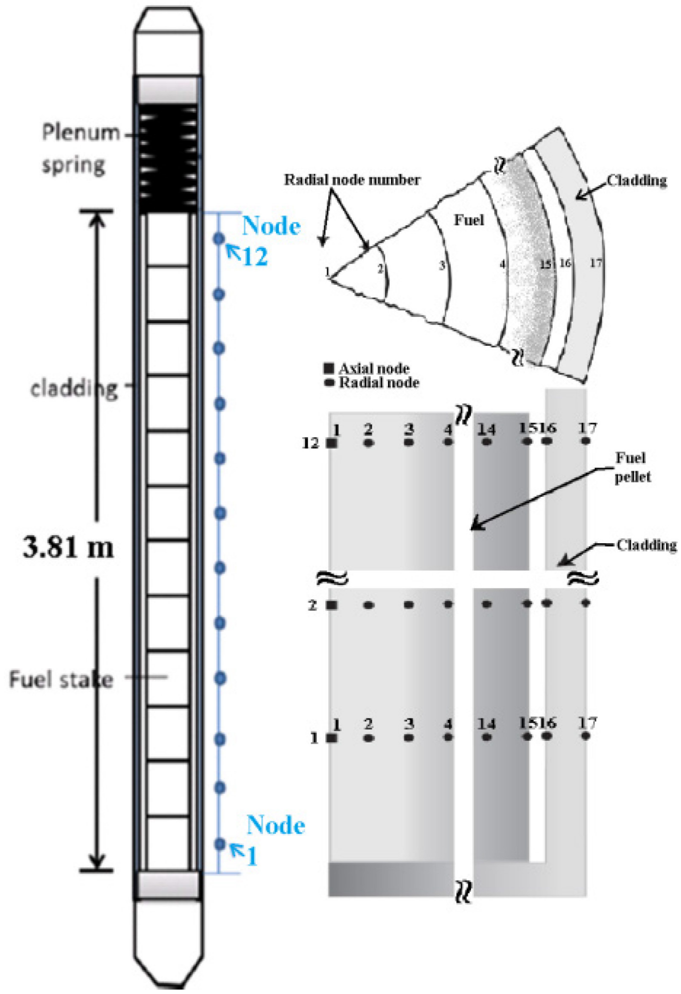


Figure 5 Schematic of fuel rod geometry in FRAPTRAN

3 INITIAL CONDITIONS AND RESULTS

3.1 Assumptions and Initial Conditions

The assumptions and initial conditions of the analysis are as follows:

- Initial reactor power was 4005 MWt (102% rated power).
- Double-ended MSLB break occurred at 0sec. The broken area of the RPV side was $0.0984m^2$ (flow limiter area). And the broken area of the main steam header side was $0.319m^2$ (main steam line area).
- MSIVs started to close at 0.5sec and fully closed at 5.0sec after MSLB.
- Initial pressure and temperature of DW were 5.17kPaG and 57.2℃, respectively.
- Initial pressure and temperature of WW were 5.17kPaG and 35℃, respectively.
- The initial suppression pool (SP) level was at 7.1m from the SP bottom.

3.2 TRACE/PARCS Calculation Results

3.2.1 Blowdown Conditions

Figure 6 and Figure 7 show the blowdown conditions at both RPV side and main steam header side. The blowdown conditions of GOTHIC code at RPV side are generated from two different ways: a) obtained by RELAP5 transient analysis (GOTHIC_1); b) calculated by a simplified RPV in GOTHIC (GOTHIC_2). Note that, in FSAR analysis (not shown), the RPV side and main steam header side are lumped as one single break (a time-varied broken area) on RPV side. The TRACE/PARCS results show the same trends with case GOTHIC_1, but the case GOTHIC_2 displays extremely different behaviors at RPV side. That is because the assumption of GOTHIC_2 is according to FSAR: because of RPV pressure drop, the core water level would swell and reach the elevation of main steam line at 2sec (RPV swell time) after MSLB. In other words, before 2sec, RPV side provides the single-phase flow only. After 2 sec, a lot of liquid water would blow down into DW from RPV via main steam line.

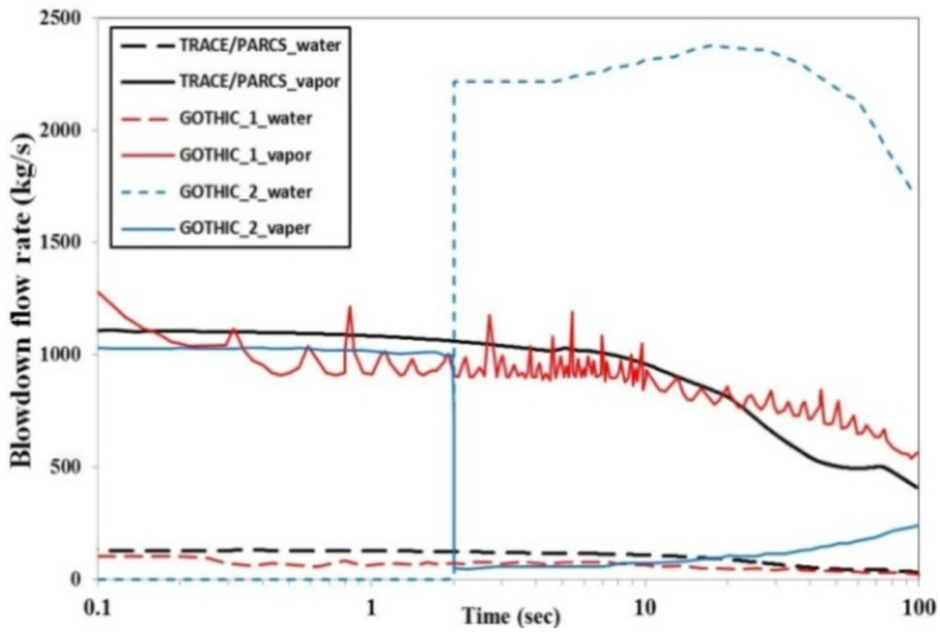


Figure 6 Blowdown condition of RPV side

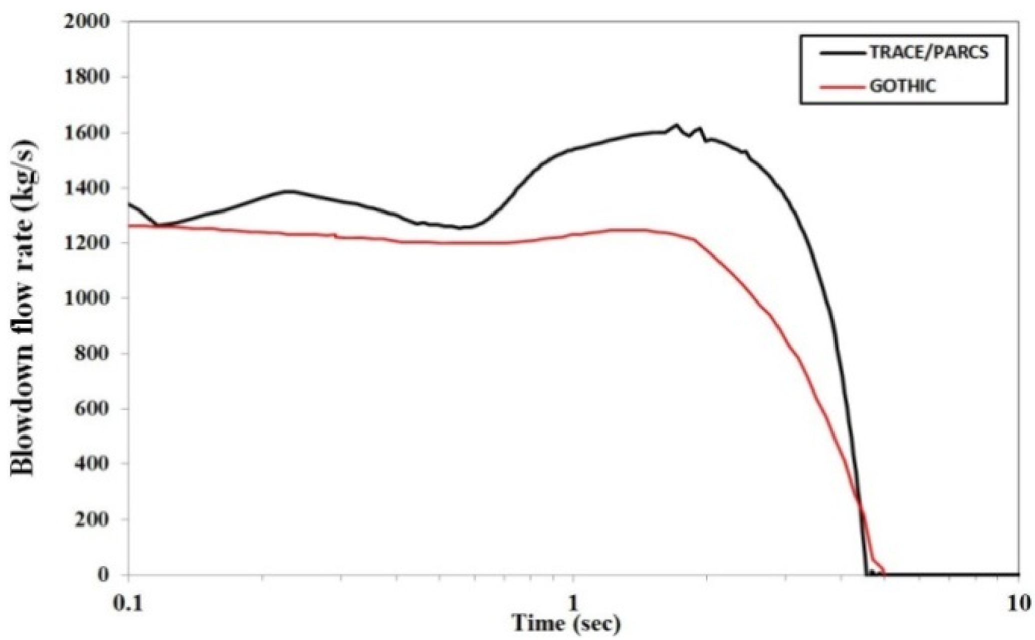


Figure 7 Blowdown condition of main steam header side

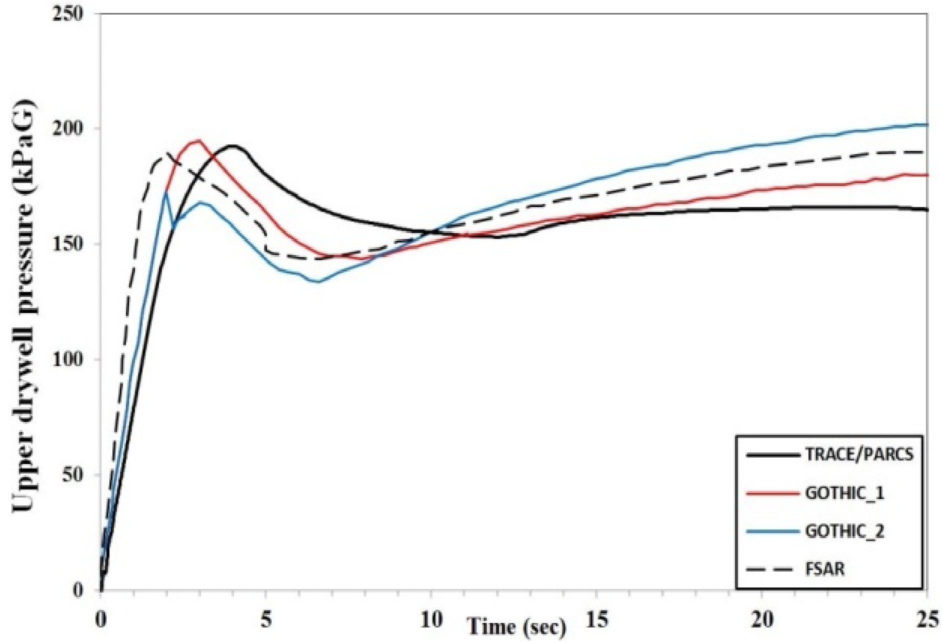
3.2.2 Pressure and Temperature Responses of Containment

Figure 8 and Figure 9 show the pressure and temperature responses of UDW and LDW. The TRACE/PARCS results show the same trends with case GOTHIC_1, but both pressure and temperature transfer delay-times are slightly longer than GOTHIC_1. That is because both FSAR and GOTHIC analyses, for conservative assumption, assume the DW volume to be the sum of UDW and 50%LDW. Thus, the transmissions of pressure and temperature in both FSAR and

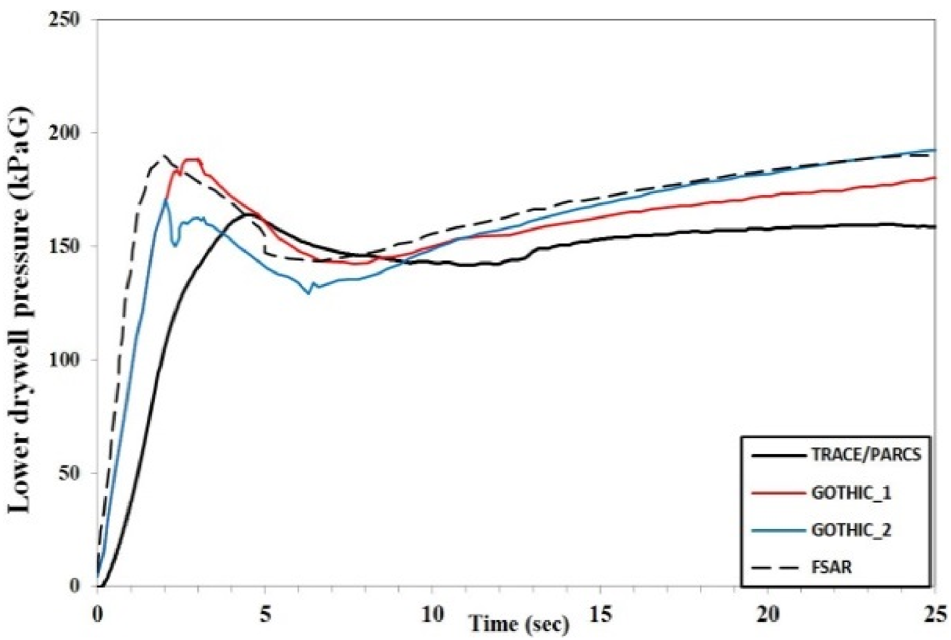
GOTHIC are faster than TRACE/PARCS. In addition, because FSAR and GOTHIC_2 make the same assumption of RPV swell time (2sec), as mentioned in 4.1.1, both pressure and temperature of DW drop obviously after a large amount of liquid water blow down into DW. Moreover, in FSAR analysis, the results of UDW and LDW are the same because FSAR treats UDW and LDW as one volume.

Figure 10 and Figure 11 show the pressure and temperature responses of WW. The TRACE/PARCS results show the same trends with both FSAER and GOTHIC except the WW airspace temperature, because FSAR assumes WW to be homogeneous mixture and steam to be completely condensed by SP.

According to TRACE/PARCS calculation, the peak of RPV dome pressure is 7.03MpaG (Figure 12, 10.342MPaG for criteria); the peaking values of pressure and temperature in DW are 192.44kPaG and 158.82℃(309.9kPaG and 171.1℃ for criteria, respectively); the peaking values of WW pressure, WW airspace temperature, and SP temperature are about 100kPaG, 80℃ and 38℃(309.9kPaG, 97.2℃ and 124.0℃ for criteria, respectively). And the peak of DW-WW pressure difference is 130.561kPaD(+172.6kPaD for criteria).

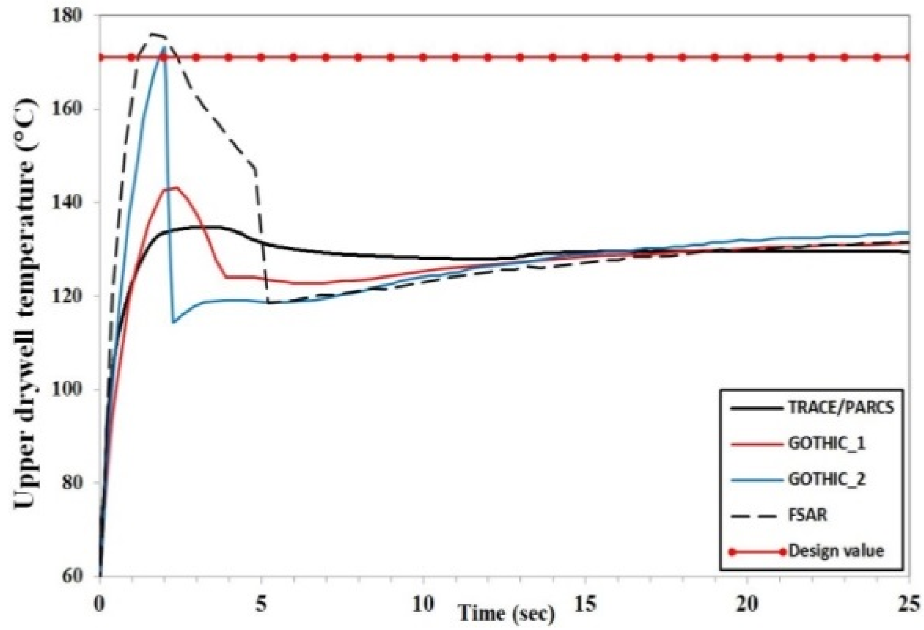


(a)

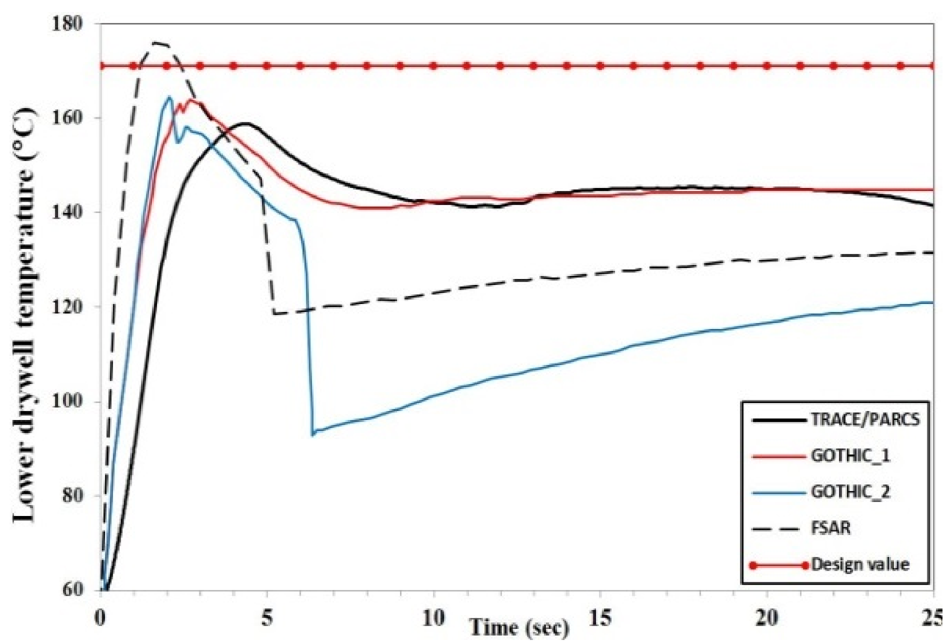


(b)

Figure 8 Pressures of (a) UDW and (b) LDW



(a)



(b)

Figure 9 Temperatures of (a) UDW and (b) LDW

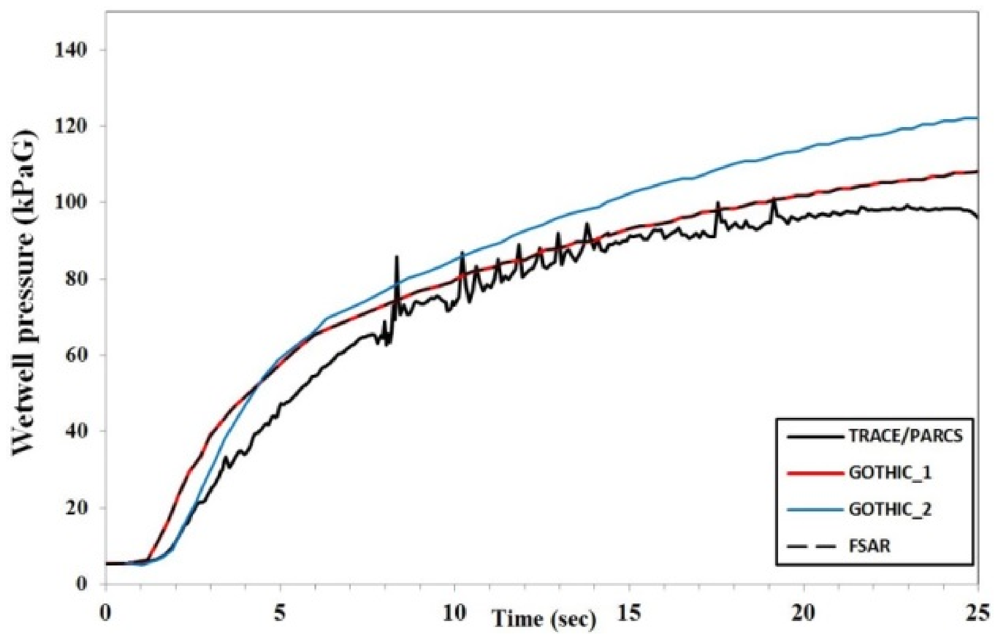
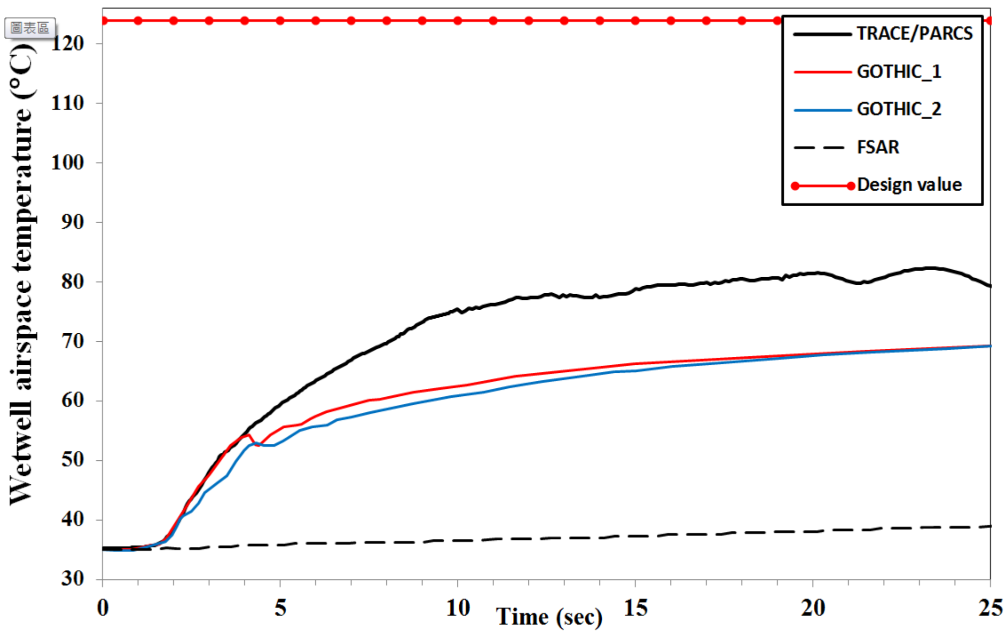
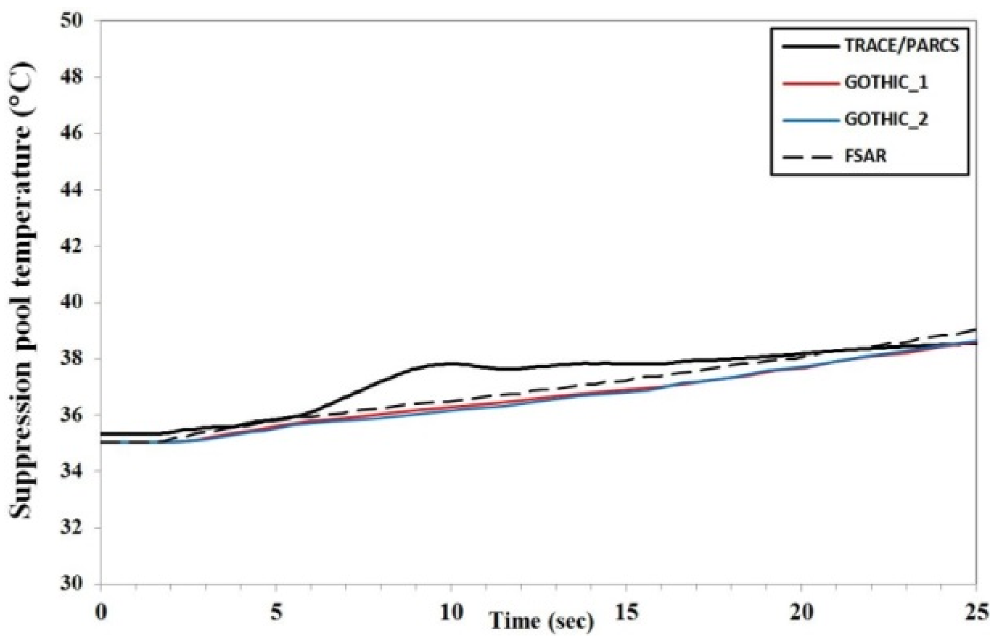


Figure 10 Pressure of WW



(a)



(b)

Figure 11 (a) Airspace and (b) SP Temperatures of WW

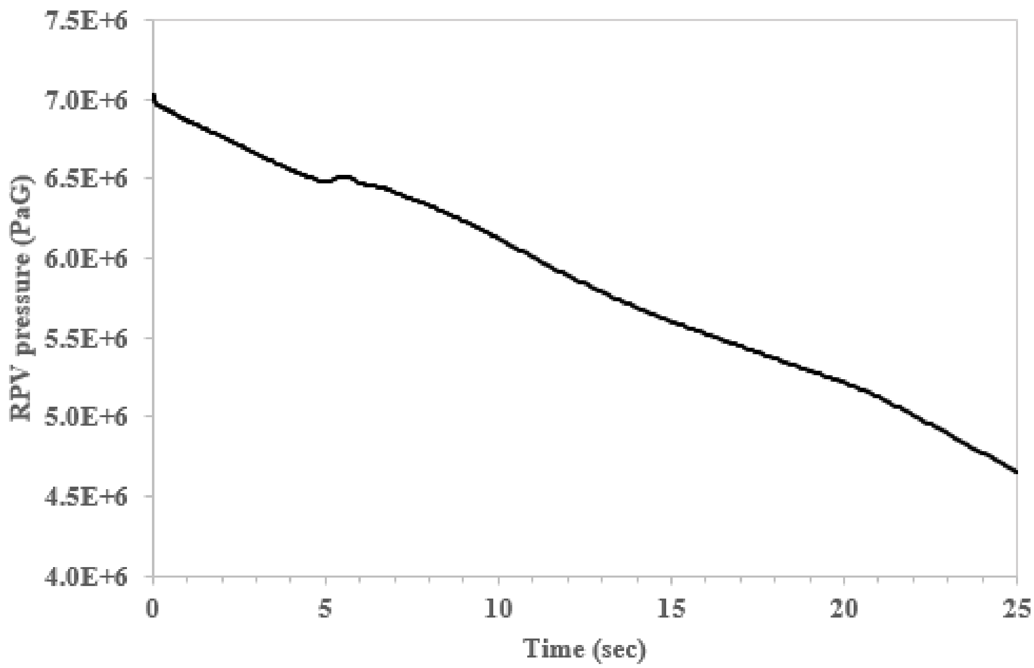


Figure 12 Pressure of RPV

3.3 FRAPTRAN Calculation Results

Before FRAPTRAN analysis, the cladding outside temperature calculated by FRAPTRAN must be compared with that calculated by TRACE/PARCS to re-confirm the correctness of input data, as shown in Figure 13. Note that, in FRAPTRAN analysis, MSLB was started at 200sec. Thus, the transient started time of FRAPTRAN, x-axis, was shifted to 0sec for comparison with TRACE/PARCS data.

Figure 14 and Figure 15 show the hoop strains of fuel surface and cladding. The main factor influencing the fuel surface hoop strain is reactor power. As Figure 12 shows, the fuel surface hoop strain decreases (i.e., fuel pellet contracts) after reactor power scrammed. The cladding hoop strain was calculated based on the following equation:

$$\varepsilon_\theta = \left[\frac{1}{E}(\sigma_\theta - \upsilon\sigma_z)\right] + \left[\varepsilon_\theta^P + d\varepsilon_\theta^P\right] + \left[\int_{T_0}^{T} \alpha\, dT\right]$$

where $\left[\frac{1}{E}(\sigma_\theta - \upsilon\sigma_z)\right]$ is due to the pressure difference between cladding inside and outside surface; $\left[\varepsilon_\theta^P + d\varepsilon_\theta^P\right]$ is plastic term; $\left[\int_{T_0}^{T} \alpha\, dT\right]$ is due to thermal expansion. The FRAPTRAN calculation indicates that the plastic term is zero. That is, there is no non-reversible change during MSLB transient. The term $\left[\frac{1}{E}(\sigma_\theta - \upsilon\sigma_z)\right]$ increases as RPV pressure drops after MSLB. Contrarily, the term $\left[\int_{T_0}^{T} \alpha\, dT\right]$ decreases after reactor power scrammed. The overall cladding hoop strain increases (i.e., cladding expands) with term $\left[\frac{1}{E}(\sigma_\theta - \upsilon\sigma_z)\right]$ due to RPV pressure drop except the duration of control rod inserted. From 200.5sec to control rod fully inserted, cladding hoop strain decreases (i.e., cladding contracts) with term $\left[\int_{T_0}^{T} \alpha\, dT\right]$ due to reactor power scram.

Figure 16 and Figure 17 show the temperatures of fuel surface and cladding inside surface, both indicating that the temperatures decrease as reactor power decreases. The peak temperatures of fuel surface and cladding inside surface are 1390.10℃ and 609.53℃(2805.0℃ and 1200.0℃ for criteria, respectively), respectively.

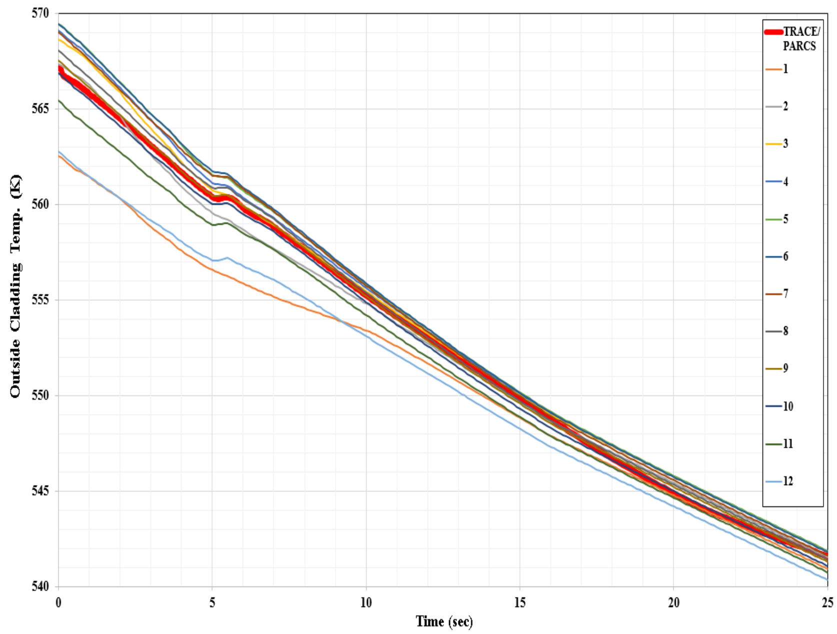


Figure 13 Cladding outside temperatures calculated by TRACE/PARCS coupling model and FRAPTRAN model

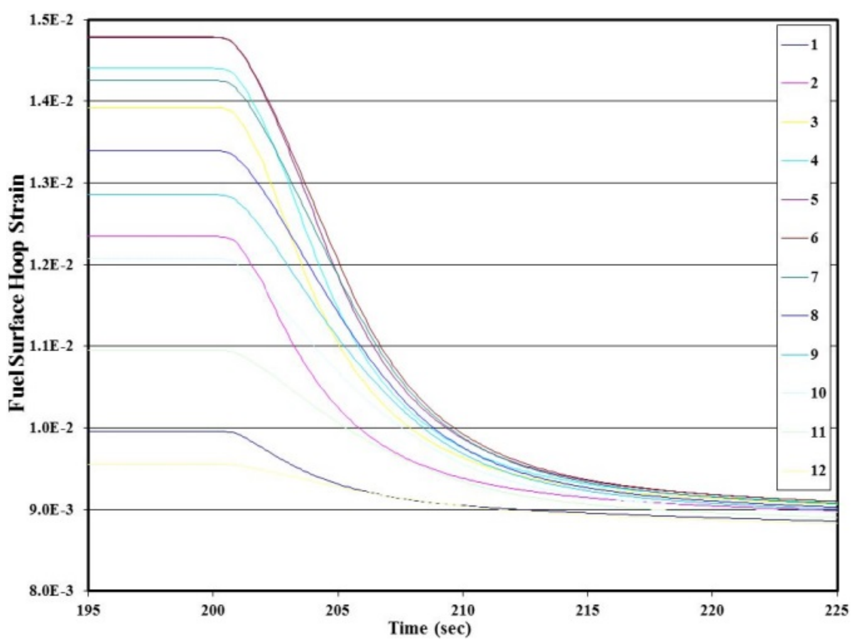


Figure 14 Fuel surface hoop strain

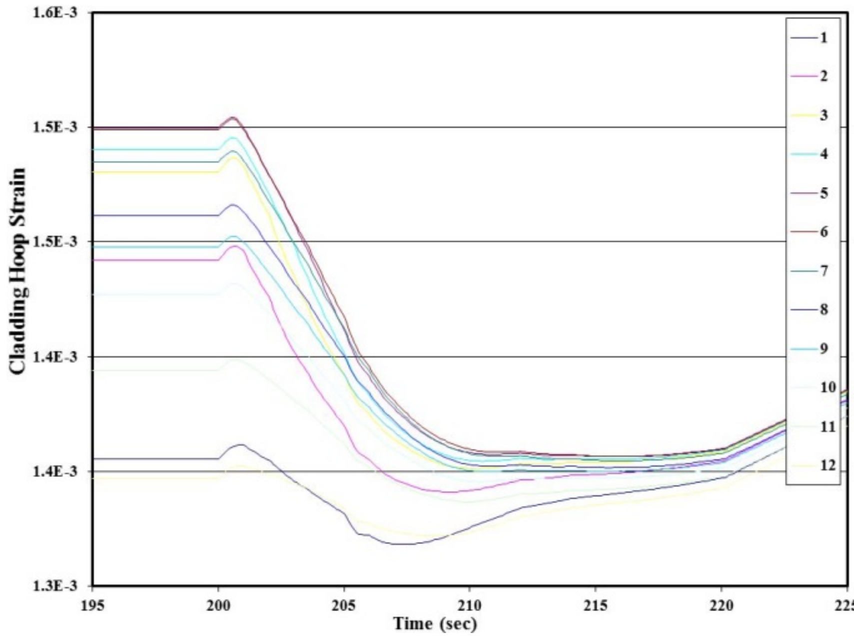


Figure 15 Cladding hoop strain

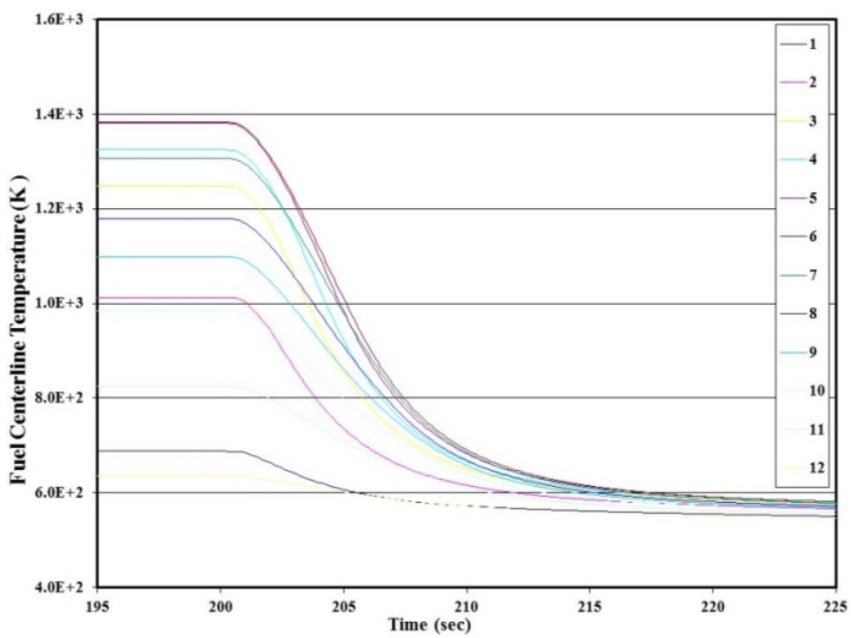


Figure 16 Fuel centerline temperature

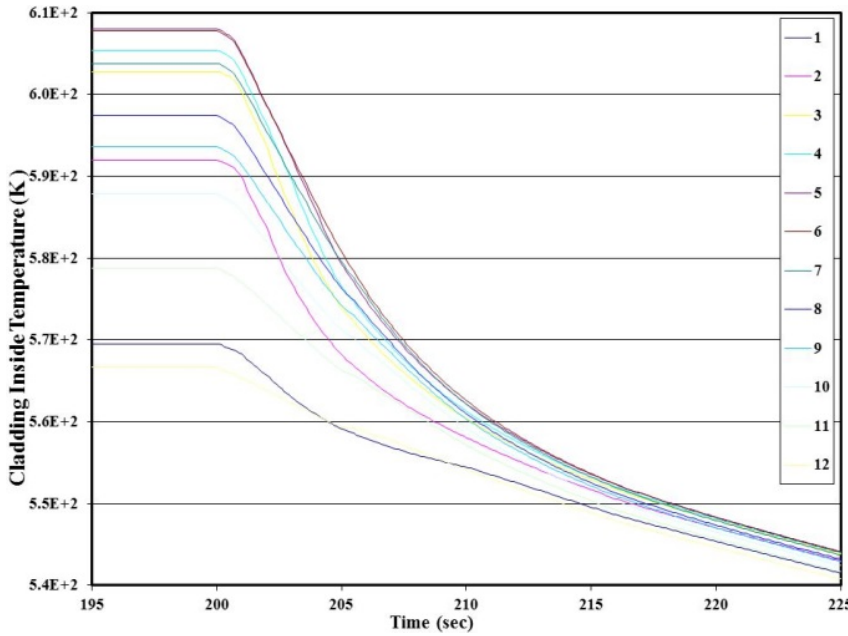


Figure 17 Cladding inside temperature

4 CONCLUSIONS

A complete flow chart for analyzing the nuclear system transient was performed. And the results of TRACE/PARCS coupling calculation were compared with those of both FSAR and GOTHIC, indicating that the TRACE/PARCS coupling model has the ability to predict the MSLB transient. According to TRACE/PARCS calculation, the peak of RPV dome pressure is 7.03MPaG (10.342MPaG for criteria); the peaking values of pressure and temperature in DW are 192.44kPaG and 158.82℃ (309.9kPaG and 171.1℃ for criteria, respectively); the peaking values of WW pressure, WW airspace temperature, and SP temperature are about 100kPaG, 80℃ and 38℃ (309.9kPaG, 97.2℃ and 124.0℃ for criteria, respectively). And the peak DW-WW pressure difference is 130.561kPaD (+172.6kPaD for criteria). Both RPV integrity and containment integrity criteria are met. According to FRAPTRAN calculation, the peak temperatures of fuel surface and cladding inside surface are 1390.10℃ and 609.53℃(2805.0℃ and 1200.0℃ for criteria, respectively), respectively. The oxidation under this temperature is insignificant. Therefore, the fuel integrity criteria are met.

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10. SUPPLEMENTARY NOTES

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11. ABSTRACT (200 words or less)

The object of this paper is to develop methodologies for analyzing the behaviors of fuel rod, vessel, and containment during main steamline break (MSLB) transient. The broken area of the RPV side was assumed to be 0.0984m2 (flow limiter). And the broken area of the main steam header side was assumed to 0.319m2 (main steam line area). According to FSAR, for conservative assumption, MSIVs started to close at 0.5sec and fully closed at 5.0sec after the transient started. The results of TRACE/PARCS coupling calculation were compared with those of both FSAR and GOTHIC data, indicating that the TRACE/PARCS coupling model has the ability to predict the MSLB transient, and both RPV integrity and containment integrity criteria are met. After that, the output data from TRACE/PARCS calculation was put into FRAPTRAN code as boundary conditions to analyze the thermo-mechanical behavior and calculate the stress, strain, oxide thickness, etc. The values of these factors were compared with the criteria. And the final results show that the fuel rod integrity criteria are met.

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