**NUREG/IA-0455**

# International Agreement Report

# Analysis of the Control Rod Drop Accident (CRDA) for Lungmen ABWR


Prepared by:
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# ABSTRACT

The purpose of this report is to understand the realistic behavior in Lungmen ABWR (Advanced Boiling Water Reactor) during a control rod drop accident (CRDA) transient. The CRDA transient would lead the reactor through an extremely fast and localized power excursion, requiring an accurate core modeling. The CRDA analysis for Lungmen ABWR was performed by coupling the 3D neutron kinetic code, PARCS, and two-phase thermal-hydraulic (T-H) code, TRACE. After TRACE/PARCS coupling calculation, the output data from TRACE/PARCS would be putted into FRAPTRAN code as boundaries, such as a function of time-dependent fuel rod power and coolant boundary conditions, to calculate the fuel damage. The CRDA analysis for Lungmen ABWR was performed for two conditions: a) case1: hot-full-power (HFP) at beginning of cycle (BOC); b) case2: hot-zero-power (HZP) at BOC. Under these conditions, the damage mechanisms of fuel rod are: 1) cladding ballooning and burst; 2) embrittlement and failure by high-temperature oxidation; 3) melting of cladding and/or fuel pellets. And the relevant quantities for fuel performance are the maximum fuel enthalpy and the melting temperatures of cladding and fuel pellet. The results of CRDA analysis show that a) case1: no fuel failure occurs under HFP condition at BOC; b) case2: the fuel rod nearby the dropped control rod failed under HZP condition at BOC, and the FRAPTRAN data exposes that the main reason of rod failure is the cladding high temperature.

# FOREWORD

The US NRC (United States Nuclear Regulatory Commission) 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 (Symbolic Nuclear Analysis Program) 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 usually used in the nuclear power plants analysis. This report showed TRACE can also do the calculation of small system such as dry-storage cask.

Taiwan and the United States have signed an agreement on CAMP (Code Applications and Maintenance Program) which includes the development and maintenance of TRACE. INER (Institute of Nuclear Energy Research, Atomic Energy Council, R.O.C.) is the organization in Taiwan responsible for the application of TRACE in thermal hydraulic safety analysis, for recording user's experiences of it, and providing suggestions for its development. 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 CRDA 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. INER 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 by INER.

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. NRC has ensured that TRACE will be the main code used in thermal hydraulic safety analysis in the future without further development of other thermal hydraulic codes, such as RELAP5 and TRAC. Besides, 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 $7.64\times10^6$ Kg/h at rated power condition. The designed rated core flow is $52.2\times10^6$ 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 initiating event for CRDA is the separation of a control rod blade from its driving mechanism. And the control rod is removed from core due to gravity and drops in a free fall. The CRDA can occur at any reactor operating condition. As control rod drops, the ability of neutron absorption decreases, causing an increase in fission rate and reactor power. This power excursion may lead to failure of the fuel rod and integrity of the reactor core, and in very severe case, even lead to disruption of the reactor.

The CRDA is the design-basis reactivity-initiated accident (RIA) in BWR. Historically, the point kinetics model or one dimensional kinetics model using core wide coefficients having a significant conservatism has been employed for the safety assessment of RIA [1]. This approach is insufficient to calculate an extremely fast and localized power excursion induced by CRDA. In order to understand the realistic reactor and fuel behavior in Lungmen ABWR, in this report, CRDA transient is performed by coupling the 3D neutron kinetic code, PARCS, and two-phase T-H code, TRACE. After TRACE/PARCS coupling calculation, the output data from TRACE/PARCS would be inputted into FRAPTRAN code as a function of time-dependent fuel rod power and coolant boundary conditions to calculate the fuel damage. And there are two cases to discuss: a) case1: HFP condition at BOC; b) case2: HZP condition at BOC.

# ABBREVIATIONS

| | |
|---|---|
| BOC | Beginning Of Cycle |
| CRDA | Control Rod Drop Accident |
| CPR | Critical Power Ratio |
| DNBR | Departure From Nuclear Boiling Ratio |
| GDC-28 | General Design Criterion 28 |
| HFP | Hot-Full-Power |
| HZP | Hot-Zero-Power |
| LOCA | Loss Of Coolant Accidents |
| PCMI | Pellet-Clad Mechanical Interaction |
| RG | Regulatory Guide |
| RIPs | Reactor Internal Pumps |
| SRVs | Safety Relief Valves |
| TCVs | Turbine Control Valves |
| TBV | Turbine Bypass Valve |

# 1. INTRODUCTION

The purpose of this report is to understand the realistic behaviour in Lungmen ABWR during CRDA transient. The analysis of CRDA for Lungmen ABWR was performed by using TRACE v5.0 p2, PARCS V3.0, and FRAPTRAN 1.4 under SNAP Configuration 2.0.6.
The fuel rod behaviour under CRDA is affected by the characteristic of the power pulse, core coolant conditions, burn-up-dependent state of fuel rod, and fuel rod design [2]. There are four potential failure modes for the fuel rod, one at low temperature:

- by pellet-clad mechanical interaction (PCMI) under the early heat-up stage,
  and three at high temperature:
- by cladding ballooning and burst,
- by disruption of the cladding upon quenching from high temperature,
- by melting the cladding and/or fuel pellets.

Low-temperature PCMI-induced cladding failure is defined by the initial fuel enthalpy increase, which may occur in high-burn-up fuel rod, but not in fresh fuel rod. Thus, for the zero-burn-up fuel rod, high-temperature failures are its limiting failure modes. Cladding ballooning and burst occurs, if clad-to-coolant heat transfer is impaired by a boiling crisis (film-boiling) and the fuel rod internal gas pressure exceeds the coolant pressure (rod internal overpressure). Fuel rod disruption under quenching is due to cladding embrittlement by high temperature oxidation under the film-boiling phase. High-temperature failure is defined by the total radial average fuel enthalpy as a function of pressure difference between internal and external pressure of fuel rod [3].

Acceptance criteria for CRDA events are based on 10CFR50 Appendix A, General Design Criterion 28 (GDC-28) requirements detailed within Regulatory Guide (RG) 1.77 and NUREG-0800 Standard Review Plan. For CRDA with zero-burn-up fuel rod at HFP and HZP, the following criteria are listed [4][5][6]:

- Fuel cladding failure criteria: for zero power conditions, a peak radial average fuel enthalpy can not be greater than 170cal/g for fuel rods with an internal rod pressure at or below system pressure and 150cal/g for fuel rods with an internal rod pressure exceeding system pressure; for intermediate and full power conditions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g. DNBR and CPR).
- Coolable core geometry criteria: peak radial average fuel enthalpy must remain below 230cal/g; peak fuel temperature must remain below incipient fuel melting conditions (fuel centerline temperature < 3078.15K); peak cladding temperature must remain below incipient cladding melting conditions (< 1473.15K).

There are two cases to discuss: a) case1: HFP condition at BOC; b) case2: HZP condition at BOC. And the results show that a) case1: no fuel failure occurs under HFP condition at BOC; b) case2: the fuel rod nearby the dropped control rod failed under HZP condition at BOC, and the FRAPTRAN data exposes that the main reason of rod failure is the cladding high temperature.

# 2. MODELS OF LUNGMEN ABWR

## 2.1 Lungmen TRACE Model

The analysis of CRDA for Lungmen ABWR was performed by using TRACE v5.0 p2. The preliminary Lungmen TRACE model is established based on the relevant documents, as shown in Figure 1 [7]~[10]. 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 [11][12].

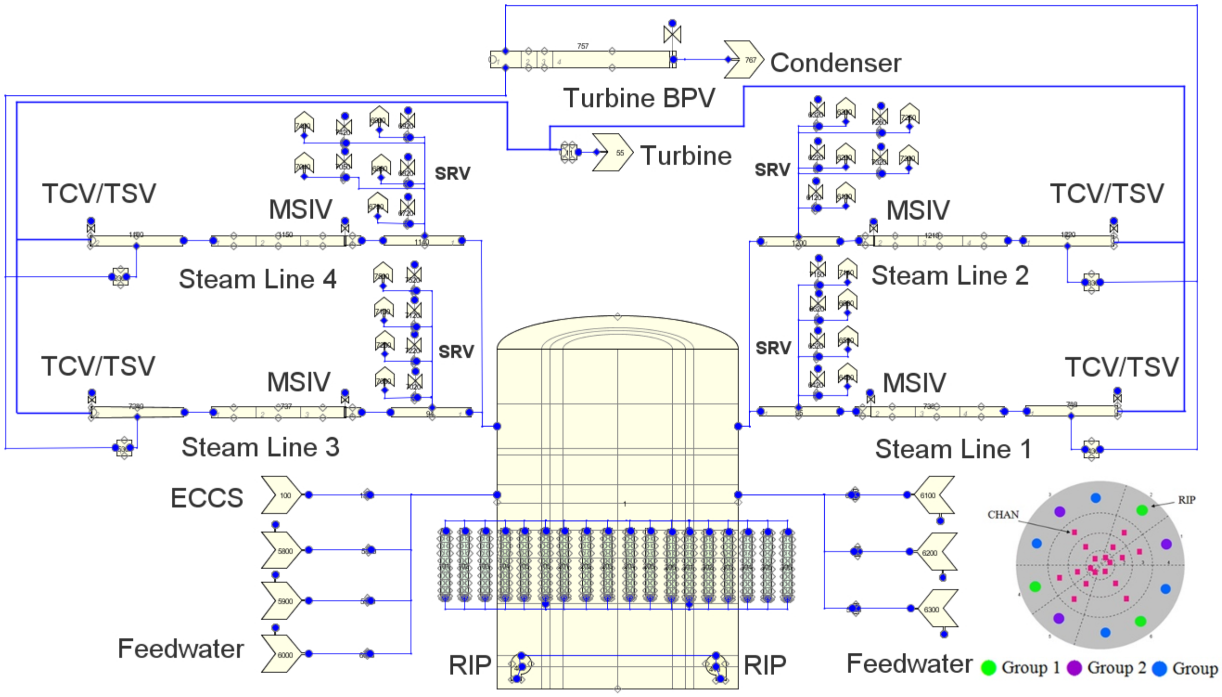



**Figure 1 Lungmen TRACE model**

## 2.2 Lungmen PARCS Model

The analysis of CRDA for Lungmen ABWR was performed by using PARCS V3.0. 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 2 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 [13].

The preliminary Lungmen PARCS model is established by our laboratory colleagues, Chen [14] and Chang [15]. 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 shows a control rod pattern in Lungmen ABWR under reactor normal operation condition. Figure 4 shows the control rod pattern at HZP condition. Normally, the 205 control rod were divided into 19 groups, each group has different initial step. For analysis with CRDA, an additional control rod group was added to control the selected control rod. Figure 5 shows the core averaged axial power shape at (a) HFP and (b) HZP condition.

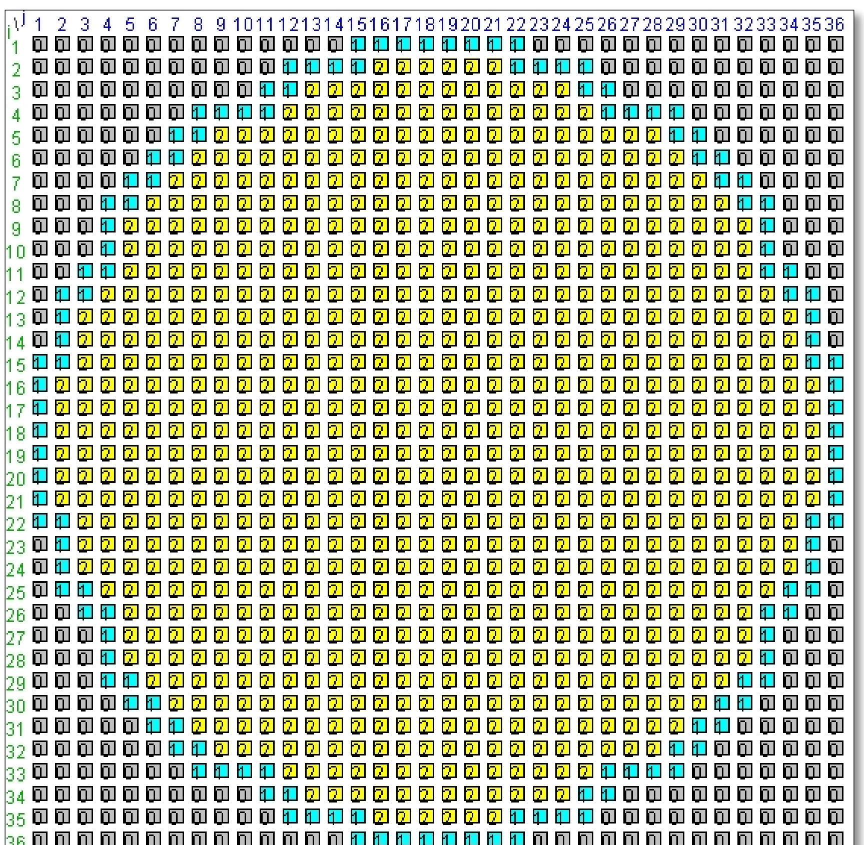


**Figure 2 Core pattern for Lungmen PARCS model**

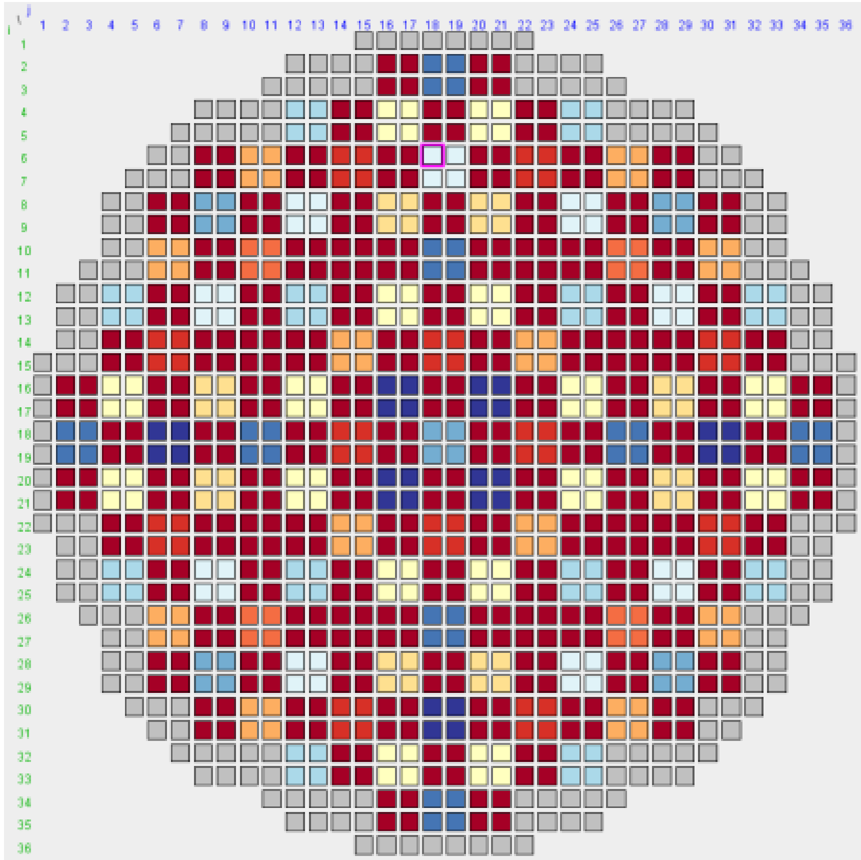


**Figure 3　Control rod pattern for case1-HFP**

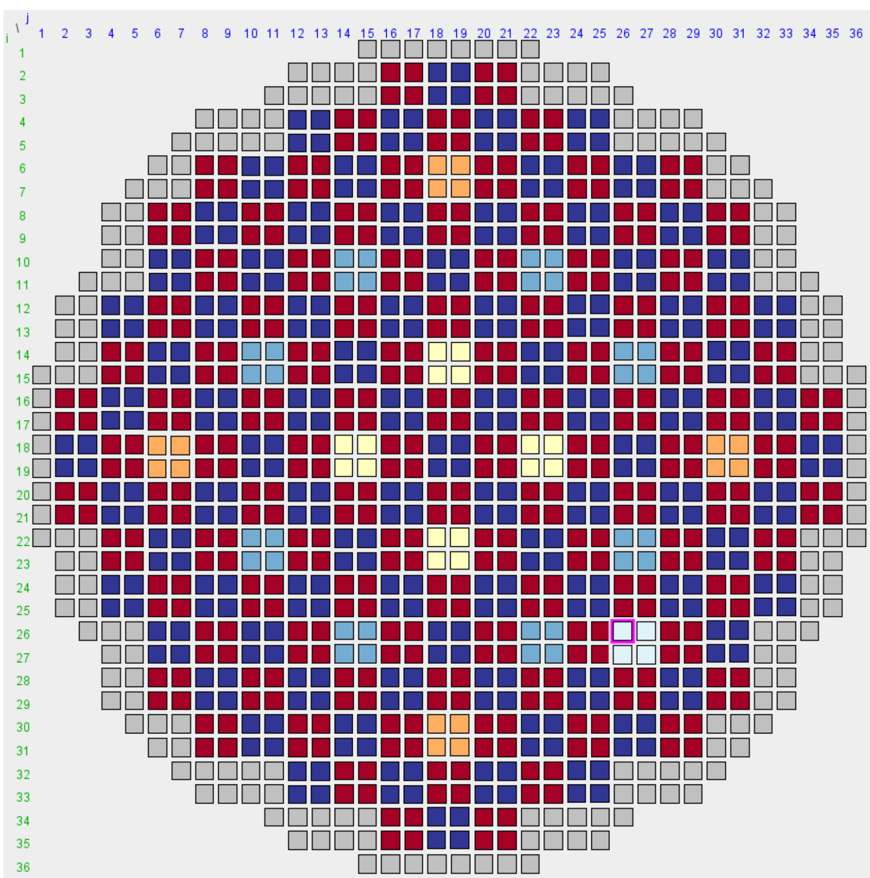
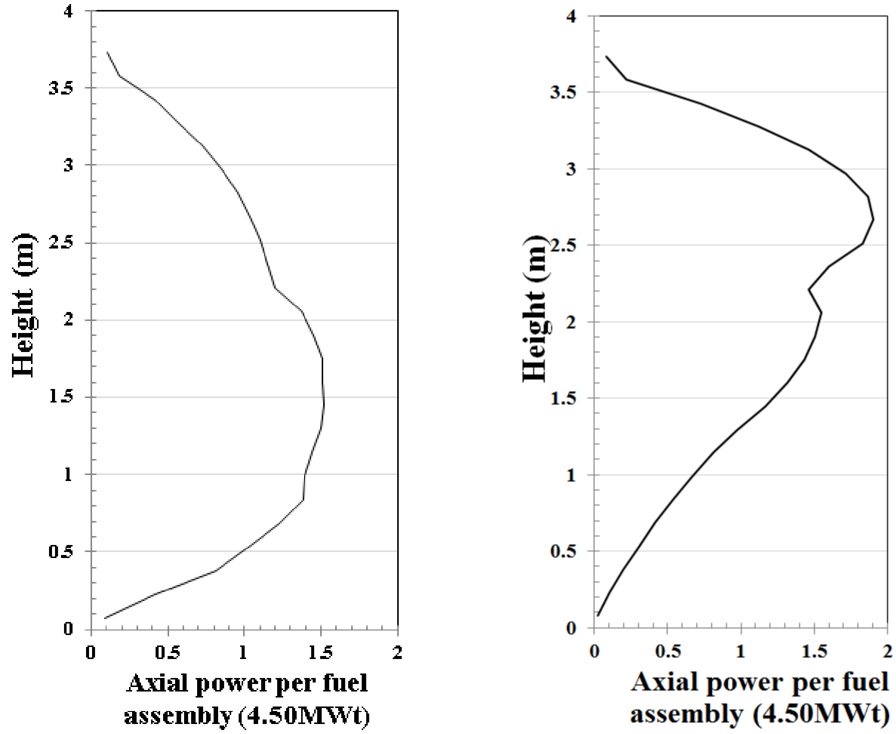


**Figure 4　Control rod pattern for case2-HZP**




(a) HFP (b) HZP

**Figure 5 Core averaged axial power shape at (a) HFP and (b) HZP condition**

## 2.3 Lungmen TRACE/PARCS Coupling Model

Note that the case ARI and FMCRD run-in was performed by TRACE/PARCS coupling model, but in case SLCS initiation we substituted PARCS for point kinetics because Lungmen PMAXS file used in core power calculation has problems in boron reactivity calculation.

Figure 6 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. [16] 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.

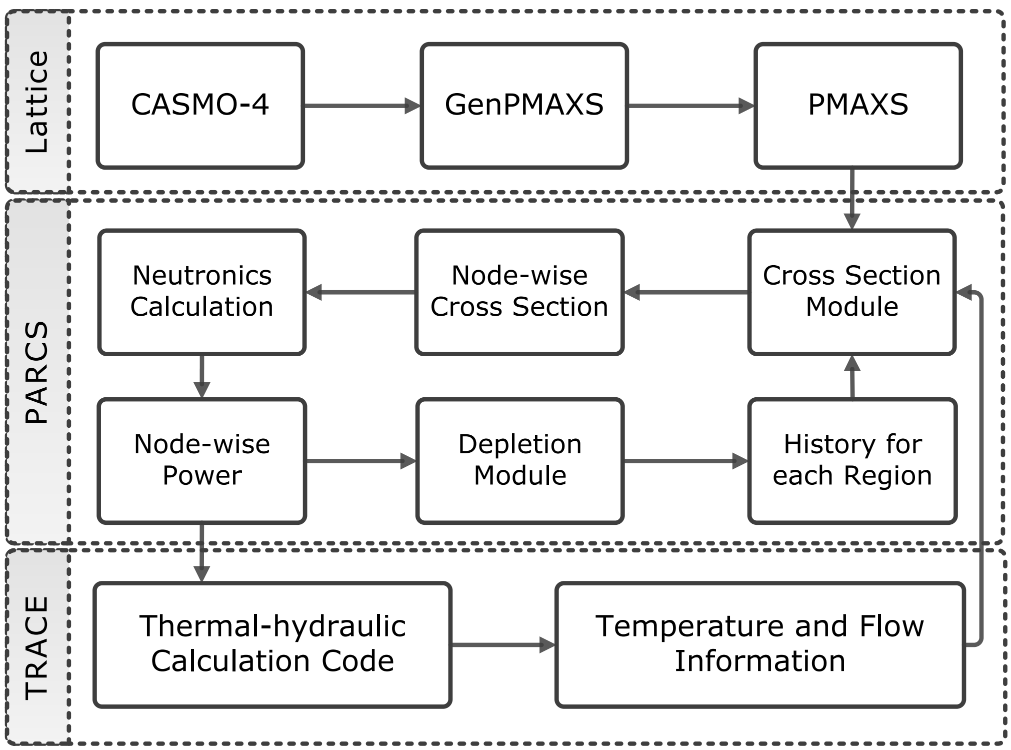



**Figure 6 The procedure of TRACE/PARCS coupling calculation [17]**

## 2.4 Lungmen TRACE/PARCS/FRAPTRAN Model

The analysis of CRDA for Lungmen ABWR was performed by using FRAPTRAN 1.4. FRAPTRAN is a computer code for analyzing the thermo-mechanical behavior of light water reactor (LWR) fuel rod under transients and accidents, such as LOCAs and RIAs [18]. Figure 7 illustrates 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 8 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 (Figure 7); 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 (Hagrman et al., 1981). 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.

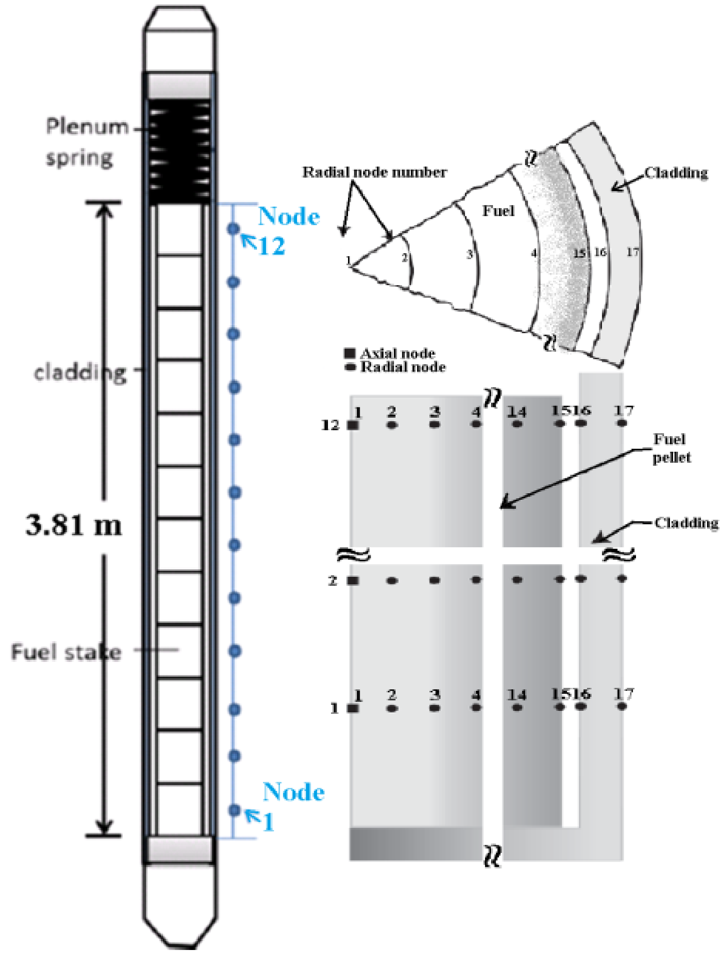



**Figure 7 Schematic of fuel rod geometry in FRAPTRAN**

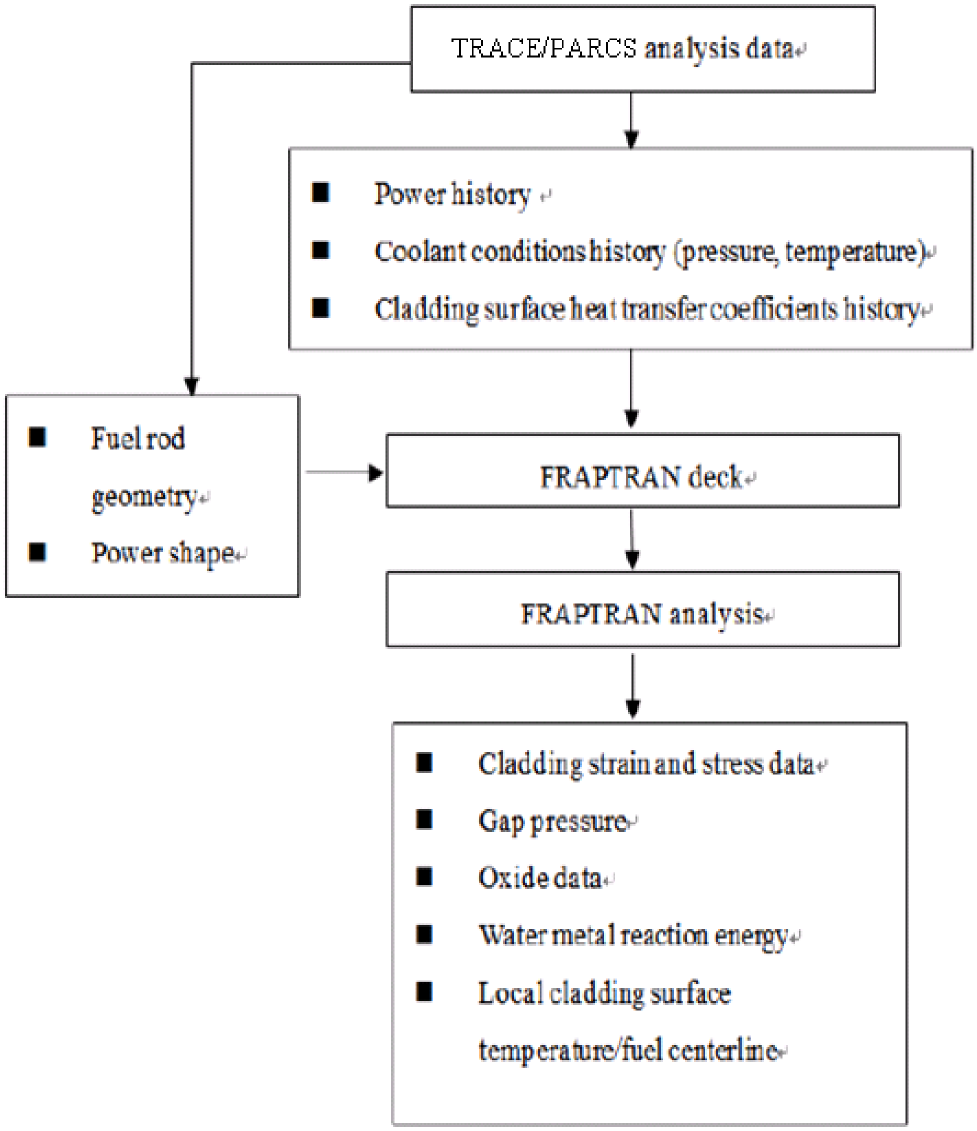



**Figure 8 Flowchart of combining TRACE/PARCS and FRAPTRAN codes**

# 3. INITIAL CONDITIONS AND RESULTS

The initiating event for CRDA is the separation of a control rod blade from its driving mechanism. And the control rod is removed from core due to gravity and drops in a free fall. The CRDA event can occur at any reactor operating condition. Two conditions were performed for CRDA analysis of Lungmen ABWR: a) case1: HFP condition at BOC; b) case2: HZP condition at BOC. The initial core conditions and parameters used in the analysis for each case were shown in Table 1. And the initial control rod patterns for these two cases were shown in Figure 3 and Figure 4, respectively. Figure 9 shows the inserted reactivity due to a control rod drop at various control-rod-drop-distances from reactor core center, indicating that the inserted reactivity of the dropped control rod located in the periphery of core is larger than that located in the center of core. Thus, in the following two cases, the CRDA analyses were performed with choosing the dropped control rod located in the periphery of core. In addition, the reactor would be scrammed if the reactor power level reaches 115% rated power (4514.9MWt).

Reactivity is a fundamental quantity, expressing the departure of a nuclear reactor from criticality as the control rod drops and describing the feedback effects of fuel and moderator during the transient. There are four important factors: control rod position, fuel temperature (Doppler), moderator temperature, and moderator void fraction. Hence, the reactivity rate of change may be written [16]:

$$\dot{\rho} = \dot{\rho_{CR}} + \frac{\partial \rho}{\partial T_f}\dot{T}_f + \frac{\partial \rho}{\partial T_m}\dot{T}_m + \frac{\partial \rho}{\partial \alpha_m}\dot{\alpha}_m = \dot{\rho_{CR}} + \dot{\rho_{TF}} + \dot{\rho_{DM}}$$

where $\dot{\rho_{CR}}$, $\dot{\rho_{TF}}$, and $\dot{\rho_{DM}}$ are the reactivity rate of change induced by control rod, fuel temperature, and moderator density. $\dot{T}_f$ and $\dot{T}_m$ are the rate of temperature change for fuel and moderator, and $\dot{\alpha}_m$ is the rate of change for the moderator void fraction.

**Table 1 Core initial conditions for two CRDA cases**

| Core condition/Parameter | Value/Assumption | |
|---|---|---|
| | Case1 | Case2 |
| Initial power level (MW) | 3926 (rated power) | 3.926E-3 (1E-6 rated power) |
| Dome pressure (MPa) | 7.17 | 6.62 |
| Coolant temperature (K) | 560.61 | 547.5 |
| Core flow (%) | 100% | 35% |
| Control rod drop speed (m/s) | 0.7 | |

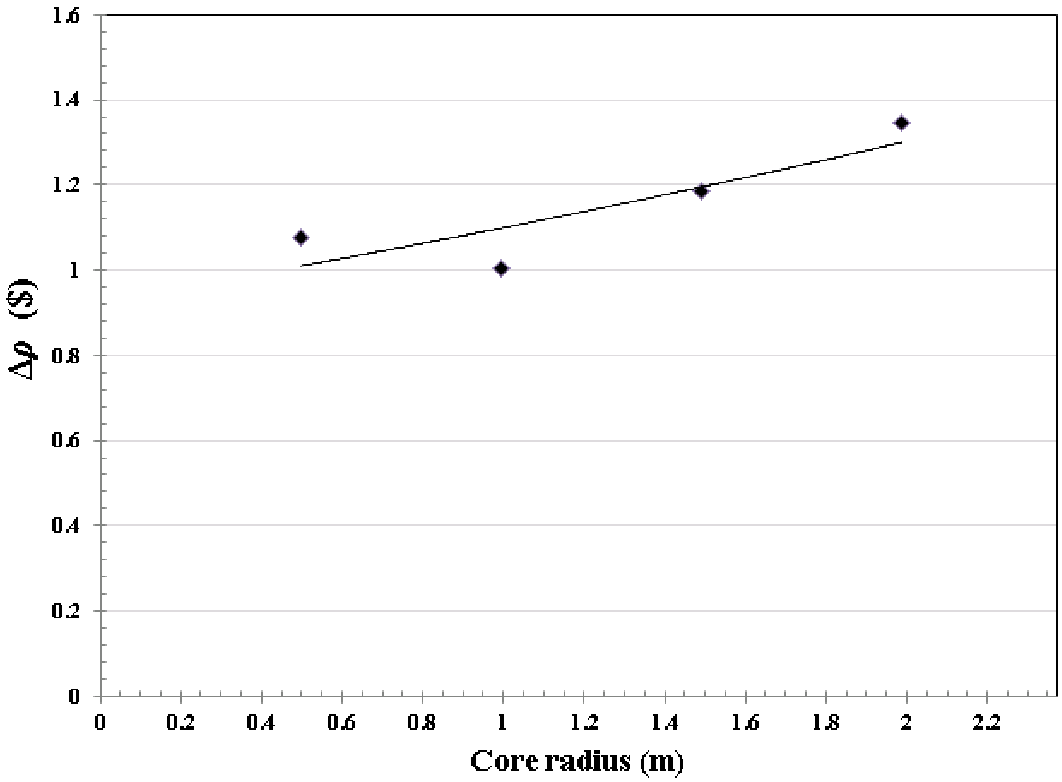



**Figure 9 Inserted reactivity due to control rod drop at various rod position**

### 3.1 <u>Case1-HFP Condition at BOC</u>

At 500sec, the selected control rod starts to fall. Figure 10 shows the trend of the different reactivity feedback components during the transient. As control rod drops, the reactivity raises. The positive reactivity leads the reactor power to increase, as shown in Figure 11, increasing the fuel temperature immediately (Figure 12). The increase of fuel temperature feedbacks a negative reactivity to system, called Doppler effect. At the same time, the moderator void fraction increases immediately when fuel temperature increases. The time lag for fuel-to-cladding-to-coolant transfer does not show an obvious effect in case1. And the increase of moderator void fraction also gives a negative feedback-reactivity. The negative feedback-reactivity from fuel and moderator suppresses the reactivity increase and lets the reactor back to criticality at higher power level.

During the whole transient, the maximum value of power is 4267MWt (about 109% rated power) at 517.2sec. No reactor scram occurs. Figure 13 and Figure 14 show the average cladding temperature and average fuel enthalpy at each node form bottom to top of fuel. The maximum value of fuel centerline temperature, cladding inside temperature, and average fuel enthalpy are 1347.2K, 587.4K, and $2.085\times10^{5}$J/kg (49.83cal/g), respectively, which all occurs at node 5 (elevations~1.43m) where the fuel rod is more reactive at HFP condition, as shown in Figure 5 (a). In addition, the fuel temperature, cladding temperature, and maximum fuel enthalpy are well below the limitation of criteria, indicating that no fuel failure occurs in case1. And it is not necessary to discuss the high-temperature failures in fuel rod.

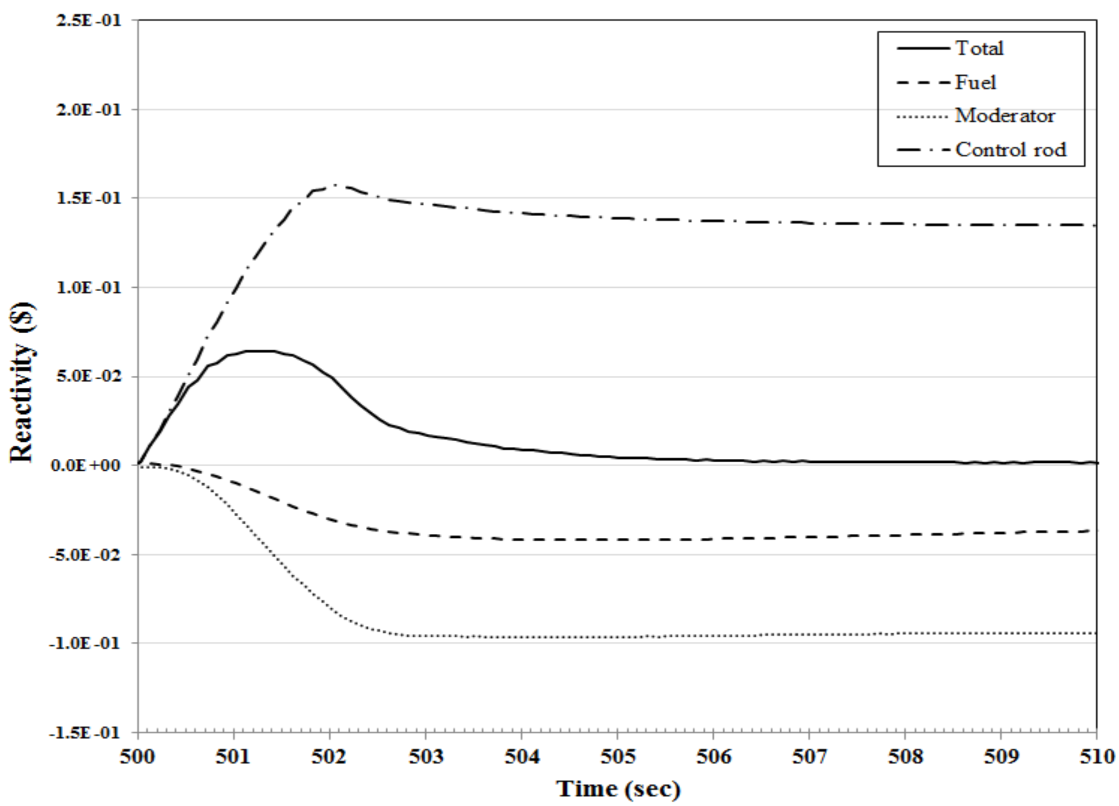



**Figure 10 Reactivity components evolution in case1**

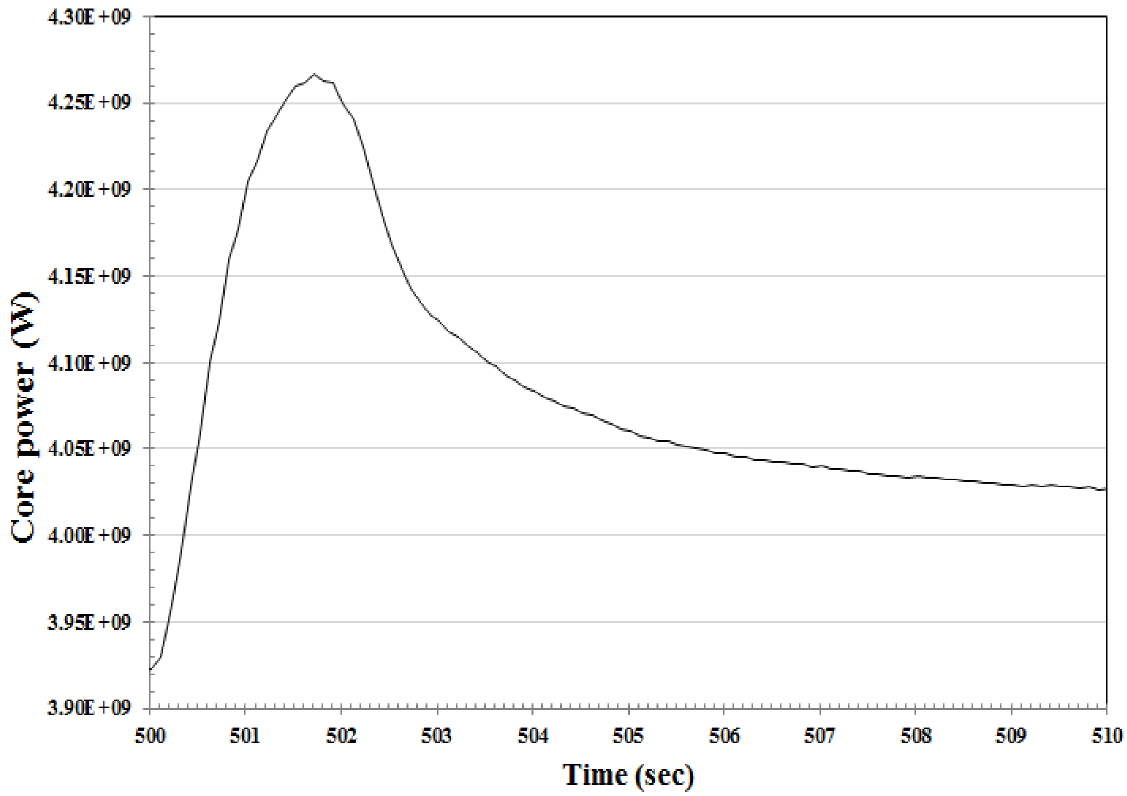



**Figure 11 Reactor thermal power evolution in case1**

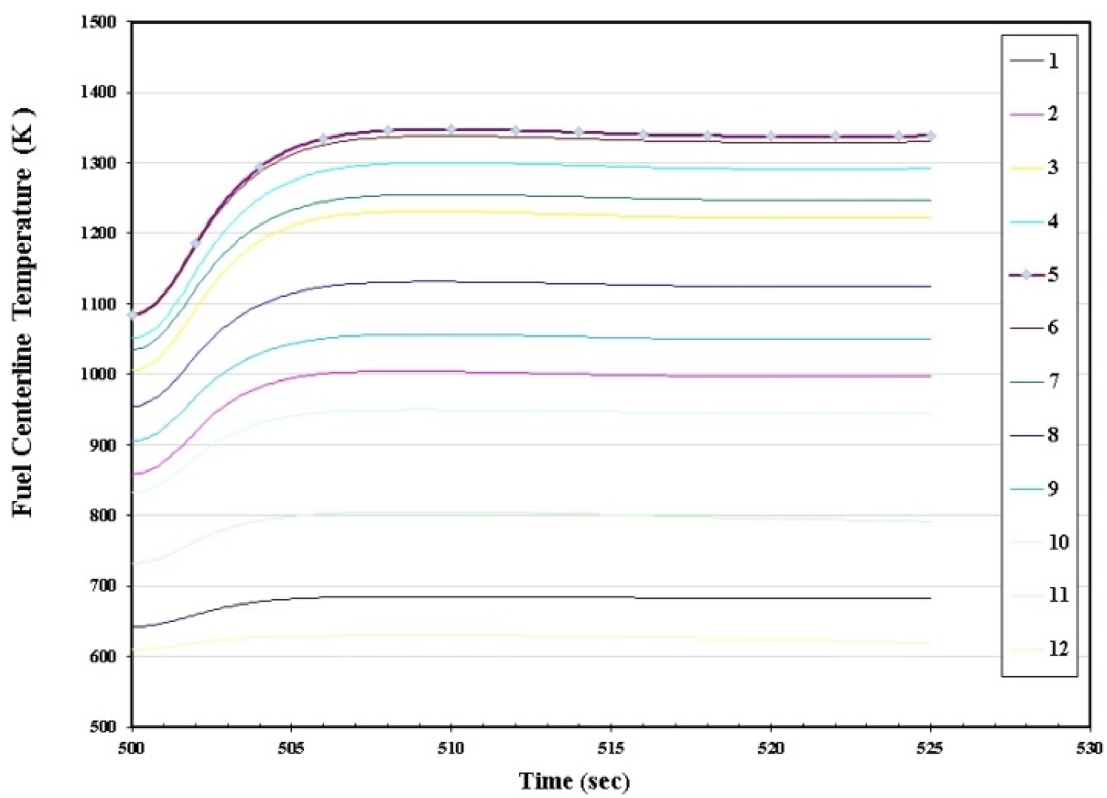



**Figure 12 Fuel centerline temperature at each node from bottom to top of fuel in case1**

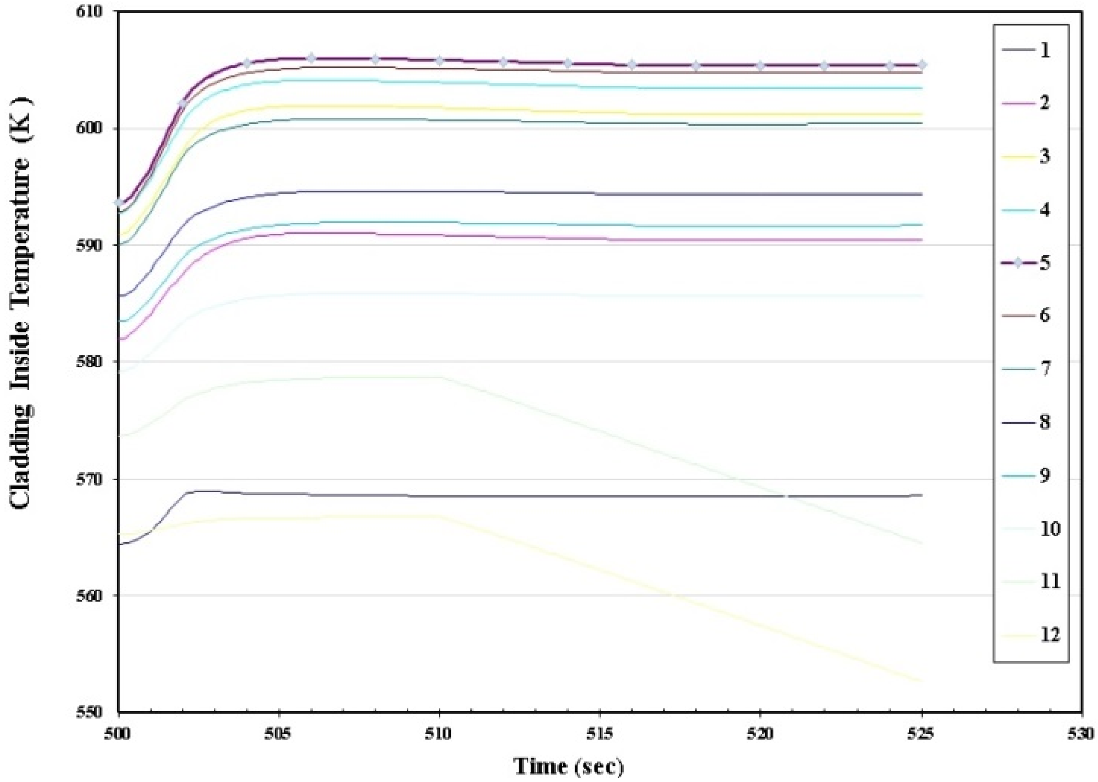



**Figure 13 Cladding Inside temperature at each node from bottom to top of fuel in case1**

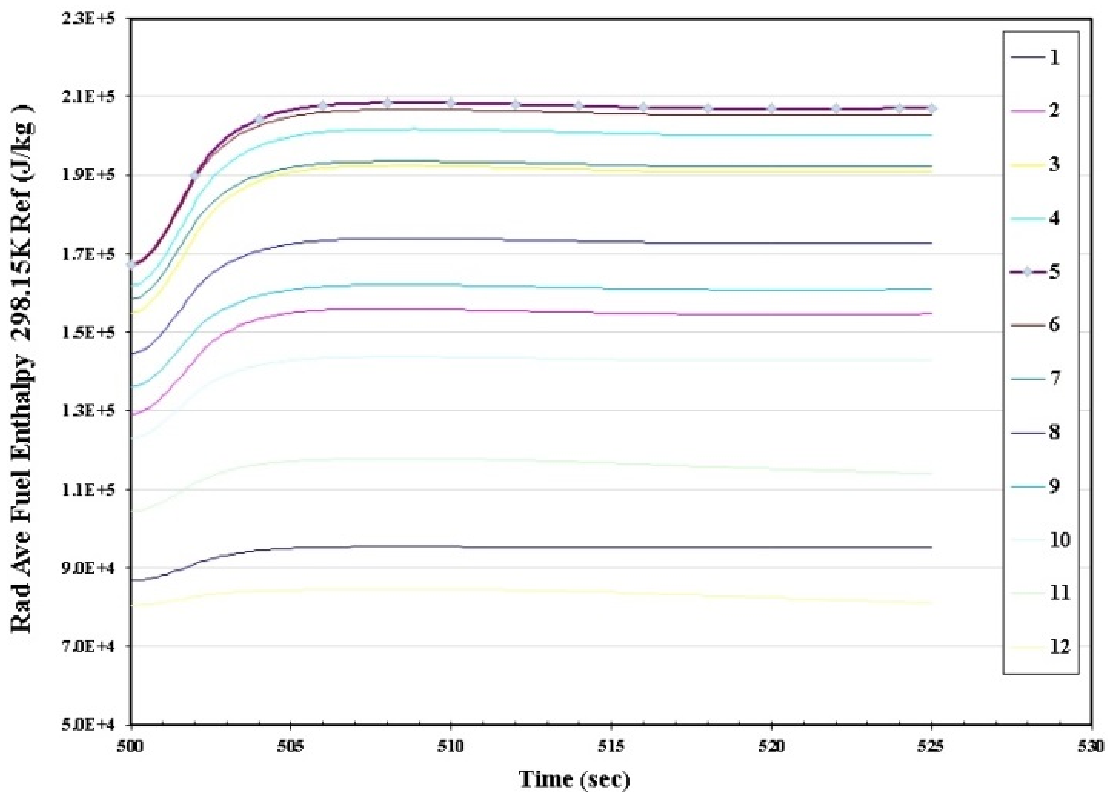



**Figure 14 Average fuel enthalpy at each node from bottom to top of fuel in case1**

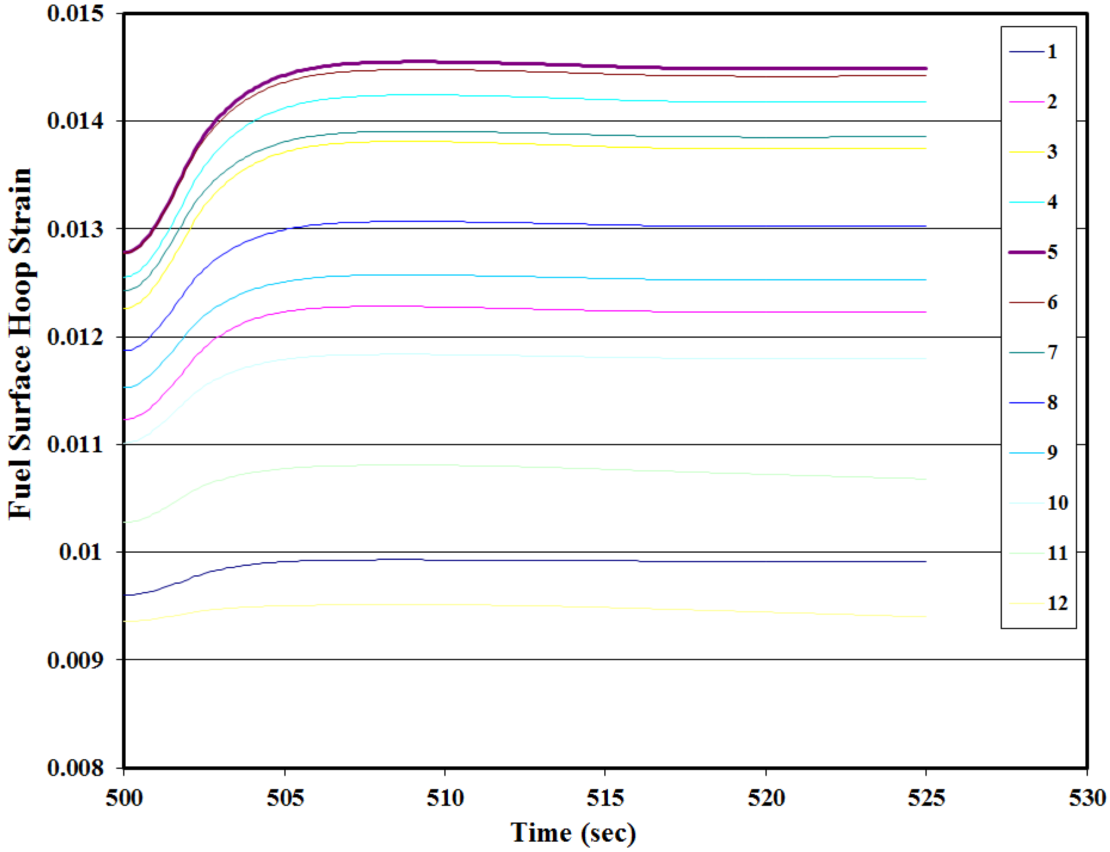



Figure 15 Fuel surface hoop strain in case1

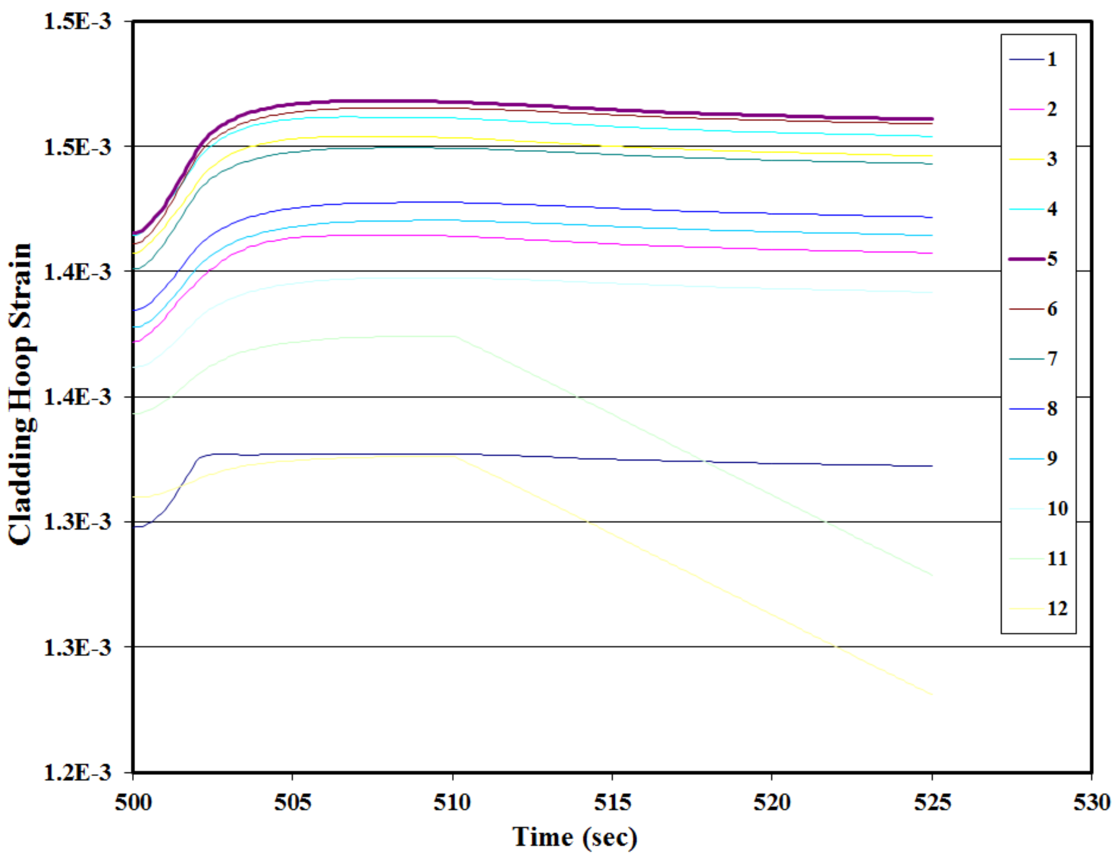



Figure 16 Cladding hoop strain in case1

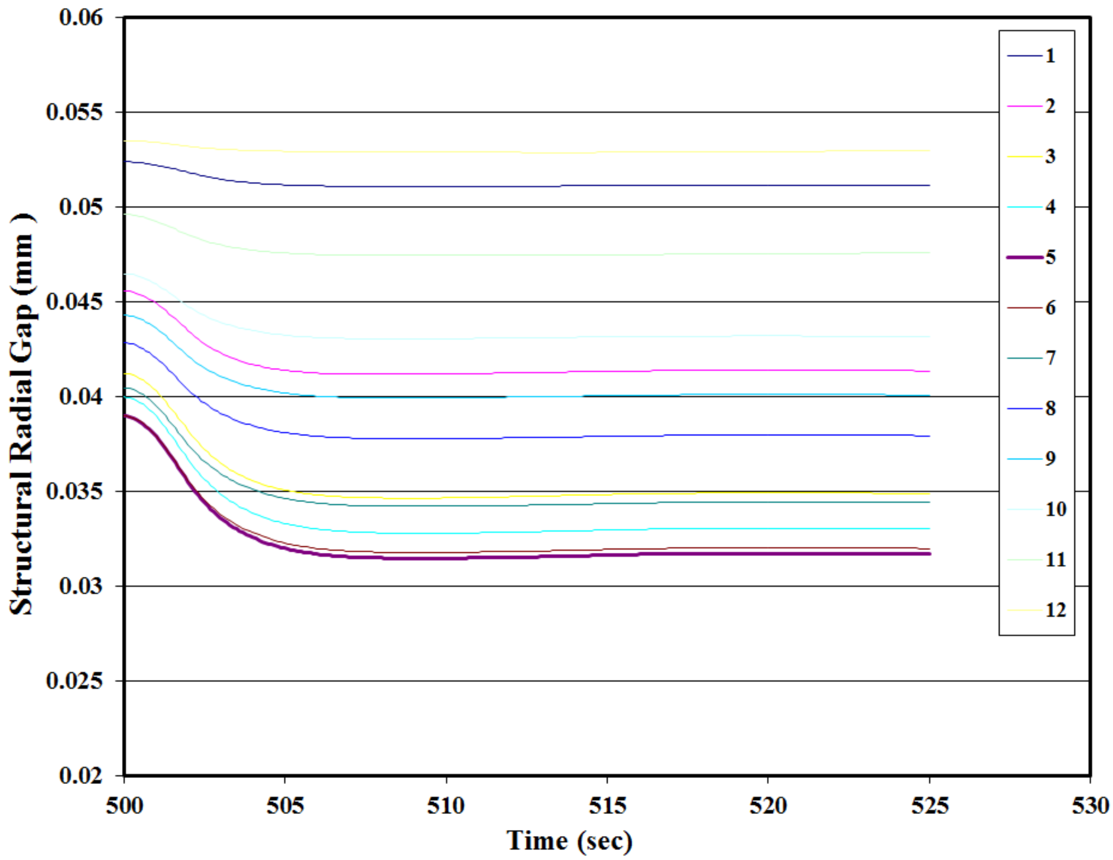



**Figure 17 Structural radial gap in case1**

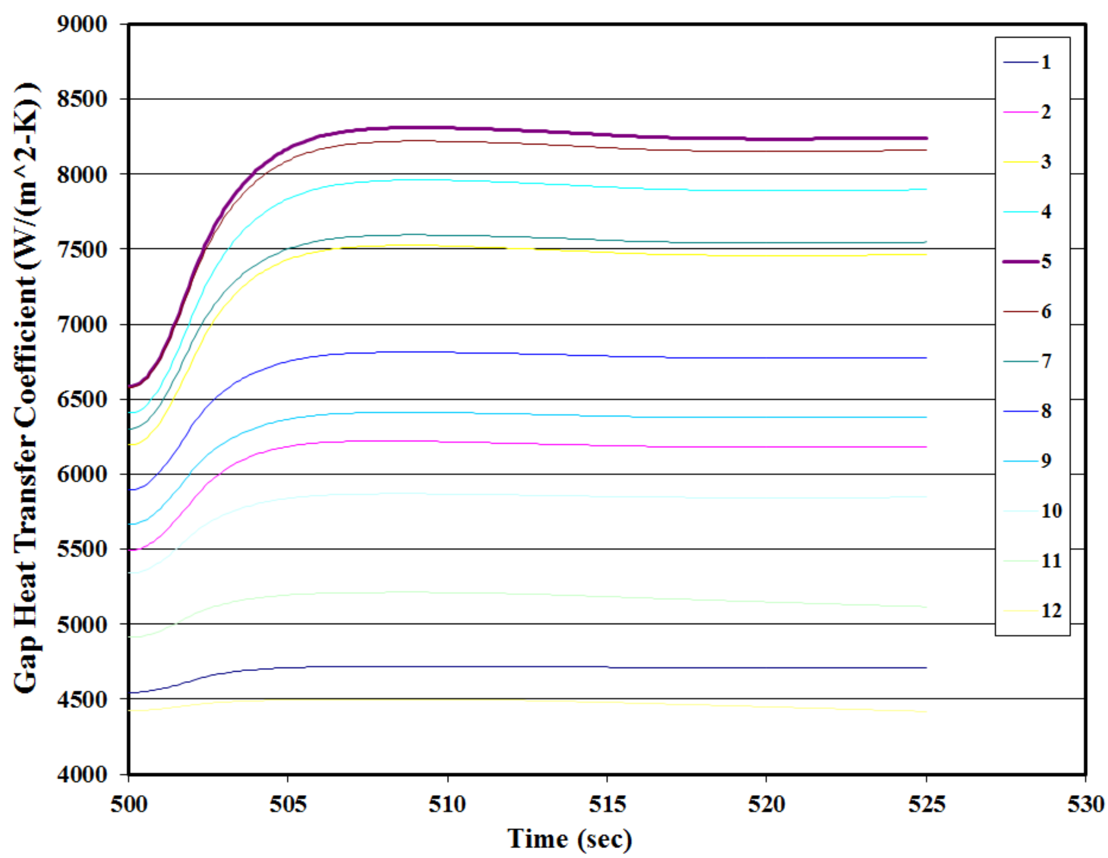



**Figure 18 Gap heat transfer coefficient in case1**

### 3.2 Case2-HZP Condition at BOC

At 200sec, the selected control rod starts to fall. Figure 15 shows the trend of the different reactivity feedback components during the transient. As control rod drops, the reactivity raises. Because of the low fuel temperature and subcooled moderator under HZP condition, compared with HFP condition, the effects of fuel and moderator feedback in reactivity are delayed, and the power level increases obviously about 3.3sec after transient starts, as shown in Figure 16. Figure 17 shows that the local boiling around the dropped control occurs at 203.4sec. At the same time, the moderator density starts to affect the reactivity. Compared with the effect of fuel temperature (starting at 202.5sec), T-H feedback seems to need a longer time, for fuel-to-cladding-to-coolant transfer, to be effective than at HFP condition. The negative feedback-reactivity from fuel and moderator suppresses the reactivity increase and lets the reactor drop below prompt critical, terminating the power excursion. The maximum value of power is 22646MWt (about 577% rated power) at 203.5sec. Finally, the reactor power is significantly reduced by reactor scram.

Figure 18, Figure 19, and Figure 20 show the fuel centerline temperature, average cladding temperature and average fuel enthalpy at each node form bottom to top of fuel. The maximum value of fuel centerline temperature, cladding inside temperature, and average fuel enthalpy are 3113.2K (at node 10; elevations~3.02m), 1405.7K (at node 8; elevations~2.38m), and $1.053\times10^{6}$J/kg (251.67cal/g at node 9; elevations~2.70m), respectively, which occurs at where the fuel rod is more reactive at HZP condition, as shown in Figure 5 (b).

The fuel temperature, cladding temperature, and maximum fuel enthalpy all reached the limitation of criteria, indicating that the fuel rod nearby the dropped control rod failed in case2. Figure 21 and Figure 22 show that both pellets and cladding expand during the transient, but the expansion of pellets is faster than that of cladding. Thus, the gas gap, between the pellets and cladding, is decreased, as shown in Figure 23. Figure 24 shows the gap heat transfer coefficient. As the gap is shortened, the heat transfer becomes better and has maximum value when the pellet touches the cladding inner-surface. Finally, the cladding melts due to high temperature (>1473.15K). In addition, the fuel centerline temperature also exceeds the criteria (3078.15K), indicating that there is pellet melting in the center of fuel.

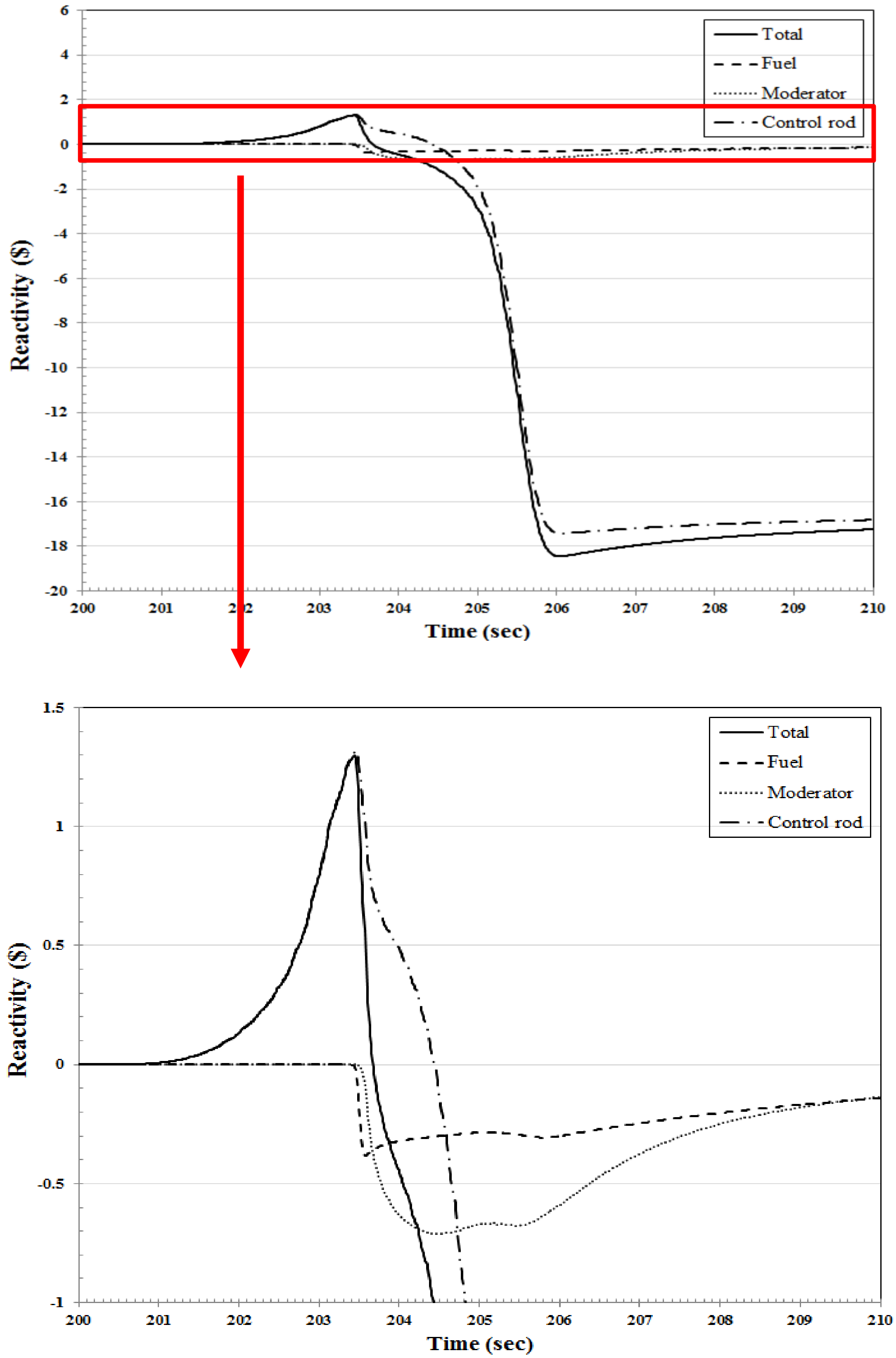



**Figure 19  Reactivity components evolution in case2**

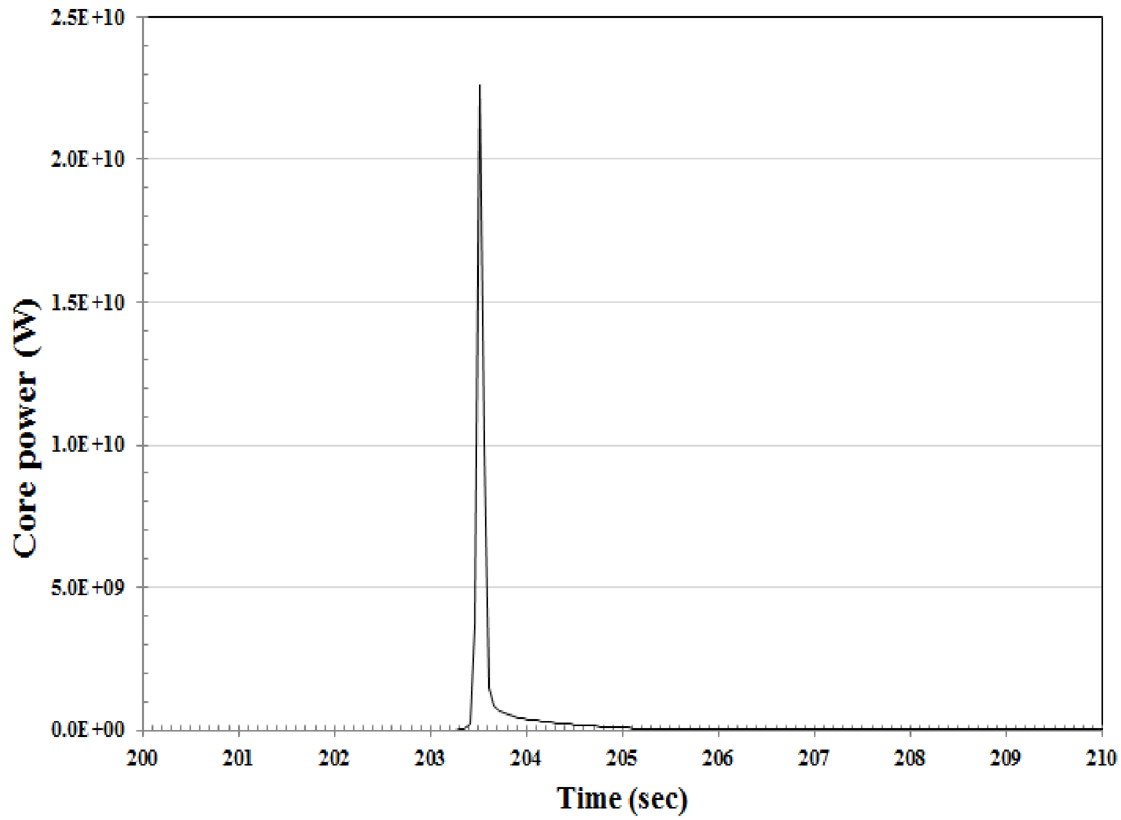



**Figure 20　Reactor thermal power evolution in case2**

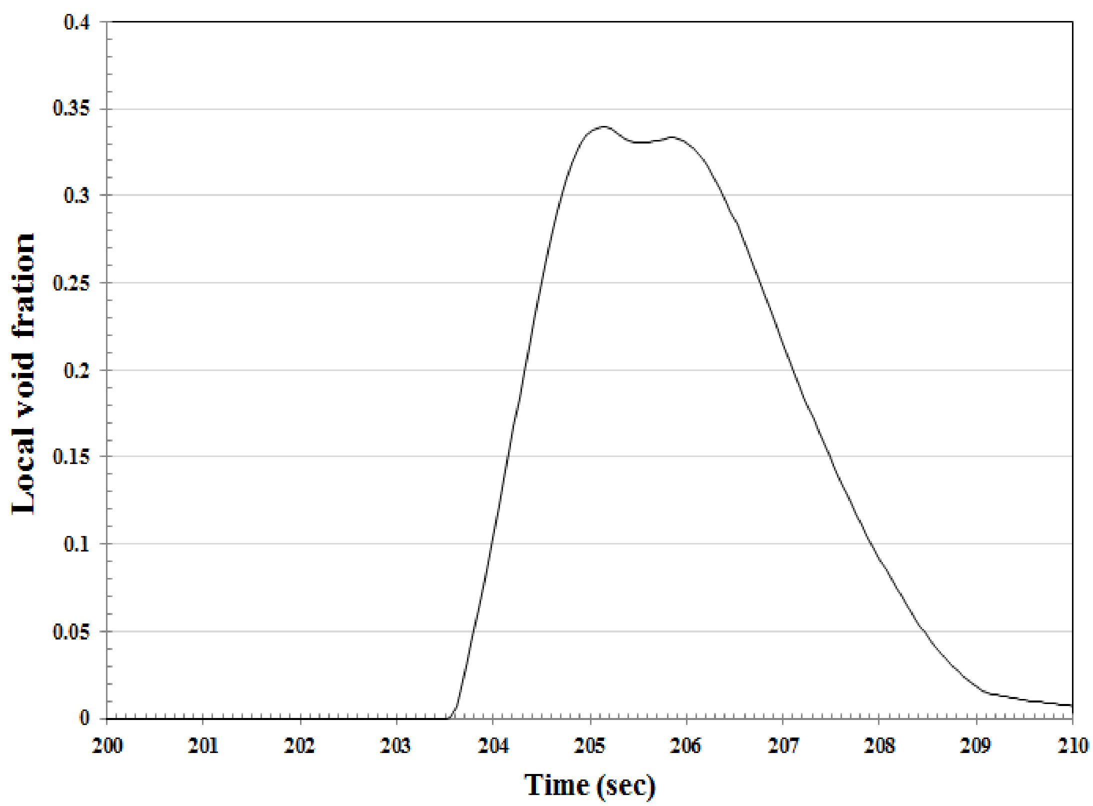



**Figure 21　Local void fraction around the dropped control rod in case2**

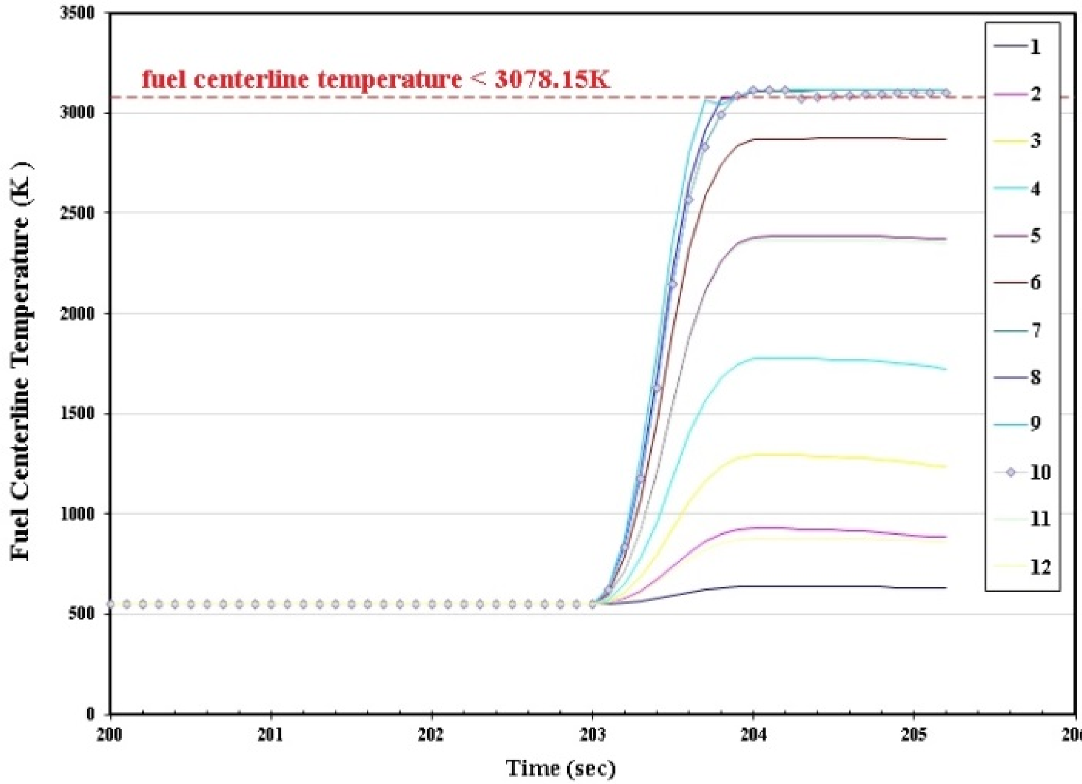



**Figure 22 Fuel centerline temperature at each node form bottom to top of fuel in case2**

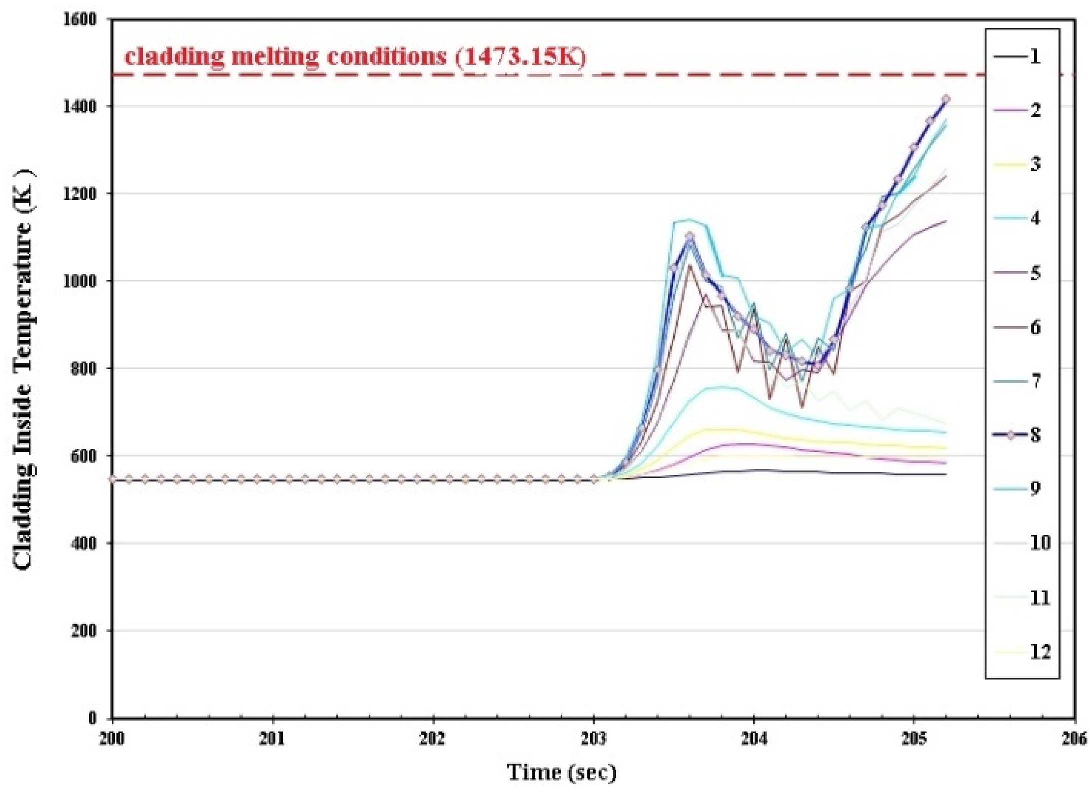



**Figure 23 Cladding inside temperature at each node form bottom to top of fuel in case2**

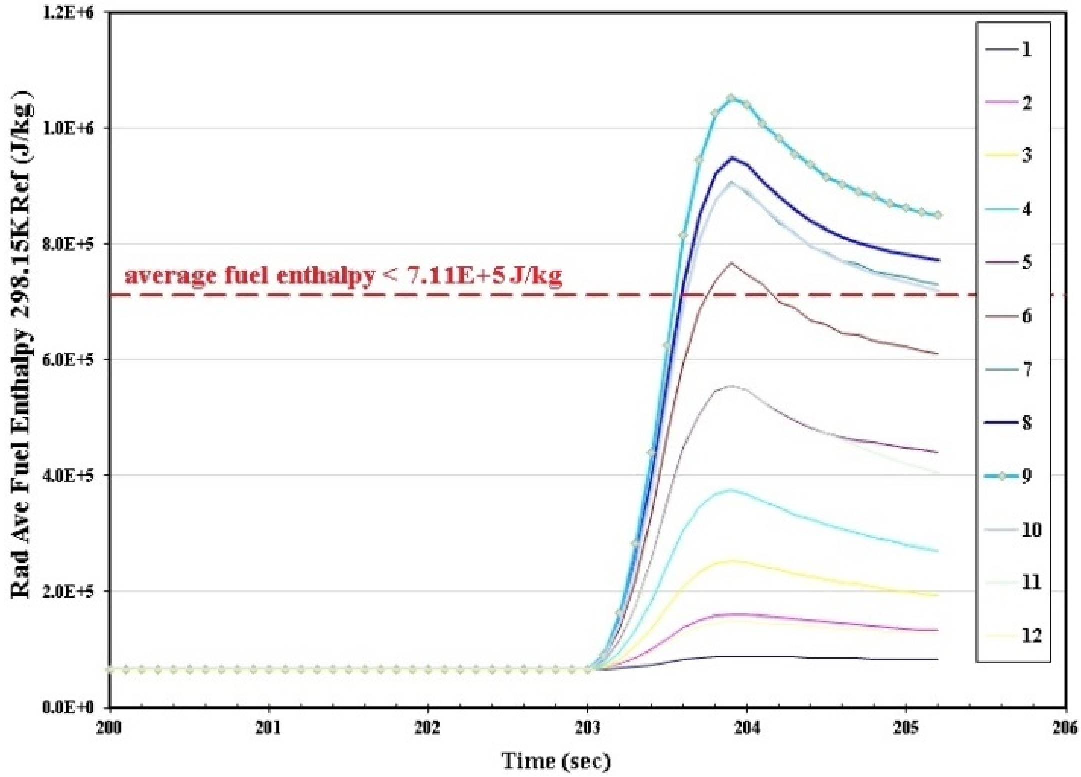



**Figure 24 Average fuel enthalpy at each node form bottom to top of fuel in case2**

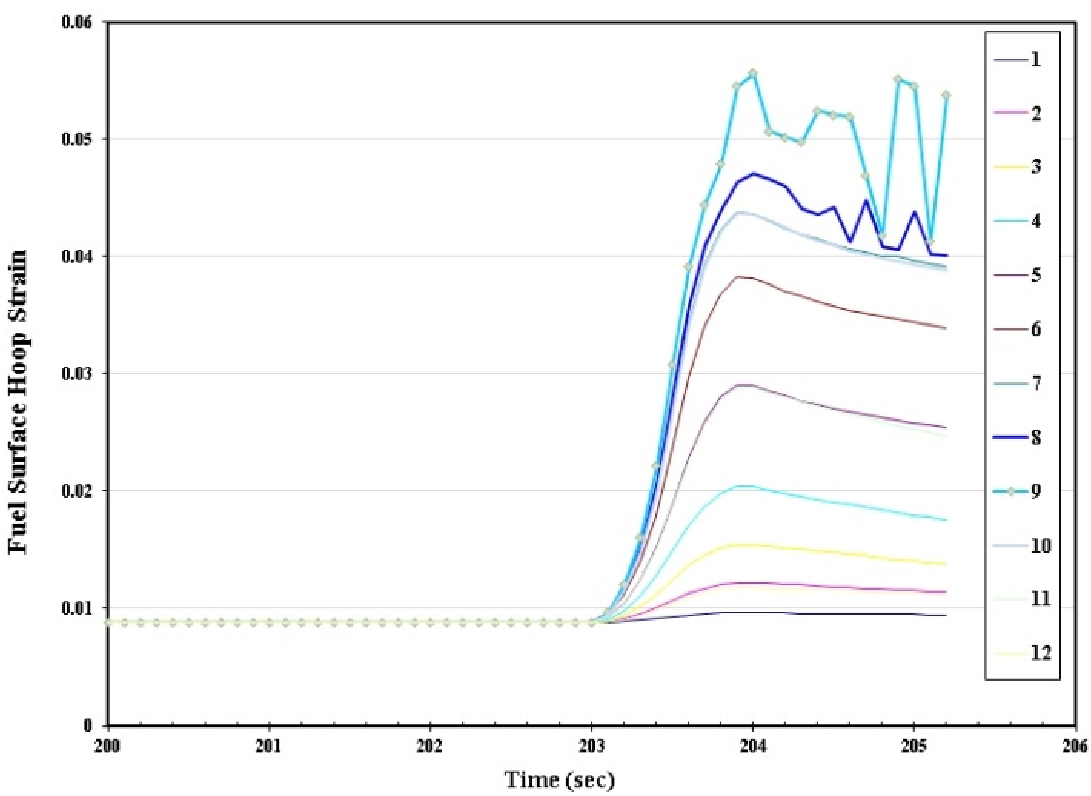



**Figure 25 Fuel surface hoop strain in case2**

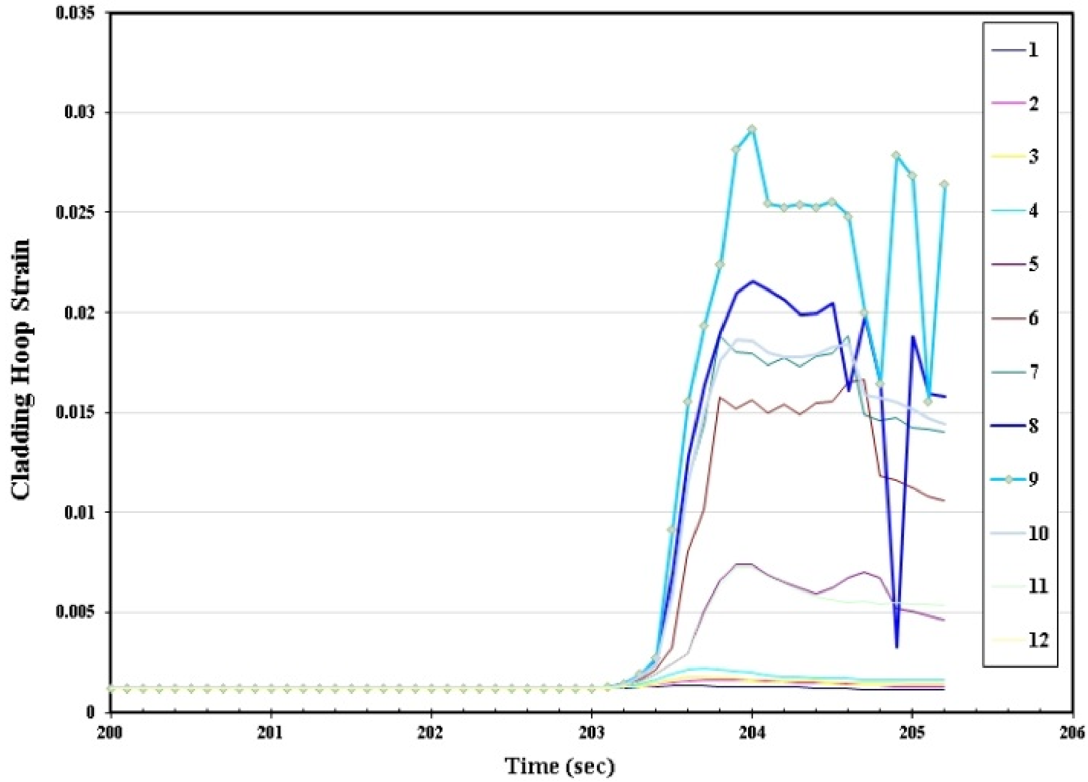



**Figure 26    Cladding hoop strain in case2**

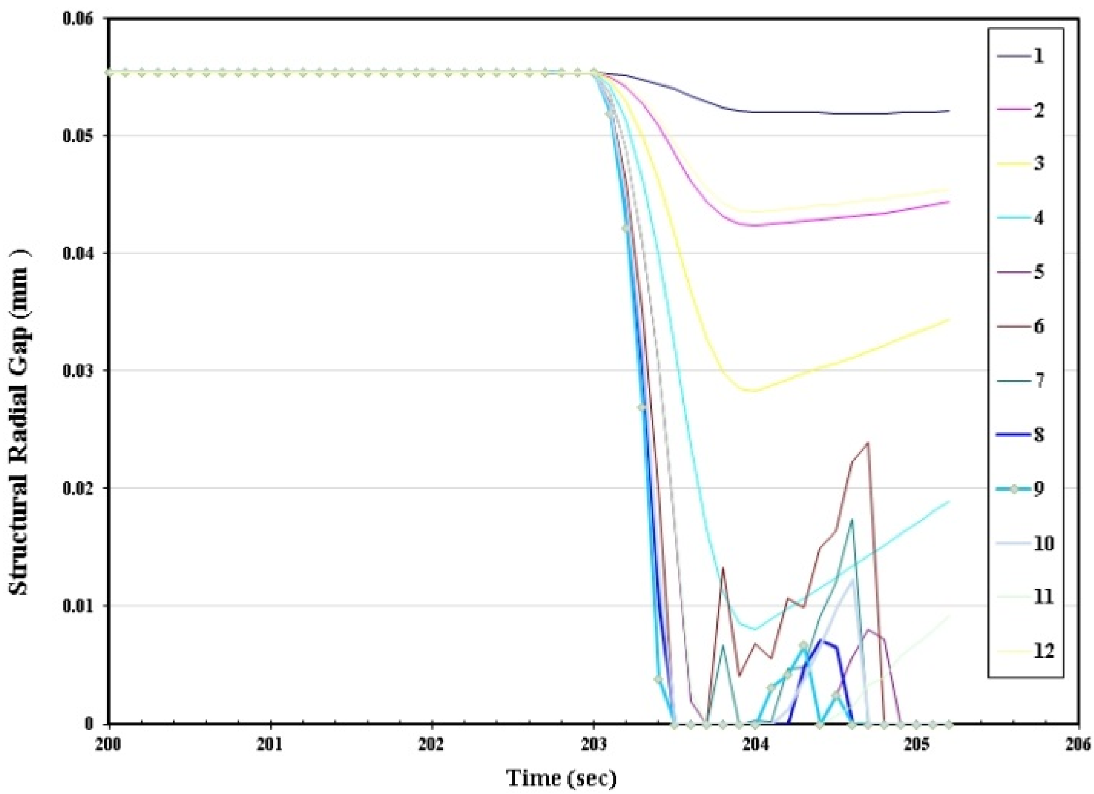



**Figure 27    Structural radial gap in case2**

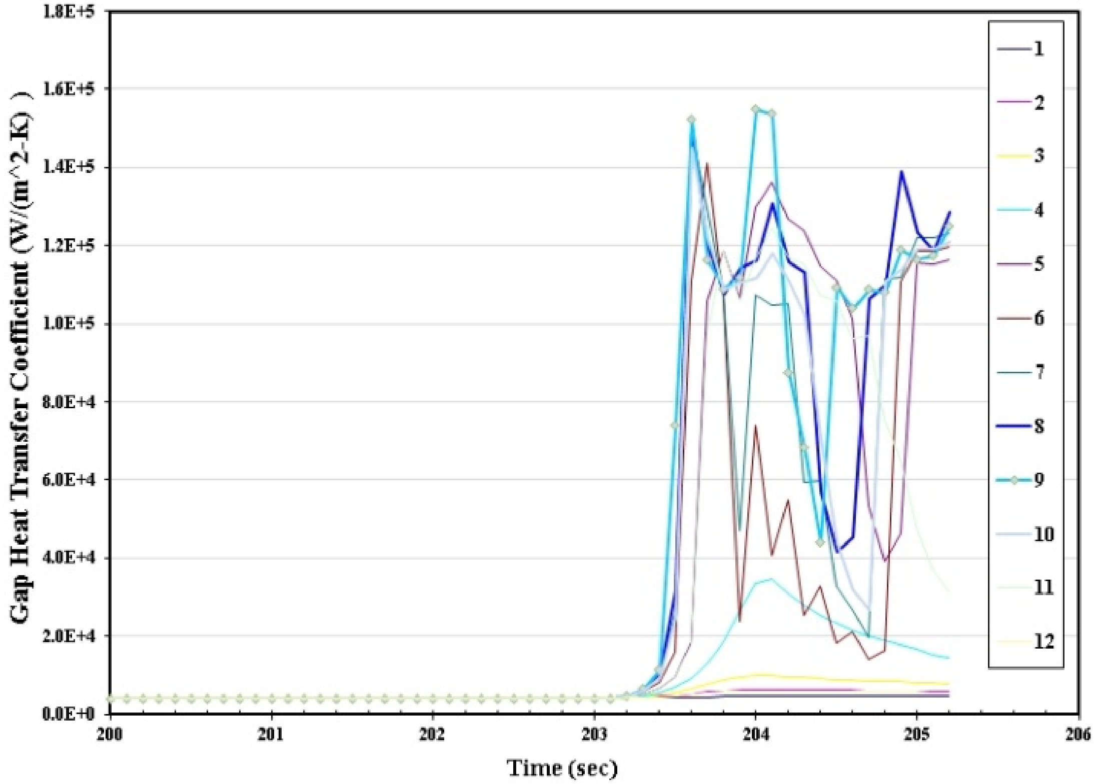



**Figure 28 Gap heat transfer coefficient in case2**

# 4. CONCLUSIONS

The purpose of this report is to understand the realistic behavior in Lungmen ABWR during a CRDA transient. The control rod drop inserts a positive reactivity, leading the reactor power to rise. Conversely, the increase of fuel temperature and decrease of moderator density feedback a negative reactivity to suppress the reactivity increase and to let the reactor drop below prompt critical, terminating the power excursion. In case2, because of the low fuel temperature and subcooled moderator under HZP condition, compared with HFP condition, the effects of fuel and moderator feedback in reactivity are delayed, and the power level increases obviously about 3.3sec after transient starts. Moreover, T-H feedback seems to need a longer time, for fuel-to-cladding-to-coolant transfer, to be effective at HZP condition than at HFP condition.

In case1, the fuel temperature, cladding temperature, and maximum fuel enthalpy are well below the limitation of criteria, indicating that no fuel failure occurs under HFP condition at BOC. In case2, the fuel rod nearby the dropped control rod was found to fail in CRDA analysis. And the FRAPTRAN data exposes that the main reason of rod failure is the cladding high temperature.

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11. ABSTRACT (200 words or less)
The purpose of this report is to understand the realistic behavior in Lungmen ABWR (Advanced Boiling Water Reactor) during a control rod drop accident (CRDA) transient. The CRDA transient would lead the reactor through an extremely fast and localized power excursion, requiring an accurate core modeling. The CRDA analysis for Lungmen ABWR was performed by coupling the 3D neutron kinetic code, PARCS, and two-phase thermal-hydraulic (T-H) code, TRACE. After TRACE/PARCS coupling calculation, the output data from TRACE/PARCS would be putted into FRAPTRAN code as boundaries, such as a function of time-dependent fuel rod power and coolant boundary conditions, to calculate the fuel damage. The CRDA analysis for Lungmen ABWR was performed for two conditions: a) case1: hot-full-power (HFP) at beginning of cycle (BOC); b) case2: hot-zero-power (HZP) at BOC. Under these conditions, the damage mechanisms of fuel rod are: 1) cladding ballooning and burst; 2) embrittlement and failure by high-temperature oxidation; 3) melting of cladding and/or fuel pellets. And the relevant quantities for fuel performance are the maximum fuel enthalpy and the melting temperatures of cladding and fuel pellet. The results of CRDA analysis show that a) case1: no fuel failure occurs under HFP condition at BOC; b) case2: the fuel rod nearby the dropped control rod failed under HZP condition at BOC, and the FRAPTRAN data exposes that the main reason of rod failure is the cladding high temperature.



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