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PRESSURE RESPONSE ANALYSIS WITH DISPERSAL OF FAILED FUEL DURING RIA USING RETRAN-3D

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1. Introduction

The main safety concerns in reactivity-initiated accidents (RIA) are loss of long-term core coolability and possible damage to the reactor pressure boundary and the core through pressure wave generation. During a RIA in PWR, a large and instantaneous amount of energy is deposited inside of fuel rods. Per Reference 2, the following factors are important to address the fuel rod behavior under a RIA:

- Characteristics of the power pulse
- Core coolant conditions
- Burnup- dependent state of the fuel rod
- Fuel rod design

And reference 2 explains 4 potential failure modes for the fuel rod under a RIA:

(1) Low-temperature failures by the pellet-cladding mechanical interaction (PCMI) under the early heat-up stage of the accident
(2) High-temperature failures by cladding ballooning and burst
(3) Failures by disruption of the cladding upon quenching from high temperature
(4) High-temperature failures by melting of the cladding

Failure modes of (1) and (3) can lead to fuel dispersal. For LWR fuel rods subjected to a RIA, the extensive experiments were performed in the NSRR, CABRI, IGR, and BIGR [2]. The experiments showed that the fuel dispersal occurs in connection with PCMI-type cladding failure. The thermal energy of fuel particles, expelled into the coolant from failed fuel rods, is rapidly converted to mechanical energy in the form of destructive pressure pulses. The coolant pressure pulses may damage nearby fuel assemblies, other core internals and ultimately also the reactor pressure vessel.

In this study, the RETRAN-3D code [1] is adopted to model numerically the pressure shock phenomena in the nuclear power plant. The RETRAN-3D code has proven to be a versatile and reliable computer program for use in best-estimate transient thermal-hydraulic analysis of light water reactor systems. The RETRAN-3D code requires numerical input data that completely describe the components and geometry of the system being analyzed. The input data include fluid volume sizes, initial flow, pump features, power generation, heat exchanger properties, and material compositions. The pressure response model for the fuel dispersal is constructed using these RETRAN-3D features. The following section will describe the detail RETRAN-3D model to analyze the pressure response due to the fuel dispersal.

2. Calculational Model

The followings will describe the important RETRAN-3D model and assumptions to analyze the pressure response due to the fuel dispersal. The constructed model contains whole nuclear power plant components such as reactor vessel, piping system, pressurizer, reactor coolant pumps and steam generators, etc.

2.1 Primary System

2.1.1 Reactor Core

The reactor core model is essential to analyze the pressure response during RIA. In this study, the extensive parametric studies were performed to search the appropriate reactor core model for the PCMI-induced pressure shock. One of these efforts is the sensitivity study with the number of axial nodes in reactor core. According to the results of sensitivity calculation, the number of axial nodes can be set by the amount of fuel failures to analyze the PCMI-induced pressure shock. In Figure 1, volumes 217, 240, 241, 105, 140 and 141 represent the coolant channels in active core. Volumes 218 and 106 means inactive core like the upper core plenum. Heat conductors 2, 240, 241, 1, 140 and 141(red-colored numbers) represent the fuel assembly groups. The left side of heat conductors represent the intact core region and the right side of heat conductors represent the failed fuel region. Therefore, the fuel fragments will be dispersed into the right side of coolant channel and the pressure wave will be generated in the same location.

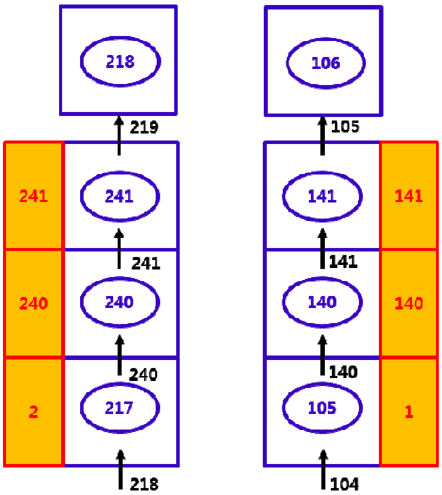

Fig. 1 Nodal scheme of reactor core

2.1.2 Reactor Coolant System (RCS)

The RCS includes the pressurizer, hot leg, steam generator tubes, cold leg and reactor coolant pumps. These components are not important with respect to the pressure shock over a very short period of time such as RIA. But these components should be considered to investigate the overall effect on the nuclear power plant system. Especially, the reactor coolant pump could make a pressure transient behavior different because it pressurizes the reactor coolant system. Therefore, the pump design data like pump homologous curves were used to construct the model of reactor coolant pumps. The pressurizer is unique due to two-phase region (steam and liquid). To simulate this unique region, the non-equilibrium and bubble rise model of RETRAN-3D code were used. The non-equilibrium model can simulate the steam and liquid regions, separately. Steam generator tubes provide the heat transfer from the primary to secondary system. The amount of heat transfer can be negligible in the pressure response analysis during the short time.

2.2 Secondary System

Steam generator model is not important with respect to the pressure response during the short time. In this study, just one secondary volume model was adopted. The pool boiling heat transfer correlation was used to simulate the heat transfer from tube metal to steam generator secondary side. In steam generator secondary region, there are steam and liquid simultaneously. To model the thermal-hydraulic condition, the bubble rise model was used. The bubble rise model provides the way to initialize the steam generator level.

2.3 Core Power

When the control rod is ejected from the core, the fast and huge positive reactivity is induced and the reactor core power can be increased sharply. The representative core power profile was obtained from another calculation. The initial core power is 1.0 MWt. Figure 2 shows the time-dependent core power used in the study.

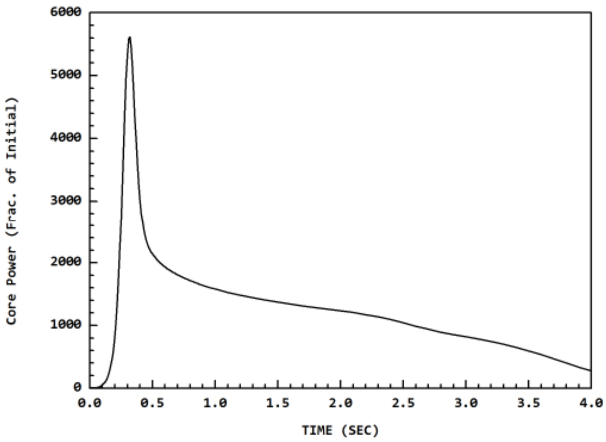

Fig. 2 Core power profile during rod ejection

2.4 Fuel Dispersal Model

PCMI-induced fuel failure produces a fine dispersal of pellet in coolant. In this situation, some of pellet fragments with high enthalpy are dispersed into the reactor coolant channel. To model this situation, non-conducting heat exchanger model is used. Using the non-conducting heat exchanger, it is possible to specify the power pulse width of fuel dispersal. The power pulse width affects on the energy transfer rate from the fuel dispersal to the reactor coolant. Fig. 3 shows the time-dependent power pulse of fuel dispersal with 100 msec. It is assumed that the most conservative fuel enthalpy and fuel failure was used to calculate the energy of fuel dispersal.

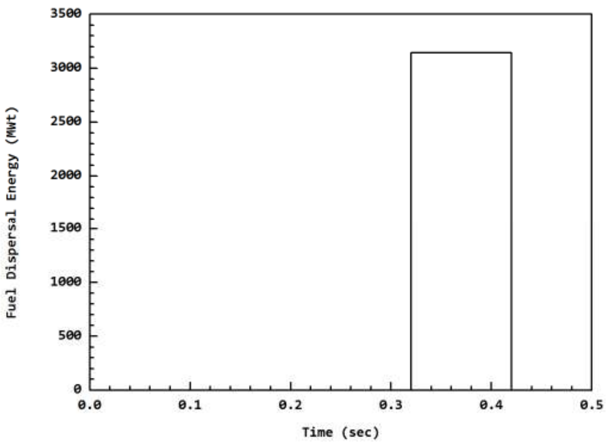

Fig. 3 Time-dependent power pulse of fuel dispersal

3. Results of Parametric Study

The parametric studies were performed for the several parameters such as the inlet core temperature, initial system pressure, initial reactor coolant flow and axial power shape. The followings show the calculation results.

3.1 Inlet Core Temperature

To investigate the effect of inlet core temperature on the pressure response, the sensitivity calculation was performed for the temperature range from 550 to 563 ˚F. Table 1 shows the peak pressure at the core lower plenum with inlet core temperature. The results showed that the peak pressure at lower plenum was increased with the rate of 16.3 psi/˚F.

Table 1. Sensitivity Results of Inlet Core Temperature

Location	Inlet Core Temperature (˚F)	
	550	563
Lower Plenum	2478	2690
Upper Plenum	2355	2442

3.2 Initial System Pressure

To investigate the effect of initial system pressure on the pressure response, the sensitivity was performed for the pressure range from 2175 to 2325 psia. Table 2 shows the peak pressure at the core lower plenum with initial system pressure. The results showed that the peak pressure at lower plenum was decreased with the rate of -0.86 psi/psi.

Table 2. Sensitivity Results of Initial System Pressure

Location	Initial System Pressure (psia)		
	2175	2250	2325
Lower Plenum	2690	2642	2561
Upper Plenum	2442	2612	2381

3.3 Initial RCS Flow

To investigate the effect of initial RCS flow on the pressure response, the sensitivity was performed for the RCS flow range from 95 to 116%. Table 3 shows the peak pressure at the core lower plenum with initial RCS flow. The results showed that the peak pressure at lower plenum was decreased with the rate of +6.71 psi/% of nominal flow.

Table 3. Sensitivity Results of Initial RCS flow

Location	Initial RCS flow (% of Nominal)		
	95	100	116
Lower Plenum	2831	2737	2690
Upper Plenum	2645	2734	2442

3.4 Axial Power Shape

To investigate the effect of axial power shape on the pressure response, the sensitivity was performed for the axial power shapes of -0.3, 0.0 and +0.3. Fig. 4 shows the axial power shapes used in sensitivity study. Table 4 shows the peak pressure at the core lower plenum with axial power shape. The results showed that the peak pressure at lower plenum occurred at +0.3 case.

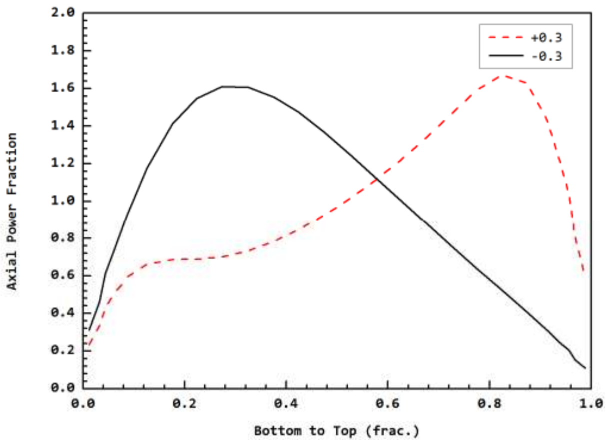

Fig. 4 Axial Power Shapes for Sensitivity Study

Table 4. Sensitivity Results of Axial Power Shapes

Location	Axial Power Shape		
	+0.3	flat	-0.3
Lower Plenum	2757	2745	2697
Upper Plenum	2465	2472	2474

3.5 Determination of Limiting Condition

According to the results of sensitivity studies, the most conservative condition can be determined as shown in Table 5. The higher temperature, lower pressure and lower flow will maximize the amount of steam generation during RIA. The amount of steam generated is most important factor in the pressurization of reactor coolant system by fuel dispersal.

Table 5. Limiting Condition

Items	Values
Inlet Core Temperature (˚F)	563
Initial System Pressure (psia)	2175
Initial RCS flow (% of Nominal)	95
Axial Power Shape	+0.3

Using the limiting conditions determined above, the final calculation was performed. Table 6 shows the peak RCS pressure at limiting condition. The maximum pressure of 2845 psia was predicted at core lower plenum and did not violate the acceptance criteria for RIA.

Table 6. Peak RCS Pressure at Limiting Condition

Location	Peak Pressure (psia)	Criteria (psia)
Lower Plenum	2845	3000
Upper Plenum	2648	3000

Fig. 5 shows the time-dependent behavior of pressure at core lower plenum. During the energy transfer from fuel dispersal to coolant, the peak pressure was produced. The pressure wave is maintained for about 1.0 second and gets stable after 1.5 seconds. Fig. 6 shows the time-dependent behavior of pressure at core upper plenum. Mostly it shows the same trend as the pressure at core lower plenum but the maximum pressure is lower than that of the lower plenum. Fig. 7 shows the time-dependent behavior of pressurizer pressure. The pressurizer pressure seems not to be affected by the fuel dispersal. It means that the pressure shock generated by the fuel dispersal is only local phenomena.

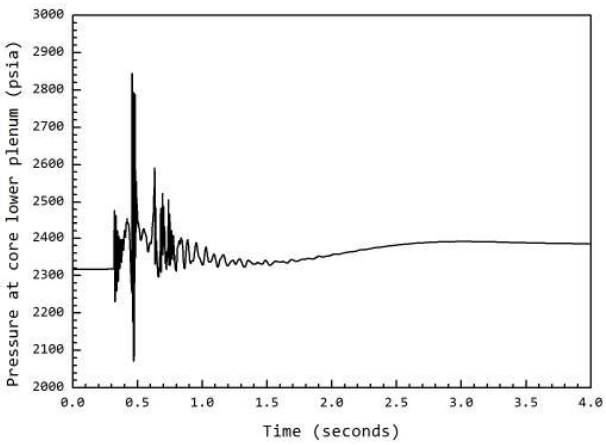

Fig. 5 Time vs. Pressure at Core Lower Plenum

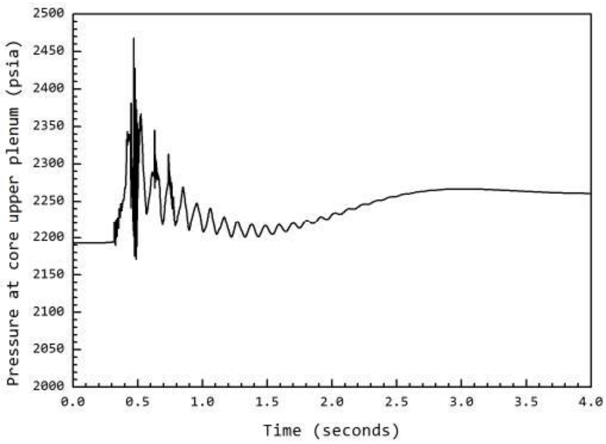

Fig. 6 Time vs. Pressure at Core Upper Plenum

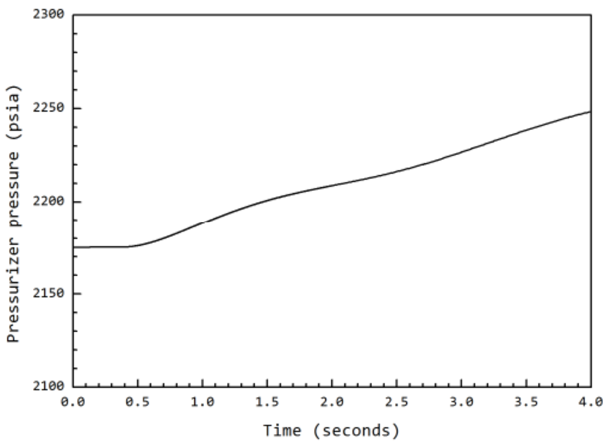

Fig. 7 Time vs. Pressurizer Pressure

Fig. 8 shows the void fraction at the fuel-dispersed location. The heat of fuel dispersal is transferred into coolant instantaneously and then 15% of total coolant volume was converted into steam. The steam is generated very fast during a moment and then collapsed. The instant generation of steam produces the pressure wave as shown in Figures 5 and 6.

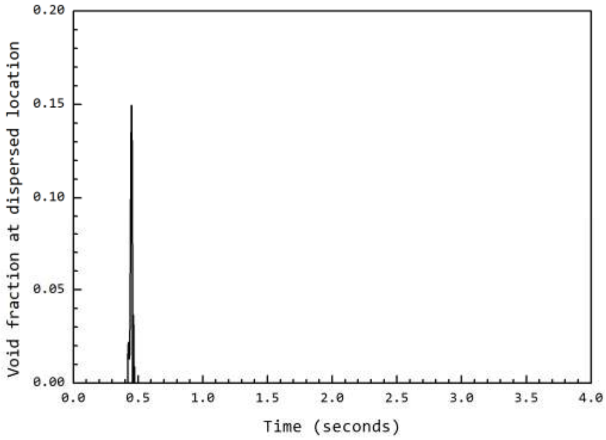

Fig. 8 Time vs. Void Fraction

4. Conclusions

In this study, the RETRAN-3D model of PWR nuclear power plant was constructed and developed to calculate the peak RCS pressure induced by PCMI fuel failure during RIA. Using this model, the parametric study was performed with the important parameters and the most limiting initial conditions and assumptions were determined. The higher temperature, lower pressure and lower flow will maximize the height of pressure pulse during RIA. Using the most conservative condition determined from the parametric study, the pressure shock analysis was performed to confirm that the maximum pressure is within the acceptance criteria. The calculation results showed that the peak RCS pressure was much lower than the allowable safety limit.

Acknowledgements

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Nomenclature

PWR	Pressurized Water Reactor
LWR	Light Water Reactor
NSRR	Nuclear Safety Research Reactor
IGR	Impulse Graphite Reactor
BIGR	Fast Impulse Graphite Reactor

References

[1] Computer Simulation & Analysis, Inc. "RETRAN-3D –A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," NP-7450(A), Volume 3 Revision 6.

[2] NEA No. 6847, "Nuclear Fuel Behaviour Under Reactivity-initiated Accident (RIA) Conditions".