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Rev.0
Japan Nuclear Technology Institute

Analysis on Criticality 'Accident' Occurred at Shika 1 of Hokuriku Electric Power Company

1. Overview

In March 2007, it was revealed that Shika 1 core had become criticality during outage due to unexpected withdrawal of 3 control rods in June 1999. Therefore, Japan Nuclear Technology Institute (JANTI) made core performance analyses using information provided from Hokuriku Electric Power Company (Hokuriku EPCO).

Analysis result showed that in the conservative condition of control rods withdrawal speed with the associated reactivity inserted (standard case), it was possible that the core had been in prompt criticality. The power decreased instantly (in 0.3 second) following rapid increase to 14% (230MW) of rated thermal power after 6 seconds of (delayed) criticality. The maximum enthalpy increase during peak power period was calculated to be 13cal/gUO_2, which is well below fuel PCMI failure threshold of 85cal/gUO_2(*1). Also, the maximum fuel enthalpy was calculated to be 49cal/gUO_2, which is below limit value of 230cal/gUO_2(*2) in accident or 92cal/gUO_2(*3) in abnormal transient during operation. In some cases such as with low control rod withdrawal speed, the core status did not result in prompt criticality, and stayed in delayed criticality.

*1) Threshold value for PCMI (Pellet-Cladding Mechanical Interaction) failure

*2) Threshold to prevent occurrence of mechanical energy by pressure impact resulted from fuel failure due to pellet melting and vaporization.

*3) Threshold to prevent fuel failure due to high temperature rapture, melting and nil-ductility of cladding

2. Analysis Condition

(1) Determination of Analysis Condition

Timeline of input parameter determination is shown in Table 1. In the determination process, sensitivity analysis was made on the associated parameter to focus on parameters with high priority, since exact value was not available at first. Then, after checking analyses condition of Hokuriku EPCO, analysis condition of standard case was determined. Also, variable range was determined for inserted reactivity and control rods withdrawal speed in analyses.

(2) Power Distribution in the Core

At Shika 1, criticality occurred as 3 out of 89 control rods were withdrawn. The situation can be understood that 'small partial core' was constituted inside the full core (Figure 1). The power distribution had shape of top peak as shown in Figure 2. In the partial core where control rods were withdrawn, 70% of power was generated in 4% of the full core volume. Kinetics importance of the partial core was estimated to be equivalent to the full core. Thus, JANTI analysis was made on the partial core shown in Figure 1.

(3) Inserted Reactivity and Control Rods Speed in Standard Case

In the analyses, inserted reactivity and control rods withdrawal speed were considered as parameter. For both parameters, basic values (standard case) were determined as follows.

- Keff in Standard Case: 1.0079
- Control Rods Speed in Standard Case: 47mm/s

Inserted reactivity (Keff) of the core in the standard case was determined based on the analysis by Hokuriku EPCO.

Control rods withdrawal speed in the standard case was determined as practically fastest speed based on mockup test by

Hokuriku EPFO.

(4) Affect due to Inserted Reactivity and Control Rods Speed

In standard case, inserted reactivity (0.0079 ΔK =1.3$) is above β (0.0060=1$). However, as extent of power increase depends on the reactivity insertion rate, it was not to be concluded that there was/was not prompt criticality occurred just based on value of inserted reactivity in excess of β. Therefore, sensitivity analyses were made to identify cases which result in prompt criticality.

For inserted reactivity, analyses were made on the core with higher/lower reactivity by 0.5$ each in addition to the standard case considering accuracy of analysis code.

- Standard Case: 1.3$
- High Reactivity: 1.8$ (0.5$ higher)
- Low Reactivity: 0.81$ (0.5$ lower)

For control rods movement, the following three withdrawal speeds were selected for analysis.

- Standard Case: 47mm/s
- High Speed: 76mm/s (normal operation speed of control rods)
- Low Speed: 16mm/s (assumed average speed of control rod (26-39) from 0pos. to 16pos. during 77 seconds between start of control rod(s) withdrawal signal and initiation of scram signal. (26-39) was at peaking power location as shown in Figure 2)

Reactivity insertion rate in each case is shown in Figure 3.

(5) Analysis Code

JANTI analyses were made using multi regional nuclear-thermal hydraulics combined kinetics code EUREKA-2. In EUREKA-2, both nuclear and thermal hydraulics feedback (Doppler feedback and coolant temperature feedback) can be treated simultaneously. In the analyses by Hokuriku EPCO, kinetics calculation was made by reactivity insertion events analysis code APEX considering Doppler feedback, followed by analysis using thermal hydraulics code SCAT with APEX result as input. Comparison of calculation method is shown in Table 3-1.

In the analyses by JANTI, EUREKA-2 code is used primarily because of its high performance during peak power period that is characteristic of reactivity insertion accidents. Meanwhile, EUREKA-2 tends to calculate conservative results for fuel enthalpy after power peak due to its assumption of constant power distribution, and due to its incapability of calculation in boiling condition etc. Therefore, in JANTI analyses, result of fuel enthalpy after power peak was considered as reference. Also, as EUREKA-2 is not capable of calculation in boiling condition, higher core pressure value was used for calculation to avoid boiling.

3. Analysis Result

(1) Analysis Result of Standard Case

Trend of power are shown in Figure 4-1 and Figure 4-2. In standard case, 1.1$ of net reactivity was inserted, and rapid power increase due to prompt criticality occurred 6 seconds after (delayed) criticality. But because of inherent reactivity feedback mechanism as shown in Figure 4-3, the power decreased instantly (in 0.3 second) following rapid increase to 14% of rated power (230 MW). Then, the power became stable about 0.3% of rated power (4MW).

The maximum enthalpy increase during peak power was calculated to be 13cal/gUO_2, which is well below fuel PCMI failure threshold of 85cal/gUO_2. The analysis results were similar to the ones by Hokuriku EPCO as shown in Table 2. The fuel enthalpy increased gradually to become maximum value of 49cal/gUO_2 after 10 seconds of rapid power increase, which is well below limit value of 230cal/gUO_2 during accident or 92cal/gUO_2 during abnormal transients during operation.

Maximum pellet temperature was about 700℃ as shown in Figure 4-4, and the maximum coolant temperature in the partial core was the boiling temperature at core outlet as shown in Figure 4-5.

(2) Affect of Inserted Reactivity

Calculation results are shown in Table 4, Figures 5-1 and 5-2. In each case, control rods withdrawal speed was considered to be 47mm/s. In the cases where reactivity of more than 1$ was inserted, the results were well below fuel PCMI failure threshold while prompt criticality was observed. In the case of large reactivity insertion, less than 0.1$ of difference was observed in the net inserted reactivity as compared with the standard case.

(3) Affect of Control Rods Movement Speed

Calculation results are shown in Table 5, Figures 6-1 and 6-2. While reactivity of 1.3$ were inserted, the net inserted reactivity was below 1$ in the case of low control rods withdrawal speed, and rapid power increase was not observed. In all cases, the results were well below fuel PCMI failure threshold.

(4) Affect of Analysis Model

Sensitivity analysis was made using "zero" coolant temperature coefficient. The result is shown in Table 3-2. No large affect due to difference in treatment of coolant temperature reactivity coefficient was observed.

4. Conclusion

Analyses were made on various conditions for criticality 'accident' occurred at Shika 1 using inserted reactivity and control rods withdrawal speed as variable parameters. Analyses result showed possibility of being prompt criticality in cases with conservative condition. On the other hand, in cases such as with low control rod withdrawal speed, the core did not result in prompt criticality, and stayed in delayed criticality. For all analyzed cases, the maximum enthalpy increase during peak power period was well below fuel PCMI failure threshold. Results of JANTI analyses were equivalent to the ones by Hokuriku EPCO. Also, the reference analyses result of maximum fuel enthalpy was below threshold value in accident or in abnormal transient during operation for all analyzed cases.

5. Attachment

(1) Summary of "Treatment of High Burn-up Fuels in Reactivity Insertion Accident of Light Water Reactor Generation Facilities"

(2) Summary of "Safety Analyses Review Guide for Light Water Reactor Generation Facilities"

(3) Summary of "Reactivity Insertion Accident Review Guide for Light Water Reactor Generation Facilities"

(4) Definition of Terms

6. Reference

"Report Regarding Criticality Accident at Shika 1 NPS"
(April 6, 2007, Hokuriku Electric Power Company)

"LWR Reactivity Insertion Accident Code EUREKA-2"
(JAERI-M 84-074, May 1984, Japan Atomic Energy Research Institute)

"BWR Analyses Method of Reactivity Insertion Accident"
(HLR-012R3, February 1999, Hitachi Co.)

Table 1 Timeline of Input Parameter Determination by JANTI

From March 20, 2007: Start consideration of core analysis

【Phase I (Preliminary analysis using 3 region core model)】

- Sensitivity Analysis using assumed Inserted Reactivity, Control Rod Withdrawal Speed, Reactivity Coefficient etc.

Parameter	Considered Value
Inserted Reactivity ($)	1.2, 1.4, 1.6
Control Rod Speed (mm/s)	10, 30, 60, 100
Doppler Coefficient (Δk/k/℃)	-2×10^{-5}
Coolant Temperature Coefficient (Δk/k/℃)	-1.0×10^{-4}
Coolant Speed (cm/s)	10

- Consideration of analysis condition of Hokuriku EPCO, analysis condition of standard case was determined as follows except for control rods withdrawal speed.

Parameter	Considered Value
Inserted Reactivity ($)	1.3
Control Rod Speed (mm/s)	Per mockup test
Doppler Coefficient (Δk/k/℃)	-2×10^{-5}
Coolant Temperature Coefficient (Δk/k/℃)	-4.0×10^{-5}
Coolant Speed (cm/s)	10*

*Determined by sensitivity analysis (4cm/s and 10cm/s).

From March 24, 2007

【Phase II (Analysis of Partial Core based on 3-D Core Analysis)】

- Setup of partial core with 5 horizontal & 10 axial regions.
- Sensitivity analysis of inserted reactivity and control rod speed.
  ① Inserted Reactivity ($): 0.81, 1.3, 1.8 etc.
  ② Control Rods Withdrawal Speed (mm/s):16, 47, 76 etc.
- Additional analysis with zero coolant temperature coefficient.

Table 2 Comparison of Analyses Input and Result

①Analyses Condition (Initial Condition)

	JANTI	Hokuriku EPCO
Analyzed Core	Partial Core (34 fuels)	Full Core (368 fuels)
Initial Power	0.7E-6%	1E-6%
Reactivity Excess	0.0079 Δk (Standard Condition)	0.0079 Δk (1)
Reactivity Feedback	Doppler Coefficient Coolant Temperature Coefficient	Doppler Coefficient

②Analyses Result

	JANTI (Standard Condition)	Hokuriku EPCO (1)	Threshold
Peak Power (Fraction to Rated Power)	14%	15%	–
Max. Enthalpy Increase during Peak Power [cal/gUO_2]	13	13	85 (2)
Max. Fuel Enthalpy [cal/gUO_2]	49	41	230 (3) 92 (3)

(1) Among conditions considered by Hokuriku EPCO, reactivity excess estimated by cold criticality test results was selected.

(2) Threshold value for fuel PCMI failure per "Treatment of High Burn-up Fuels in Reactivity Insertion Accident of Light Water Reactor Generation Facilities"

(3) Limit value in accidents/abnormal transients during operation per "Reactivity Insertion Accident Review Guide for Light Water Reactor Generation Facilities"

Table 3-1　Comparison of Analysis Method

Analysis by JANTI	Analysis by Hokuriku EPCO
・ Calculation of both nuclear and thermal hydraulics feedback (Doppler coefficient and coolant temperature coefficient) by EUREKA-2. ・ Input value of pressure was increased as calculation needed to be done with no void condition. ・ Accuracy of heat removal calculation is not comparable to SCAT. ・ Power distribution is constant.	・ Calculation of Nuclear feedback (Kinetics Calculation considering Doppler Coefficient) was calculated by APEX. ・ Input APEX result into thermal hydraulics code SCAT to calculate fuel thermal power. ・ Power distribution change is considered.

Table 3-2 Sensitivity Analysis of Affect of Coolant Temperature Coefficient

	Standard Condition	With "Zero" Coolant Temperature Coefficient
Net Inserted Reactivity [$]	1.11	1.11
Peak Power (Fraction to Rated Power)	14%	15%
Max. Enthalpy Increase during Peak Power [cal/gUO_2]	13	13
(Reference) Max. Fuel Enthalpy [cal/gUO_2]	49	53

Table 4 Analyses Result (Affect of Reactivity Inserted to the Core)

Analyzed Case	Large	Standard	Small
Inserted Reactivity	1.8$	1.3$	0.81$
Net Inserted Reactivity [$]	1.15	1.11	0.81
Peak Power (Fraction to Rated Power)	23%	14%	1%
Max. Enthalpy Increase during Peak Power [cal/gUO_2]	15	13	—
(Reference) Max. Fuel Enthalpy [cal/gUO_2]	66	49	—

- In all cases, control rods withdrawal speed is 47mm/s。

- In all cases, maximum enthalpy increase is well below 85cal/gUO_2, threshold value of fuel PCMI failure during reactivity insertion accident.

- Larger reactivity being inserted, increase of net inserted reactivity is small due to reactor core inherent feedback effect.

(Analysis Result by Hokuriku EPCO)

- Control Rod Withdrawal Speed: 47mm/s
- Peak Power: 15%
- Max. Enthalpy Increase during Peak Power: 13cal/gUO_2
- Max. Fuel Enthalpy : 41cal/gUO_2

Table 5 Analyses Result (Affect of Control Rod Withdrawal Speed)

Analyzed Case	High Speed	Standard	Low Speed
Control Rod Withdrawal Speed	76mm/s	47mm/s	16mm/s
Net Inserted Reactivity [$]	1.17	1.11	0.98
Peak Power (Fraction to Rated Power)	28%	14%	3%
Max. Enthalpy Increase during Peak Power [cal/gUO_2]	17	13	—
(Reference) Max. Fuel Enthalpy [cal/gUO_2]	50	49	—

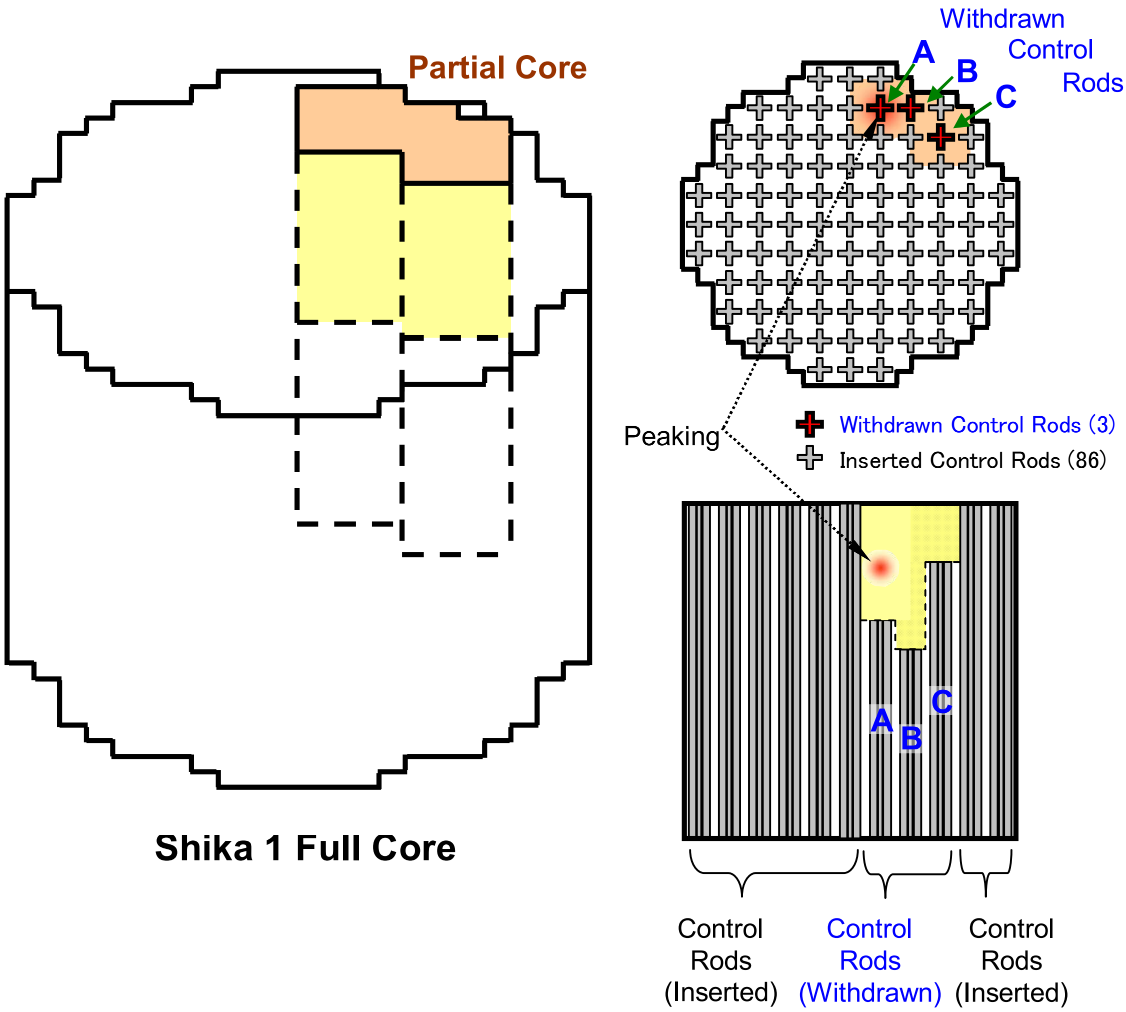
- In all cases, inserted reactivity is 1.3$.

- In all cases, maximum enthalpy increase is well below 85cal/gUO_2, threshold value of fuel PCMI failure during reactivity insertion accident.

- Even if reactivity of more than 1 $ is inserted by control rods withdrawal, net inserted reactivity can be below 1 $ based on withdrawal speed of control rods. (reactivity insertion speed)

(Analysis Result by Hokuriku EPCO)

- Control Rod Withdrawal Speed: 47mm/s
- Peak Power: 15%
- Max. Enthalpy Increase during Peak Power: 13cal/gUO_2
- Max. Fuel Enthalpy : 41cal/gUO_2



Analyzed Core (Partial Core)

- Number of Fuels : 34 of 368 (9%)
- Height : 10/24 from Top (40%)
- Volume : 4% of Full Core
- Heat Generatiion : 70% of Full Core
- Peaking of Partial Core : 3.5

Figure 1 Analyzed Partial Core

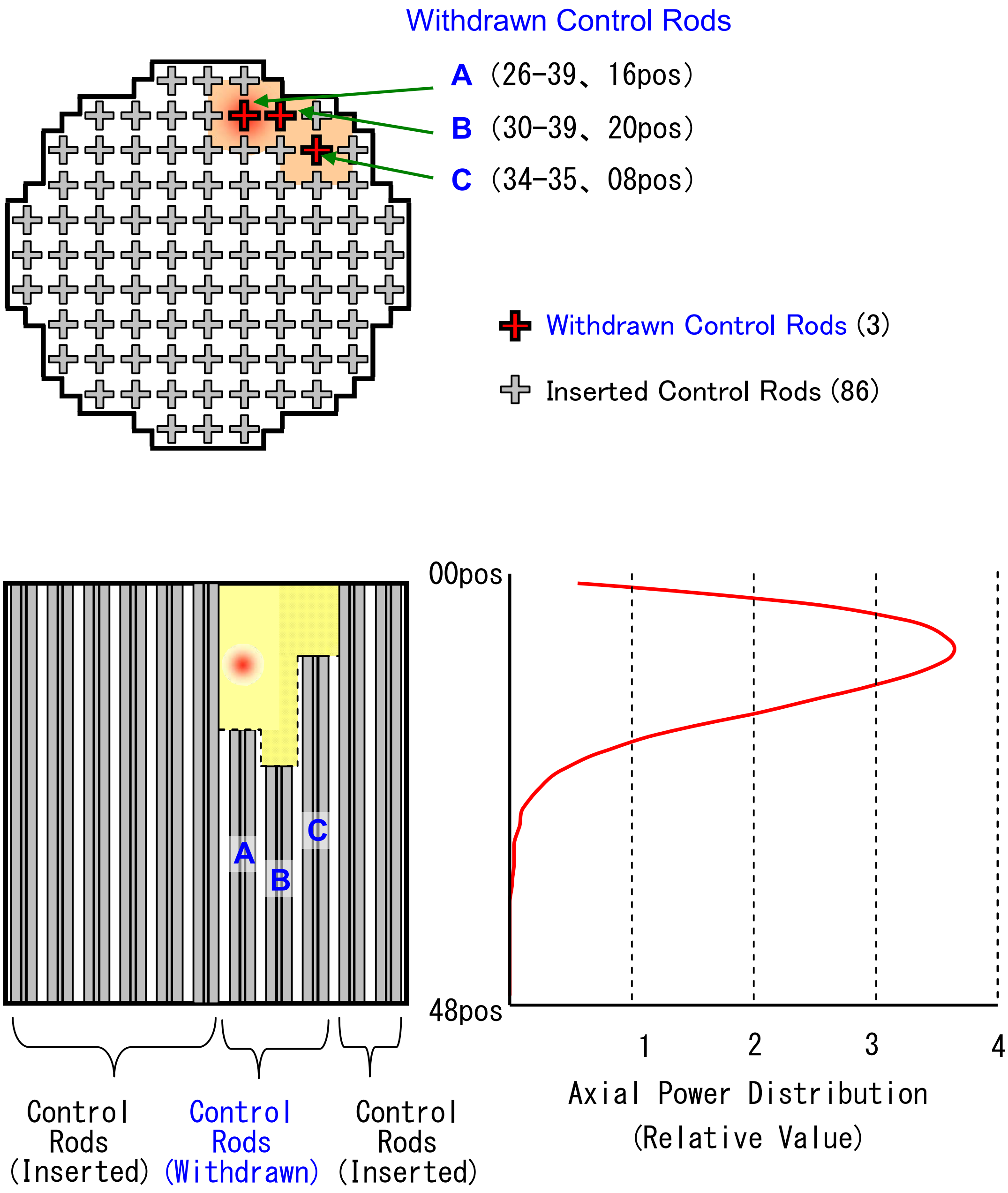


Figure 2 Power Distribution of the Core

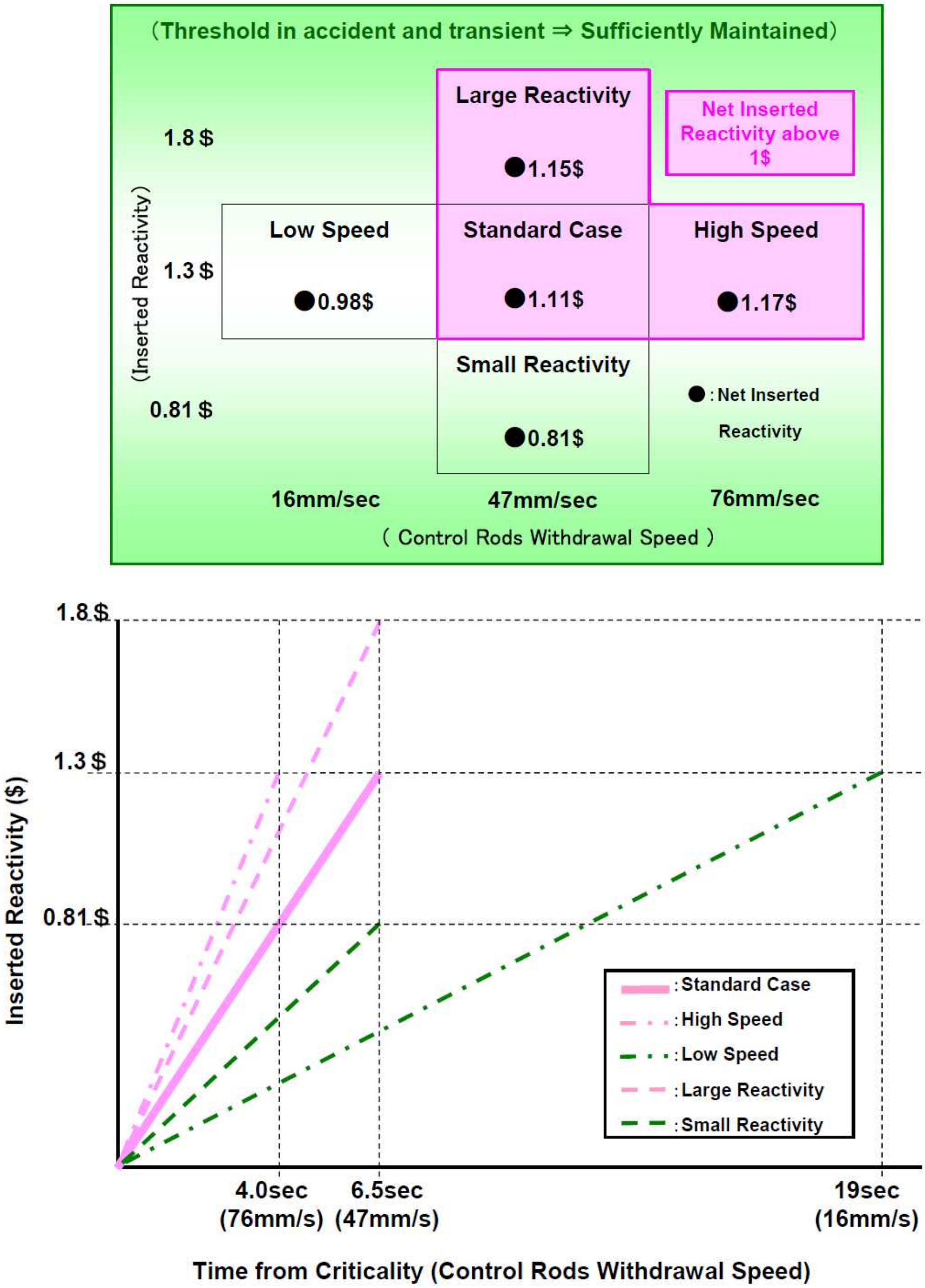


Figure 3　Reactivity Insertion Speed

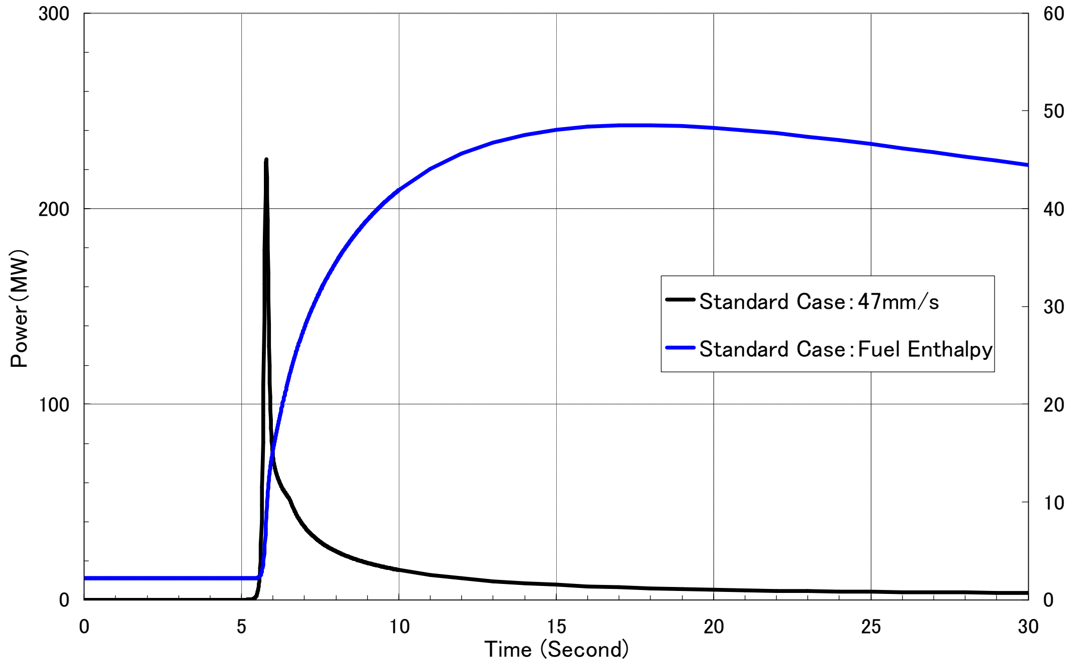


Figure 4-1　Trend of Power (Standard Case)

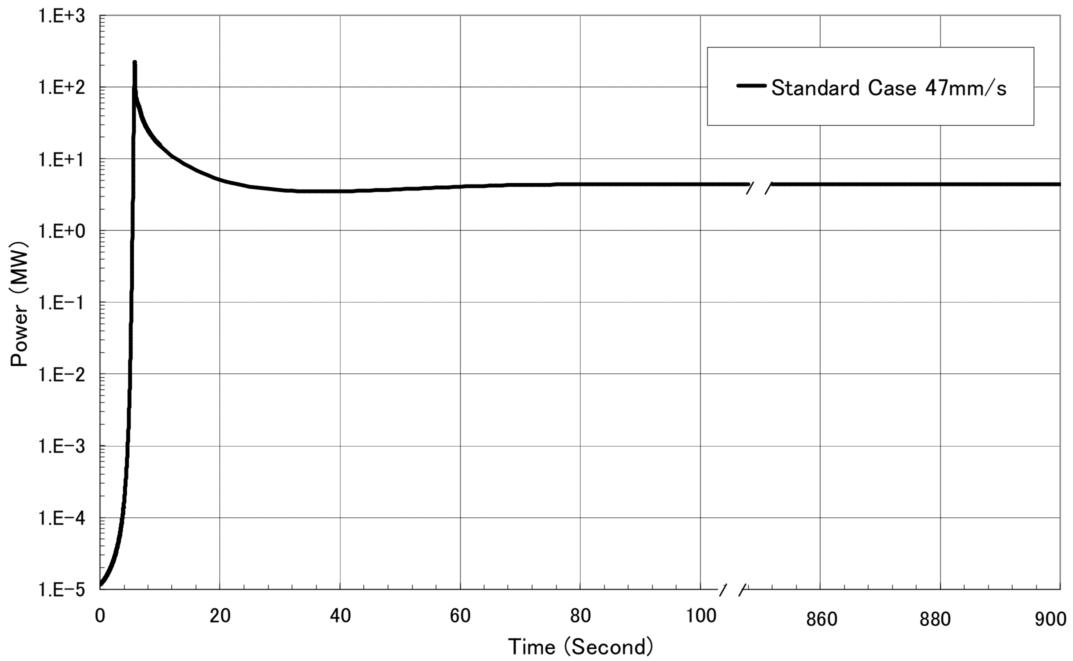


Figure 4-2　Trend of Power (Standard Case)

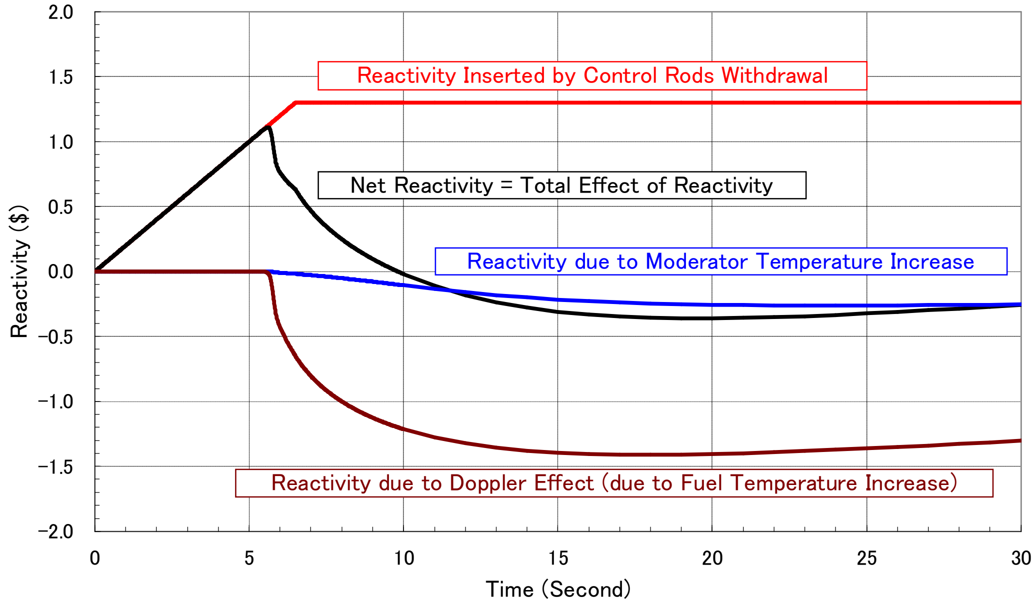


Figure 4-3　Effect of Reactivity Feedback (Standard Case)

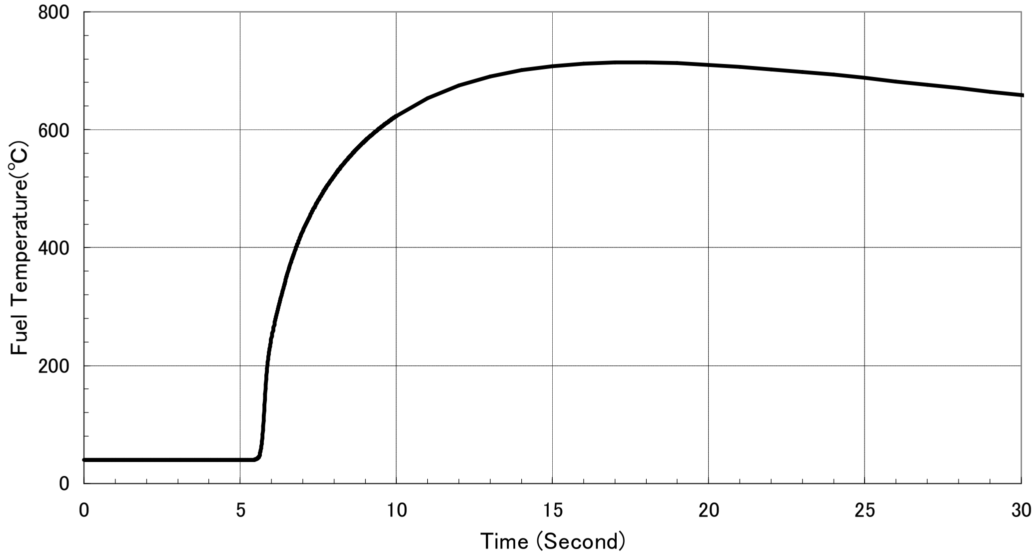
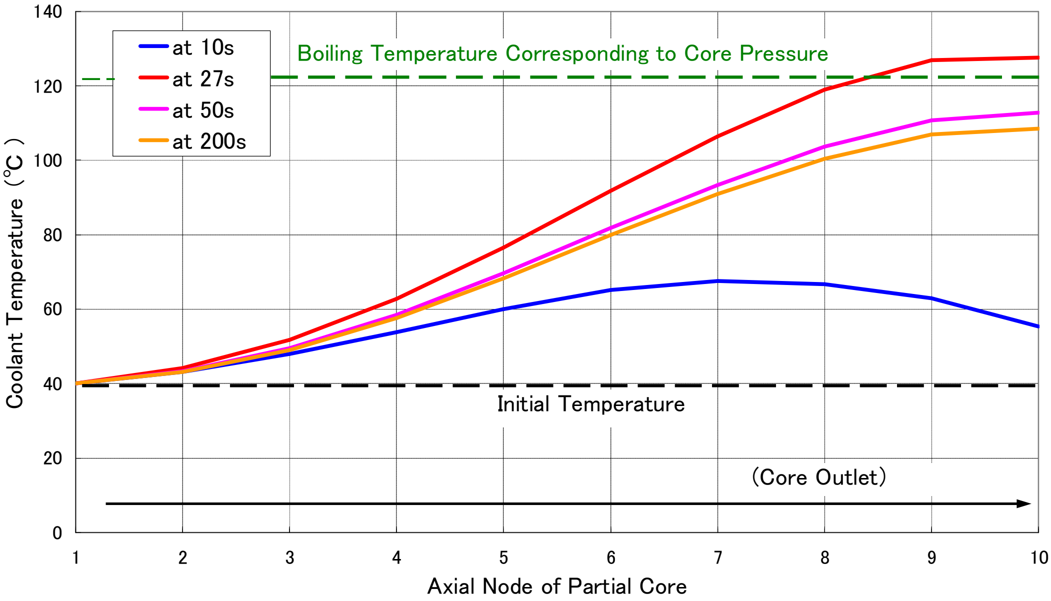


Figure 4-4　Trend of Pellet Temperature (Standard Case)



(Note) As higher core pressure value was used to avoid boiling, portion of coolant temperature appears to be above boiling temperature.

Figure 4-5 Axial Distribution of Coolant Temperature in Partial Core (Standard Case)

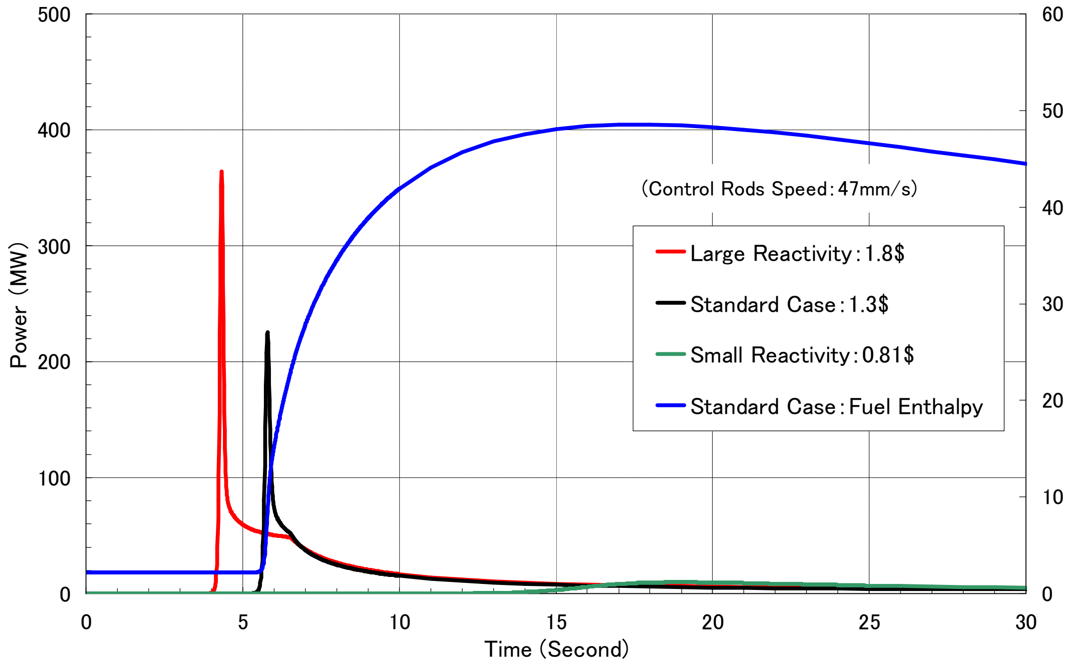


Figure 5-1 Trend of Power (Affect of Reactivity Inserted to the Core)

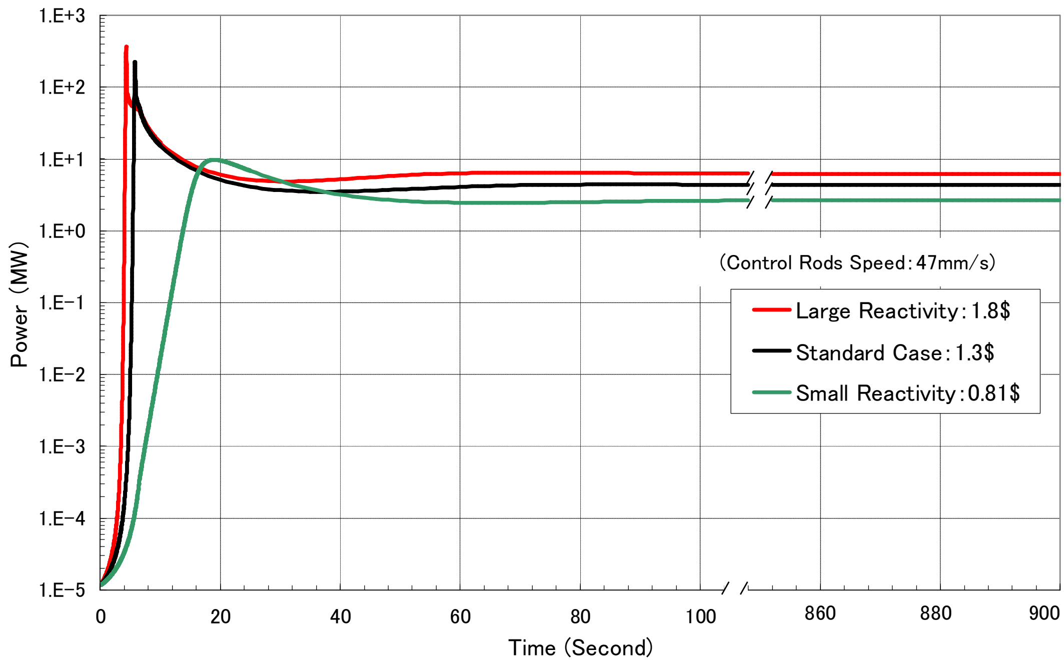


Figure 5-2 Trend of Power (Affect of Reactivity Inserted to the Core)

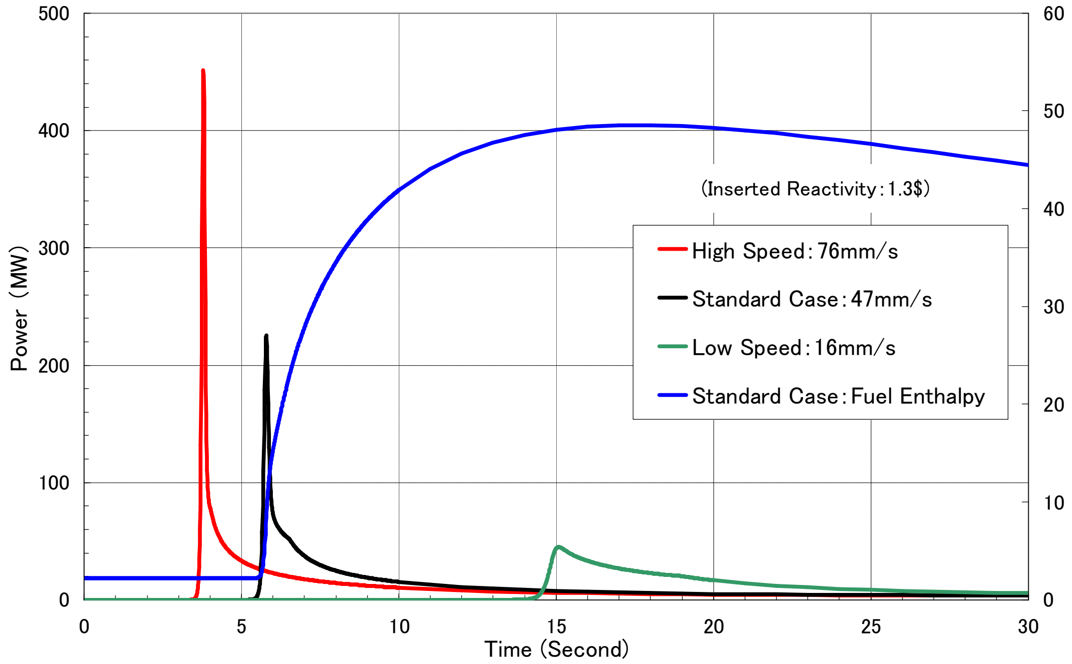


Figure 6-1 Trend of Power (Affect of Control Rod Withdrawal Speed)

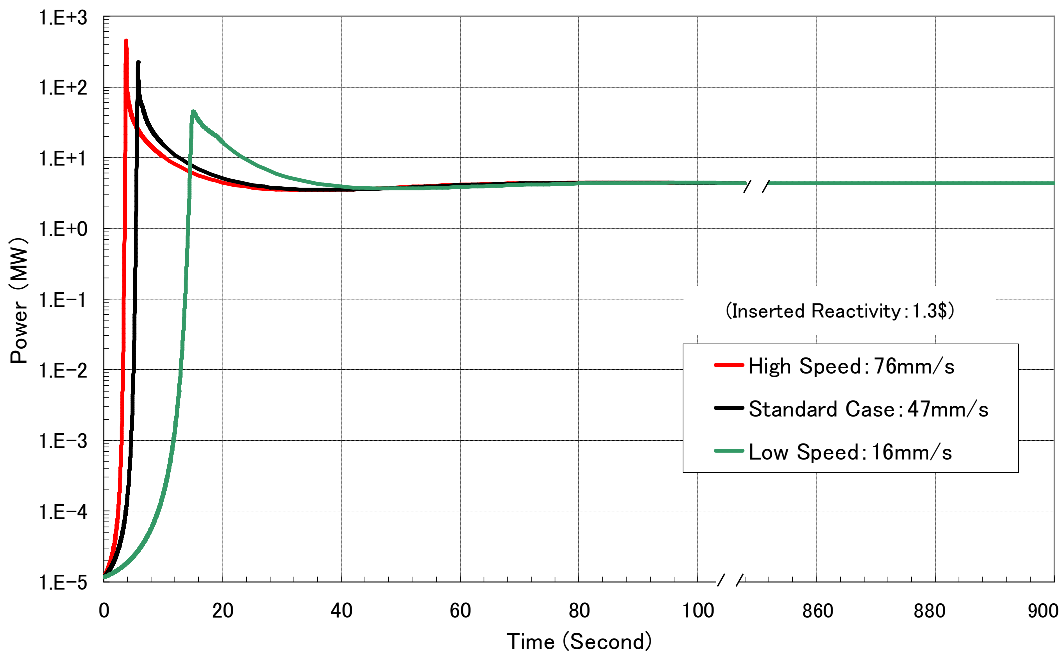


Figure 6-2 Trend of Power (Affect of Control Rod Withdrawal Speed)

Attachment 1

Summary of "Treatment of High Burn-up Fuels in Reactivity Insertion Accident of Light Water Reactor Generation Facilities"

This document was approved by Nuclear Safety Commission of Japan on April 13, 1998. For reference, summary of this document was translated as follows;

Background

For the future safety review, consideration was made on treatment of high burn-up fuels in reactivity insertion accident of light water reactors based on detailed investigation result of domestic research outcome and overseas examination results by safety standard sub-committee of nuclear safety commission to finalize the conclusion in this report.

Threshold Value of Fuel Failure

The threshold value of fuel failure due to PCMI (Pellet-Cladding Mechanical Interaction) is estimated as shown in the following table. The threshold values are presented as maximum enthalpy increase during peak power in conjunction with the associated pellet burn-up.

Pellet Burn-up	Maximum Enthalpy Increase During Peak Power
Below 25,000MWd/t	110cal/g・UO_2
Between 25,000MWd/t & 40,000MWd/t	85cal/g・UO_2
Between 40,000MWd/t & 65,000MWd/t	50cal/g・UO_2
Between 65,000MWd/t & 75,000MWd/t	40cal/g・UO_2

"Threshold Value Used in JANTI Analyses"

85cal/g・UO_2 was used based on the report by Hokuriku EPCO.

Attachment 2

Summary of “Safety Analyses Review Guide for Light Water Reactor Generation Facilities”

This review guide was approved by Nuclear Safety Commission of Japan on August 30, 1990, and revised on March 29, 2001. For reference, summary of this review guide was translated as follows;

II Safety Analysis Review

1. Purpose of Safety Analysis Review

Appropriateness of fundamental principle for safety design of nuclear facilities is reviewed per “Safety Design Review Guide.” “Safety Design Review Guide” requires that structures, systems and components of nuclear facilities should function as expected to maintain safety both during normal operation and during abnormal condition. Therefore, review and analyses of “abnormal transients during operation” and “accidents” are needed to confirm appropriateness of fundamental principle for safety design of nuclear facilities. This guide presents events to be considered for safety design review, threshold of analyses results, and conditions to be considered in analyses.

2. Scope of Analyses

2.1 Abnormal Transients during Operation

During reactor operation, events resulted from single failure/ malfunction of equipment or single operator error that is expected during operational life of nuclear facilities, or resulted from another contributors to be expected in equivalent frequency are considered.

2.2 Accidents

“Accidents” are the abnormal conditions that exceed “abnormal transients during operation.” In spite of low frequency, events should be regarded as “accidents” if there is potential of nuclear material release from the facility, and consideration is needed in the standpoint of safety review.

3. Selection of Events to be Reviewed

For both “abnormal transients during operation” and “accidents”, events for safety review should be selected appropriately in accordance with above mentioned purpose and scope of safety design review.

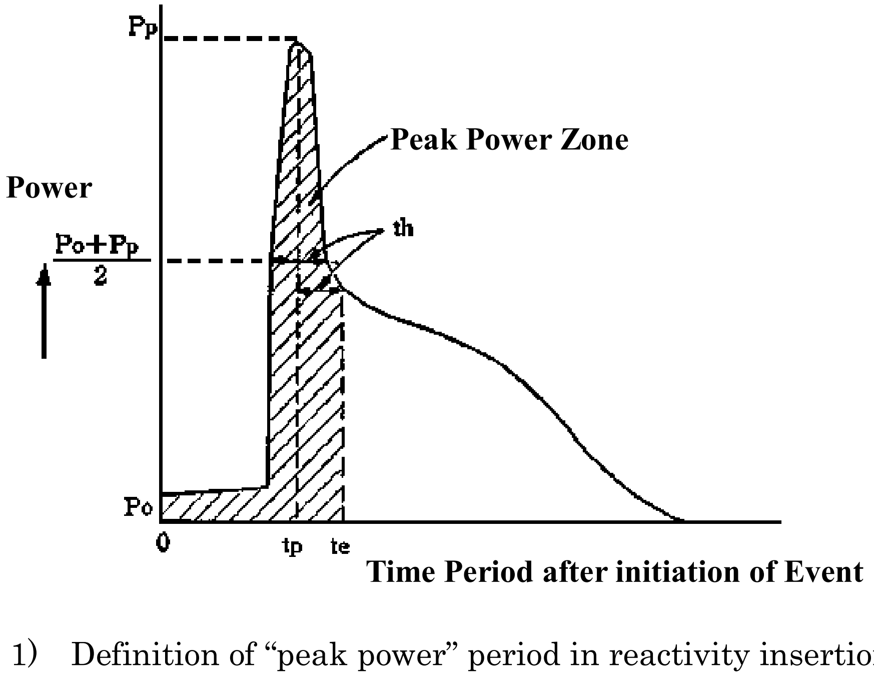
Attachment 3

Summary of “Reactivity Insertion Accident Review Guide for Light Water Reactor Generation Facilities”

This review guide was approved by Nuclear Safety Commission of Japan on January 19, 1984, and revised on August 30, 1990. For reference, summary of this review guide was translated as follows;

<u>Definition of Terms</u>

- Reactivity Insertion Accidents: Events accompanied by increase of reactor power and the associated increase of fuel enthalpy due to rapid insertion of basically more than 1$ of reactivity into reactor in or near criticality.
- Fuel Enthalpy: Radial average enthalpy of pellet. Summation of initial enthalpy and increased enthalpy obtained by analysis of the event. Fuel enthalpy is base value at 0℃.
- Definition of “peak power” period is shown in (Figure 1). “Po” is initial power, and “Pp” is peak power. “th” is the period while power is above (Po+Pp)/2. “tp” is the time of peak power. Peak power period “te” is defined as tp+th (time period of slashed zone.)



(Figure 1) Definition of “peak power” period in reactivity insertion accidents

Purpose

Analyze increase of reactor power and the associated increase of fuel enthalpy due to rapid insertion of basically more than 1$ of reactivity into reactor in or near criticality in order to confirm integrity of core and reactor coolant pressure boundary during “abnormal transient during operation” and “accident”.

Threshold

(1) “Abnormal Transient during Operation”

1) Maximum fuel enthalpy should be within “Fuel Design Limit” shown in (Figure 2).

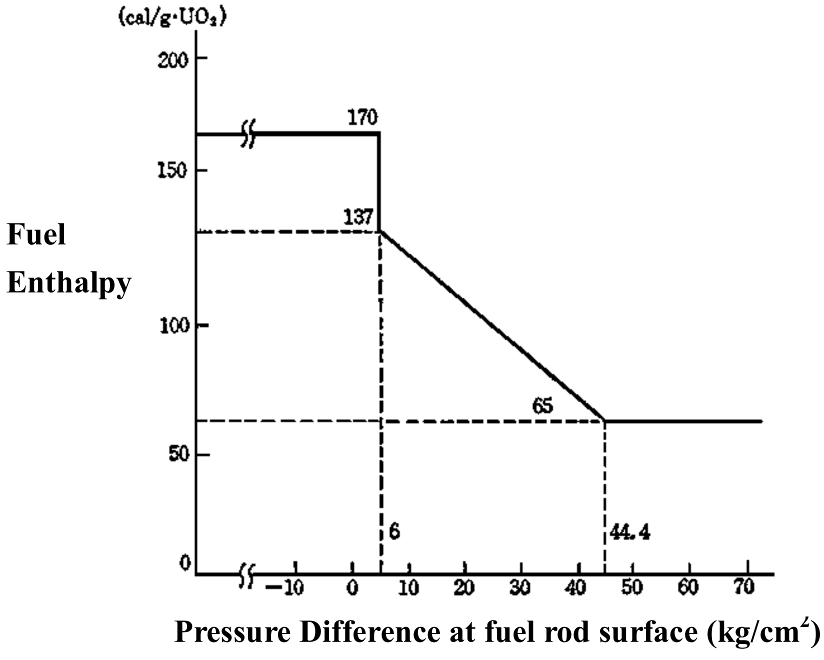
2) Pressure at reactor coolant pressure boundary should be within 110% of Maximum Operational Pressure.

(2) “Accident”

1) Maximum fuel enthalpy should be within 230cal/g・UO_2.

2) Pressure at reactor coolant pressure boundary should be within 120% of Maximum Operational Pressure.

(3) During “abnormal transient during operation” and “accident”, reactor shutdown capability and integrity of reactor pressure vessel should not be affected by disturbance such as pressure impact resulted from rupture of fuel with water intrusion.



(Figure 2)　Fuel design limit at reactivity insertion accident

Attachment 4

Definition of Terms

Terms	Explanation
Criticality	Status of generated neutron from fission and disappeared neutron from the core is in balance, and chain reaction is maintained. Status when Effective Criticality Factor (Keff) is 1.
Effective Criticality Factor (Keff)	Number of generated neutron from fission divided by number of neutron disappeared from the core.
Prompt Criticality	Status when criticality is maintained with no contribution of delayed neutron.
Delayed Criticality	Status when criticality is maintained with contribution of both prompt and delayed neutrons.
Prompt Neutron	Neutron emitted almost simultaneously (within 10^{-4} second) during fission.
Delayed Neutron	Neutron emitted from collapse of fission products after 0.4 second to 50-60 seconds of the original fission.
Delayed Neutron Fraction	Fraction of Delayed Neutron from total number of neutrons emitted from fission.
Reactivity	Value of (Keff-1)/Keff, indicator of deviation from criticality.
Reactivity Excess	Value of (Keff-1), reactivity of over criticality.

Inserted Reactivity	Reactivity inserted in the core.
Net Inserted Reactivity	Inserted reactivity subtracted by feedback reactivity.
Feedback Reactivity	Reactivity such as Doppler Reactivity and Moderator Temperature Reactivity which has suppression effect on the inserted reactivity.
Reactivity Coefficient	Coefficient of reactivity change due to change of fuel temperature or moderator temperature.
Doppler (Reactivity) Coefficient	Reactivity change per fuel temperature change. When fuel temperature increases, reactivity tends to decrease because of increased neutron absorption rate isotopes such as U-238.
Moderator Temperature (Reactivity) Coefficient	Reactivity change per moderator temperature change.
Core Inherent Safety Feature (Self Control Feature)	When reactor power increases, reactivity will decrease due to Doppler effect and others, which will lead to decrease of reactor power.
Fuel Enthalpy	Amount of heat accumulated per weight of fuel.
Power Peaking	Maximum power divided by average power.