Reactivity insertion transient analysis for KUR low-enriched uranium silicide fuel core

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Abstract

The purpose of this study is to realize the full core conversion from the use of High Enriched Uranium (HEU) fuels to the use of Low Enriched Uranium (LEU) fuels in Kyoto University Research Reactor (KUR). Although the conversion of nuclear energy sources is required to keep the safety margins and reactor reliability based on KUR HEU core, the uranium density (3.2 gU/cm^3) and enrichment (20%) of LEU fuel (U_3Si_2-AL) are quite different from the uranium density (0.58 gU/cm^3) and enrichment (93%) of HEU fuel (U-Al), which may result in the changes of heat transfer response and neutronic characteristic in the core. So it is necessary to objectively re-assess the feasibility of LEU silicide fuel core in KUR by using various numerical simulation codes. This paper established a detailed simulation model for the LEU silicide core and provided the safety analyses for the reactivity insertion transients in the core by using EUREKA-2/RR code. Although the EUREKA-2/RR code is a proven and trusted code, its validity was further confirmed by the comparison with the predictions from another two thermal hydraulic codes, COOLOD-N2 and THYDE-W at steady state operation. The steady state simulation also verified the feasibility of KUR to be operated at rated thermal power of 5MW. In view of the core loading patterns, the operational conditions and characteristics of the reactor protection system in KUR, the accidental control rod withdrawal transients at natural circulation and forced circulation modes, the cold water injection induced reactivity insertion transient and the reactivity insertion transient due to removal of irradiation samples were conservatively analyzed and their transient characteristic parameters such as core power, fuel temperature, cladding temperature, primary coolant temperature and departure from nucleate boiling ratio (DNBR) due to the different ways and magnitudes of reactivity insertions were focused in this study. The analytical results indicate that the quick power excursions initiated by the reactivity insertion can be safely suppressed by the reactor protection system of KUR in various initial power levels and different operational modes (natural circulation and forced circulation modes). No boiling and no burnout on fuel cladding surface and no blister in the fuel meat happens and KUR is safe in all of these reactivity insertion transients if the reactor protection system of KUR works in its minimum degree.

Keywords: EUREKA-2/RR; Reactivity insertion transient analysis; Control rod withdrawal transient; Cold water injection transient; Irradiation samples removal transient

1. Introduction

Kyoto University Research Reactor (KUR) is a light-water moderated tank-type reactor of 5MW power. Since the first criticality attainment in 1964, KUR had been successfully operated over than 40 years using high enriched uranium (HEU) fuel and served as one of important facilities to help the development of nuclear science and technology in Japan. In order to reduce the proliferation potential of research reactor HEU fuels and increase the assurance of continued fuel availability in the face of probable restrictions on the supply of highly enriched uranium, KUR converted its partial core from the use of HEU fuel elements to the use of low enriched uranium (LEU) silicide (U_3Si_2-Al) fuel elements in 1992, carefully evaluated the feasibility of the operating KUR in a mixed mode and successfully irradiated them to their maximum burnup (35%-U235) in the following years (Unesaki et al., 2010). The present work is due to safety analysis needs of the full core conversion from the use of HEU fuel to the use of LEU fuel in KUR, since the direct conversion of nuclear energy sources may cause the change of reactor responses to various postulated events during its service life.

The HEU and the LEU plate-type MTR (material test reactor) fuel elements are of identical geometry but differ in their fuel meat composition. KUR typical working cores consist of 16-29 standard fuel elements, 5 special fuel elements for four shim control rods and one regulating control rod, one hydraulic irradiation port, 3 pneumatic tube ports, water plug elements and aluminum-canned graphite reflector elements. KUR LEU silicide cores with minimum and maximum numbers of fuel elements are shown in Figs. 1 and 2 respectively. The standard and special fuel elements respectively consist of 18 and 9 curved fuel plates with 13.97 cm radius of curvature. The curved fuel plate is made of 20%-enriched U_3Si_2-Al dispersion fuel with aluminum cladding. The details of the standard and special fuel elements are shown in Table 1. The water plug and graphite reflector elements are usually loaded around the fuel elements. The water plug elements are for regulating the excess reactivity of the core and performing the long term neutron irradiation experiments. The graphite reflector elements are for reducing the neutron leakage from the core. The whole core is housed in an aluminum tank with 2 meters in diameter and 8 meters in depth, which is convenient for heat removal and makes various neutron beam experiments possible. The water behaves as neutron moderator, coolant and shielding material in KUR.

The cooling system of KUR comprises 2 primary circulating pump, 3 heat exchangers and a secondary cooling system. The primary coolant flows downwardly through the reactor core at the flow rate of about 900 m^3/hr in the forced circulation mode at the thermal powers of 0~5MW. The corresponding average flow velocity of the coolant in the core is about 3.5 m/s, which is much smaller than the core critical flow velocity resulting in the buckling and collapsing of the curved fuel plates, 14.5 m/s (KURRI, 2008). When the primary circulating pump stops, the convective flow valve opens automatically by the gravity force and the primary coolant driven by the heat from the core flows reversely and circulates naturally in the aluminum tank. So at low thermal powers, KUR can be operated in natural convection mode. When the reactor is shut down, the decay heat generated in the core is usually removed by the water natural circulation in the tank.

The fuel meat compositions of LEU and HEU fuel are U_3Si_2-Al and U-Al alloys respectively. The thermal conductivity of LEU fuel meat (97.5W/(mK)) is about 40% smaller than that of HEU fuel meat (163W/(mK)), which may result in the slower heat transfer responses to various transients and accidents in the LEU core than those in HEU core. The neutronic characteristic comparisons between the HEU and LEU fuel cores are shown in Table 2. The maximum power density of the LEU fuel core (164W/cm^3) is about 14% greater than that of the HEU fuel core (144W/cm^3). The absolute value of the minus reactivity coefficients are greatly increased in the LEU fuel core. The great increase of the LEU fuel temperature coefficient due to the significant ratio-increasing of its ^{238}U in the LEU fuel would help suppress the abnormal rise of the fuel temperature in the core. Sano et al., 2010 also confirmed that the thermal neutron flux in the neutron spectrum of the KUR LEU fuel core decreases, comparing with the thermal neutron flux in the neutron spectrum of the KUR HEU fuel core. In view of the property and neutronics characteristics changes in the conversion from HEU fuel core to LEU fuel core and the unique facility features of KUR, the feasibility of LEU fuel core in KUR must be objectively re-assessed.

The objective of this study is to analyze the responses of KUR to various postulated events and to ensure that the steady state and transient parameters in KUR LEU silicide core do not exceed its safety limits to protect the public from the release of the radioactive materials contained within KUR. The following 3 safety criteria for anticipated operational transients in KUR cores are used here.

(1) Minimum DNBR (viz. CHFR (critical heat flux ratio)) is greater than 1.5.

(2) Maximum temperature in fuel meat is less than 400 ^{o}C.

(3) No boiling happens in the core.

The first and second criteria are to ensure that no burnout on fuel cladding surface and no blister in the fuel meat happen respectively to keep the cladding as the healthy first barrier against the radioactive products release. The third criterion is to ensure that no high pressure due to the generation of large volumes of vapor happens in primary coolant system to keep the tank and piping as the healthy second barrier against the radioactive products release.

2. Neutronic analysis of KUR core and its evaluation core selection

Thermal hydraulic analysis has to be coupled with neutronics analysis in the nuclear reactor safety analysis since heat energy generated in the reactor core is induced by the fission neutrons. Neutronics data for the transient analyses (Nakajima et al., 2010) were obtained from three-dimensional neutron transport and diffusion calculations for the KUR LEU core by using SRAC, a general purpose neutronics code system applicable to core analyses for various types of reactors (Okumura et al., 1996). The SRAC system includes major public neutron data libraries (JENDL-3.2, JENDL-3.1, JENDL-2, ENDF/B-VI, ENDF/B-V, ENDF/B-IV) and integrates several modular codes (collision probability calculation module (PIJ) for 16 types of lattice geometries, Sn transport calculation modules (ANISN, TWOTRAN) and diffusion calculation modules (TUD, CITATION) and so on) for neutron transport and diffusion calculation. The energy group structure of the current public libraries consists of 107 groups. With the coarse group cross-section data generated by the collision probability calculation module, PIJ and the diffusion calculation module, CITATION, the effective multiplication factor, reactivity worth of the shim and regulating control rods, the power distribution of the fuel elements (power peaking factors), reactor kinetics and the feedback reactivity due to moderator temperature effect, fuel temperature effect and void effect can be obtained by the SRAC-CITATION calculation for the KUR core.

In order to analyze the neutronic characteristics in KUR cores, we selected two loading patterns for the core, namely the core with minimum fuel elements and the core with maximum fuel elements (shown in Figs. 1 and 2 respectively) as the representative cores. They are the minimum and maximum cores respectively among all of possible core loading patterns with necessary excess reactivity for high power (5MW) operation. The minimum core corresponds to the fresh core loaded with the fresh LEU fuel elements at 0%-U235 burnup at the beginning of fuel life and the maximum core corresponds to the core loaded with the LEU fuel elements at average 25%-U235 burnup at the end of fuel life. The neutronic characteristics of the minimum and maximum cores evaluated by the SRAC calculations were shown in Table 2. The comparison of the calculated neutronic characteristics with their safety limit values (KURRI, 2008) in Table 2 shows that both of the minimum and maximum cores meet the requirements, but the maximum power density of the minimum core (164 W/cm^3) is much greater than that of the maximum core (102 W/cm^3). So the minimum core with 16 fresh standard fuel elements and 5 fresh special fuel elements is selected as the evaluation core for thermal hydraulic analysis in this study. Since the SRAC calculation also shows that the highest heat generation happens in the standard fuel element at RO-5, the hottest channel in RO-5 fuel element is the hot channel in KUR minimum core. The calculated vertical axial power distribution along the height in the minimum core is illustrated in Fig. 3. The horizontal cross-sectional and vertical axial power peaking factors are 1.44 and 1.40 respectively.

3. Computer code employed and KUR core model

A computer code for reactivity accident analysis, EUREKA-2/RR (Kaminaga, 1996), which has been developed to analyze neutronic, thermal and hydrodynamic transient behaviors in a water cooled reactor, was employed to carry out the present steady-state and transient thermal hydraulic analysis for KUR core. The code has successfully predicted the thermal and hydraulic characteristics of JRR-3 (Kaminaga et al., 1997), JRR-4 (Kaminaga et al., 1991), JMTR (Nagaoka et al., 1992), Muti-purpose research reactor RSG-DG Siwabessy (Kaminaga et al., 1991) and TRIGA Mark-II research reactor (Badrun et al., 2012). The code considers a thermal and hydraulic system as a series of interconnecting user defined (control) volume. It solves the mass and energy balances for volumes assumed to contain one-dimensional homogeneous fluid with the vapor and liquid phases in thermodynamic equilibrium and solves the momentum balance at the interfaces or junctions between control volumes. A special “Heat Transfer Package” was included to calculate heat transfer coefficient, ONB (onset of nucleate boiling) temperature, critical heat flux and so on for research reactors using plate-type fuel elements in EUREKA-2/RR. In the special “Heat Transfer Package”, (1) the heat transfer coefficients of single-phase flow are calculated by Dittus-Boelter correlation (Dittus and Boelter, 1930) and Collier correlation (Collier, 1972) for forced convection and natural convection respectively, (2) the heat transfer coefficients of two-phase flow are predicted by Chen correlation (Chen, 1963) and Rohsenow correlation (Rohsenow, 1972) for low and high flow conditions respectively, (3) the ONB (onset of nucleate boiling) is determined by comparing the heat flux predictions from Dittus-Boelter correlation (Dittus and Boelter, 1930) and the modified Chen correlation (Chen, 1963), and (4) the critical heat flux is evaluated by Sudo-Kaminaga correlations (Sudo and Kaminaga, 1993). The heat conduction model is based on the method of one-dimensional time dependent heat conduction equations. Since the automatic reactivity system of the reactor can be simulated by the continuous reactivity control model in EUREKA-2/RR, the code can analyze the reactivity response of the core due to the automatic power change, transient response of the core due to the reactivity insertion caused by control rod withdrawal, coolant flow change and/or coolant temperature change and so on.

The whole minimum core including the fuel region and the upper and lower plenums was considered for the core simulation model in EUREKA-2/RR calculation. The simulation model is shown in Fig.4. The core was divided into two regions, viz. the hot channel and the average channel, which differ from each other by power generation and coolant mass flow rate. The hot channel is in the standard fuel element at RO-5 and the average channel is composed of the channels in the fuel elements except for the hot channel. Each flow channel in the fuel region is axially discretized into 15 volumes connecting by 16 junctions, as shown in Fig. 4. The fuel plate in a channel are axially nodalized into 15 heat conductors referred to as heat slabs, corresponding to the volumes in the channel. KUR core model in total consists of 40 volumes, 42 junctions and 30 heat slabs. Junction 40 is used as the fill junction to simulate the injection of primary coolant flow into the core.

After the arrangement of the volumes, junctions and heat slabs for KUR core model, the initial coolant temperature and pressure for each volume, the power fraction and feedback reactivity weighting factors for each slab have to be prepared as the preliminary data before the EUREKA-2/RR calculation. Kaminaga (1991) developed three supporting codes, viz. DISSUE, ICETEA and PREDISCO, to produce the necessary input data for EUREKA-2/RR calculation. DISSUE calculates power fraction and void, Doppler, clad expansion and coolant temperature reactivity weighting factors for each heat slab based on neutronics calculation. ICETEA and PREDISCO produce coolant temperature distribution and pressure distributions respectively for the coolant control volumes.

In order to incorporate the uncertainties, the multiplicative method, in which all the worst conditions occurs simultaneously at the same point, was adopted to make the computed results be conservative enough. So the following severe measures were taken in the EUREKA-2/RR calculations for KUR reactivity insertion transient analyses.

(1) All negative reactivity feedbacks due to moderator temperature effect, fuel temperature effect and void effect were neglected in most reactivity insertion transients except for the cold water injection induced reactivity insertion transient. The reactivity coefficients from KUR maximum core was used in the analysis for the cold water injection induced reactivity insertion transient, since the absolute values of the reactivity coefficients from KUR maximum core are greater

than those from KUR minimum core (see Table 2).

(2) The effective delayed neutron fraction and prompt neutron mean life from KUR maximum core were used, since the kinetics constants from KUR maximum core are smaller than those from KUR minimum core (see Table 2).

(3) At the high thermal powers of 1 MW and 5 MW, the primary coolant flows through the reactor core at the flow rate of about 900 m^3/hr in the normal forced circulation mode. If only 2 out of 3 heat exchangers are in service, the primary coolant flow rate will reach its minimum value of 800 m^3/hr (KURRI, 2008). So the minimum value of 800 m^3/hr was used in the calculation for the operation with forced circulation mode.

(4) KUR is operated with the primary coolant temperature at the core outlet, ranging from 10~55 ^oC and 10~45 ^oC at high power forced circulation mode and low power natural circulation mode respectively (KURRI, 2008). So the highest core inlet temperatures, which corresponding to the maximum core outlet temperatures of 55 ^oC and 45 ^oC at forced and natural circulation modes respectively, and the lowest core water temperature of 10 ^oC in the cold water injection transients were used in the calculation to make the simulated transients and accidents be severe.

(5) When KUR is shut down emergently, 4 shim rods fall with the force of the gravity. The reactor is subcritical by the shutdown margin of 2 %Δk/k in the event of a scram, even if the maximum worth control rod fails to insert into the core (KURRI, 2008). For conservative analysis of the reactivity insertion transients in the calcualtion, KUR would be shut down by the margin of 1 %Δk/k in the event of a scram and the scram reactivity with the delay time of 0.1 s is inserted into the core in S-shaped increasing way as shown in Fig. 5 within the time of 1 s. All electromotor-driving shutdowns of the reactor are neglected in the analyses.

(6) Bulk water temperature rising, heat flux and heat transfer coefficient hot channel factors were conservatively introduced to take into account of the effects of the uncertainties relating to variations in flow distribution, fuel element and coolant channel geometry, and neutron flux spatial distribution upon the thermal performance. Bulk water temperature rising and heat flux hot channel factors took the values of 1.54 and 1.34 respectively (KURRI, 2008). The heat transfer coefficient hot channel factor was 1.37 for Dittus-Boelter correlation (Dittus and Boelter, 1930) in downward flow or 1.43 for Collier correlation (Collier, 1972) in upward flow (KURRI, 2008).

In addition, physical properties of fuel plates used in the present EUREKA-2/RR analyses are shown in Table 3.

4. KUR core analysis by EUREKA-2/RR code

4.1 Steady state thermal hydraulic analysis

Steady state thermal hydraulic analysis for the LEU silicide of KUR were carried out with EUREKA-2/RR under forced convection cooling at the rated thermal power of 5MW. In view of the water depth of the core in the tank, the pressure at the core top was set to be 0.17 MPa. The calculated fuel temperatures, cladding temperatures and bulk coolant temperatures in the hot and average channels are shown in upper figure in Fig. 6. The calculated heat fluxes in the hot and average channels are shown in middle figure in Fig. 6. The DNBRs in the hot and average channels are shown in lower figure in Fig. 6. The minimum DNBR, the maximum fuel temperature and maximum primary coolant temperature are 4.91, 90.51^{o}C and 61.56 ^{o}C respectively in the LEU silicide of KUR. All of these parameters are far below safety limit of KUR. So it is safe for KUR to operate at steady state rated thermal power of 5MW.

In order to ensure the reliability of the EUREKA-2/RR calculation, the EUREKA-2/RR calculation is compared with the calculations of another two thermal hydraulic codes, viz. COOLOD-N2 (Kaminaga, 1994) and THYDE-W (Asahi et al., 1990) at steady state rated thermal power of 5MW (Shen et al. 2010). All analytic results with the 3 codes are shown in Table 4. Table 4 shows that the important parameter predictions of EUREKA-2/RR agree very well with those of the other two codes, COOLOD-N2 and THYDE-W.

4.2 Control rod withdrawal transients

Accidental continuous withdrawal of the control rods due to equipment malfunction or operator error can be a potential danger to KUR. The reactor protection system fortunately provides automatic shutdown of the reactor core and thus limits the extent of accidental control rod withdrawal transients to protect the core. KUR was designed to be automatically shut down when its period from the compensated ionization chamber instrumentation system (referred to as “period meter”) is less than 5 seconds and/or when its linear power from the compensated ionization chamber instrumentation system (referred to as “linear power meter”) exceeds 120% of each power range setting and/or when its safety power from the uncompensated ionization chamber instrumentation system (referred to as “safety power meter”) exceeds 120% of the nominal power or 300KW at natural circulation mode. If the period meter is assumed not to function for conservatism, the fast scram initiated by the period signal cannot be expected in the accidental control rod withdrawal transients. The scram initiated by the linear power meter or the safety power meter shows that KUR shutdown set-points differ with the reactor operation modes, viz. forced circulation mode and natural circulation mode, in which their setting and nominal powers are different, and the reactor responses to the accidental control rod withdrawal transients change with different initial power levels of KUR. The operation of the shim rod withdrawal is limited to one shim rod due to the interlock arrangement in KUR. So the maximum rate of reactivity insertion is 0.015%Δk/k/s in the accidental control rod withdrawal transients of KUR.

4.2.1 Accidental control rod withdrawal transient at natural circulation mode

When the required output power of KUR is less than 100KW, the reactor is operated at natural circulation mode. In this mode, KUR requires that the primary coolant pumps are not in operation, that the convective flow valve is open for the naturally circulating flow and that the water temperature in core is less than 45^{o}C. Since KUR may be operated at the maximum power of 100KW with the natural circulation mode, the maximum scram power, viz. the power limit for emergency shutdown of the reactor, will be 120KW (=100KW×120%) at natural circulation mode. In order to make the analysis results for the accidental control rod withdrawal transient at natural circulation mode be conservative enough, the initial power, flow rate and reactivity insertion conditions and so on are summarized in Table 5 as the input data for EUREKA-2/RR calculation. The initial powers of 0.001W and 1.0W are the minimum power (source level power) and maximum power respectively at the start-up stage of KUR. Since the velocity difference in the low water velocity region has very low impacts on heat transfer and critical heat flux (Sudo et al., 1984), the initial average flow rate of the primary coolant is assumed to be 18 m^3/h (5 cm/s in the core) at the natural circulation mode.

The analytical results from the EUREKA-2/RR calculations for the accidental control rod withdrawal transient at natural circulation mode are shown in Fig. 7 and Table 5. The left upper, right upper, left middle, right middle, left lower and right lower figures in Fig. 7 indicate the calculated data of power, maximum fuel temperature, maximum cladding temperature, maximum water temperature, minimum DNBR and inserted reactivity respectively in the transients. Fig.7 shows that the linear additions of reactivity to the core result in the power excursions when one control rod starts to be withdrawn inadvertently at the time of 0 s and the rising powers are suppressed by the control rod insertion (scram) when the powers from the linear power meter or safety power meter reach its scram value of 120KW in KUR. The minimum

DNBRs are 94.54, 79.92 and 19.16 and happen at the time of 42.99s, 39.39s and 0.003s after the rod withdrawals for the initial powers of 0.001W, 1W and 100KW respectively. Since the minimum DNBRs are much greater than the safety criterion of 1.5, it is not possible for the burnout to happen on the cladding surface. The highest fuel and cladding temperatures are 82.26^{o}C and 82.22^{o}C respectively at the initial power of 100KW in these transients. Since the fuel temperature is much lower than the blister happening limit of 400^{o}C, the fuel meat is in good state in these transients. The cladding temperature is much lower than the onset of nucleate boiling of 117^{o}C, which is obtained from the prediction of Bergles-Rohsenow correlation (Bergles and Rohsenow, 1964) with the primary coolant temperature of 45^{o}C. So no boiling happens in the core. In view of these analyses, KUR is safe in the accidental control rod withdrawal transient at natural circulation mode.

4.2.2 Accidental control rod withdrawal transient at forced circulation mode

To meet the experimental needs for high neutron flux in the field of medical care, material development and so on, KUR is often operated at high thermal powers of 1 and 5 MW with pump-driving forced circulation mode. The high power forced circulation mode requires that 2 primary coolant pumps and 3 heat exchangers are in operation at the flow rate of about 900 m^{3}/hr and the primary coolant temperature is less than 55^{o}C at core outlet. If only 2 heat exchangers are in service, the primary coolant flow rate must be kept to be greater than 800 m^{3}/hr. If the linear power meter is further assumed to fail for conservatism, the scram will be initiated by the safety power meter at high power forced circulation mode when the power of the core reaches 6MW(=5MW×120%). Since KUR is allowed to be operated with the power fluctuation of 10%, its power may change from 4.5MW to 5.5MW at rated thermal power of 5MW. In view of the high power forced circulation operations at start-up stage, 1MW and 5MW, the initial powers of 0.001W, 1.0W, 1MW, 4.5MW, 5MW and 5.5MW are selected in the analyses for the accidental control rod withdrawal transients at forced circulation mode. The important input data are summarized in Table 6 for EUREKA-2/RR calculations in these transients.

The analytical results from the EUREKA-2/RR calculations for the accidental control rod withdrawal transients at force circulation mode are shown in Fig. 8 and Table 6. The left upper, right upper, left middle, right middle, left lower and right lower figures in Fig. 8 indicate the calculated data of power, maximum fuel temperature, maximum cladding temperature, maximum water temperature, minimum DNBR and inserted reactivity respectively in the transients. When one control rod starts to be withdrawn inadvertently at the time of 5s, the linear additions of reactivity to the core result in the power excursions. The rising powers are suppressed by the control rod insertion (scram) when the powers from the safety power meter reach its scram value of 6MW in KUR. The minimum DNBRs are 2.2898, 2.9551, 3.6616, 3.6813, 3.6814 and 3.6815 and happen at the time of 44.05s, 41.43s, 21.13s, 7.44s, 5.44s and 3.16s after the rod withdrawals for the initial powers of 0.001W, 1W, 1MW, 4.5MW, 5MW and 5.5MW respectively. Since the minimum DNBRs are greater than the safety criterion of 1.5, the burnout does not happen on the cladding surface. The highest fuel and cladding temperatures are 129.39^{o}C and 125.61^{o}C respectively at the initial power of 0.001W in these transients. Since the fuel temperature is much lower than the blister happening limit of 400^{o}C, the fuel meat is in good state in these transients. The cladding temperature is lower than the onset of nucleate boiling of 127^{o}C, which is obtained from the prediction of Bergles-Rohsenow correlation (Bergles and Rohsenow, 1964) with the primary coolant temperature of 55^{o}C. So the boiling does not happen in the core. In view of these analyses, KUR is safe in the accidental control rod withdrawal transient at forced circulation mode.

4.3 Cold water injection induced reactivity insertion transients

When KUR is operated at natural circulation mode, the primary coolant pump suddenly gets started running due to its malfunction. The core is reflooded by the cold water driven the pump. The cold water reflooding due to the forced circulation will insert considerable positive reactivity into the core, which means that the power in the core may overshoot to a considerably high level and the generated heat may cause damage to the fuel plates if the control rods fail to insert. The reactor will undergo a cold water injection induced reactivity insertion transient if the core is not subcritical.

When the power of KUR rises to the scram power levels in the cold water injection induced reactivity insertion transient, the reactor protection system will emergently shut down the reactor to protect the core. Since the transient happens at the natural circulation operation mode, the scrams of KUR may be initiated by (1) the period meter scram happening when the period is less than 5 seconds, (2) the linear power meter scram happening when the linear power from the compensated ionization chamber instrumentation system exceeds 120% of each power range setting (maximum 100KW) and (3) the safety power meter scram happening when the safety power from the uncompensated ionization chamber instrumentation system exceeds 120% of the nominal power or 300KW. If the period meter scram and the linear power meter scram at the range less than 100KW are ignored for conservatism, the reactor may be shut down by the scram at the power of 120KW initiated by the linear and safety power meters or the scram at the power of 300KW initiated by the safety power meter (if the 120KW scram fails) in this transient. Since the lowest coolant temperature is 10 ^{o}C and the coolant temperature coefficient is about -0.018%Δk/k/^{o}C at the water temperature range of 10-45^{o}C in the core, the cold water reflooding will insert the positive reactivity of 0.630%Δk/k (=0.018%Δk/k/^{o}C×(45-10) ^{o}C) in a step increasing way in the transient. In view of the operation of KUR with the power fluctuation of 10%, its power may change from 90KW to 110KW in the severest condition of 100KW power natural circulation operation. So the initial powers of 90KW, 100KW and 110KW are selected in the analyses for the cold water injection induced reactivity insertion transients. The important input data are summarized in Table 7 for EUREKA-2/RR calculations in these transients.

The analytical results from the EUREKA-2/RR calculations for the cold water injection induced reactivity insertion transients are shown in Fig. 9 and Table 7. The left upper, right upper, left middle, right middle, left lower and right lower figures in Fig. 9 indicate the calculated data of power, maximum fuel temperature, maximum cladding temperature, maximum water temperature, minimum DNBR and inserted reactivity respectively in the transients. When the cold water flooding starts at the time of 0s, the step additions of reactivity to the core result in the quick power excursions. The rising powers are suppressed by the control rod insertion (scram) when the powers from the linear and safety power meters reach their scram value of 120KW or the powers from the safety power meter reach its scram value of 300KW. When the scrams happen at 120KW, the initial power of 110KW is the severest case. If the scram at 120KW fails, the reactor will be shut down by the scram at the power of 300KW. The comparison of the analytical results from the 2 transients of 110KW initial power with different scram powers of 120KW and 300KW shows that the transient with scram power of 300KW is severer than that with scram powers of 120KW. In the severest case, the minimum DNBR is 11.73 and happens at the time of 0.562s after the cold water reflooding. Since the minimum DNBR is greater than the safety criterion of 1.5, the burnout does not happen on the cladding surface. The highest fuel and cladding temperatures are 95.28^{o}C and 95.22^{o}C respectively in the severest transient. Since the fuel temperature is much lower than the blister happening limit of 400^{o}C, the fuel meat is in good state in the severest transient. The cladding temperature is lower than the onset of nucleate boiling of 117^{o}C. So the boiling does not happen in the core. In view of these analyses, KUR is safe in the cold water injection induced reactivity insertion transient.

4.4 Reactivity insertion transients due to removal of irradiation samples

The samples may be inserted into or removed from the irradiation facilities of 8 beam holes, 3 pneumatic transfer tubes and 1 hydraulic conveyor in the high power forced circulation operation of KUR. The maximum inserted positive reactivity was evaluated to be 0.13 %Δk/k if all of the beam holes are accidentally injected with water (KURRI, 2008). Since the 3 irradiation chambers of the pneumatic samples are in the outer ring of the core (see Figs. 1 and 2), the positive reactivity additions due to the removal of their samples are much smaller than that of hydraulic conveyor, which locates in the center of the core (see Figs. 1 and 2). The addition of the positive reactivity due to the removal of irradiation samples in the hydraulic conveyor is limited to be less than 0.2 %Δk/k in KUR (KURRI, 2008). So the reactor will undergo the severest transient with the core power overshooting to a considerably high level when the positive reactivity of 0.2 %Δk/k due to the removal of the irradiation sample in the hydraulic conveyor is inserted in step way into the core. When the power rises to the scram power, the reactor will be emergently shut down by its reactor protection system. If the scrams from the period meter and the linear power meter do not function, the reactor will be shut down by the safety power meter at the power of 6MW. In view of the high power operation of 1MW and 5MW and the possible power fluctuation of 10% the initial powers of 1MW, 4.5MW, 5.0MW and 5.5MW are investigated in the analyses for the reactivity insertion transients due to removal of irradiation samples. The important input data are summarized in Table 8 for EUREKA-2/RR calculations in these transients.

The analytical results from the EUREKA-2/RR calculations for the reactivity insertion transients due to removal of irradiation samples are shown in Fig. 10 and Table 8. The left upper, right upper, left middle, right middle, left lower and right lower figures in Fig. 10 indicate the calculated data of power, maximum fuel temperature, maximum cladding temperature, maximum water temperature, minimum DNBR and inserted reactivity respectively in the transients. When the removals of irradiation samples start at the time of 2s, the step additions of reactivity to the core result in the quick power excursions. The rising powers are suppressed by the control rod insertion (scram) when the power from the safety power meter reaches its scram value of 6MW. The minimum DNBRs are 3.677, 3.707, 3.361 and 3.062 and happen at the time of 24.58s, 0.32s, 0.30s and 0.29s after the sample removals for the initial powers of 1MW, 4.5MW, 5MW and 5.5MW respectively. Since the minimum DNBRs are greater than the safety criterion of 1.5, the burnout does not happen on the cladding surface. The highest fuel and cladding temperatures are 111.02^{o}C and 108.13^{o}C respectively at the initial power of 5.5MW in these transients. Since the fuel temperature is much lower than the blister happening limit of 400^{o}C, the fuel meat is in good state in these transients. The cladding temperature is lower than the onset of nucleate boiling of 127^{o}C. So the boiling does not happen in the core. In view of these analyses, KUR is safe in the reactivity insertion transient due to removal of irradiation samples.

5. Summary and conclusion

This study presents a safety assessment for the LEU silicide core in KUR to realize its full core conversion from the use of HEU fuels to the use of LEU fuels. A detailed simulation model for the LEU silicide core was established and the safety analyses for the reactivity insertion transients in the core were performed by using EUREKA-2/RR code. The steady state analysis for KUR was firstly carried out when KUR is operated at its rated thermal power of 5MW. Its safety was confirmed by the EUREKA-2/RR calculation results. The calculated results from EUREKA-2/RR code were also compared with the predictions from another two thermal hydraulic codes, COOLOD-N2 and THYDE-W. The validity of EUREKA-2/RR code was accordingly verified by these steady state calculations. The accidental control rod withdrawal transients at natural circulation and forced circulation modes, the cold water injection induced reactivity insertion transient and the reactivity insertion transient due to removal of irradiation samples were chosen in the present transient analyses. In view of the core loading patterns, the operational conditions and characteristics of the reactor protection system in KUR, the conservative input data were used in the simulated calculations. The characteristics of different parameters such as core power, fuel temperature, cladding temperature, primary coolant temperature and DNBR due to the different way and magnitude of reactivity insertions were focused in these transients. It is found that the reactor powers have overshot to a certain high level when the reactivity is inserted and the quick rising power has been safely suppressed by the reactor protection system of KUR in various initial power levels and different operational modes (natural circulation and forced circulation modes). Their minimum DNBRs are greater than the safety criterion of 1.5. Their fuel temperature is much lower than the blister happening limit temperature. Their cladding temperature is lower than the onset of nucleate boiling. So KUR is safe in all of these reactivity insertion transients if the reactor protection system of KUR works in its minimum degree.

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Fig. 1 KUR core loading pattern with minimum fuel elements.

Fig. 2 KUR core loading pattern with maximum fuel elements.

Fig. 3 Axial power distribution along the distance from fuel top in KUR minimum core.

Fig. 4 Schematic diagram of reactor core nodalization.

Fig. 5 The S-shaped increasing curve of the inserted scram reactivity.

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Fig. 9 Cold water injection induced reactivity insertion transients.

Fig. 10 Reactivity insertion transients due to removal of irradiation samples.

Table 1 Standard and special fuel elements

Item		Standard fuel element	Special fuel element
Fuel element type		Plate-type MTR fuel element	
Overall dimension (width×width×length) (mm)		80×75×870	
Number of fuel plate in each element		18	9
U-235 mass contained in each element (g)		213	106.5
Fuel-element-averaged maximum fuel burnup (%-Uranium)		35	
Fuel meat	Category	U_3Si_2-Al dispersion fuel material	
	Main material	U_3Si_2-Al	
	Uranium enrichment (%)	20	
	Uranium density (gU/cm^3)	3.2	
	Thickness (mm)	0.5	
	Width (mm)	63	
	Length (mm)	594	
Fuel cladding	Material	Aluminum alloy, JIS A1200	
	Thickness (mm)	0.51	
Fuel plate	Thickness (mm)	1.52	
	Width (mm)	71	
	Length of inner fuel plate (mm)	625	
	Length of outer fuel plate (mm)	676	
Coolant channel	Channel number (-)	17	9
	Thickness (mm)	2.81	
	Width (mm)	66.5	
	Length (mm)	594	

Table 2 Neutronic characteristics of KUR cores

Item	KUR HEU core (KURRI, 1996)	KUR LEU core		
		Minimum core at 0%-U235 burnup	Maximum core at 25%-U235 burnup	KUR safety limit
Reactor shutdown margin (%Δk/k)				
• When all shim rods are inserted	-	11.6	7.4	>1.0
• When the shim rod with maximum worth is completely withdrawn	-	5.9	4.1	
Maximum reactivity insertion rate (%Δk/k/s)				
• Shim rod	-	0.014	0.011	<0.015
• Regulating rod	-	0.010	0.009	<0.030
Maximum excess reactivity (%Δk/k)	-	3.1	3.3	<5.0
Reactivity coefficient				
• Fuel temperature coefficient (%Δk/k/^{o}C)	$(-1.8\sim-1.3)\times10^{-6}$ for [300~600K]	$(-1.9\sim-1.4)\times10^{-3}$ for [300~700K]	$(-2.2\sim-1.6)\times10^{-3}$ for [300~700K]	<0
• Coolant temperature coefficient (%Δk/k/ ^{o}C)	$(-1.0\sim-1.4)\times10^{-4}$ for [313~338K]	$(-1.7\sim-2.8)\times10^{-2}$ for [300~400K]	$(-1.8\sim-2.7)\times10^{-2}$ for [300~400K]	<0
• Coolant void reactivity coefficient (%Δk/k/%void)	$(-1.7\sim-2.2)\times10^{-1}$ for [0~20%void]	$(-2.4\sim-5.8)\times10^{-1}$ for [0~50%void]	$(-2.1\sim-5.8)\times10^{-1}$ for [0~50%void]	<0
Kinetics constants				
• Mean lifetime of prompt neutron(s)	8.1×10^{-5}	7.46×10^{-5}	7.00×10^{-5}	-
• Effective delayed neutron fraction(-)	0.008	0.00735	0.00698	-
Power peaking factors				
• Radial direction (-)	-	1.44	1.47	-
• Axial direction (-)	-	1.40	1.38	-
• Maximum power density (W/cm^3)	144	164	102	-

Table 3 Physical properties of fuel plates

Materials	Density (Kg/m^3)	Thermal conductivity(W/(mK))		Volumetric heat capacity ($KJ/(m^3K)$)	
		Temperature (^oC)	Thermal conductivity	Temperature (^oC)	Volumetric heat capacity
Fuel meat	5385	97.5		10	2353.4
				100	2462.3
				300	2705.1
Fuel Cladding	2700	20	130	20	2487.0
		100	142	100	2600.0
		300	173	300	2826.1

Table 4 Analysis results for KUR core at rated thermal power of 5MW

Codes	Eureka-2/RR		Coolod-N2		Thyde-W	
Channels	Average	Hot	Average	Hot	Average	Hot
Coolant velocity (m/s)	3.017		3.017		3.000	
Pressure difference (MPa)	0.0157		0.0130		0.0350	
Minimum DNBR (-)	9.55	4.91	9.47	4.91	9.47	4.87
Maximum heat flux (W/cm^2)	28.12	54.69	28.14	54.29	28.10	54.67
Coolant temperature at core outlet (^{o}C)	55.30		55.04		55.09	
Maximum fuel temperature (^{o}C)	71.39	90.51	71.57	91.58	71.77	88.45
Maximum cladding temperature (^{o}C)	69.83	87.52	69.46	88.56	70.52	86.63
Maximum coolant temperature (^{o}C)	56.29	61.56	55.84	63.42	56.13	63.87

Table 5 EUREKA-2/RR calculations for accidental control rod withdrawal transients at natural circulation mode

	Items	Case 1	Case 2	Case 3
Input data	Initial power(W)	0.001	1.0	100,000
	Rate of reactivity insertion(%Δk/k/s)	0.015		
	Inserted reactivity(%Δk/k)	-1.64	-1.58	-1.31
	Initial temperature of primary coolant(^{o}C)	45		
	Pressure at core outlet(MPa)	0.17		
	Initial flow rate of primary coolant(m^3/h)	18		
	Scram power(KW)/Scram reactivity inserting way	120/S-shaped inserting way		
	All negative reactivity feedbacks	Neglected		
	Mean lifetime of prompt neutron(s)	7.00×10^{-5}		
	Effective delayed neutron fraction(-)	0.00698		
	Bulk water temperature rising hot channel factor(-)	1.54		
	Heat flux hot channel factor(-)	1.34		
	Heat transfer coefficient hot channel factor(-)	1.43		
	Cross-sectional power peaking factor(-)	1.44		
	Axial power peaking factor(-)	1.40		
Results	Time of arrival of the scram power(s)	42.44	38.86	5.28
	Maximum power(MW)/Arrival time(s)	0.19/42.65	0.15/39.07	0.12/5.38s
	Maximum fuel temperature(^{o}C)/Arrival time(s)	50.53/43.05	51.73/39.46	82.26/5.75
	Maximum cladding temperature(^{o}C)/Arrival time(s)	50.52/43.06	51.72/39.46	82.22/5.76
	Temperature at hot channel outlet(^{o}C)/Arrival time(s)	45.37/52.3	45.45/47.7	57.71/7.2
	Minimum DNBR(-)/Arrival time(s)	94.54/42.99	79.92/39.39	19.16/0.003

Table 6 EUREKA-2/RR calculations for accidental control rod withdrawal transients at forced circulation mode

	Items	Case 1	Case 2	Case 3	Case 4	Case 5	Case 6
Input data	Initial power(W)	0.001	1.0	1,000,000	4,500,000	5,000,000	5,500,000
	Rate of reactivity insertion(%Δk/k/s)	0.015					
	Inserted reactivity(%Δk/k)	-1.66	-1.62	-1.31	-1.11	-1.08	-1.05
	Initial coolant temperature at core inlet(^{o}C)	55	55	53.9	50.2	49.6	49.0
	Pressure at core inlet(MPa)	0.17					
	Primary coolant flow rate(m^3/h)	800					
	Scram power(MW)/Scram reactivity inserting way	6/S-shaped inserting way					
	All negative reactivity feedbacks	Neglected					
	Mean lifetime of prompt neutron(s)	7.00×10^{-5}					
	Effective delayed neutron fraction(-)	0.00698					
	Bulk water temperature rising hot channel factor(-)	1.54					
	Heat flux hot channel factor(-)	1.34					
	Heat transfer coefficient hot channel factor(-)	1.37					
	Cross-sectional power peaking factor(-)	1.44					
	Axial power peaking factor(-)	1.40					
Results	Time of arrival of the scram power(s)	43.75	41.15	20.92	7.27	5.28	3.02
	Maximum power(MW)/Arrival time(s)	11.36/43.97	8.60/41.36	6.11/21.04	6.02/7.37	6.02/5.38	6.01/3.11
	Maximum fuel temperature(^{o}C)/Arrival time(s)	129.39/44.05	116.38/41.43	104.83/21.13	102.16/7.45	101.76/5.44	101.36/3.17
	Maximum cladding temperature(^{o}C)/Arrival time(s)	125.61/44.05	113.41/41.44	102.40/21.13	99.74/7.45	99.34/5.44	98.94/3.17
	Temperature at hot channel outlet(^{o}C)/Arrival time(s)	78.85/44.12	74.18/41.5	69.86/21.19	66.13/7.51	65.53/5.51	64.93/3.24
	Minimum DNBR(-)/Arrival time(s)	2.2898/44.05	2.9551/41.43	3.6616/21.13	3.6813/7.44	3.6814/5.44	3.6815/3.16

Table 7 EUREKA-2/RR calculations for cold water injection induced reactivity insertion transients

	Items	Case 1	Case 2	Case 3	Case 4
Input data	Initial power(KW)	90	100	110	110
	Coolant temperature coefficient(%Δk/k/°C)	-0.018			
	Inserted reactivity(%Δk/k)/Reactivity inserting way	0.630/Step inserting way			
	Initial temperature of primary coolant(°C)	45			
	Pressure at core top(MPa)	0.17			
	Initial flow rate of primary coolant(m^3/h)	18			
	Scram power(KW)/Scram reactivity inserting way	120/S-shaped inserting way			300/S-shaped inserting way
	The other negative reactivity feedbacks	Neglected			
	Mean lifetime of prompt neutron(s)	7.00×10^{-5}			
	Effective delayed neutron fraction(-)	0.00698			
	Bulk water temperature rising hot channel factor(-)	1.54			
	Heat flux hot channel factor(-)	1.34			
	Heat transfer coefficient hot channel factor(-)	1.37			
	Cross-sectional power peaking factor(-)	1.44			
	Axial power peaking factor(-)	1.40			
Results	Time of arrival of the scram power(s)	0.0041	0.0025	0.0012	0.0230
	Maximum power(MW)/Arrival time(s)	0.92/0.233	1.02/0.231	1.11/0.230	1.19/0.250
	Maximum fuel temperature(°C)/Arrival time(s)	86.14/0.605	89.84/0.604	93.53/0.603	95.28/0.626
	Maximum cladding temperature(°C)/Arrival time(s)	86.09/0.608	89.78/0.607	93.47/0.605	95.22/0.629
	Temperature at hot channel outlet(°C)/Arrival time(s)	57.53/6.1	57.63/5.7	57.73/5.4	57.78/5.3
	Minimum DNBR(-)/Arrival time(s)	14.82/0.541	13.39/0.540	12.21/0.538	11.73/0.562

Table 8 EUREKA-2/RR calculations for reactivity insertion transients due to removal of irradiation samples

	Items	Case 1	Case 2	Case 3	Case 4
Input data	Initial power(MW)	1.0	4.5	5.0	5.5
	Inserted reactivity(%Δk/k)	0.2			
	Reactivity inserting way	Step inserting way			
	Initial coolant temperature at core inlet(^{o}C)	53.9	50.2	49.6	49.0
	Pressure at core inlet(MPa)	0.17			
	Primary coolant flow rate(m^3/h)	800			
	Scram power(MW)/Scram reactivity inserting way	6/S-shaped inserting way			
	All negative reactivity feedbacks	Neglected			
	Mean lifetime of prompt neutron(s)	7.00×10^{-5}			
	Effective delayed neutron fraction(-)	0.00698			
	Bulk water temperature rising hot channel factor(-)	1.54			
	Heat flux hot channel factor(-)	1.34			
	Heat transfer coefficient hot channel factor(-)	1.37			
	Cross-sectional power peaking factor(-)	1.44			
	Axial power peaking factor(-)	1.40			
Results	Time of arrival of the scram power(s)	24.41	0.032	0.012	0.005
	Maximum power(MW)/Arrival time(s)	6.03/24.52	6.29/0.172	6.97/0.156	7.66/0.150
	Maximum fuel temperature(^{o}C)/Arrival time(s)	104.65/24.58	101.72/0.31	106.28/0.30	111.02/0.30
	Maximum cladding temperature(^{o}C)/Arrival time(s)	102.23/24.59	99.32/0.32	103.63/0.30	108.13/0.30
	Temperature at hot channel outlet(^{o}C)/Arrival time(s)	69.84/24.65	65.89/0.39	66.89/0.37	67.97/0.37
	Minimum DNBR(-)/Arrival time(s)	3.677/24.58	3.707/0.32	3.361/0.30	3.062/0.29

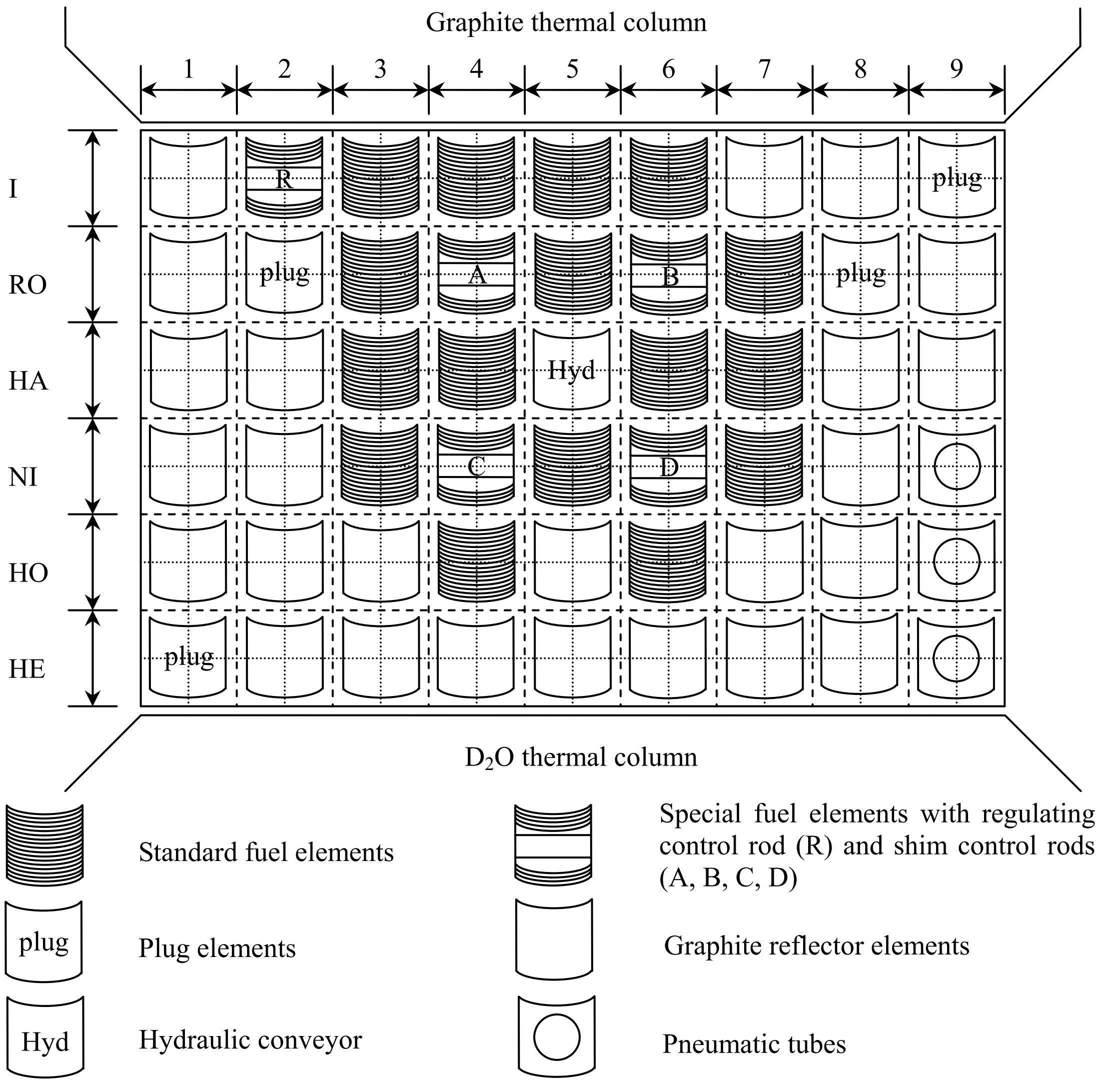

Fig. 1 KUR core loading pattern with minimum fuel elements.

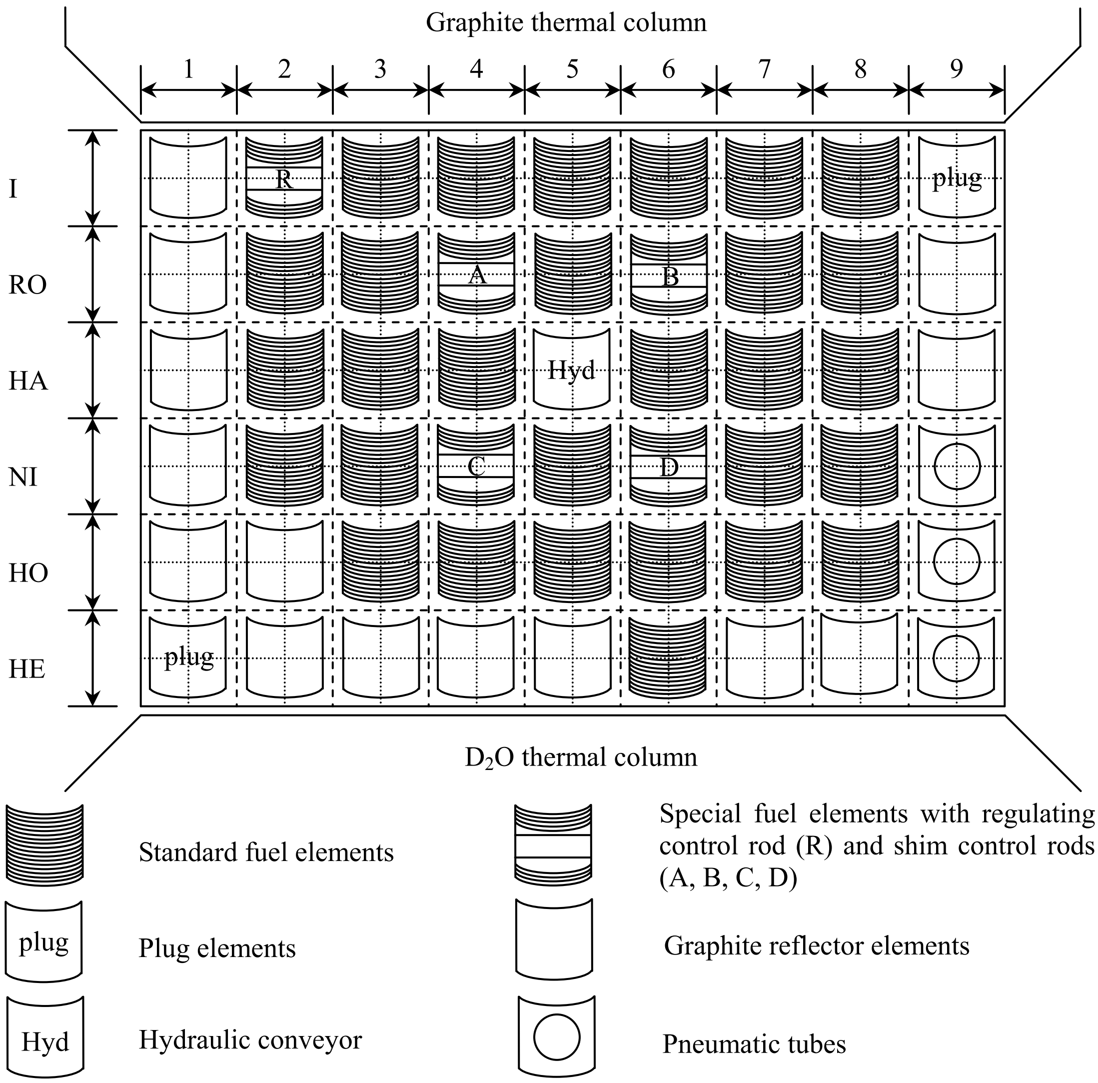

Fig. 2 KUR core loading pattern with maximum fuel elements.

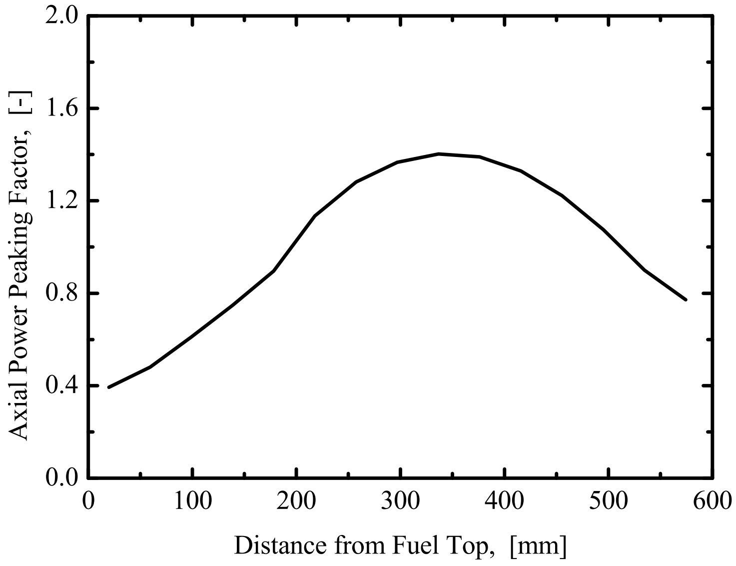

Fig. 3 Axial power distribution along the distance from fuel top in KUR minimum core.

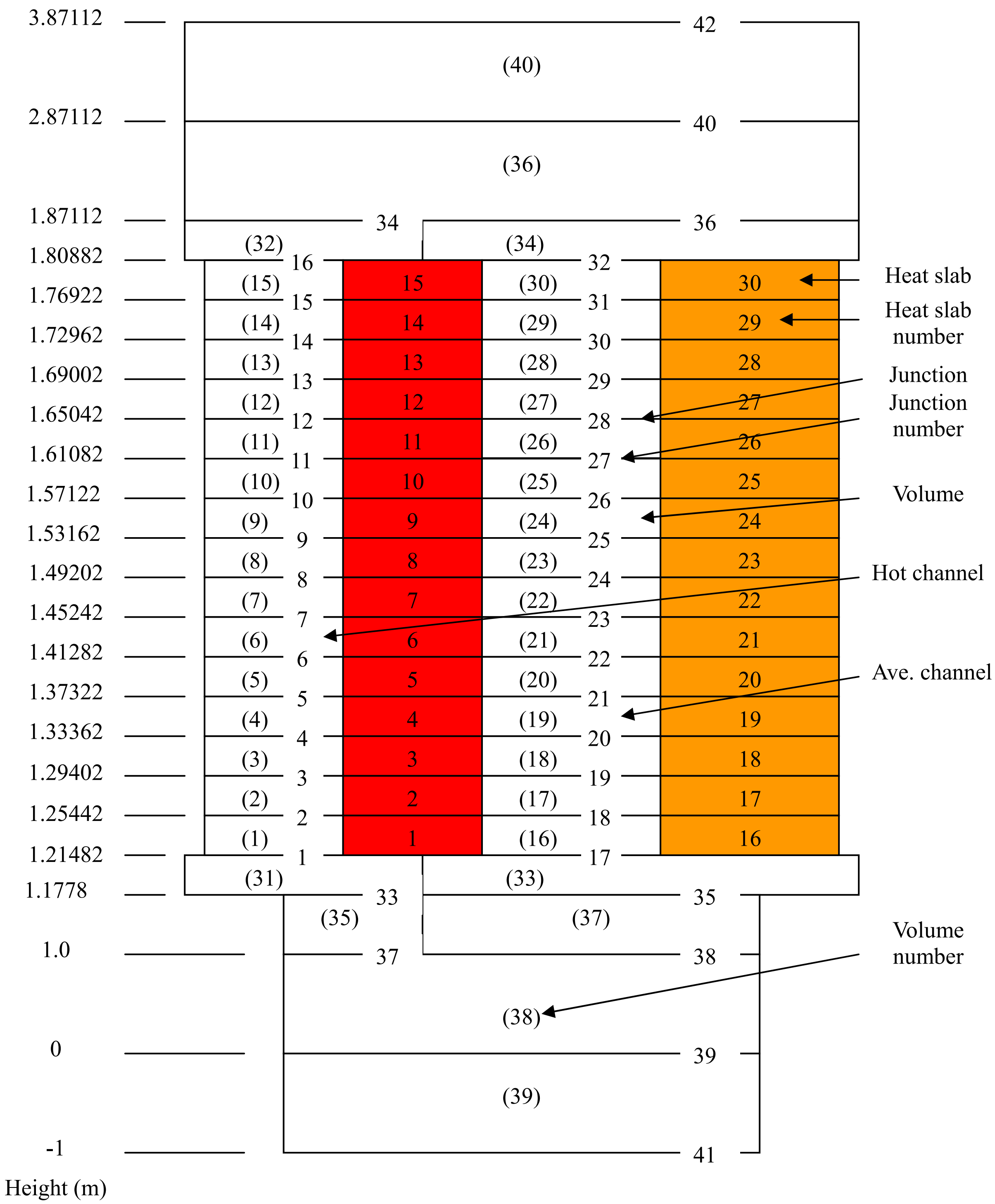

Fig. 4 Schematic diagram of reactor core nodalization.

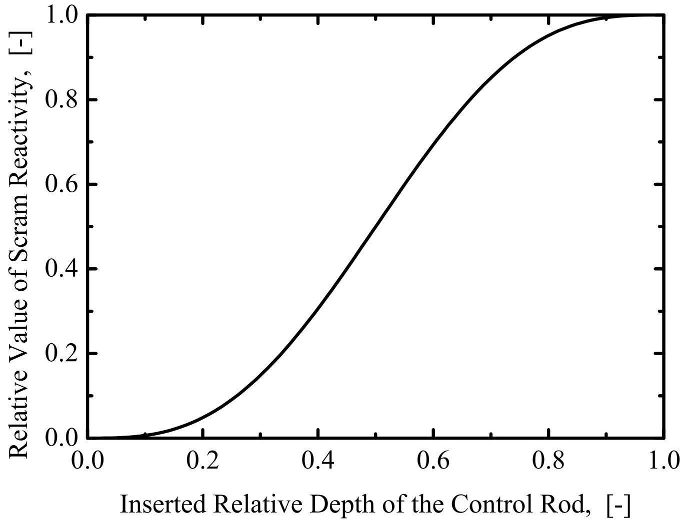

Fig. 5 The S-shaped increasing curve of the inserted scram reactivity.

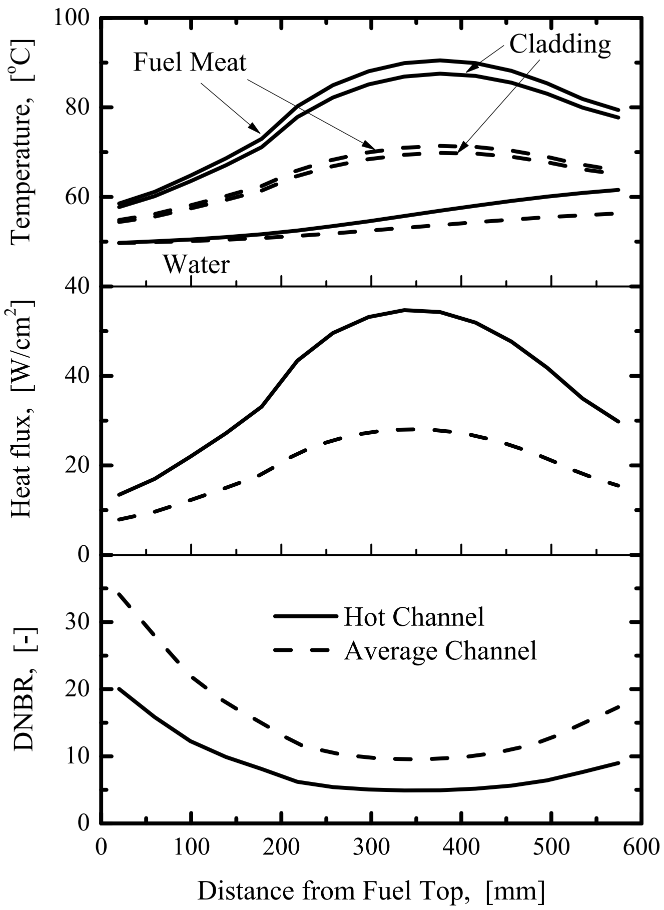

Fig. 6 Steady state calculation results at rated power of 5MW.

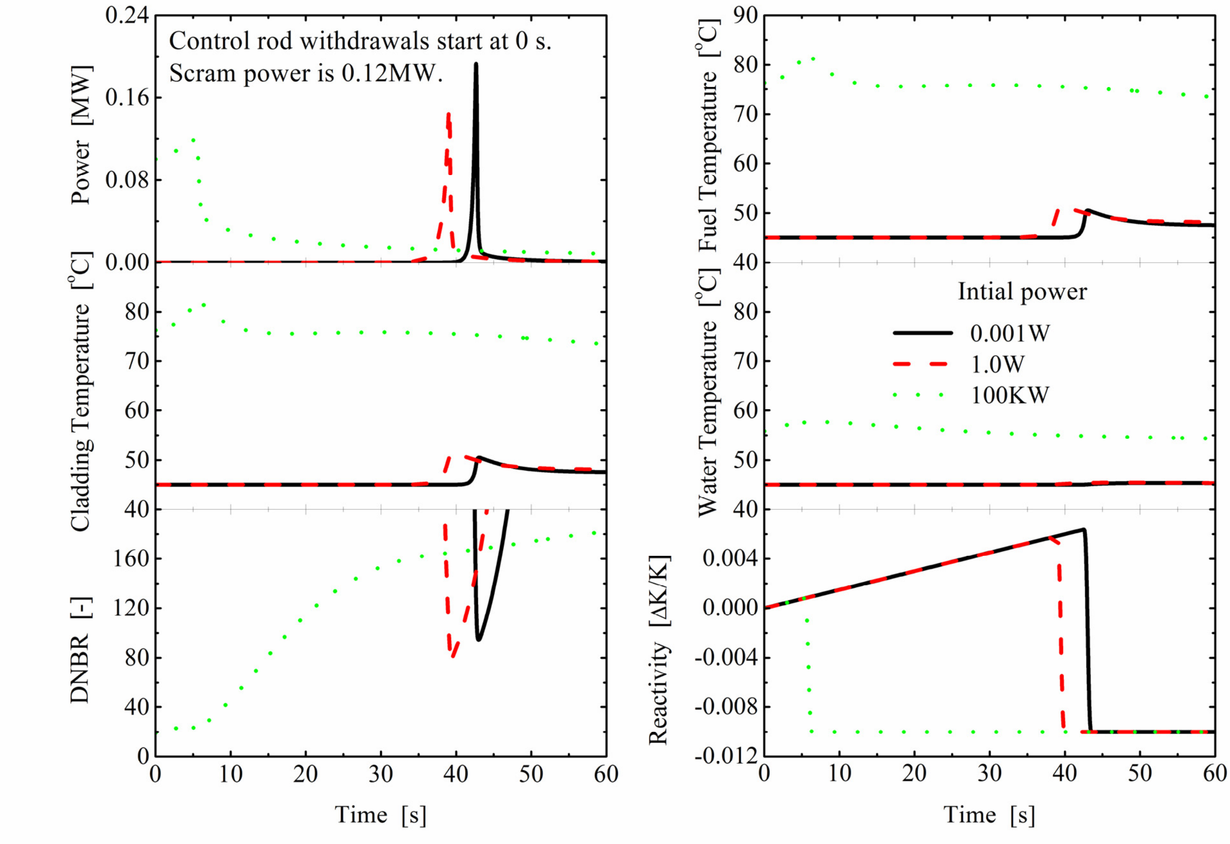

Fig. 7 Accidental control rod withdrawal transients at natural circulation mode.

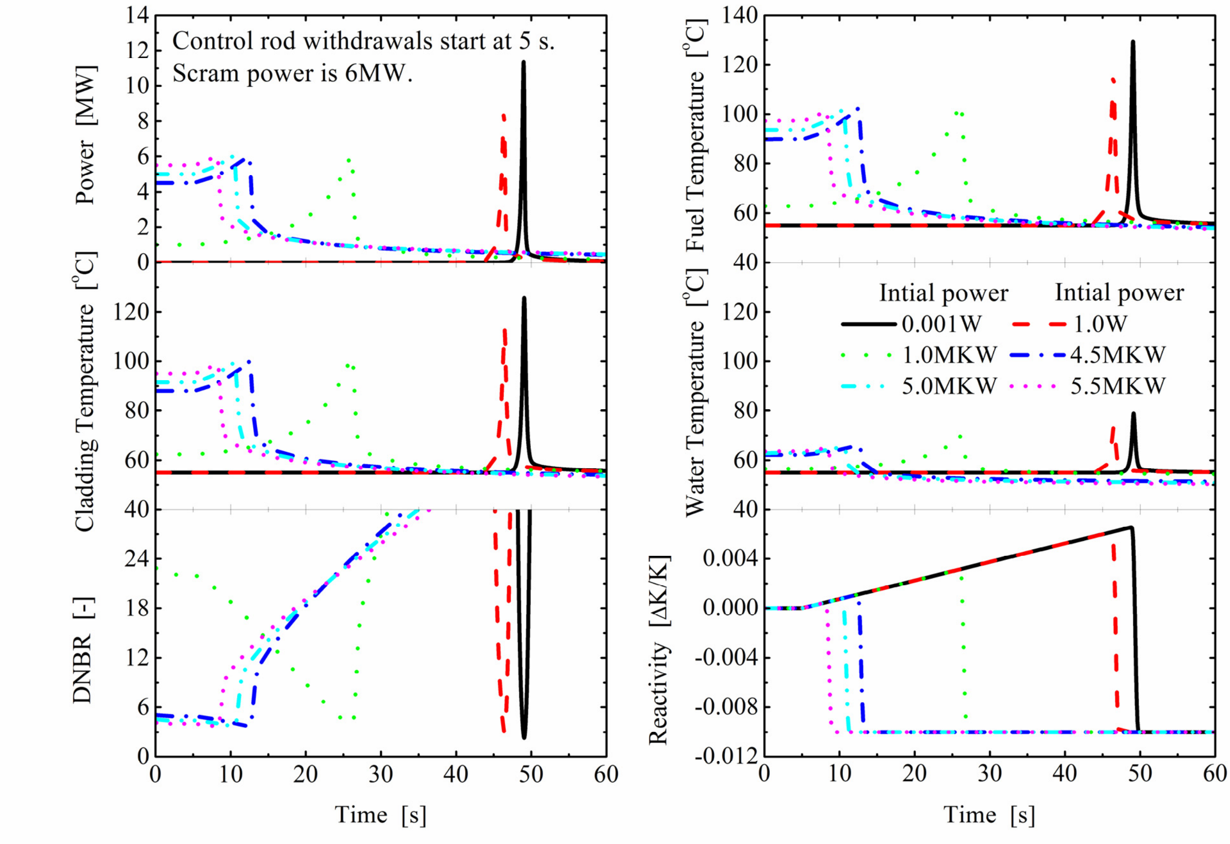

Fig. 8 Accidental control rod withdrawal transients at forced circulation mode.

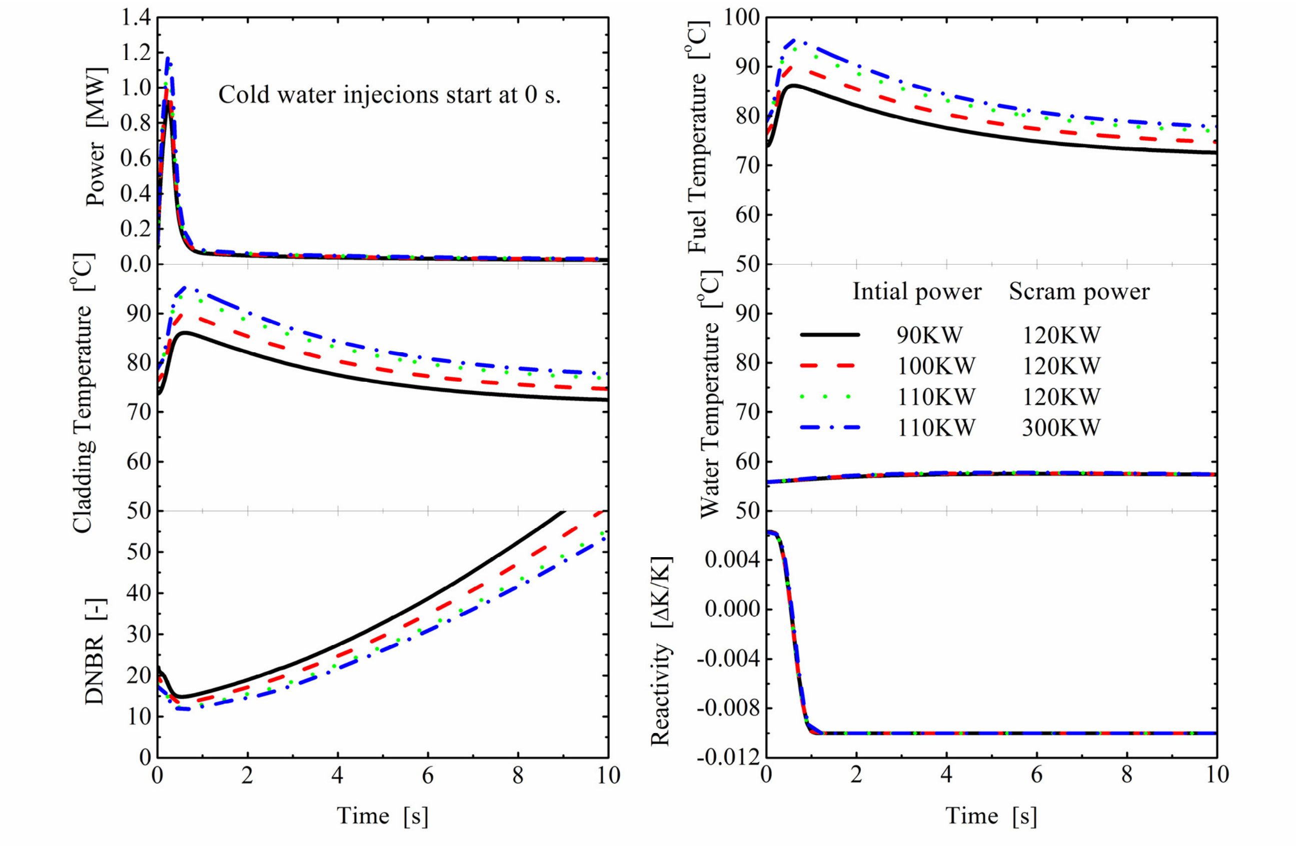

Fig. 9 Cold water injection induced reactivity insertion transients.

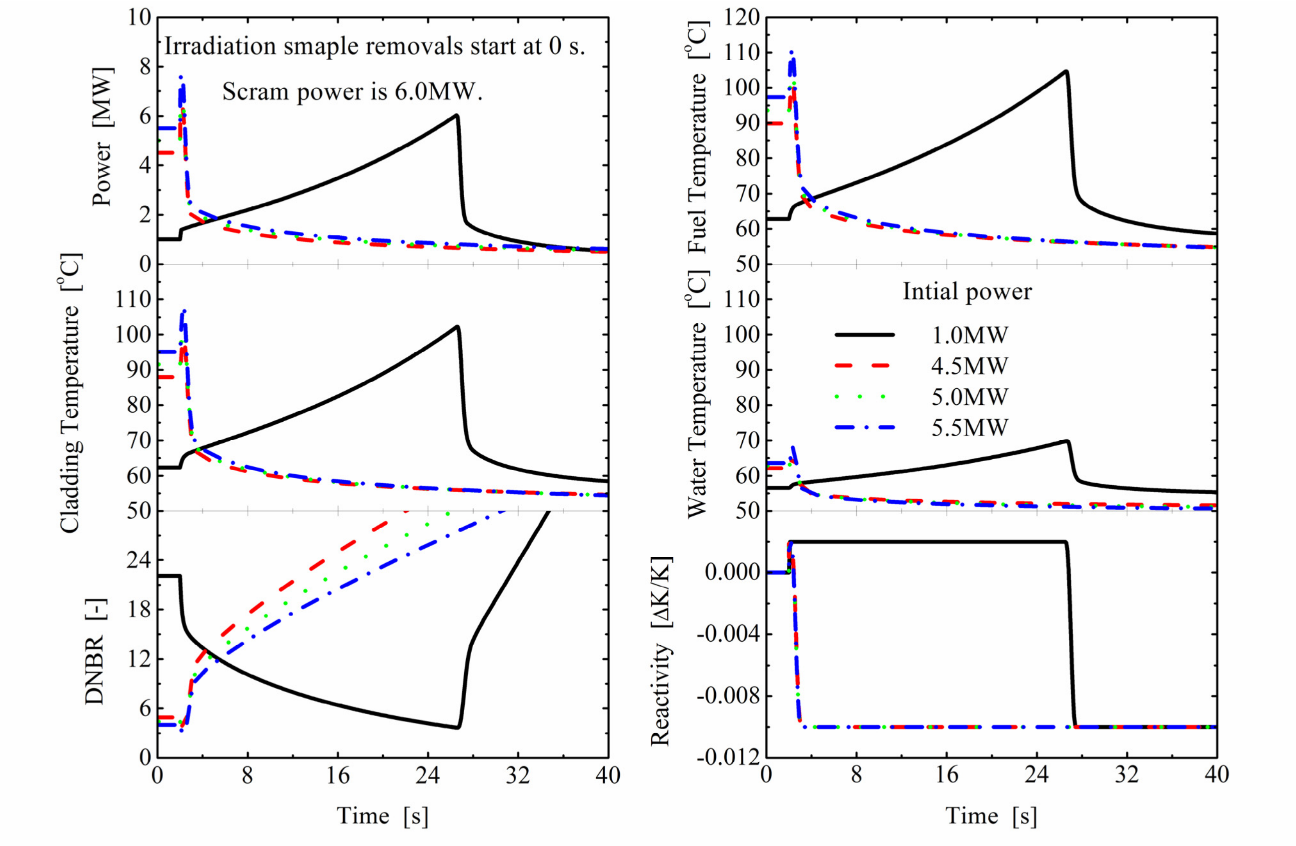

Fig. 10 Reactivity insertion transients due to removal of irradiation samples.