HYPOTHETICAL ACCIDENT ANALYSES OF THE PROPOSED SPLIT CORE AT NIST USING ANL-PARET CODE

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ABSTRACT

Preliminary design basis accident analyses have been performed on the proposed split core to examine the thermal-hydraulics (T/H) safety characteristics of the new design. The multi-channel T/H safety analysis code PARET, developed by Argonne National Laboratory, is employed to perform the transient analyses with the power and kinetics parameters provided by neutronics calculations. Two common design basis transient overpower phenomena, control rod withdrawal accident during the core start-up and maximum reactivity insertion accident at the full power operation condition, are investigated in the paper. The accident scenarios are investigated at the start-up (SU) and end-of-cycle (EOC) core conditions in an equilibrium cycle to examine the T/H performance characteristics at different stages of the cycle. The postulated accidental transients are simulated in PARET with real-time monitoring of the fuel cladding temperature, power rate and mass flow rate. It has been shown that the new design can achieve reasonable T/H safety margins by comparing the minimum critical heat flux ratio and the peak clad temperature to the safety criteria specified for typical low enriched uranium (LEU) test reactors.

1. Introduction

In anticipation of the eventual retirement of the research reactor (NBSR-National Bureau of Standards Reactor) at the National Institute of Standards and Technology (NIST), research efforts are ongoing at NIST Center for Neutron Research (NCNR) to design a new research reactor. The primary purpose of the new reactor is to optimize cold neutron production for scientific neutron scattering experiments. The new design has two high quality cold neutron sources (CNS). The thermal power of the new reactor is 20 MW and the operating cycle of the equilibrium core is set to be around 30 days. Low enriched uranium (LEU) fuel - U_3Si_2/Al fuel – with U-235 enrichment less than 20 wt.% is used to comply with nuclear non-proliferation agreements.

Neutronics studies have been performed to demonstrate the viability of the design. A full core model has been developed using MCNP6 [1]. The core consists of 18 MTR-type fuel elements, which are arranged in a compact pattern and are horizontally split into two halves. The core is cooled and moderated by light water but reflected by heavy water to create a large thermal neutron flux trap in the reflector region between the core halves. To maximize the scientific utilization of neutrons, the reactor vessel in the current design is equipped with two horizontal cold neutron source beams and four tangential thermal neutron beams. The CNS's were positioned to keep the heat load of each below 4 kW and to optimize cold neutron brightness performance. The neutronics feasibility of the new design is justified by the effective multiplication factor of the system and the maximum thermal flux performance. The superiority of the new design is demonstrated by the cold neutron spectrum brightness in the CNS per unit reactor power [2, 3].

In this paper, the preliminary design basis accident analyses are performed on the split core to examine the thermal-hydraulics safety characteristics. Hypothetical transient overpower accidents are analyzed using the system safety analysis code PARET [4, 5], which was developed by Argonne National Laboratory. The accidental scenarios in both the start-up (SU) and end-of-cycle (EOC) core status for an equilibrium cycle are examined to compare the T/H performance characteristics at different conditions. In the PARET model, the active core is simplified as a two-channel model with kinetics and feedback parameters provided by neutronics calculations. The overpower transients are simulated with the real-time monitoring of the fuel cladding temperature, power rate and mass flow rate. The thermal-hydraulics (T/H) safety margins of the new design are verified by the minimum critical heat flux ratio and the peak clad temperature.

2. Overview of the Split Core Design

The proposed NIST reactor employs the standard 'tank-in-pool' design pattern, in which a cylindrical heavy water tank 2.5 m diameter and 2.5 m height is placed in the center of a large light water pool that functions as the thermal and biological shields. The core design embraces the compact core concept and creates a thermal flux trap in an easily accessible location in the reflector tank to maximize the flux. The reactor core is enclosed by two zirconium core boxes to separate heavy water and light water. To maximize useful flux trap volume in the reflector, an innovative horizontally split core is employed in the design such that the thermal flux trap between the core halves would provide ideal locations to place cold neutron sources (CNS). Two vertical liquid deuterium CNS are placed in the flux trap located in the north and south sides of the core. Two CNS beam tubes are connected to the CNSs with guides pointing in the north and south directions. Four thermal beam tubes are placed in the east and west sides of the core at different elevations (20 cm above and below core mid-plane) with the pointing direction tangential to the core face. A schematic view of the reactor components and the fuel element radial layout in the split core is illustrated in Fig. 1.

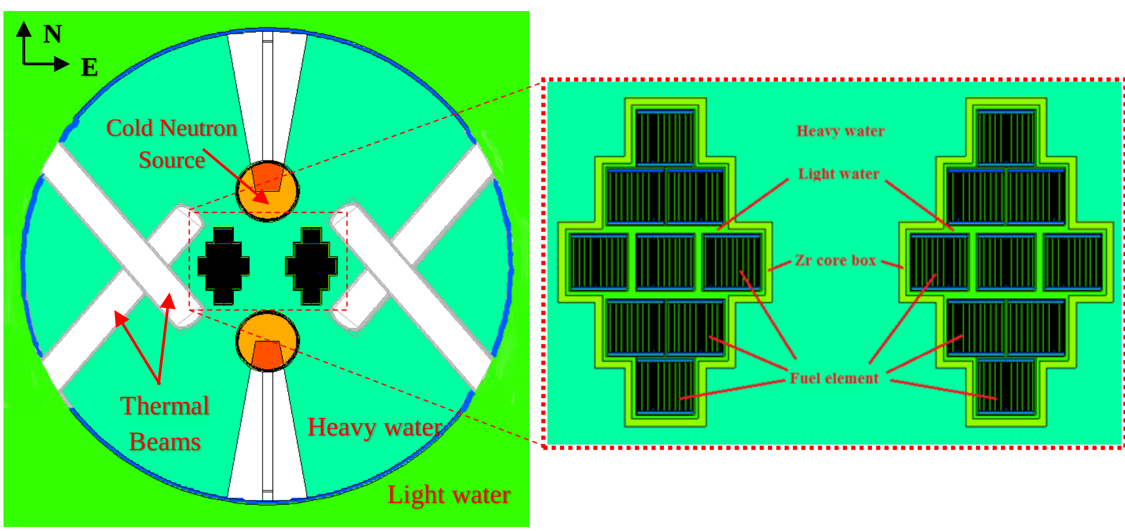

Fig. 1. A schematic plan view of the mid-plane of the reactor with horizontally split cores [3].

The power distribution in the fuel is required for reactor safety analyses to specify the initial heat source profile for the heat structure in the T/H model. In this study, the power density for a given position in a core is calculated by MCNP6, in which we conservatively assume that all the recoverable fission energy is deposited at the point of fission, and the power density is proportional to fission density. In order to obtain a detailed power distribution for the core, the fuel plate (really the fuel meat inside the plate) is evenly divided into 3 stripes, and each stripe is evenly divided into 30 axial pieces. Fig. 2 depicts the axial power distributions along the hottest and average coolant flow channel in SU and EOC core, respectively. The hot channel here refers to a coolant channel that contains the fuel plate producing the largest amount of power, whereas the average channel represents the core average power effects.

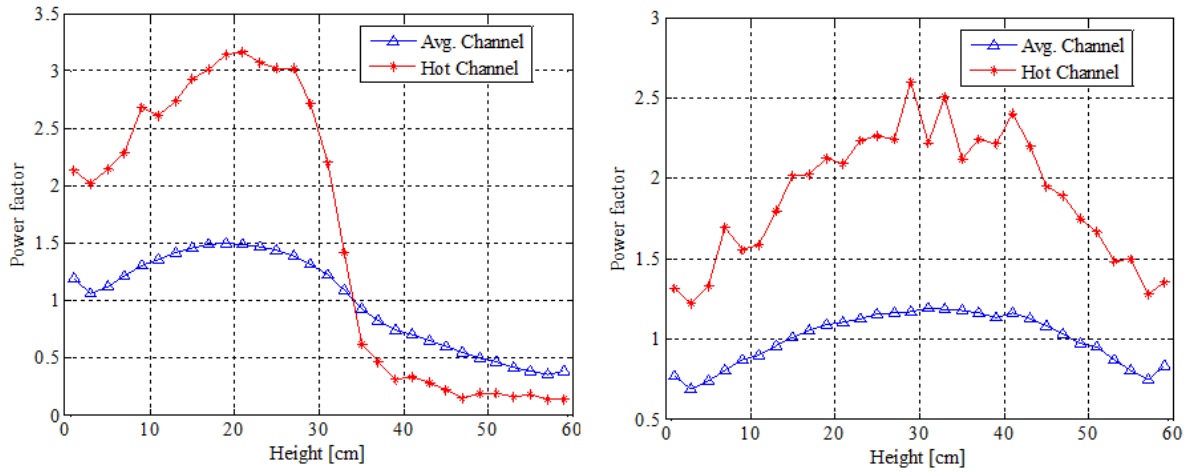

Fig. 2. Steady-state axial power distribution in the SU (left) and EOC (right) core.

The results shown in Fig. 2 represent the steady-state axial power distributions that have been normalized to the total reactor power, and serve as initial power profiles for all point kinetics reactor model based transient analyses. Since the control elements are partially inserted from the top of the core to compensate the large excess reactivity at SU, the peak power is significantly skewed to the bottom region of the core at SU (see the left figure in Fig. 2). This asymmetry vanishes in EOC case as the controls are fully withdrawn out the core at this status (see the right figure in Fig. 2).

The necessary kinetics parameters required for safety analysis, the prompt neutron generation time and the effective delayed neutron fraction, in SU and EOC are summarized in Table 1. They are calculated with Monte Carlo code MCNP6 using the adjoint-weighted tally methodology [1]. The uncertainty shown in the table represents the standard deviation of the corresponding parameter estimated by the MCNP6 tallies.

Table 1. Kinetics parameters calculated by MCNP6

Kinetics Parameter	**SU (+/-)**	**EOC (+/-)**
Prompt neutron generation time - Λ (μs)	202.61 (4.60)	203.82 (4.42)
Effective delayed neutron fraction (β_{eff})	0.00740 (0.00047)	0.00717 (0.00041)

Four sets of H-shaped hafnium control blades are utilized as both criticality and safety control elements for the reactor. Due to the limited space in the core, all control blades are about 0.5 cm thick and 60 cm long (the same length as the active fuel length). The blades are controlled by a mechanical driver located at the bottom of the core but with the fully withdrawn positions at the top of the core. Reactivity worths of the control rods are needed by the safety analysis code to determine the correct negative reactivity inserted to the core after scram. Fig. 3 shows the reactivity worths of the control rods in the split core at SU and EOC. Due to different excess reactivity existing at different stages of the cycle, the critical control rod position at SU and EOC are different. As indicated in Fig. 3, the differential reactivity worth at the critical position for SU is higher than the one for EOC. This will result in a greater reactivity insertion rate for SU than EOC at the time of scram, if the control rod insertion speed is assumed to be constant and the reactor is assumed to be operating at critical status. Therefore a faster power reduction after scram is expected for the SU case, and this fact is verified by the PARET simulation results discussed in the result section.

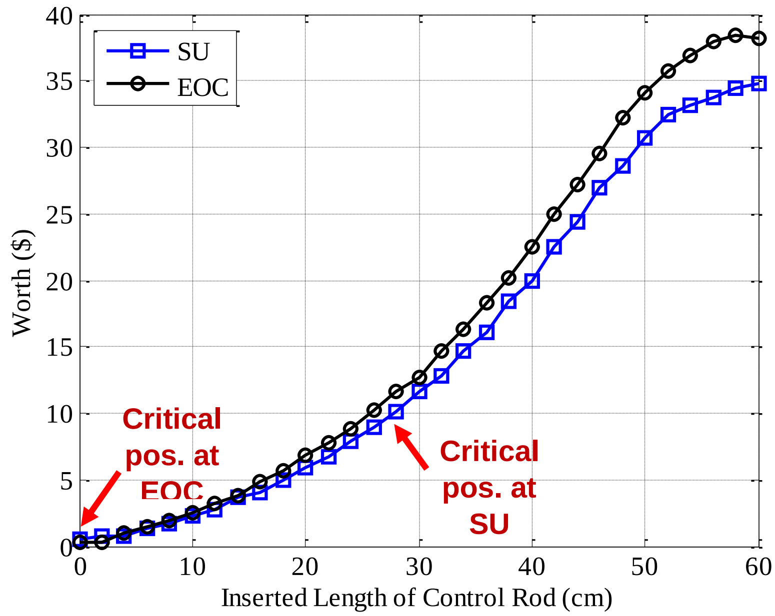

Fig. 3. Reactivity worths of the control rods at SU and EOC.

3. The PARET Model and Safety Analysis Criteria

The ANL-PARET code is intended primarily for safety analysis of research and test reactors that use plate-type fuel elements, or round fuel pins. The code employs 1-D hydrodynamics, 1-D heat transfer and point kinetics model with considerations of proper reactivity feedback. The hydrodynamics equations and heat transfer equations are numerically solved simultaneously to obtain the temperature distributions for fuel, clad and coolant along the axial direction. The solutions to these equations also yield the pressure drop across the core, as well as point-wise fluid enthalpies, pressures, and mass flow rates in the coolant channels. PARET was initially written for nondestructive reactivity accident analyses (i.e., overpower transient analyses) [4], and was recently extended to provide an ability to follow a loss-of-flow (LOF) transient with down flow initially, through flow reversal and finally through the establishment of natural convection cooling [5]. All these features sufficiently meet the requirements of the safety analysis on the split core. The work presented here is mainly focused on transient overpower analyses, and the LOF analyses will be investigated in future research.

For simplicity, a two-channel PARET model is developed in this study to account for physical conditions in the hot and average channel, respectively. Each channel includes a 1-D slab geometry of fuel plate, extending from the plate centerline to the coolant centerline on both sides of the plate. Appropriate volume fractions are weighted for each channel to account for proper heat source transferred in the channel. A diagram to illustrate a T/H channel is shown in Fig. 4 with the corresponding dimensions provided in Table 2.

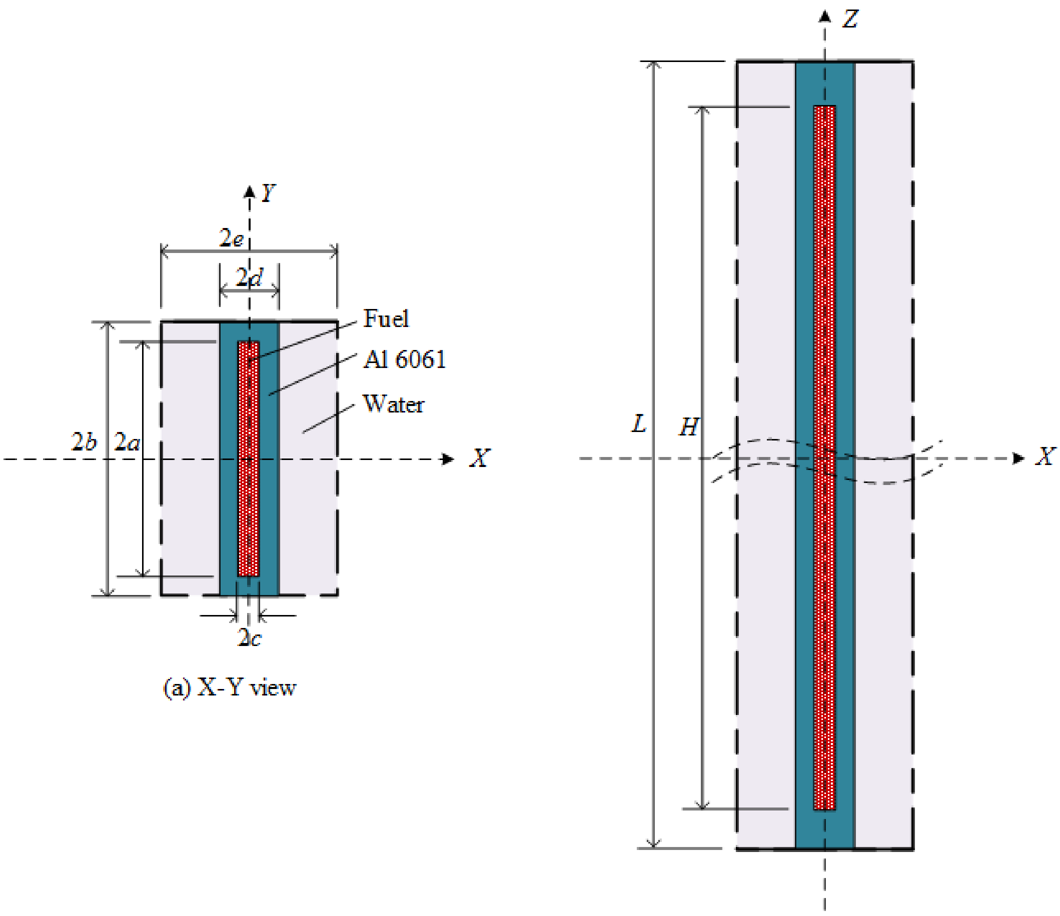

Fig. 4. Diagram of the flow channel model in the plate-type core.

Table 2. The dimensions of the channel model in Fig. 4.

Parameter	**Size (cm)**
Half width of the fuel meat (*a*)	3.067
Half width of the fuel plate (*b*)	3.3325
Half thickness of the fuel meat (*c*)	0.033
Half thickness of the fuel plate (*d*)	0.0635
Half pitch of the fuel plates (*e*)	0.211
Length of the fuel meat (*H*)	60
Length of the channel (*L*)	67.28

PARET is not able to model the entire primary coolant system of the reactor, but rather develops equivalent T/H characteristics of a flow channel in the core region by providing proper boundary conditions. For the split core design, a downward flow with a total flow rate 8000 gal/min or 1817 m^3/hour was assumed and the inlet coolant temperature was set at 37 °C. With these conditions, the temperature rise along the average channel was about 10 K based on energy conservation. The core was assumed to be operated at atmospheric pressure and the outlet pressure was assumed to be 135 kPa. All these T/H conditions were designated with the intention to achieve T/H performance similar to the existing reactor at NIST [6]. A summary of the required T/H boundary conditions and parameters based on the channel dimensions is outlined in Table 3.

Table 3. T/H Conditions and parameters used in the PARET model.

Conditions and Parameters	**Values**
Outlet pressure (kPa)	135
Inlet temperature (°C)	37
Inlet volumetric flow rate (gpm)	8000
Flow area of the channel (cm^2)	1.9662
Heated surface area of the channel (cm^2)	736
Rectangular channel width (cm)	6.67
Wetted perimeter of the channel (cm)	13.63
Hydraulic diameter (cm)	0.58

To satisfy safety concerns, two thermal constraints are examined during the course of the transients. The first one is the peak clad temperature (PCT), which is a direct indicator of the physical damage to the fuel plate. For reactor designs, it is required the PCT must not reach the fuel blister temperature, which is taken as **515 °C to 575 °C** for silicide LEU fuel [7]. Another constraint is on the critical heat flux (CHF), which characterizes the departure of nucleate boiling (DNB) occurring at the surface of the fuel cladding. The DNB may significantly reduce the heat transfer coefficient and subsequently cause the flow burnout phenomena. A metric to address the CHF constraint is the minimum critical heat flux ratio (MCHFR) that is defined as the DNB heat flux estimated from an appropriate correlation divided by the expected heat flux. The limit of MCHFR is set **1.32**, which is also obtained from the safety report of the existing NIST reactor [5]. To conform to the available options in PARET, the Mirshak DNB correlation [8] is used to estimate the DNB critical heat flux.

4. Transient Overpower Accident Analyses

4.1 Steady-State Conditions

The PARET inputs have been run to establish the steady-state conditions for the core at full power (20 MW). Table 4 shows the respective results for SU and EOC. It can be seen that the PCT and MCHFR at steady-state conditions for both SU and EOC satisfy the thermal constraints as specified previously.

Table 4. T/H performance characteristics at steady-state conditions

Core Status	**SU**	**EOC**
Coolant outlet temperature [°C]	52.96	56.09
Peak clad temperature [°C]	107.05	95.51
Peak fuel temperature [°C]	120.46	106.18
MCHFR	2.21	2.77

4.2 Control Rod Withdrawal Start-up Accident

The control rod withdrawal start-up accident is modeled with a slow ramp reactivity insertion to a critical core from a very low power with selected assumptions made to examine the severity of the event. The core condition is considered at both the start-up (SU) and the end-of-cycle (EOC) of an equilibrium cycle to show the different power transient behaviors under the accidental scenarios. The reactor is initially critical and operating at a power of 2 W (0.01% of the full power). The ramp reactivity is assumed to be inserted with a rate 0.1$/s to mimic the slow reactor start-up procedure.

The reactor scram occurs with a power trip at 24 MW (120% of the full power). A time delay constant 25 ms is defined in the model to account for the finite time required for the safety rods to start the movement after scram. The control rods are assumed to move with a constant rate 1.2 m/s for scram. At this moment, all reactivity feedback coefficients (including fuel, coolant density, and moderator void) and the period trip are neglected in the analyses.

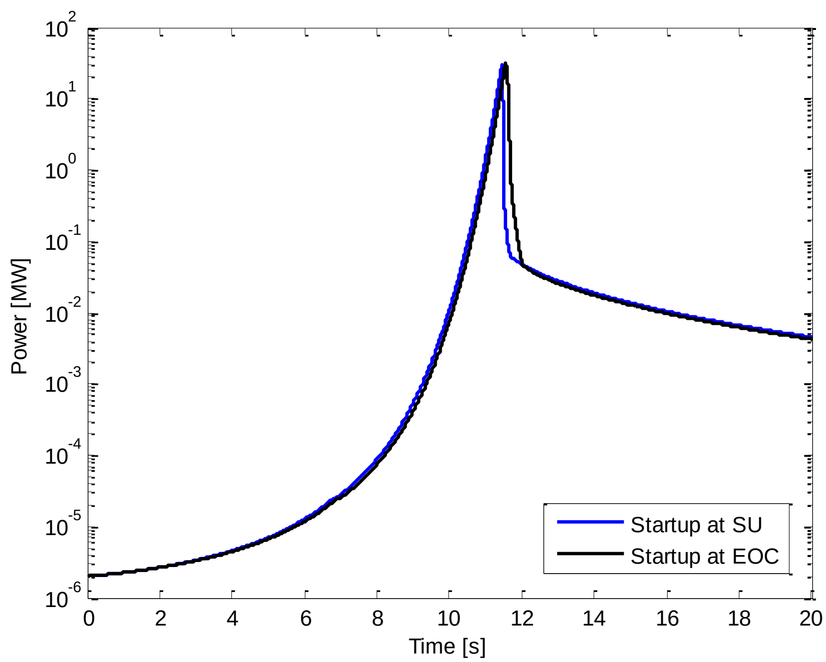

Fig. 5. Reactor power in start-up accidents.

Fig. 5 shows the reactor power transient behavior of first 20 s in the start-up accidents. Because the kinetics parameters for SU and EOC have only slight differences (see Table 1), the power increases with identical rates in start-up transients for SU and EOC, and both reach the maximum power around 30 MW at about 12 s into the accidents. They then both quickly drop off to the decay heat power level after the scram. The power reduction curve for the SU, however, exhibits a shorter time constant, this is due to the higher differential reactivity worth of control rods at the critical position in SU (see Fig. 3). Since the core is initially operating at critical and the control rods are assumed to move with a constant rate, the SU core thereby obtains larger negative reactivity than the EOC core in a period following the scram. As a result, the power at the SU case decreases faster after the scram.

The corresponding PCT and MCHFR behavior are shown in Fig. 6 and 7, respectively. The specified thermal limits for these two parameters are also shown as red dashed lines in the figures. It can be clearly seen that the safety criteria are satisfied during the entire transient. In fact, large safety margins are observable in the PCT figure (Fig. 6). The extreme quantities of the power, PCT and MCHFR and the corresponding time of occurrence in the accidents are summarized in Table 5.

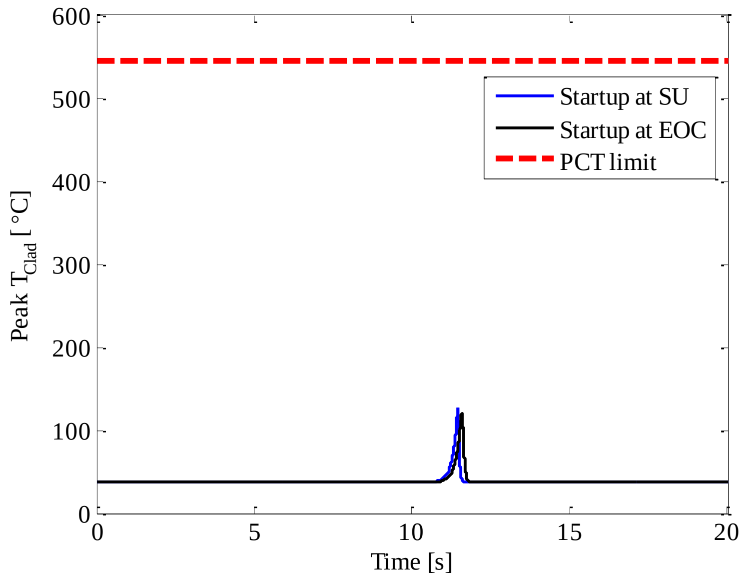

Fig. 6. Peak clad temperature in start-up accidents.

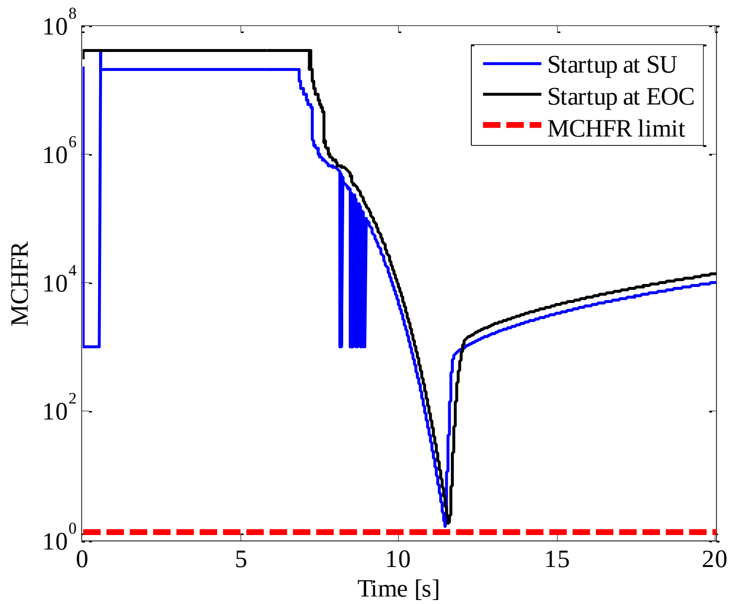

Fig. 7. Minimum critical heat flux ratio in start-up accidents.

Table 5. Peak quantities and their occurring time in start-up accidents

Core Status	SU	EOC
Peak Power [MW]	29.12	31.05
Peak power time [s]	11.45	11.56
Power trip time [s]	11.42	11.51
Peak clad temperature [°C]	127.38	119.49
PCT time [s]	11.46	11.57
MCHFR	1.66	1.86
MCHFR time [s]	11.46	11.57

4.3 Maximum Reactivity Insertion Accident (MRIA)

The maximum reactivity insertion accident models the power excursion with a large positive reactivity inserted in the core that may be caused by experiments removed from the core. Both SU and EOC core are considered for the accident. The reactor is assumed to be initially operated at a full power of 20 MW. A large positive reactivity 1.5$ was inserted to the core in 0.5 s. The scram set point, time delay constant for the scram and the constant control rod movement speed are all assumed to be the same as the start-up accident case. For conservatism, all reactivity feedback coefficients are assumed to be zero. As shown in Fig. 8, the power, for both SU and EOC, increases to a maximum power about 26 MW in in about 120 ms in MRIA. The PCT and MCHFR, however, remain well below the thermal limits as shown in Fig. 9 and 10, respectively. The extreme quantities of the related parameters and the corresponding time of occurrence in the transients are summarized in Table 6.

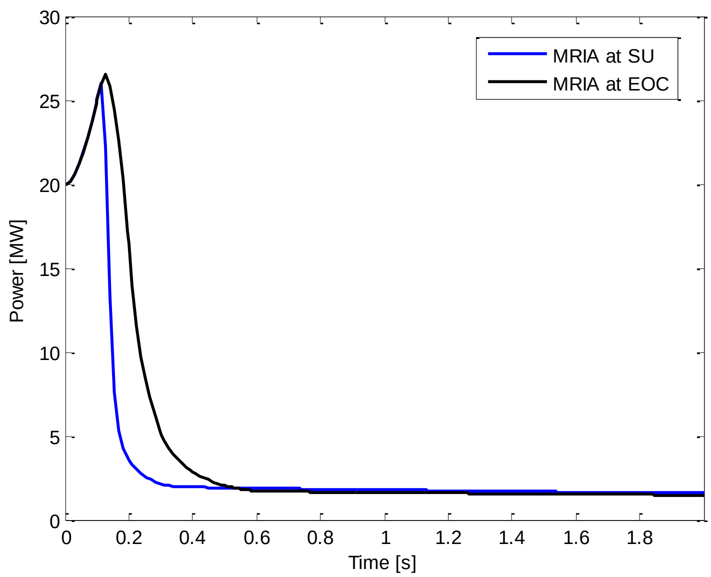

Fig. 8. Reactor power in maximum reactivity insertion accidents.

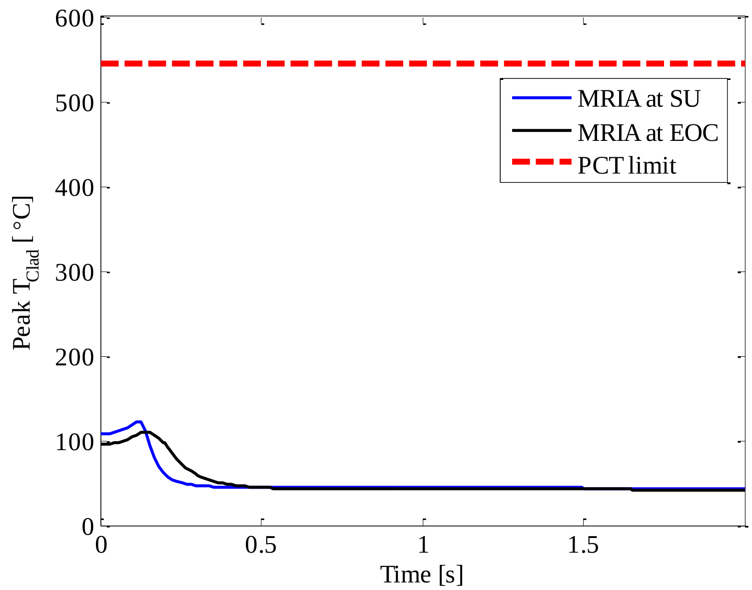

Fig. 9. Peak clad temperature in maximum reactivity insertion accidents.

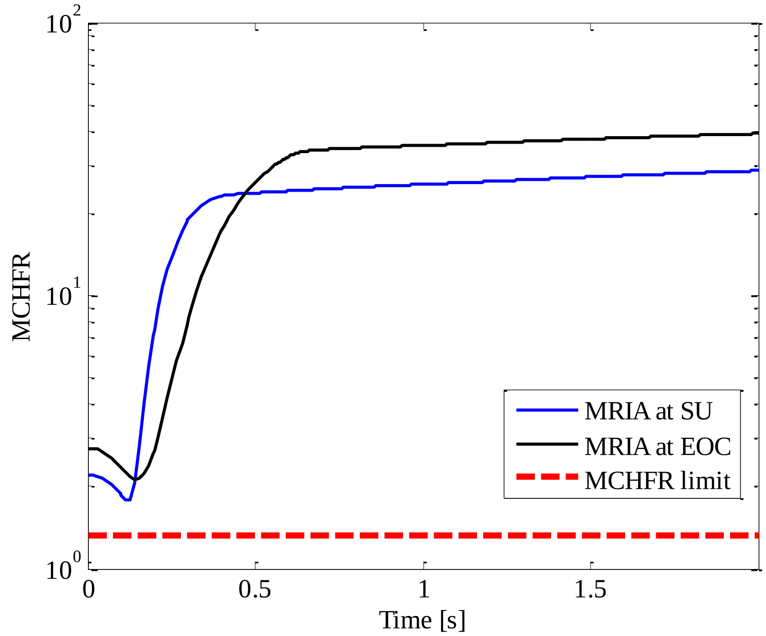

Fig.10. Minimum critical heat flux ratio in maximum reactivity insertion accidents.

Table 6. Peak quantities and their occurring time in maximum reactivity insertion accidents

Core Status	SU	EOC
Peak Power [MW]	26.03	26.51
Peak power time [s]	0.1126	0.1266
Power trip time [s]	0.0985	0.0985
Peak clad temperature [°C]	121.99	109.99
PCT time [s]	0.1267	0.1406
MCHFR	1.78	2.12
MCHFR time [s]	0.1267	0.1406

5. Summary and Conclusions

Preliminary design basis protected transient overpower accident analyses for the NIST's proposed LEU fueled split core reactor are performed using the ANL-PARET safety analysis code. The control rod withdrawal start-up accident and the maximum reactivity insertion accident are modeled with selected assumptions to maximize the severity of the event. The accidents are analyzed at respective SU and EOC conditions of an equilibrium cycle. The peak clad temperature and minimum critical heat flux ratio are examined during the transients to fully satisfy the safety criteria. The DNB critical heat flux is estimated by the Mirshak correlation in this study. The safety analysis results indicate reasonably sufficient safety margins were achievable in both start-up and MRIA accidents.

In the near future, the loss-of-flow (LOF) accidents will be investigated also using the PARET code to assess the safety margins as well as the natural circulation heat removal capability for the design under these conditions. The loss-of-coolant accident (LOCA) will also be analyzed with a more comprehensive model including the entire primary system of the reactor. A more systematic T/H system analysis code such as RELAP5 [9] will be used for such studies.

6. References

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