Preliminary Safety Analyses on a Conceptual LEU Fueled Research Reactor at NIST

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INTRODUCTION

Feasibility studies on a new low enriched uranium (LEU) fueled beam-type research reactor are underway at the NIST Center for Neutron Research (NCNR). The primary purpose of the reactor is to provide quality neutron sources, particularly cold neutron sources (CNSs), for experimental instruments. The new design is targeting at least two high quality cold neutron sources. The thermal power of the new reactor is designated at 20 MW and the operating cycle of the equilibrium core is set to be around 30 days. A horizontal split compact core with a large D_2O reflector tank is proposed and studied in the first phase of the project, with the expectation of achieving better thermal and cold neutron performance than the present NIST reactor (NBSR) [1]. A preliminary core design has been completed with MCNP modeling and simulation. The performance characteristics of the new core at the end of the cycle (EOC) indicates the new design is competitive with recently developed advanced research reactors around the world [2].

The study on the new reactor is continued by adding more components into the MCNP model. Two vertical liquid deuterium CNSs are placed in the flux trap between the two core halves. The distance between center of the CNS and the reactor center is 40 cm, which is a tradeoff result between the cold neutron performance and the heat load estimation of the CNS. Two CNS beam tubes are attached to the CNS as close as possible, with the guides pointing the north and south direction, respectively. Four thermal beam tubes are located in the east and west side of the core at different elevations for the purpose of closely reaching the core face without intersection. The split core consists of a total of 18 fuel elements which are evenly distributed into two regions. The core is cooled with H_2O and moderated by D_2O in the reflector. Control elements are also considered for the compact core. As a reference study, the control elements are designed as control blades surrounding the core boxes. The control blades are made of material with large thermal neutron absorption cross-section and are placed as close as possible to the core when they are inserted into the core. The control blades are functioned for both shim control and safety shutdown control purposes. A schematic view at the mid-plane of the reactor is depicted in Fig. 1.

With all these considerations, the material inventories of an equilibrium core at the startup (SU) and end of cycle (EOC) state were generated using methodologies discussed in previous studies [3]. The steady-state core performance characteristics for both states were produced by MCNP calculations. The criticality of the core was achieved by adjusting the control rod positions.

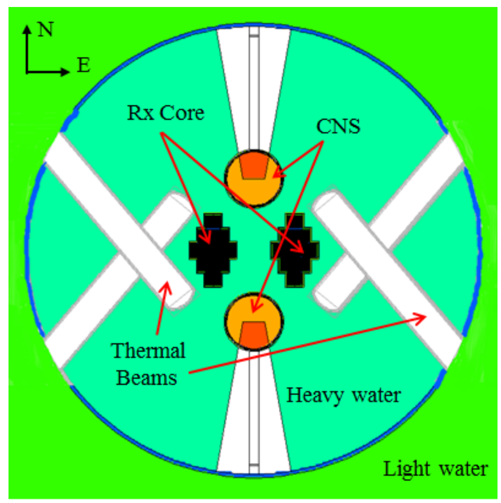

Fig. 1. Schematic of the reactor components.

Using the neutronics analysis results, preliminary safety analyses at SU and EOC were performed based on a single-channel thermal-hydraulics (T/H) model and point kinetics model (PKM) [4]. T/H related safety examination was conducted by evaluating minimum critical heat flux (CHF) ratio and minimum onset of flow instability (OFI) ratio using the Sudo-Kaminaga correlations [5] and Saha-Zuber criteria [6], respectively. Both the steady-state operational condition and the reactivity insertion accident scenario for both SU and EOC of the equilibrium core were investigated.

SAFETY ANALYSIS METHODOLOGIES

As mentioned above, the safety related fuel integrity during both steady-state operational and postulated accidental conditions is examined by investigating the minimum critical heat flux ratio (MCHFR) and minimum onset flow instability ratio (MOFIR) as safety performance indication parameters. As a matter of expediency for this study, the limitations of these parameters for the LEU core are obtained from the NBSR conversion safety analysis report [7], though more realistic limiting conditions specifically for the new reactor may be obtained from a statistical hot channel analysis approach [8].

Critical Heat Flux

The Sudo-Kaminaga correlation [5] is used to calculate CHF because it is developed for reactors with plate-type fuels and considered to be appropriate for the geometry and flow condition for our core design. The correlation was originally developed for vertical rectangular channels in Japan Research Reactor Unit 3 (JRR-3) based on CHF experiments. The effect of mass flux, inlet and outlet subcooling, flow direction, pressure, and channel configuration are taken into account in the CHF experiments. Based on a defined dimensionless mass flux G^*, the flux is categorized into three regions: low, medium and high regions, respectively. The mass flux boundary values for each region can be calculated based on flow fluid properties and the flow channel conditions. The CHF correlation, which is mass flux and flow direction dependent, is subsequently provided for each mass flux region. The scheme of the Sudo-Kaminaga correlation in different flow regions is depicted in Fig. 2.

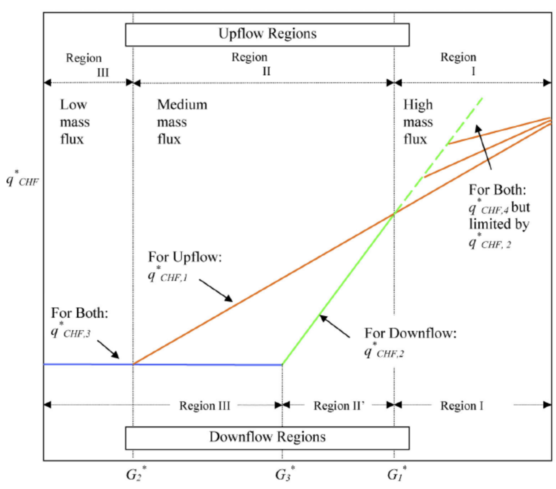

Fig. 2. Sudo-Kaminaga correlation scheme [8].

The G_1^*, G_2^*, G_3^* in Fig. 2 are the dimensionless mass flux boundary values for the regions, and $q^*_{\text{CHF},1}$, $q^*_{\text{CHF},2}$, $q^*_{\text{CHF},3}$ are the dimensional CHF calculated in the correlation associated with each region. The CHFR is thereby evaluated as

$$\text{CHFR} = \frac{q''_{\text{CHF}}}{q''_{\text{Model}}} \tag{1}$$

where q''_{CHF} is the CHF evaluated from the correlation, and q''_{Model} stands for the calculated heat flux in the flow channel predicted by the physics models. In this study, q''_{Model} is approximated from the neutronics calculation results and the total heat transfer areas in flow channels.

Onset of Flow Instability

Excursive flow instability, which is indicative of OFI, can occur in channels of reactor with plate-type fuels when the pressure drop is reduced due to the reduction of flow rate and significant amounts of vapor build up in channels. At this point, the overall pressure drop in the hot channel will increase and the flow will be reduced. This will result in a rapid loss of adequate cooling for the hot channel.

The OFI is determined by assuming the onset of net vapor generation is a conservative threshold for OFI and the Saha-Zuber criteria [6] are used. The heat fluxes for OFI are calculated based on the low- and high-mass flow rates that are determined by the Péclet number. The OFIR is thereby evaluated as

$$\text{OFIR} = \frac{q''_{\text{OFI}}}{q''_{\text{Model}}} \tag{2}$$

where q''_{OFI} is the heat flux at OFI evaluated by Saha-Zuber criteria, and q''_{Model} stands for the heat flux in the flow channel predicted by the physics models.

RESULTS

Steady-State Operational Condition

A single channel model with equivalent T/H characteristics of the average flow channel in the core is used in the preliminary safety analyses. The total flow rate was assumed to be constant (8000 gal/min or 1817 m^3/hour) 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 NBSR [9]. The heat flux in the hot channel was obtained from the axial power distribution in the hot stripe of the core [1] and the total heat transfer surface area along the channel. The behavior of the heat flux in the hot channel associated with CHF and OFI heat flux predicted by the correlations at SU and EOC are illustrated in Fig. 3 and Fig. 4, respectively. The evaluated MCHFR and MOFIR for both states are summarized in Table I.

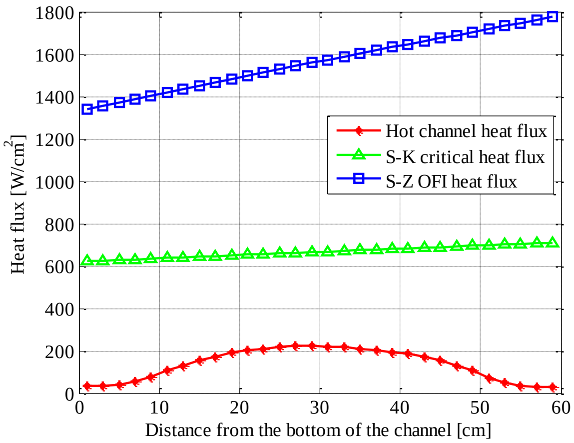

Fig. 3. Heat flux along the vertical channel at SU.

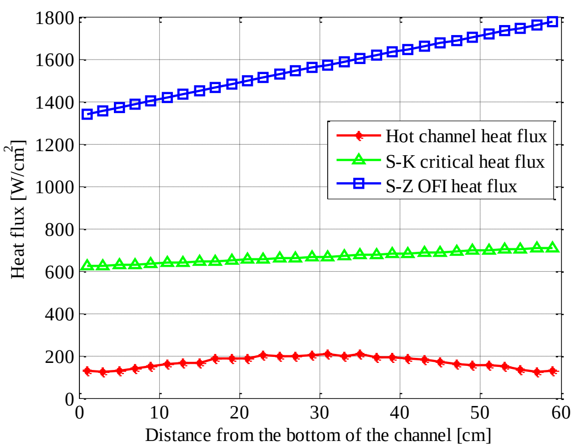

Fig. 4. Heat flux along the vertical channel at EOC.

Table I. Calculated MCHFR and MOFIR at SU and EOC

CASE	SU	EOC
MCHFR	2.94	3.22
MOFIR	6.85	7.54

The hot-channel limits set for a LEU fueled core in Ref. [7] was 1.32 for MCHFR and 1.27 for MOFIR, both of which are within 90% confidence level. As can be seen here, satisfactory large safety margins exist in both MCHFR and MOFIR for the new core in steady-state operational conditions.

Reactivity Insertion Accident

The reactivity insertion accident power excursion is analyzed using the point reactor kinetics model with six delayed neutron precursors [4]. The kinetics parameters for the SU and EOC core were obtained from MCNP calculations based on the adjoint-function weight approach. The main kinetics parameters used in the study are given in Table II.

Table II. Kinetics Parameters for the SU and EOC

Kinetics Parameter	**SU**	**EOC**
Prompt neutron lifetime - l_p (μs)	97.15	160.69
Effective delayed neutron fraction (β_{eff})	0.00837	0.00722

For conservatism, the calculation does not consider any fuel or moderator temperature reactivity feedback. However, the effect of negative feedback from fuel and moderator temperature coefficients is believed to be quite small in such type of research reactors. [7] The reactor is assumed to be operated initially at full power rate (P=1). During the accident, a ramped reactivity with the insertion rate 500 pcm/s is introduced starting at t=0 s until t=0.5 s. This amount of reactivity inserted is a mimic to the reactivity changes due to removal of experimental samples from the core. The over-power level trip is set to 120% of nominal operating power (P=1.2) in the accident. A delayed time constant (τ=140 ms) was included in the model to account for the delay in response of the trip circuits and the finite time for safety rod insertions. Upon reactor scram, the control rods are all inserted with the assumption that the initial control rod position corresponds to where the reactor is critical at the full power.

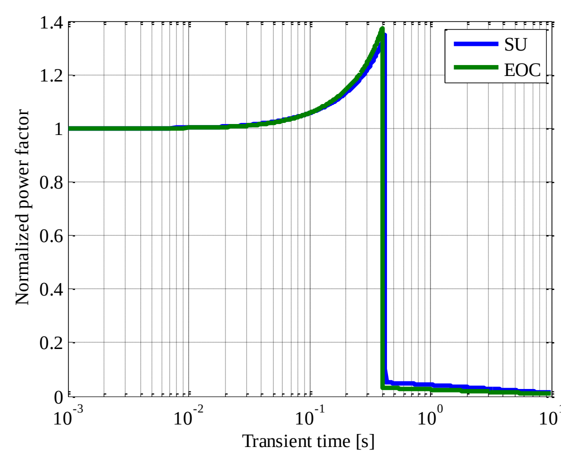

Fig. 5. Reactor power in maximum reactivity insertion accident.

The power transient behavior predicted by the point kinetics model for SU and EOC core are illustrated in Fig. 5. The time-step used in all transient calculations was ~1 ms. As can be seen, in both cases, the power starts to increase exponentially at time zero and reaches peak at around 0.4 s. Then the power decreases suddenly as the safety rods are inserted into the core region after a reactor trip signal is generated. The exact peak power, peak occurring time and trip time for both SU and EOC case are summarized in Table III.

Table III. Peak Power and Time in MRIA

CASE	Peak power	Peak time (s)	Trip time (s)
SU	1.35	0.421	0.281
EOC	1.37	0.399	0.259

As observed from Fig. 5, the power rises slightly faster at EOC than at SU, this is due to the difference existing in the kinetics parameters of both states (see Table II). Because of large amount of uranium is depleted at EOC, the core averaged spectrum is slightly shifted to the thermal range, which leads to a greater prompt neutron lifetime and a smaller delayed neutron fraction.

As shown in Table III, the peak power estimated at EOC is slightly higher than the one at SU. This is attributed to the fact that the initial control rod positions at trip are different at EOC and SU. Due to fuel burnup and poison buildup within the cycle, control rods are noticeably withdrawn out of the core at EOC comparing to their positions at SU. It results in a smaller initial negative reactivity insertion rate after the reactor trip at EOC.

CONCLUSION

Preliminary safety analyses on NIST's proposed LEU fueled beam reactor were performed using a single-channel T/H model and point kinetics model. The safety analysis results indicates reasonably sufficient safety margins were achievable on the MCHFR and MOFIR estimated in both steady-state operational condition and over-power reactivity insertion transient situation for SU and EOC cores. These are preliminary feasibility checks for the conceptual LEU fueled research reactor, additional accident scenarios will be analyzed in the second phase of the project.

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