



Neutronics and Safety Studies on a Research Reactor Concept for an Advanced Neutron Source

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Abstract — This paper presents preliminary neutronics and thermal hydraulics safety analysis results for a low-enriched uranium (LEU) fueled research reactor concept being studied at the National Institute of Standards and Technology (NIST). The main goal of this research reactor is to provide advanced sources for neutron scattering experiments with a particular emphasis given to high intensity cold neutron sources (CNSs). A tank-in-pool type reactor with an innovative horizontally split compact core was developed in order to maximize the yield of the thermal flux trap in the reflector area. The reactor concept considered a 20 MW thermal power and a 30-day operating cycle. For non-proliferation purposes, a LEU fuel (U_3Si_2-Al) with 19.75 wt% enrichment was used. The core performance characteristics of an equilibrium cycle with several representative burnup states—including startup and end of cycle—were obtained using the Monte Carlo-based code MCNP6. The estimated maximum perturbed thermal flux of the core is $\sim 5.0 \times 10^{14}$ n/cm²-s. The calculated brightness of the CNS demonstrates an average gain factor of ~ 4 compared to the current source operated at the existing NIST reactor. Sufficient reactivity control worth and shutdown margins were provided by hafnium control elements. Reactivity coefficients were evaluated to ensure negative feedback. Thermal hydraulics safety studies of the reactor were performed using the multi-channel safety analysis code PARET. Steady-state analysis shows that the peak cladding temperature and minimum critical heat flux ratio are less than design limits with sufficient safety margins. Detailed transient analyses for a couple of hypothetical design-basis accidents show that no fuel damage or cladding failure would occur with the protection of reactor scrams. All these study results suggest this new research reactor concept offers a demonstrable potential to greatly expand the cold neutron capability with a 20 MW power and certified LEU fuels.

Keywords — Research reactor, low-enriched uranium, cold neutron source.

Note — Some figures may be in color only in the electronic version.

I. INTRODUCTION

The National Bureau of Standards Reactor (NBSR) (Ref. 1) is a 20 MW thermal reactor that currently operates at the Gaithersburg campus of the National Institute of Standards and Technology (NIST). Since it was built in 1960s, the NBSR has evolved into a major neutron source facility hosting over 2000 guest researchers

annually. As of December 2016, NBSR provided beams to 28 neutron research instruments for various neutron scattering, imaging, and fundamental physics experiments. 21 of these instruments use cold neutrons, which are neutrons slowed down by a cold moderator to energies less than 5 meV (wavelength greater than 4 Å). Cold neutrons have become increasingly important for scattering experiments that probe large structures using Small Angle Neutron Scattering (SANS), and for very high energy resolution inelastic neutron scattering in complex materials.

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The NBSR went first critical on December 7, 1967, and was originally licensed to operate for 20 years. The initial license for 10 MW operation was replaced by a new 20 MW license in 1984, and then was renewed again in 2009 for an additional 20 years, and the current intent is to renew it again in 2029. Nevertheless, the reactor will eventually need to be replaced. The demand for experimental time and the number of users of the NBSR have continued to grow in the past decade, particularly after the addition of a new guide hall served by five new cold neutron guides in 2012. Since the reactor is still operated with high-enriched uranium (HEU) fuel, a plan for the safe conversion of the NBSR to low-enriched uranium (LEU) fuel has been submitted, but various challenges have appeared in the development and fabrication of the high density LEU fuel (U-10Mo monolithic fuel). Conversions of U.S. high performance research reactors such as the NBSR have been delayed by at least a decade.²

Under these circumstances, there is strong interest to build a new neutron beam facility at NIST in order to maintain and enhance the neutron scattering and other beam capabilities when the NSBR is shut down. A reactor replacement study was therefore initiated, and efforts to design a new research reactor fueled with LEU optimized for cold neutron sources (CNSs) are currently underway at the NIST Center for Neutron Research (NCNR). Feasibility studies are being carried out to demonstrate the capability of the reactor as an advanced neutron source. The primary purpose of the proposed new reactor is to provide bright and reliable cold neutron beams for scientific experiments. The design currently under study incorporates two high quality CNSs and four thermal neutron beams. To leverage the existing site license and knowledge gained from the NBSR, the new reactor was chosen to be of similar scale to the existing one but will incorporate the latest research reactor design features that have been used in recent research reactor projects, e.g., Australia OPAL (Ref. 3) and Germany FRM-II (Ref. 4). The material testing reactor (MTR)-type fuel element was used in the conceptual design of the new reactor. LEU fuel with ²³⁵U enrichment less than 20 wt% was used to comply with non-proliferation requirements. The feasibility of an innovative horizontally split compact core, cooled and moderated by light water while reflected by heavy water, is being investigated to provide cold neutron beams 4 times the intensity of existing source and twice as many beams.^{5,6} The new reactor was designed for a 20 MW thermal power and a 30-day refueling cycle for an equilibrium core condition to provide a similar cycle

to the current NBSR and to ensure licensability with current site boundary requirements.

As part of the reactor design efforts, neutronics and thermal hydraulics (T/H) safety studies were performed to demonstrate the high intensity neutron production from the core design using a qualified LEU fuel while satisfying safety-related thermal limits during normal operation and abnormal events. The neutronics calculations were performed using the code MCNP6 (Ref. 7) with an explicit geometric representation of the core. Specifically, a multi-cycle equilibrium core configuration with several representative burnup states was developed using the burnup feature in MCNP6. Detailed physics calculations were performed using the equilibrium core model to demonstrate the flux performance characteristics. Reactivity control and feedback were assessed to satisfy the standard reactivity control criteria and negative feedback requirements. The T/H safety calculations were performed using the channel-based safety analysis code PARET (Ref. 8). Both steady-state operating conditions and some hypothetical accident scenarios were examined to ensure the T/H safety of the core at different conditions.

In this paper, we present the core performance and T/H safety characteristics of the proposed design based on neutronics and T/H studies. An overview of the LEU core design is outlined in Sec. II. Section III briefly summarizes the study objective and safety requirements, while the neutronics and T/H safety study procedures and results are presented in Secs. IV and V, respectively. Some summary and concluding remarks are offered in the last section.

II. CORE DESIGN OVERVIEW

Recently developed or proposed neutron beam research reactors^{3,9–12} have been based on compact cores,¹³ characterized by a small core with a high power density. A compact core is capable of producing a high thermal neutron flux in a large volume outside of the reactor core such that beam tubes can be readily placed in this region to extract neutrons for scattering experiments. Characteristics of a compact core include: the active core volume is made as small as possible for a given reactor power; the core is surrounded with a reflector of high quality and large volume to maximize the thermal flux production, and the reactor power is set as high as possible to maximize the flux. Our core design employs a split compact core to create a thermal flux trap in an easily accessible location in the reflector tank.

Figure 1 is a schematic view of the reactor components and the fuel element radial layout at the mid-plane of the split core. The commonly recognized “tank-in-pool” design pattern was used in the new design. A cylindrical heavy water tank—2.0 m in diameter and 2.0 m in height—is placed in the center of a large light water pool, which provides thermal and biological shielding to the reactor. To maximize the useful flux trap volume in the reflector, an innovative horizontally split core was employed in the design such that the thermal flux trap between the core halves provides ideal locations to place CNSs (Ref. 5). The core itself is cooled and moderated by light water and surrounded by the heavy water reflector. The core halves are enclosed in two zirconium core boxes, which separate heavy water and light water. Two vertical liquid deuterium (LD₂) CNSs are placed in the flux traps located in the north and south sides of the core. The distance between the center of the CNS and the reactor center is 40 cm, which is a tradeoff between the cold neutron performance and the estimated heat load for the CNS. Two CNS beam tubes are connected to the CNSs with guides pointing north and south. Four tangential thermal beam tubes are placed in the east and west sides of the core at different elevations. They are placed 20 cm above and below the core mid-plane in the present design. This number, however, might be increased if desired.

As shown in Fig. 1, the split core consists of 18 MTR-type fuel elements in two horizontally split regions. Each region consists of 9 fuel elements and represents one-half of the reactor core. Each fuel element has 17 inside fuel plates and 2 unfueled end plates. All plates have LEU fuel clad with Al. The fuel used in this study is U₃Si₂-Al dispersion fuel¹⁴ with ²³⁵U enrichment 19.75 wt%, which is currently the highest density LEU fuel certified by the U.S. Nuclear Regulatory Commission (NRC). The fuel meat has a rectangular shape with dimensions of 60 cm long, 6.134 cm wide, and 0.66 mm (26 mil) thick. In this design, the ²³⁵U mass in a fresh fuel element is 391 g. Note that the central fuel elements are separated by 1 cm water gaps (see the split core plot in Fig. 1) for the purpose of accommodating control elements.

For reactivity control, hafnium (Hf) was selected as the control material due to its high neutron absorption cross sections in both thermal and epithermal energy ranges and excellent corrosion resistance in a light water environment. Some post-irradiation tests have indicated that Hf is an adequate long-life neutron absorber material for light water reactors in terms of burnup, corrosion, tensile properties and fatigue behavior.¹⁵ Four ‘#’ shaped hafnium control blades (see Fig. 2) are utilized for both criticality and safety control in the reactor. The top and side views of the critical core with control elements are depicted in Fig. 2. The

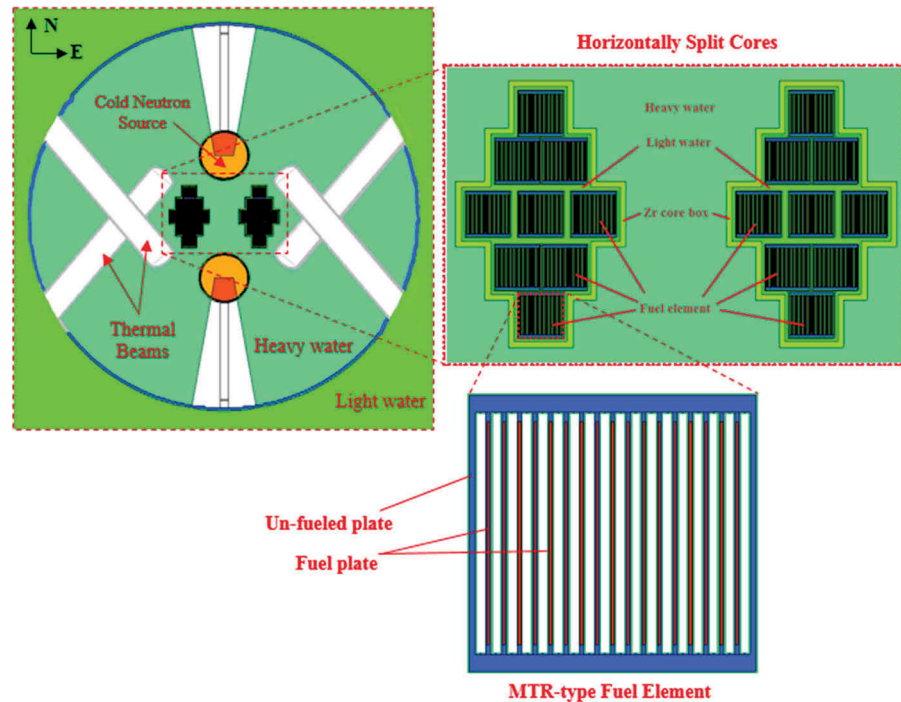


Fig. 1. A schematic view of the mid-plane of the reactor with horizontally split cores and MTR-type fuel elements. (The thermal beams are actually located above and below the mid-plane.).

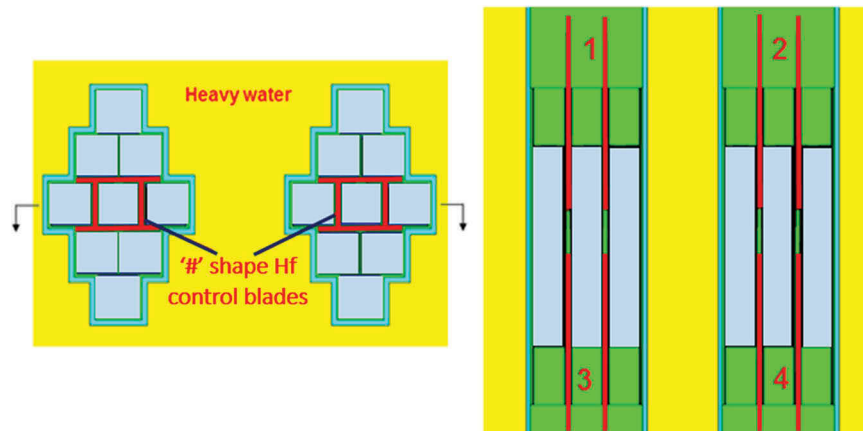


Fig. 2. The top (left) and side (right) views of the core with control elements.

numbers shown in the side view are control blade identifiers, which would be used in the shutdown margin calculation discussed in Sec. IV.B.4. Due to the limited space in the core, all control blades are made 0.5 cm thick and 60 cm long (the same length as the active fuel length). To maintain an axially symmetric flux profile, the top (No. 1 and 2 in Fig. 2) and bottom (No. 3 and 4 in Fig. 2) control elements are assumed to operate simultaneously in opposite directions. More discussion on the movement and critical positions of the control elements during the operational fuel cycle can be found in Sec. IV.A along with the discussion of a multi-cycle equilibrium core generation.

III. OBJECTIVE AND SAFETY REQUIREMENTS

The primary utilization of the new reactor is determined by the neutron scattering research community, where demand for thermal and cold neutrons has steadily increased in recent years. Therefore, the principal objective of this study is to demonstrate the excellent neutron flux performance characteristics of the new design compared to

the existing state-of-the-art designs. One figure of merit to quantify the flux feature of a neutron beam reactor is its quality factor, which is defined as the ratio of the maximum thermal flux (MTF) to the total thermal power of the reactor. It should be noted that MTF here refers to the maximum value in regions of a reactor that are close to the beams, and not necessarily the MTF across the reactor because only neutrons that are accessible to beams will make contributions to neutron scattering experiments. The quality factors of several well-established research reactors, including the NBSR, are shown in Table I [data were obtained from the online IAEA research reactor database (RRDB)]. The aim of the new reactor is to produce comparable or superior neutron fluxes to those already in existence. Note that some of the reactors shown in Table I still use high-enriched fuels and are currently undergoing fuel conversion studies. The flux performance of these reactors will possibly decrease after the fuel is converted to LEU unless other design changes are made.

The primary emphasis of the control element study is for neutronic and T/H safety considerations, with some speculation on a practical operating control system. From

TABLE I
Performance Characteristics of Selected Research Reactors

Reactor	NBSR (Ref. 1)	HFIR (Ref. 16)	BR-2 (Ref. 17)	FRM-II (Ref. 4)	OPAL (Ref. 3)	CARR (Ref. 9)
Country	U.S.	U.S.	Belgium	Germany	Australia	China
Power (MW _{th})	20	85	60	20	20	60
Fuel	HEU	HEU	HEU	HEU	LEU	LEU
Max Φ_{th} ($\times 10^{14}$ n/cm ² -s)	2	12	10	6.4	2.5	8
Quality factor ($\times 10^{13}$ MTF/MW _{th})	1.00	1.41	1.67	3.20	1.25	1.33

a safety perspective, the reactor is required to be able to shut down under any circumstances including normal and abnormal operating scenarios. As such, the total control worth and shutdown margin must be sufficient to meet this requirement. In this study, the design requirements of the control elements used for the proposed new research reactor are summarized in Table II. These criteria are based on the NBSR technical specifications¹⁸ and some previous publications on the same subject.¹⁹

To meet these safety criteria, two thermal limits were considered in the safety studies. The first is the peak clad temperature (PCT), which is a direct indicator of the physical damage to the fuel plate. The PCT must not reach the fuel blister temperature, 515°C for the uranium silicide LEU fuel.¹⁴ The second constraint accounts for the critical heat flux (CHF), which characterizes the departure from nucleate boiling (DNB) occurring at the surface of the fuel cladding. The DNB will significantly weaken the heat removal capability and subsequently cause the PCT to exceed the blister temperature. One metric to address the CHF is the minimum critical heat flux ratio (MCHFR), defined as the DNB heat flux estimated from an appropriate correlation divided by the expected heat flux. The limit of MCHFR for a reactor should be determined by the T/H characteristics of the reactor. In this study, the MCHFR is required to be greater than 1.78, a figure with 99.9% confidence level considered from a statistical hot channel analysis of the NBSR (Ref. 20), which has similar thermal and flow conditions to those of the new reactor. Of the available options in PARET, the Mirshak DNB correlation²¹ is the most appropriate one for plate-type fuels and thereby was used to estimate the DNB CHF in the safety study.

IV. NEUTRONICS STUDY

The Monte Carlo-based code MCNP6 was extensively used in the neutronics calculations. All the components in the core as well as the cold neutron moderator assemblies were modeled explicitly, except the fuel plates

were modeled without curvature for simplicity. The realistic MTR-type fuel plate is slightly curved to prevent buckling due to thermal expansion, and the curvature has nearly no impact on neutronics performance. The neutronics study started with an iterative search scheme to generate fuel inventories at four representative burnup states of a multi-cycle equilibrium core and then continued with a refined calculation to obtain the physics performance characteristics of the core, with a particular interest on the CNS performance. The power profiles and kinetics parameters required for safety analyses are also provided. All the calculations performed in MCNP were criticality calculations (KCODE mode) with nuclear data from the ENDF/B-VII.1 library. For computational efficiency, the statistical uncertainties on the k_{eff} convergence at the equilibrium core search stage were much larger than the ones used for the detailed physics calculations. The k_{eff} statistical 1- σ error is ~ 100 pcm (per cent mille) for the iterative search procedure and ~ 10 pcm at the detailed calculation stage. In both stages, however, sufficient inactive fission cycles are skipped to ensure the convergence of fission source distribution.

IV.A. Multi-Cycle Equilibrium Core

A three-batch fuel management scheme was employed to achieve a multi-cycle equilibrium core. Figure 3 shows the fuel shuffling scheme for the 18 fuel elements. For each numeric pair indicated in the fuel, the first number stands for the fuel batch number and the second one represents the ID number of the fuel element in the batch. The fuel elements in the first batch have fresh fuels in the startup (SU) core, and the elements in the second batch have once-burnt fuels, while those in the third batch will be discarded at the end of cycle (EOC). Therefore, there will be 6 fuel elements being replaced at the end of each cycle under this fuel management scheme.

To achieve a multi-cycle equilibrium core, an iterative search procedure was performed following the methodology introduced by Hanson and Diamond.²² In the depletion calculation, each fuel element was divided into 6 axial zones, resulting in $18 \times 6 = 108$ fissionable zones in the entire core. Four representative burnup states were considered in a cycle in the search procedure:

1. the startup (SU) state, which is initiated with all fresh fuel
2. the beginning of the cycle (BOC) state, which has burned 1.5 days into the cycle and assumed to have equilibrium xenon concentration

TABLE II

The Reactivity ($\% \Delta k/k$) Control Requirements

Parameter	Criteria
Excess reactivity	<15
Total control worth	>25
Shutdown margin	>3

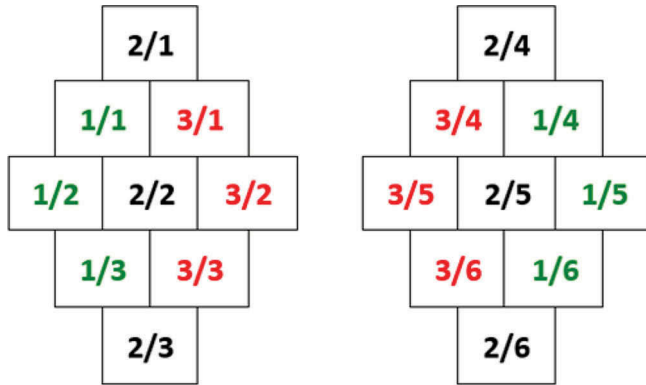


Fig. 3. Three-batch fuel element shuffling scheme.

- the middle of the cycle (MOC) state, which has burned 15 days into the cycle
- the end of cycle (EOC) state, which has burned a full cycle length (30 days).

The fuel compositions after EOC will then decay for about 1 week before the elements are shuffled into the SU core for the next iteration (except the discarded fuels). The iterative procedure continues until k_{eff} converges for each state. Note that the control blade positions for each state must be adjusted properly to maintain critical status during the equilibrium search process. A diagram briefly illustrating the search procedure is shown in Fig. 4, and the critical positions of the control blades at different burnup states in an equilibrium cycle are depicted in Fig. 5.

All control blades are operated in a ganged fashion within the cycle, with the top and bottom control elements moving in opposite directions. To achieve critical status for each state, the inserted length of control blades has to be adjusted correspondingly to compensate the reactivity loss due to the burnup of fissile material and the buildup of fission product poisons. The search procedure begins with critical control blade positions

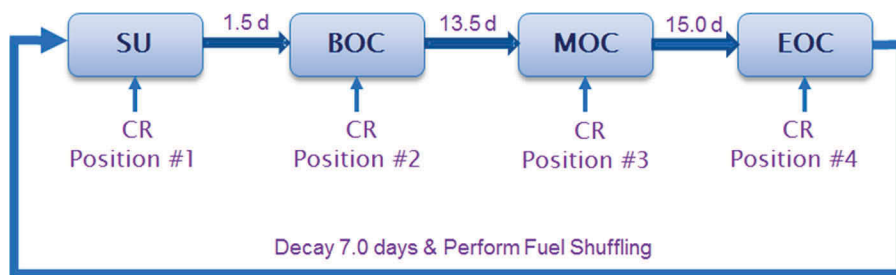


Fig. 4. Flow diagram of the multi-cycle equilibrium core search procedure.

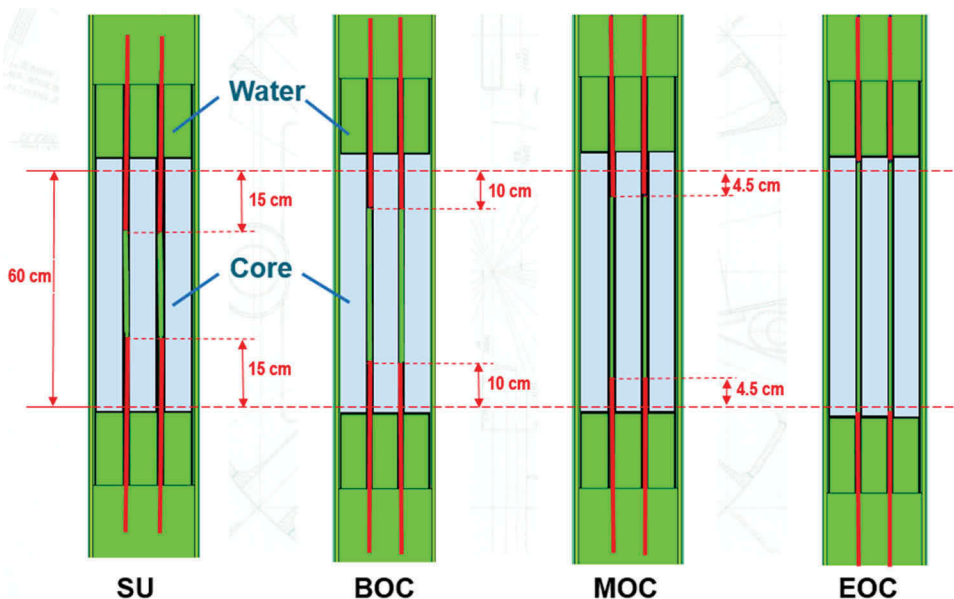


Fig. 5. Critical positions of the control blade at different burnup states.

approximately estimated for each state based on prior knowledge on the control worth. If large off-criticality is identified for any burnup state in middle of the search procedure, the critical position will be manipulated to ascertain the critical status of that state. The iterative search continues until the criticality holds at each state, which indicates an equilibrium cycle has been realized. The converged k_{eff} values and the control blade withdrawal lengths for the four states are summarized in Table III.

IV.B. Physics Performance Characteristics

After obtaining the material inventories of the equilibrium core, many key physics performance characteristics of the core such as neutron flux and fission rate can be subsequently calculated by MCNP6. Thus, all the calculations performed thereafter were based upon the equilibrium core obtained in Sec. IV.A. In order to reflect the absolute flux information, MCNP tallies must be normalized to the actual reactor power (20 MW; in this study). With the assumption that the recoverable energy per fission is approximately 200 MeV and the average number of neutrons generated per fission is 2.44 (Ref. 23), the total source of neutrons can be calculated as follows:

$$\begin{aligned} \text{Total source} &= (2.44 \text{ neutrons/fission})(20 \times 10^6 \text{ J/s}) / \\ &\quad [(200 \text{ MeV/fission})(1.602189 \\ &\quad \times 10^{-13} \text{ J/MeV})] \\ &= 1.523 \times 10^{18} \text{ neutrons/s} \end{aligned}$$

This number worked as a normalization factor to estimate the absolute neutron flux and fission rates in the core.

IV.B.1. Neutron Flux

The flux was obtained via the standard MCNP FMESH tally. Figure 6 shows the axial distribution of the perturbed

flux (including fast flux and thermal flux) at the center of the core for the four different states. Here “perturbed” means the reactor includes cold and thermal neutron beams as shown in Fig. 1. The cutoff energy for thermal neutrons is 0.625 eV. Due to the movement of the control blades, the axial behavior of the flux varies at different states during the cycle. As can be seen in Fig. 6, the maximum thermal flux that is achievable in the reflector area can reach $\sim 5.0 \times 10^{14}$ n/cm²-s. The maximum beam accessible thermal flux is $\sim 4.0 \times 10^{14}$ n/cm²-s, since the core is presently designed at 20 MW; the quality factor of the neutron source is therefore approximately 2.0×10^{13} MTF/MW, which exceeds nearly all the well-known neutron sources summarized in Table I except the FRM-II reactor. Note that the results shown in Fig. 6 represent the flux in the axial channel where the maximum thermal flux occurs. The standard deviation of the flux was also obtained with MCNP mesh tally. While they are not shown in Fig. 6, the relative errors associated with the flux tallies are all about 1% in these calculations.

IV.B.2. Cold Neutron Performance

Cold neutrons have kinetic energies less than 5 meV and wavelengths greater than 4 Å. They can be transported over tens of meters through super-reflecting neutron guides with minimal losses and, thereby, provide high intensity beams to a large number of special scientific experimental instruments. Intense beams of cold neutrons can be obtained from a cryogenic moderator such as LD₂ that further slows down thermal neutrons produced in the reactor. Figure 7 illustrates a generic vertical CNS in which a small volume of gaseous deuterium (GD₂) offers a re-entrant hole between the CNS and beam port that facilitates cold neutron transport to the guides.

Simulations of cold neutron production and transport depend heavily on the scattering kernels [cross sections for low energy neutrons, or $S(\alpha, \beta)$ data] of the cold moderators. The recently released ENDF/B-VII.1 libraries that include continuous energy and angle data in $S(\alpha, \beta)$ data²⁴ are used in the calculation. For LD₂ simulation at low temperature ~ 20 K, the ratio of ortho-LD₂ (LD₂ molecule with odd angular momentum) to para-LD₂ (LD₂ molecule with even angular momentum) in the ortho-para LD₂ mixture matters because the para-LD₂ has a smaller scattering cross section than the ortho-LD₂ for low energy neutrons. Based on the spectroscopy measurements of the LD₂ source at SINQ (the spallation neutron source at the Paul Scherrer Institut in Switzerland²⁵) the LD₂ used in this CNS calculation was conservatively assumed to contain 67% ortho-LD₂.

TABLE III

Critical Conditions of the Converged Equilibrium Core

Burnup State	k_{eff}	Control Position ^a
SU	1.00139 ± 0.00014	30 cm
BOC	1.00348 ± 0.00012	20 cm
MOC	1.00281 ± 0.00013	9 cm
EOC	1.00229 ± 0.00012	0 cm

^aIndicated as the total control blade inserted length in the active fuel region (shown in Fig. 5).

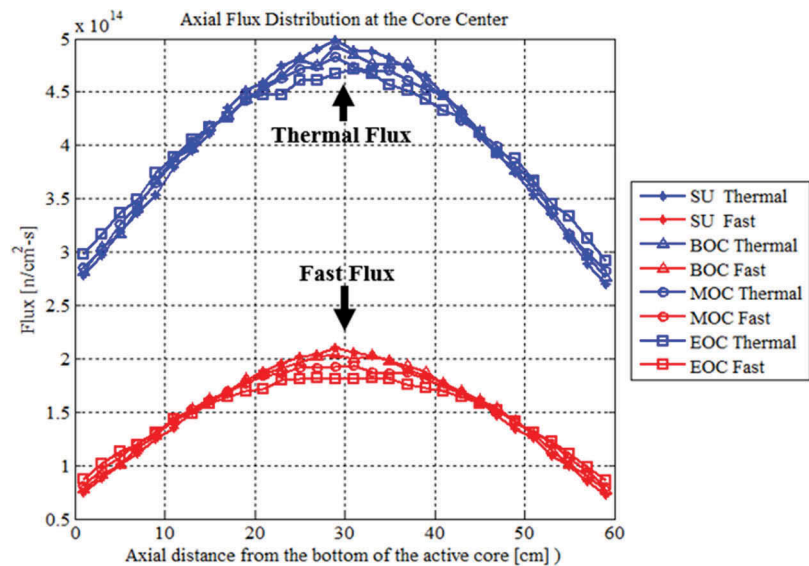


Fig. 6. The axial flux distribution at the center of the core.

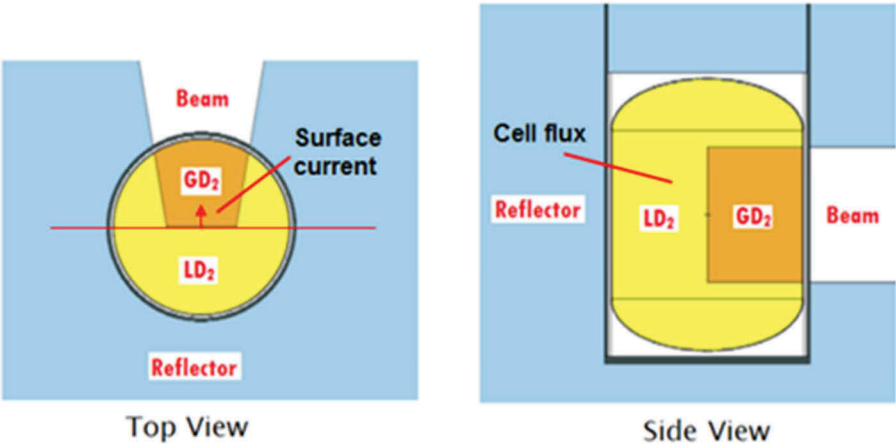


Fig. 7. Schematic view of the vertical CNS.

One of the important measures of the CNS performance is the cold neutron surface current (in the unit of $n/cm^2\cdot s$) at the exit surface of the re-entrant hole as shown in the left plot in Fig. 7. The estimated surface current for the CNSs of the new reactor is summarized in Table IV. The result is compared to the value at the exit surface of the existing liquid hydrogen CNS in NBSR (Ref. 26). As can be seen, the surface current for both north and south CNS have marginal differences between SU and EOC, and they both obtain a gain factor of ~ 6 compared to that of NBSR operated with the same power rate (i.e., 20 MW).

A better figure of merit to evaluate the performance of a CNS is the “brightness” of the source in the direction of the guides to various instruments. The brightness, either in units of $neutrons/cm^2\cdot s\cdot \text{\AA}\cdot ster$ or $neutrons/cm^2\cdot s\cdot meV\cdot ster$, can be obtained from the current tallies across a surface within a

TABLE IV
Cold Neutron Surface Current ($n/cm^2\cdot s$) at the Exit Surface of the CNS Beams

	North CNS ^a	South CNS	NBSR
SU	5.50×10^{11}	5.49×10^{11}	8.88×10^{10}
EOC	5.25×10^{11}	5.27×10^{11}	8.18×10^{10}

^aAll tallies are performed with neutron energy less than 5 meV and $\cos \theta$ greater than 0.99, where θ is the angle between the neutron streaming direction and the normal direction of the exit surface (see Fig. 7). The relative errors of the tallies are all less than 0.3%.

DXTRAN sphere in MCNP, and its value should not depend on the distance of the tally surface from the source if the tally angle is chosen properly.

Figure 8 presents the calculated brightness (in the unit of neutrons/cm²-s-meV-ster) of the vertical CNS in the split core. It is compared to the performance of the large liquid hydrogen (LH₂) CNS at NIST (Ref. 26). Figure 8 clearly shows the substantial gains achieved in brightness with respect to the NBSR LH₂ CNS over the entire low energy range from 0 to 30 meV. For the cold neutron energy range with $E < 5$ meV, an average gain factor of ~ 4 is achieved. Since the present NIST liquid hydrogen CNS has comparable performance in terms of flux per unit power to almost all existing worldwide cold sources, the preliminary results indicate the performance of the vertical CNS in the split core has significant gains compared to all currently available CNSs.

IV.B.3. Power Density and Kinetics Parameters

The power distribution in the fuel was used to determine the source profile for the heat structure in the T/H model. In this study, the power density for a given position in the core was calculated by MCNP6 by conservatively assuming that all the recoverable fission energy was deposited at the point of fission and the power density was proportional to fission density. In order to obtain a detailed power distribution for the core, the fuel meat was evenly divided into 3 stripes, and each stripe was evenly divided into 30 axial pieces. As a result, the smallest spatial mesh for power calculation is 2×2 cm² and has a volume about 0.264 cm³. The core-averaged axial power distribution at different burnup states is shown in Fig. 9. Being similar to the axial flux behavior (Fig. 6), the axial

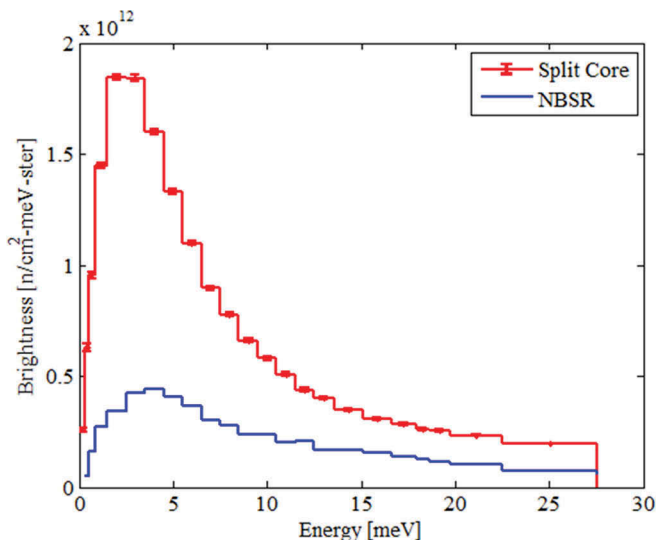


Fig. 8. The cold neutron brightness for the NBSR and proposed split core reactor.

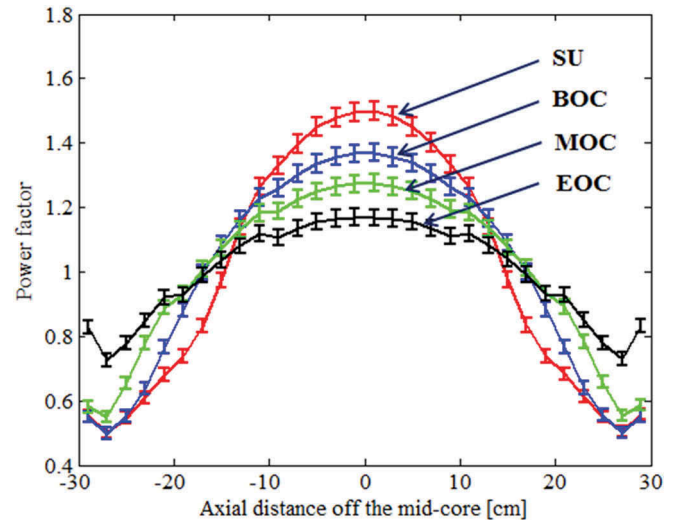


Fig. 9. The core-averaged power distribution at different burnup states.

power curve shifts toward the core center at every state, whereas the distributions have a tendency to be flattened out from SU to EOC because of the movements of control elements and fuel burnup effects.

Table V summarizes the power peaking factors (PPF^a) estimated for the core at different states. The different type of PPF (hot-spot PPF, hot-stripe PPF, etc.) shown in the table was produced by defining different unit mesh in the power distribution calculation. For example, the hot-stripe PPF was obtained with the power distribution evaluated using the stripe as a unit mesh. Hot spots generally occur in the end fuel plates of the fuel elements at core periphery because neutrons in these places receive enhanced moderation in the reflector. As can be seen in Table V, the greatest PPF is hot-spot peaking power factor (also referred to total peaking power factor) for SU. However, the value is less than 2.5, which is generally taken as a limiting factor for total peaking factors for research reactors. Moreover, the peaking factors may be further mitigated with more optimized studies on the design.

The kinetics parameters, namely the prompt neutron generation time and the effective delayed neutron fraction, are required for safety analyses. Kinetics parameters can be calculated with MCNP6 using the adjoint-weighted tally methodology.⁷ Table VI summarizes the kinetics parameters for the SU and EOC cores. The uncertainty shown in the table represents the 1- σ standard deviation of the corresponding parameter. It is seen the kinetics

^a The PPF is defined by the peak power divided by the average power in an associated domain.

TABLE V
Power Peaking Factors at Different Core States

Core State	SU	BOC	MOC	EOC
Hot-spot PPF	2.47	2.35	2.16	1.99
Hot-stripe PPF	1.93	1.90	1.84	1.76
Plate-wise PPF	1.86	1.82	1.76	1.69
Fuel element-wise PPF	1.10	1.13	1.13	1.14

TABLE VI
Kinetics Parameters Calculated by MCNP6

Kinetics Parameter	SU ($\pm$) ^a	EOC (+/-)
Prompt neutron generation time Λ (μ s)	250.06 (2.31)	252.63 (2.22)
Effective delayed neutron fraction (β_{eff})	7.24E-3 (0.21)	7.18E-3 (0.19)

^a1- σ standard deviation with 68.2% significance.

parameters for SU and EOC have nearly negligible differences, indicating that similar kinetic behaviors will be expected during reactor transient activities at SU and at EOC.

IV.B.4. Reactivity Control and Feedback

As mentioned in Sec. II, four “#” shaped hafnium control blades were utilized for both criticality and safety control 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). Reactivity worths of the control blades are needed by

safety analysis codes to determine the correct negative reactivity inserted into the core after a reactor scram.

Figure 10 shows the *integral* and *differential* reactivity worth curves of the control blades in the split core at SU and EOC. The *integral* reactivity worths were obtained by placing the control blade at different axial positions and calculating the reactivity changes from the case with control blades fully inserted. The *differential* reactivity worth represents the worth change over a unit length change of the control blade. It is usually expressed in $\$/\text{cm}$ or $\text{¢}/\text{cm}$. The differential worth at each control position can be readily calculated by the differential value of that point in the integral worth curve. As seen in the right plot in Fig. 10, 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 a scram, if the control blade insertion speed is assumed to be constant and the reactor is assumed to be operating at critical status. Therefore, a faster power reduction after a scram is expected for the SU case. This fact was verified by the PARET simulation results discussed in Sec. V.

The excess reactivity, total control worth, shutdown reactivity, and shutdown margins of the control system for SU and EOC core were estimated and are summarized in Table VII. The shutdown margin was estimated with the assumption that one of four “#” shaped control blades is stuck in the full out position (the stuck-rod criterion) and all three of the others are fully inserted in the core. The identifying numbers of the control blades are shown in Fig. 2. The results shown in Table VII indicate all reactivity control requirements specified in Table II have been satisfied with the present design.

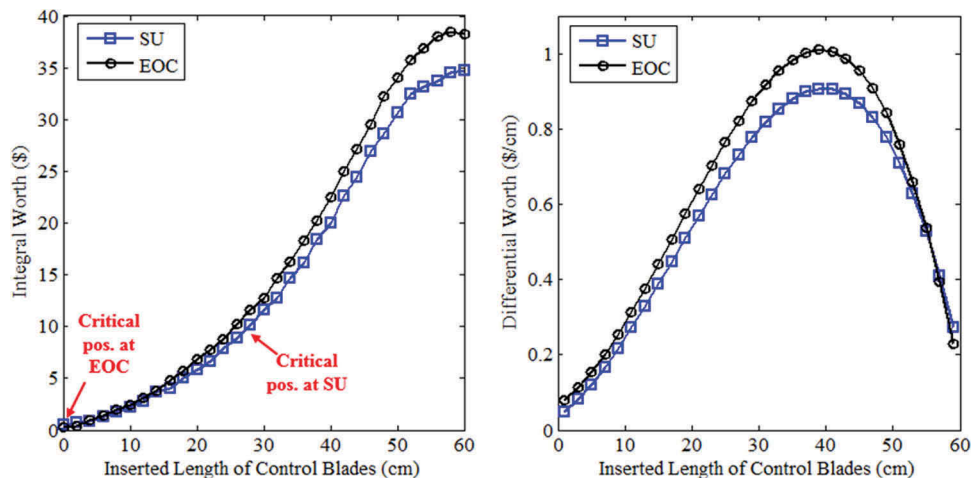


Fig. 10. Integral (left) and differential (right) reactivity control worth curves at SU and EOC.

TABLE VII
Reactivity (%Δk/k) Control Assessment

	SU	EOC
Excess reactivity	7.18	0.00
Total control worth	25.63	27.51
Shutdown reactivity	-18.44	-27.45
Shutdown margin 1 ^a	4.20	12.13
Shutdown margin 2	4.34	12.07
Shutdown margin 3	4.12	12.07
Shutdown margin 4	4.20	12.21

^aIndicating the control blade with the identifier number 1 was stuck.

The reactivity coefficient addresses reactor safety concerns as well as provides valuable quantitative parameters for reactor safety analyses. In this study, three typical reactivity coefficients for the low-power LEU core were calculated using MCNP6: the moderator temperature coefficient (MTC), the fuel temperature coefficient (FTC), and the void coefficient (VC). A direct perturbation methodology was applied to calculate all these coefficients. The perturbed case was produced by manually perturbing a single physical property with respect to the reference case, in which the moderator and fuel temperature were assumed to be the standard room temperature (20°C). The corresponding reactivity coefficient was thereby determined by dividing the change in reactivity over the change in the temperature or void,

$$\frac{\partial \rho}{\partial x} \approx \frac{\Delta \rho}{\Delta x} = \frac{\rho_{pert} - \rho_{ref}}{x_{pert} - x_{ref}},$$

where x denotes the perturbed parameter.

To calculate the MTC, the water density was decreased with elevated temperature. The physical temperatures of the coolant cells (specified by TMP card in MCNP) were also changed correspondingly. The temperature dependent scattering kernel impact on k_{eff} was taken account separately by considering the reactivity changes with scattering kernels of two different temperatures (the reference temperature and 76°C). The impacts of the density variation and scattering kernel change were combined to determine the MTC. To calculate the FTC, the reference model was used to generate perturbed cases by modifying only the fuel temperature and Doppler-broadened cross section to determine the reactivity changes in the perturbed cases. With the available data in MCNP6, a set of ENDF-B7.1 libraries generated for 250 K, 293.75 K, and 600 K were used for the perturbation. The TMP card for fuel was modified accordingly, whereas the thermal expansion was neglected in the

calculation due to the compact nature of the core. In order to examine the spatial characteristic of the coolant VC, a series of perturbed cases were generated with voids created at different axial zoning of the coolant channels. The voiding height of each zone is 6 cm. The reactivity change was calculated for each voided case, and the corresponding VC was determined by dividing the reactivity change by the volume of void. More detailed information and results on the reactivity coefficient calculation for the new reactor can be found in Ref. 27. Here we only summarize the major reactivity feedback coefficients for SU and EOC core in Table VIII.

As can be seen, all the reactivity coefficients are negative, which is required for reactivity control in reactor power transients. We have not taken credit for these coefficients in the transient safety studies presented in Sec. V primarily because the MTCs and FTCs are very small. Since the variation of the moderator and fuel temperature will not be exceedingly high for such a low power reactor, the negative reactivity feedback contributions from the moderator and fuel are negligible. In addition, the time delay effect needs to be considered if one wants to incorporate moderator feedback properly in the transient analysis. In nearly all situations, the coolant is operated in deeply subcooled conditions, and thus the void fraction is very small, so as the void feedback effect. Since the reactivity coefficients are negative, neglecting these effects actually provides more conservative estimations in the safety studies.

V. THERMAL HYDRAULICS SAFETY STUDY

Preliminary safety analyses were performed at both steady-state and design-basis accident conditions using the safety analysis code PARET (Refs. 8 and 28). The accident scenarios at the SU and EOC states were examined to compare the T/H performance characteristics at different conditions. SU and EOC states are the two most limiting conditions during an equilibrium cycle. At SU, fresh fuel elements are loaded and the power is highly concentrated in the center portion of the core due to the

TABLE VIII
Major Reactivity Coefficients for LEU Core

	MTC (pcm/°C)	FTC (pcm/°C)	VC (pcm/liter)
SU	-6.7	-1.3	-416.3
EOC	-4.5	-1.3	-328.5

effect of the control blades, which makes SU a limiting state point for many transient events. At EOC, the control blades are fully withdrawn and provide the lowest differential worths to the core, which makes EOC another limiting state point at some events. The PARET code is a computational tool developed by Argonne National Laboratory (ANL) with particular suitability for plate-type research reactor safety analyses. It consists of a one-dimensional (1-D) T/H model and a point-kinetics model to couple the neutronics and thermal hydrodynamics effects on reactor behavior during normal and off-normal conditions. A reactivity model is integrated in the code to provide proper thermal feedback from the T/H model to the neutronics model.

For simplicity, a two-channel PARET model was developed in this study to account for physical conditions in the hot and average channels, respectively. Each channel includes a 1-D slab geometry of a 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. 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 defining proper boundary conditions. For the split core design, a downward flow with a total flow rate 8000 gal/min or 1817 m³/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°C based on energy conservation. The reactor was designed with an open pool. The height from the outlet of the coolant channel to pool surface is about 10 m. Thus, the outlet pressure was assumed to be 200 kPa. Under these conditions, the calculated pressure drop along the coolant channel is ~65 kPa. All the T/H conditions were designated with the intention to have T/H performance similar to the NBSR (Ref. 1). A summary of the required T/H boundary conditions and parameters based on the channel dimensions is outlined in Table IX.

As mentioned in Sec. III, two thermal constraints were considered in the safety studies as safety satisfactory criteria: the PCT and the MCHFR. In the present study 515°C is the maximum allowed PCT of silicide LEU, and 1.78 is set for the MCHFR. The T/H performance characteristics of the split core in both normal and off-normal (hypothetical accident scenarios) conditions were examined against these thermal limits to ensure the fuel integrity and operation confidence of the core design.

In the PARET model, the active core was simplified as a two-channel model with thermal heat profiles and

TABLE IX

T/H Conditions and Parameters Used in the PARET Model

Conditions and Parameters	Value
Outlet pressure (kPa)	200
Inlet temperature (°C)	37
Inlet volumetric flow rate (m ³ /hour)	1817
Flow area of the channel (cm ²)	1.966
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

kinetics parameters provided by neutronics calculations. Besides steady-state performance, two hypothetical transients, accounting for a reactivity insertion accident (RIA) and a loss of flow accident (LOFA), were considered with the real-time monitoring of the fuel cladding temperature, power, and mass flow rate. The T/H safety margins of the new design were justified by examining the MCHFR and the PCT to the thermal constraints defined in Sec. III.

V.A. Steady-State Conditions

The PARET inputs have been run to establish the steady-state conditions for the core at full power (20 MW). Table X summarizes the T/H characteristics of the *hot* and *average* channel for SU and EOCs. 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.

V.B. Reactivity Insertion Accident

The RIA models the power excursion with a large positive reactivity inserted into the core that may be caused

TABLE X

T/H Performance Characteristics at Steady-State Conditions

Core State Point	SU		EOC	
	Hot	Avg.	Hot	Avg.
Coolant outlet temperature [°C]	48.7	47.0	54.6	47.1
Peak clad temperature [°C]	92.9	74.2	86.6	67.7
Peak fuel temperature [°C]	103.4	80.5	94.7	72.6
MCHFR	3.37	5.49	4.19	7.03

by experiments removed from the core. Both SU and EOC cores were considered for the accident. The reactor was assumed to be initially operated at a full power of 20 MW. A very large positive reactivity of 1.5 \$ was inserted into the core in 0.5 s. The reactor scram occurred with a power trip at 24 MW (120% of the full power). A time delay constant (25 ms) was defined in the model to account for the finite time required for the safety rods to start to move after scram. The control blades were assumed to move with a constant rate 1.2 m/s for the scram. Reactivity feedback coefficients (including fuel, coolant density, and moderator void) and the period trip were neglected in these analyses.

Figure 11 shows the reactor power transient behavior of first 2 s in the RIA. Because the kinetics parameters for SU and EOC have only slight differences (see Table VIII), the power increases with identical rates in the transients for SU and EOC, and both reach the maximum power around 26 MW at about 0.2 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, due to the higher differential reactivity worth of control blades at the critical position in SU (see Fig. 10). Since the core is initially operating at critical and the control blades start to move at critical positions after scram, the SU core thereby obtains larger negative reactivity than the EOC core in a same period following the scram. As a result, the power at the SU case decreases faster after the scram.

The corresponding PCT and MCHFR behavior for the hot channel are shown in Figs. 12 and 13, respectively. The specified thermal limits for these two

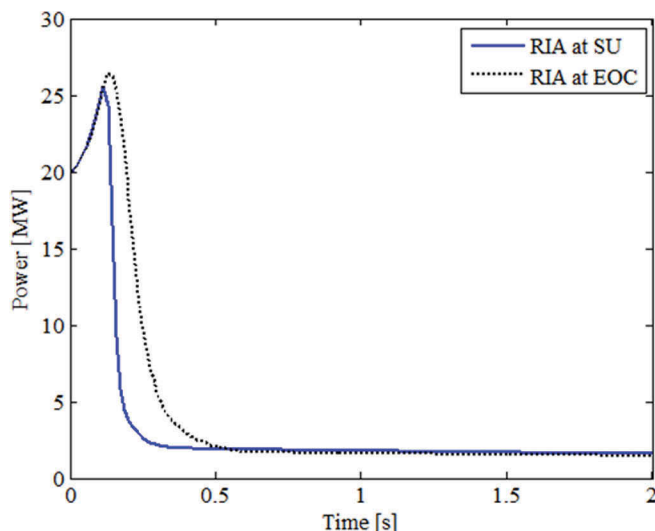


Fig. 11. Reactor power behavior in RIA.

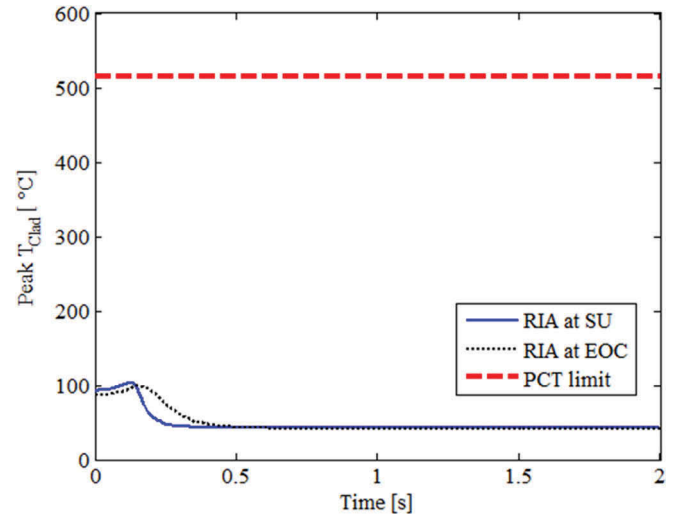


Fig. 12. Peak clad temperature transient profile in RIA.

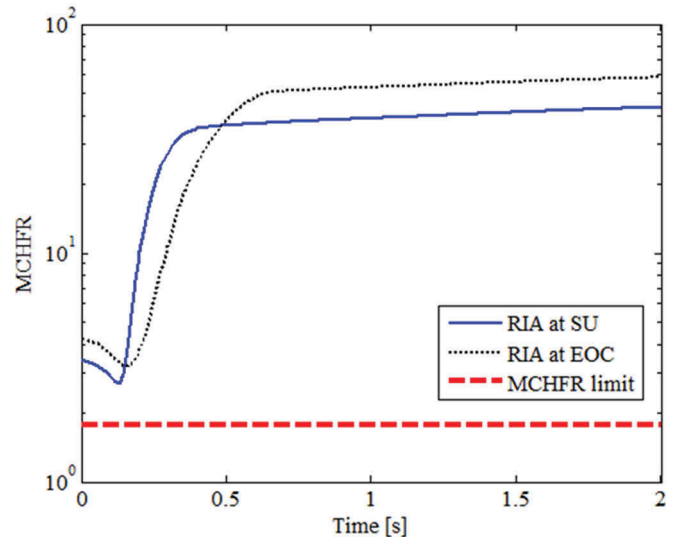


Fig. 13. Minimum critical heat flux ratio transient profile in RIA.

parameters are also shown as dashed lines in the figures. It can be clearly seen that the safety criteria are satisfied during the entire transient.

V.C. Loss of Flow Accident

The LOFA is a malfunction of the cooling system while the reactor is operating at its full power. Such accidents involve strong T/H interactions between the core and the coolant loop. In this study, the flow decay due to pump coast down was modeled as an exponential $[exp(-t/T)]$ decrease with a period $T = 1$ s to mimic a fast loss of flow (LOF) scenario. The reactor was assumed to

be initially operated at full power with downward cooling flow conditions. The scram was tripped when the flow was reduced by 15%, with a response delay time of 0.2 s. In addition, when the flow reached 15% of its initial value, the natural convection valve (NCV) opened to allow flow reversal and the establishment of passive decay heat removal process by natural circulation flow. Other parameters used in the LOF were similar to those in the RIA case.

The power transient behavior for the LOFA is illustrated in Fig. 14, and the corresponding PCT and MCHFR behavior in the hot channel are shown in Figs. 15 and 16, respectively.

As shown in Fig. 14, the power behaviors for SU and EOC are nearly identical. After the scram, there is a sharp

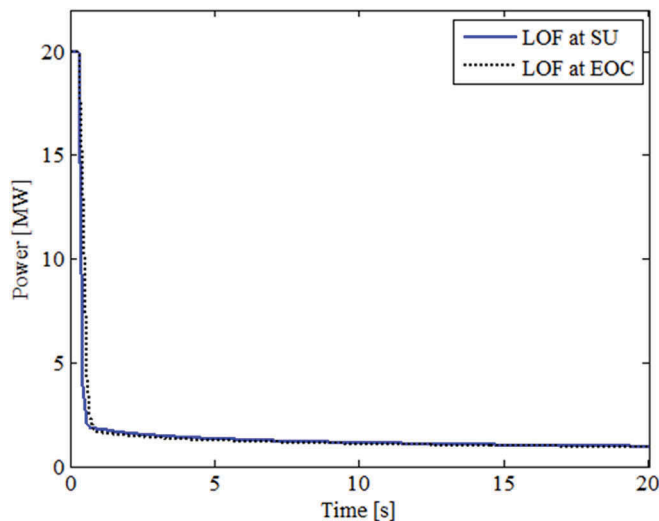


Fig. 14. Reactor power behavior in LOFA.

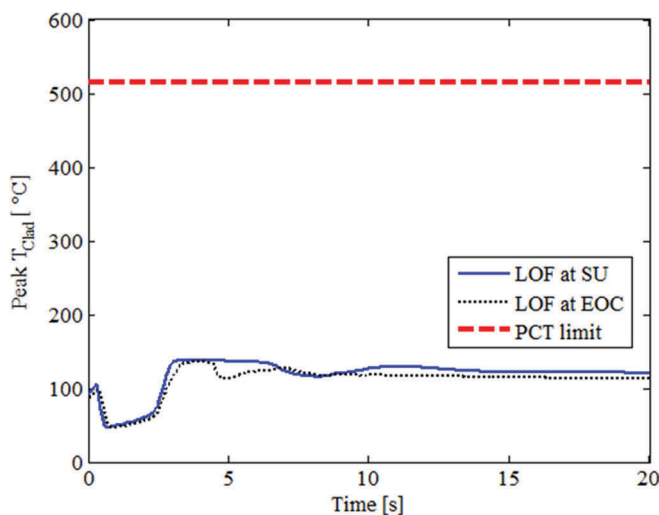


Fig. 15. Peak clad temperature transient profile in LOFA.

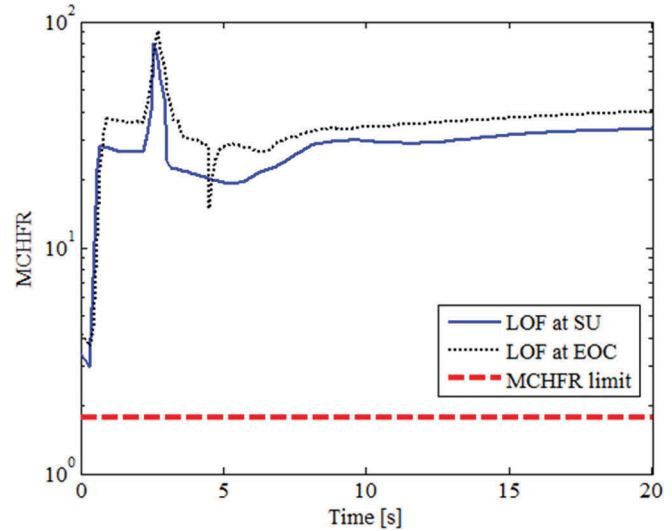


Fig. 16. Minimum critical heat flux ratio transient profile in LOFA.

decrease in reactor power (as well as core temperatures including fuel, cladding, and coolant) at less than 1 s into the transient, followed by a constantly slow decrease with the power level dominated by the radiative decay heat. Nevertheless, the temperature of the core after the scram exhibits a steady-state rise due to the degradation of the core cooling capacity during the flow decay (see Fig. 15). Again, large thermal safety margins in terms of PCT and MCHFR can be observed in Figs. 15 and 16 during the LOFA.

Figure 17 illustrates the mass flux transient characteristics along the hot channel (the average channel has similar behavior) for the LOFA. As can be seen, at

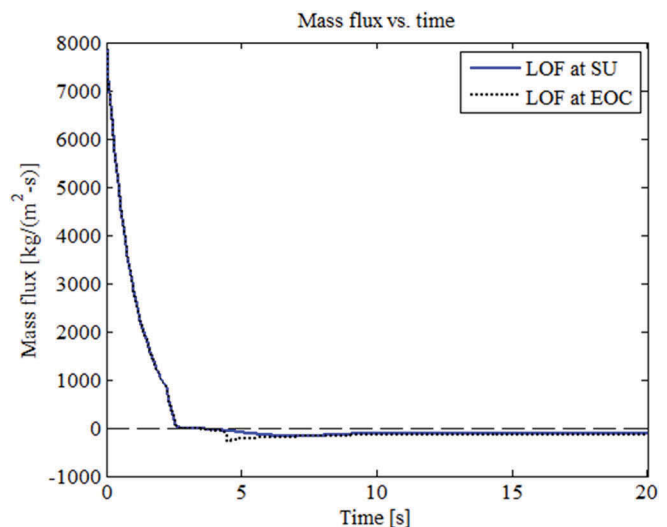


Fig. 17. The mass flux transient behavior in LOFA.

some point, the buoyant forces, due to the coolant heatup by the decay heat, exceed the flow coast down inertia. As a result, a mixed convection flow is established followed by a flow reversal and natural circulation regime 4 or 5 s into the transient. Consequently, the core temperatures exhibit a second rise due to the combined effect of constant decay heat and continued reduction of the core flow rate. The increase is further sustained as the flow regime passes to laminar regime and to mixed flow when the NCV opens to allow flow and the establishment of passive decay heat removal process by natural circulation flow. The core temperatures begin to decrease only when the natural circulation flow is fully established as indicated by the cladding temperature profile shown in Fig. 15.

VI. SUMMARY AND CONCLUSIONS

Neutronics and safety studies for a proposed new LEU-fueled research reactor optimized for cold neutron production at NIST have been performed. The reactor core has two horizontally split halves, and each half consists of 9 MTR-type fuel elements. The core is surrounded by a heavy water reflector that provides a large volume thermal flux trap. Two cold neutron beams and four thermal neutron beams are located in the reflector area.

Neutronics studies were performed using MCNP6. A multi-cycle equilibrium core was achieved based on a three-batch fuel management scheme and an iterative search procedure. The core performance characteristics at four representative burnup states were presented and discussed. The results demonstrated the superiority of the new design with respect to the thermal and cold neutron flux performance of the existing NBSR neutron source. The maximum perturbed thermal flux in the core reaches $\sim 5.0 \times 10^{14}$ n/cm²-s, and the current of cold neutrons at the exit hole of the source has a gain of a factor of ~ 6 compared to that of the existing NBSR source. The estimated brightness of the cold neutron demonstrates an average gain factor of ~ 4 with respect to the NBSR cold neutron performance. The control elements provide sufficient reactivity control and large shutdown margins. Reactivity coefficients were evaluated to ensure the negative feedback.

Preliminary safety studies were performed using the PARET code to evaluate the safety features of the reactor at steady-state and some design-basis protected transient conditions. The RIA and LOFA were modeled with conservative assumptions to maximize the severity

of the event. The accidents were analyzed at both SU and EOC conditions of an equilibrium cycle. The PCT and MCHFR during the transients were shown to fully satisfy the safety criteria. The DNB CHF was estimated by the Mirshak correlation²¹ in this study. The safety analysis results indicated that adequate safety margins were achieved in steady-state conditions. Detailed transient analyses for the postulated accidents showed that no fuel damage or cladding failure would occur with the protection of reactor scram.

All the preliminary feasibility studies suggest the proposed research reactor concept offers a demonstrable potential to greatly expand the cold neutron capability with a 20 MW power and certified LEU fuels. The new reactor design is currently an ongoing project at NIST. Several important tasks will be performed in the near future. For example, safety analyses will be expanded to the whole primary loop with more detailed flow conditions described. This work will be performed with a RELAP5 model.²⁹ The U-10Mo LEU fuel (a uranium alloy with 10% molybdenum by weight) is not yet qualified, but its high uranium density is of great interest in research reactor community.³⁰ This fuel will be investigated in the next stage to assess the neutronics feasibility and safety performance under the split core concept. Research efforts will continue on the CNS geometry to achieve the maximum cold neutron gain under the physical constraints.

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