IMPACT ON SAFETY OF THE CONVERSION OF THE NIST RESEARCH REACTOR

D.J. Diamond, J.S. Baek, A.L. Hanson, L-Y. Cheng, A. Cuadra, N.R. Brown
Brookhaven National Laboratory, New York, USA
diamond@bnl.gov

Abstract

The NIST research reactor is heavy-water moderated and cooled and fueled with high-enriched uranium fuel. It operates at 20 MW and provides thermal and cold neutrons for researchers. A program is underway to convert the reactor to low-enriched uranium (LEU) fuel using a U-Mo alloy. An LEU core has been designed that minimizes changes to the fuel elements and maintains the current optimum fuel cycle length, but incurs a penalty to researchers because the additional ^{238}U in the core reduces the neutron flux that goes into the beam tubes. In the current study, the safety of the converted core is analyzed for normal operation and under postulated reactivity-initiated and loss-of-flow accidents. Neutronic parameters as a function of burnup are obtained from the three-dimensional Monte Carlo code MCNP and transient analysis is done with the system thermal-hydraulic code RELAP5. The parameters that are calculated to assure safety include shutdown margin, reactivity feedback coefficients, critical heat flux ratio, onset of flow instability ratio, and clad temperature. The results show that the conversion will not lead to significant changes in the safety analysis and there is adequate margin to fuel failure during accidents.

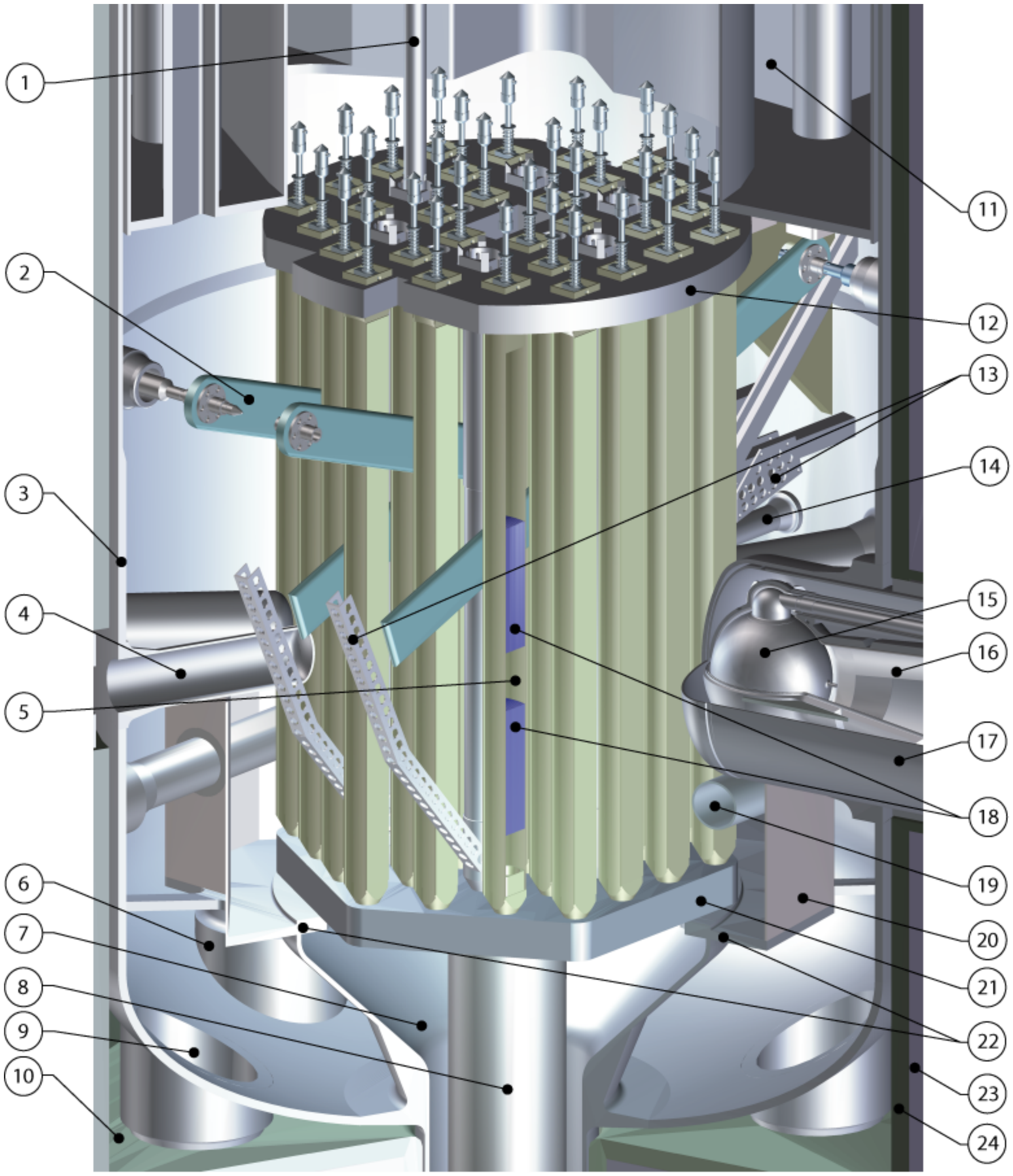
1. Introduction

1.1 Background

The Global Threat Reduction Initiative at the National Nuclear Security Administration is working toward the conversion of the research reactor at the National Institute of Standards and Technology (NIST) from high-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel. This is being made possible by the development of a U-Mo alloy (10% Mo) that has the high density needed to compensate for the reduction in fuel enrichment and the ability to withstand the temperatures and irradiation in the reactors that would use it. A design for the converted core has been developed [1] to minimize changes to the fuel elements and maintain the optimal 38.5-day fuel cycle of the reactor (known as the NBSR). In this paper the focus is on the safety analysis to assure that the reactor can be operated with similar margins as the HEU core.

1.2 Description of the NBSR

The NBSR is a heavy water (D_2O) cooled, moderated, and reflected research reactor that operates at a design power of 20 MWth. The NBSR is cooled by forced upward circulation through two concentric plena below the lower grid plate of the reactor. There are thirty fuel elements in the core on a triangular pitch. The large volume and spacing within the core provides flexible capabilities for thermal neutron irradiation. There are four shim arms providing reactivity control that swing from a pivot point above the core. A regulating rod of aluminum that moves vertically is present for fine reactivity control. Figure 1 is a drawing of the basic features of the reactor.



1	Poisoned Hold Down Tube (1 of 7)	13	Shim Arm Guide Extension # 3
2	Shim Arm #2	14	RT-3 Pneumatic Tube
3	Reactor Vessel	15	Cold Source Moderator Chamber
4	BT-2 Thimble	16	Cold Source Insulating Vacuum
5	Element Unfueled Gap	17	Cold Source D_2O Cooling Jacket
6	D_2O Outlet (1 of 2)	18	Fuel Plates, Upper and Lower Sections
7	D_2O Inlet - Outer Plenum	19	Through Tube (North)
8	D_2O Inlet – Inner Plenum	20	Heavy Water Holdup Pan
9	Fuel Transfer Chute Opening	21	Lower Grid Plate
10	CO_2 Filled Region	22	Outer Plenum Flange
11	Inner Reserve Tank	23	Steel (Thermal Shield)
12	Upper Grid Plate w/30 Elements	24	Lead (Thermal Shield)

Figure 1 NBSR Vessel Internals and Reactor Core

The fuel elements are split axially into two halves with a 7 in (17.8 cm) gap located between the two halves at the mid-plane. This mid-plane gap maximizes thermal neutrons to the beam tubes for thermal and cold neutron scattering research while minimizing "contamination" from fast neutrons and gamma rays. Insertion of eight radial beam tubes and two cold neutron sources into the plane of the fuel gap allows high intensity, low energy beams of neutrons to be extracted. The gap can be seen on the fuel element drawing in Figure 2. Each half-element encapsulates 17 curved fuel plates (material test reactor (MTR) curved plate geometry) as shown in the figure.

Presently the NBSR is fueled with HEU with a nominal ^{235}U enrichment of 93%. The fuel is U_3O_8 in an aluminum powder dispersion that is clad in aluminum alloy. The fuel meat for the LEU conversion of the NBSR is U10Mo metal foils with the same aluminum alloy cladding and with a zirconium interlayer between the fuel and the clad to improve fuel behavior under irradiation.

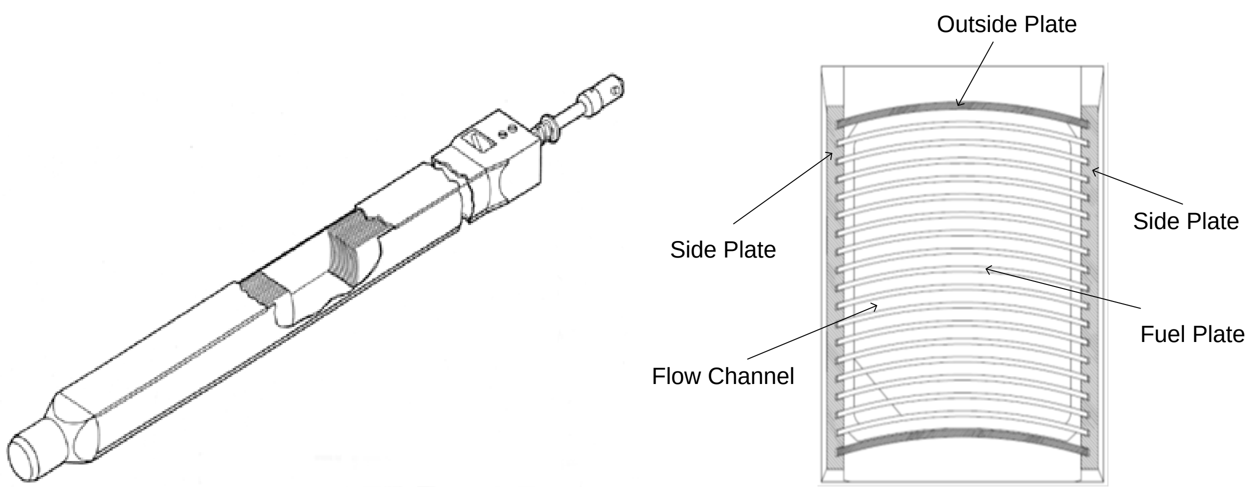


Figure 2 Cutaway Isometric Drawing of Fuel Element and Cross Sectional View

2. Calculational Methodology

2.1 Neutronics

A detailed three-dimensional Monte Carlo model for both MCNPX and MCNP5 [1, 2] had been used for the HEU core. This model with appropriate changes was used to calculate the uranium loading needed for the LEU fuel for the 38.5-day fuel cycle. Then the composition of each half fuel element (treated uniformly) was calculated at distinct points in the fuel cycle. Key physics parameters could then be calculated and compared with those obtained for the HEU core. The model includes a plate-by-plate (1020 total) representation of each fuel element (FE), the water gap at the axial mid-plane, beam tubes (BTs), shim arms, regulating rod, axial and radial reflectors, cold neutron sources, and other structures internal to the NBSR. A planar cross section at the core midplane is shown in Figure 3.

The four state points used for the analysis were start-up (SU), beginning-of-cycle (BOC), middle-of-cycle (MID), and end-of-cycle (EOC). The SU core has fresh fuel in four locations and the short-lived fission product poisons, such as ^{135}Xe, in the previously irradiated fuel have decayed away during the refueling period since the end of the previous cycle. In the BOC core, all the short-lived fission products, including ^{135}Xe, are at equilibrium concentrations. EOC is the point at which the shim arms are completely removed and the MID point is halfway between BOC and EOC.

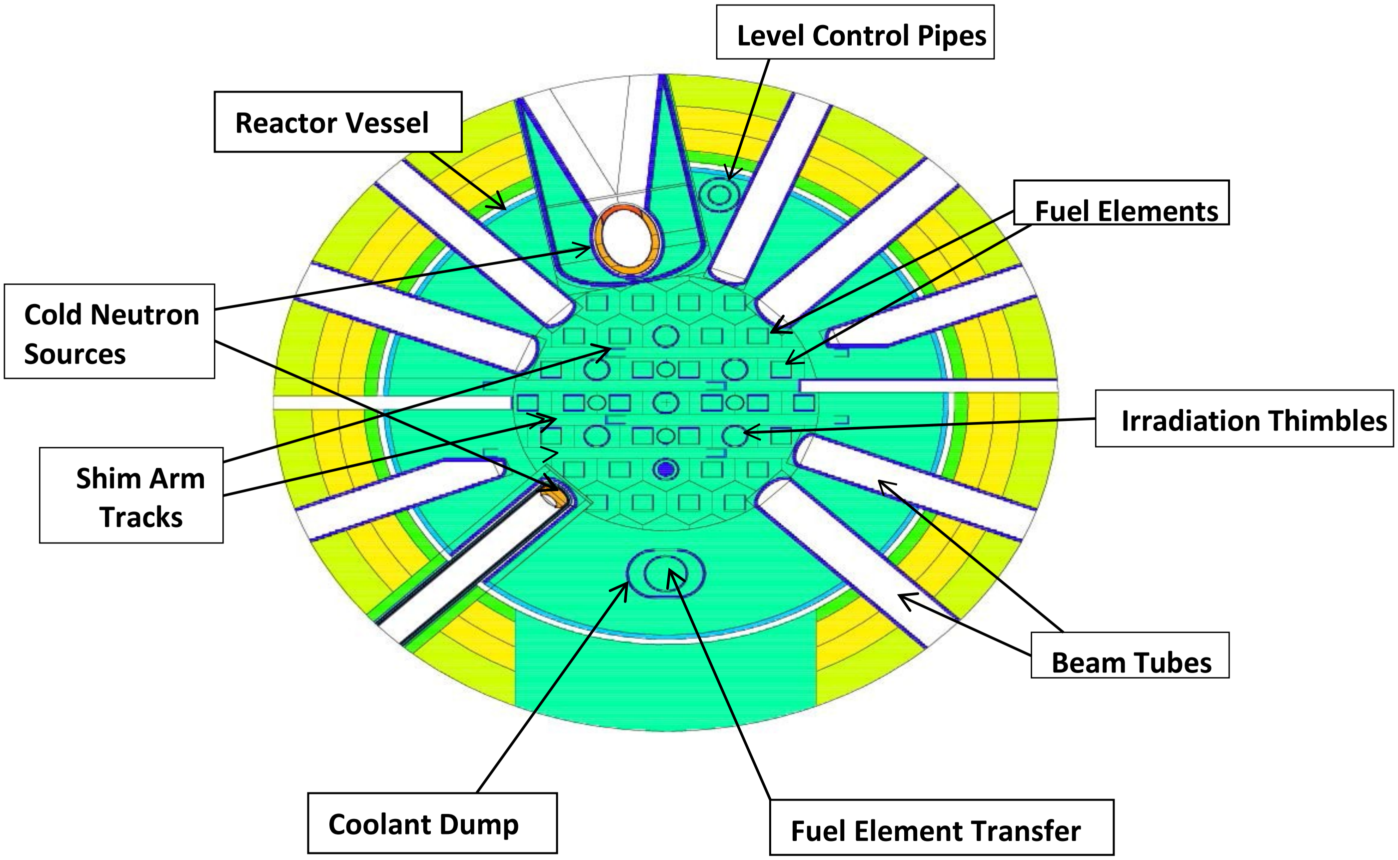


Figure 3 Planar View of MCNP Model at Core Midplane

2.2 Transient Analysis

The RELAP5/MOD3.3 model of the NBSR [3] simulates the transport of heat and coolant in the primary system for various accidents. Figure 4 shows a nodal diagram of the NBSR. The reactor vessel is divided into a number of interconnected hydrodynamic volumes and heat structures with internal heat generation used to model the fuel plates. In the nodal diagram, hydraulic components are described by numbers with the background color of light gray and heat structures are represented by the red background color.

A "hottest cell" channel is defined as the channel containing the axial mesh interval with the highest power density calculated by MCNP and a "hottest stripe" channel is defined as the channel representing the highest axially integrated (along one-third of the plate width) power density calculated by MCNP.

The inner six fuel elements, with coolant from the inner plenum, are modeled as an inner group and the outer 24 fuel elements as an outer group. The inner group is divided into five different channel types, each with a different heating rate and flow area. The five types of channels are the hottest cell channel and hottest stripe channel with no mixing of coolant in the mid-plane (central unfueled) gap; the hottest cell channel with mixing of coolant from the other channels in the fuel element; a channel for 16 non-hot fuel plate channels with mixing of coolant from the hottest cell channel in the hot fuel element; and a channel for non-hot (average) channels in five elements. Similarly, the outer group is divided into five channel types, and three additional channels corresponding to eighteen average elements in subsets of six fuel elements.

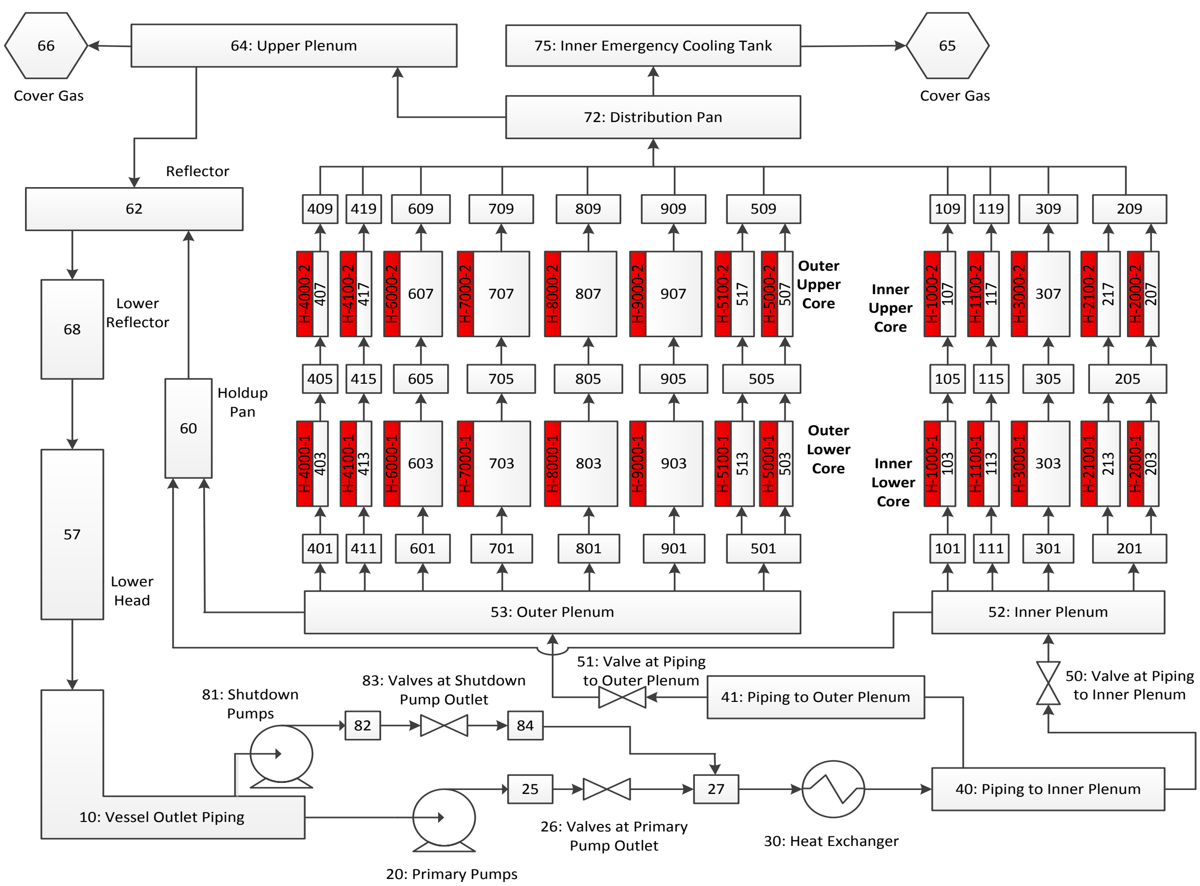


Figure 4 Nodal Diagram of RELAP5 Model for NBSR

3. Safety Analysis

3.1 Nuclear and Thermal-Hydraulic Design Parameters

Results for key parameters that influence safety are obtained at two points in the fuel cycle, SU and EOC. The SU condition is the most reactive point in the cycle and has the highest power peaking; therefore, it is the bounding point in the fuel cycle for most analyses. The EOC state point is bounding in transients in which the reactor shutdown is particularly important since the initial differential shim arm worth after reactor trip is lowest when the shim arms are withdrawn at EOC.

A summary of the significant parameters is given in Table 1 for both the HEU and LEU cores. In all cases shown, the change from HEU to LEU does not have a significant impact on the safety margin.

Items 1 and 2 in the table relate to the Technical Specifications [4] that state that the core cannot be loaded such that the excess reactivity will exceed 15% Δk/k and that the NBSR shall not be operated if it cannot be kept shutdown with the most reactive shim arm fully retracted. Items 3 and 4 refer to a backup shutdown system. The NBSR has a pipe, referred to as the moderator dump, whose entrance is just above the fueled portion of the core. If an emergency situation requires it, the pipe can be used to drain the coolant to that dump level leaving the core with no upper reflector. The lack of an upper reflector results in the reactor becoming subcritical. Items 5-8 show that reactivity control of the reactor will not be impacted.

Items 9-14 are reactivity coefficients that must be negative to provide safe feedback. The void coefficients are a function of where the void is placed. Similarly, Items 15-18 show that if voided regions (at the cold neutron source, CNS, or beam tubes, BT) are flooded with heavy water, the result will not be an excessively large reactivity insertion. The reactivity insertion is less than the 0.5 %Δk/k used to analyze the maximum reactivity insertion accident as discussed in Section 3.2.

Items 19-22 are kinetics parameters [5] that are used in transient analysis. The equilibrium LEU core has contributions to the fission rate from ^{238}U and ^{239}Pu which are not significant contributors in the HEU core. This reduces both the delayed neutron fraction and the neutron lifetime; however, as is seen in the discussion of transients in Section 3.2 the effect is not significant.

Items 23-25 show the changes in power in a half-element. Although there appears to be a significant change at SU, as will be shown below, this does not have a significant impact on thermal limits. The most important change in power is the shift in the power peaking from the periphery of the core to the central region of the core due primarily to the additional amount of ^{238}U in the LEU core relative to the HEU core. The isotope acts as an absorber, reducing the leakage out of the core. The consequence of the reduced leakage from the core into the beam tubes and CNS is a penalty to the users of the NBSR but not an important safety consideration.

Table 1 Summary of Core Parameters

Parameter	HEU Core	LEU Core
1. Excess reactivity (%Δk/k)	6.7	6.3
2. Shutdown margin with highest worth shim arm out (%Δk/k)	-10.1	-10.8
3. k_{eff} with moderator at dump level, SU	0.9857	0.9849
4. k_{eff} with moderator at dump level, EOC	0.9124	0.9215
5. Shim arm worth, SU (%Δk/k)	24.9	24.2
6. Shim arm worth, EOC (%Δk/k)	27.2	26.0
7. Regulating rod worth, SU (%Δk/k)	0.50	0.53
8. Regulating rod worth, EOC (%Δk/k)	0.45	0.43
9. Moderator temperature coefficient, SU (%Δk/k/°C)	-0.0313	-0.0280
10. Moderator temperature coefficient, EOC (%Δk/k/°C)	-0.0275	-0.0228
11. Void coefficient, all thimbles voided, SU (%Δk/k/liter)	-0.047	-0.040
12. Void coefficient, all thimbles voided, EOC (%Δk/k/liter)	-0.039	-0.049
13. Void coefficient, all FEs voided, SU (%Δk/k/liter)	-0.016	-0.018
14. Void coefficient, all FEs voided, EOC (%Δk/k/liter)	-0.019	-0.015
15. Reactivity insertion for CNS flooded, SU (%Δk/k)	0.24	0.19
16. Reactivity insertion for CNS flooded, EOC (%Δk/k)	0.25	0.20
17. Reactivity insertion for flooding one tangential BT, SU (%Δk/k)	0.27	0.28
18. Reactivity insertion for flooding one tangential BT, EOC (%Δk/k)	0.20	0.19
19. Delayed neutron fraction, SU	0.00665	0.00650
20. Delayed neutron fraction, EOC	0.00661	0.00648
21. Recommended prompt neutron lifetime, SU (μs)	650	600
22. Recommended prompt neutron lifetime, EOC (μs)	750	700
23. Peak half-element relative power, SU	1.28	1.35
24. Peak half-element relative power, EOC	1.18	1.15
25. Peak half-element relative power with misloaded FE	1.93	1.83
26. Steady state minimum CHFR, SU	3.78	3.87
27. Steady state minimum CHFR, EOC	4.08	4.05
28. Steady state minimum OFIR, SU	5.58	5.69
29. Steady state minimum OFIR, EOC	6.38	6.30

Any limitations on power are reflected in the thermal-hydraulic design-basis. No fuel damage is allowed during normal operation and fuel integrity must also be assured during any credible accident. For normal operating conditions, the acceptance criterion is that heat transfer to the primary coolant shall not exceed critical heat flux (CHF) conditions, including any excursive instability; the latter being defined by “onset of flow instability” (OFI). This would preclude blistering and the potential for fuel damage. The temperature at which blistering might occur is the Safety Limit in the Technical Specifications and hence, also a criterion for fuel damage. For HEU fuel it is 723 K and for LEU fuel it is expected to be similar.

In order to determine how close the reactor operates to CHF or OFI a statistical methodology [6, 7] is first used to determine acceptable limits. Cumulative distribution functions are obtained for the CHF ratio (CHFR), and OFI ratio (OFIR). The correlation used for CHF is from Sudo-Kaminaga [8] and the correlation for OFI is that of Saha-Zuber [9]. These correlations are discussed in [3] along with their application. Results to preclude CHF or OFI with a given probability are given in Table 2. Items 26-29 in Table 1 are the results for the minimum CHFR and OFIR at normal operating conditions. They are clearly much larger than the values needed to show acceptance with more than a 99.9% probability.

Table 2 Statistical Analysis Results for CHFR and OFIR

Probability Level	CHFR		OFIR	
	HEU	LEU	HEU	LEU
90%	1.30	1.30	1.31	1.31
95%	1.39	1.39	1.40	1.40
99.9%	1.78	1.78	1.83	1.83

3.2 Transient Analysis

3.2.1 Reactivity Initiated Accidents

Two reactivity initiated accidents need to be considered: the startup accident and the maximum reactivity insertion accident. The former is assumed to occur at an initial power level of 100 W when contrary to operating procedures and all previous training and experience, the operator withdraws the shim arms steadily without any pause, until the reactor is scrammed by a high power level trip. The accident model uses a reactivity insertion rate for the shim arm withdrawal equal to 5×10^{-4} Δk/k/s. This rate is greater than the maximum measured and calculated rate at any shim arm initial position.

The shim arms are assumed to trip from what would be their initial critical position at full power due to a high power signal. The high power level trip is set to 26 MW (130% of full power). This is conservative because the setting is actually at 125% of power. For conservatism the calculation does not consider any fuel or moderator reactivity feedback and does not consider the period scram which is active below 2 MW.

The predicted reactor power is shown in Figure 5 for 12-18 seconds (where conditions are most limiting). After reaching its peak, the power decreases suddenly as the shim arms are inserted after the reactor trip signal is generated. The initial increase in power is greater in the LEU core due to different kinetics parameters.

In all cases the clad temperature and minimum values for CHFR and OFIR are calculated and shown to be acceptable. The maximum clad temperature is less than ~400 K in all cases for this accident; well below the expected blister temperature. For CHFR and OFIR, the results are given in Table 3 and show that the requirement is satisfied with probability greater than 99.9%.

In the maximum reactivity insertion accident a reactivity insertion of 0.005 Δk/k is assumed to occur in 0.5 s. This amount of reactivity is the Technical Specification limit for the reactivity of any experiment. The reactor would go through a power increase which would be mitigated by reactor trip. The resulting minimum CHFR and OFIR is given in Table 3. For this accident, the peak clad temperature again remains below 400 K (Figure 6).

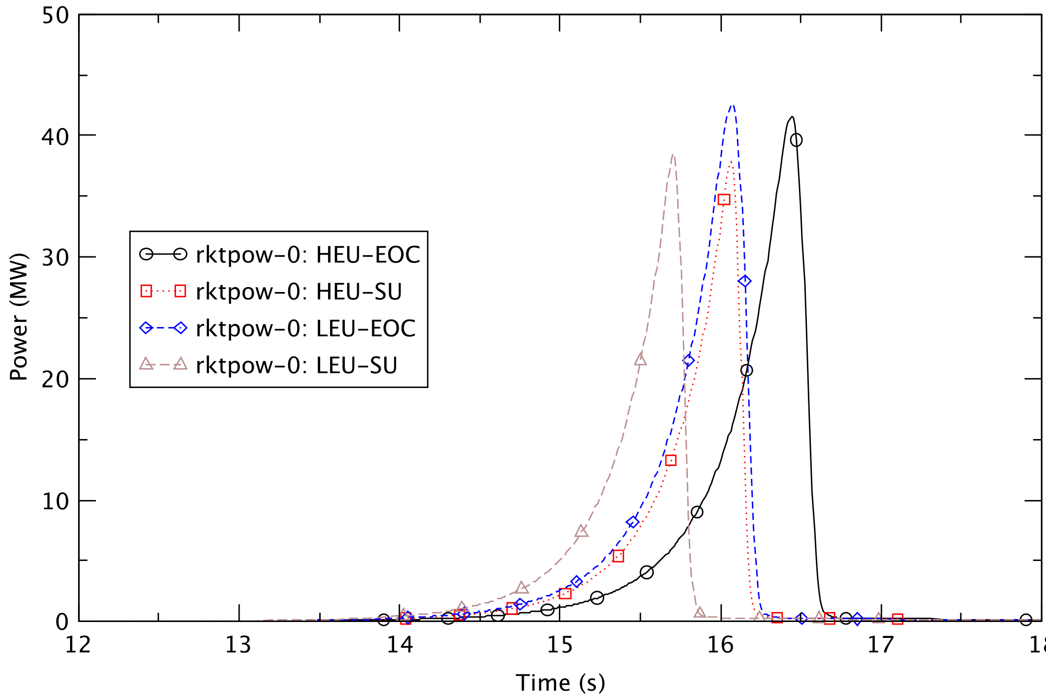


Figure 5 Reactor Power in Startup Accident

Table 3 Minimum Values for CHFR and OFIR

Event	MCHFR		MOFIR	
	HEU	LEU	HEU	LEU
Startup Accident	2.09	2.01	3.47	3.48
Reactivity Insertion Accident	2.26	2.21	3.19	3.26

3.2.2 Loss-of-Flow Accidents

The loss-of-flow accidents considered are:

- loss of offsite power (with and without shutdown pump failure)
- seizure of one primary pump
- throttling of coolant flow to the outer/inner plenum

Each of these events leads to a reactor trip on a low flow signal at either the inner or outer plenum and the concern is the power/flow ratio before the reactor is shut down. Clad temperature, CHFR, and OFIR are calculated using RELAP5 and making conservative assumptions where there is uncertainty or variability. Two examples: Energy deposition is into the fuel only whereas it is known that a sizeable fraction is deposited in the coolant, and flow coast-down is calculated conservatively relative to measured values. A typical result for peak clad temperature is shown in Figure 7 for a loss of offsite power accident. The increase in temperature is less than 30 K. In all events the minimum CHFR and OFIR are ~2.0-4.0 for both HEU and LEU cores and no fuel damage is expected.

3.2.3 Natural Circulation Cooling at Low Power Operation

A RELAP5 transient calculation was performed to simulate operation at low power without forced-flow cooling. The result indicates that operation with natural convection cooling would not lead to fuel element damage at a power level of 100 kW, which is ten times the Technical Specification limit (10 kW) for operation without forced circulation.

The general system behavior with LEU fuel is similar to that with HEU fuel. Even though significant natural circulation flow is not established in the primary system, the safety of the reactor core is maintained without any considerable change of thermal-hydraulic parameters because of the large coolant inventory in the reactor vessel. Minimum CHFRs are high enough so that CHF would not be expected with a probability greater than 99.9%.

4. Conclusion

The NBSR is expected to be converted from HEU fuel to LEU fuel. In order to perform safety analyses, a RELAP5 model has been developed for many events and an MCNP model developed for determining neutronic parameters. The results for the equilibrium LEU core have been compared to those obtained for the current HEU core using the same methodology.

Key neutronic parameters such as shutdown margin, excess reactivity, and reactivity coefficients show that the LEU properties are acceptable and not significantly different from the HEU properties. The power distribution is very different in the two cores with the HEU core having the peaking at the core periphery and the LEU core having the peaking toward the center of the core. This has an effect on neutron beam performance but because the peaking factors do not significantly change, it has little impact on safety. Both cores are shown to preclude both critical heat flux and onset of flow instability with probabilities greater than 99.9% during normal operation.

The RELAP5 analysis results for the reactivity initiated accidents and the loss-of-flow accidents are summarized below.

- The general system behavior with LEU fuel is very similar to that with HEU fuel in all postulated accidents considered in this paper.
- Reactor power increases from time zero in the reactivity initiated accidents due to insertion of positive reactivity but decreases rapidly when shim arms are inserted after a reactor trip, and remains at decay power level until the end of the simulations.
- The initial position of the shim arms at SU or EOC dictates the short term transient response of the cases that include shim arm motion. The initial rate of negative reactivity insertion is higher at SU than at EOC conditions.
- The highest peak cladding temperature of 402 K, much lower than the expected blister temperature, occurs in the HEU core at SU in the startup reactivity accident.

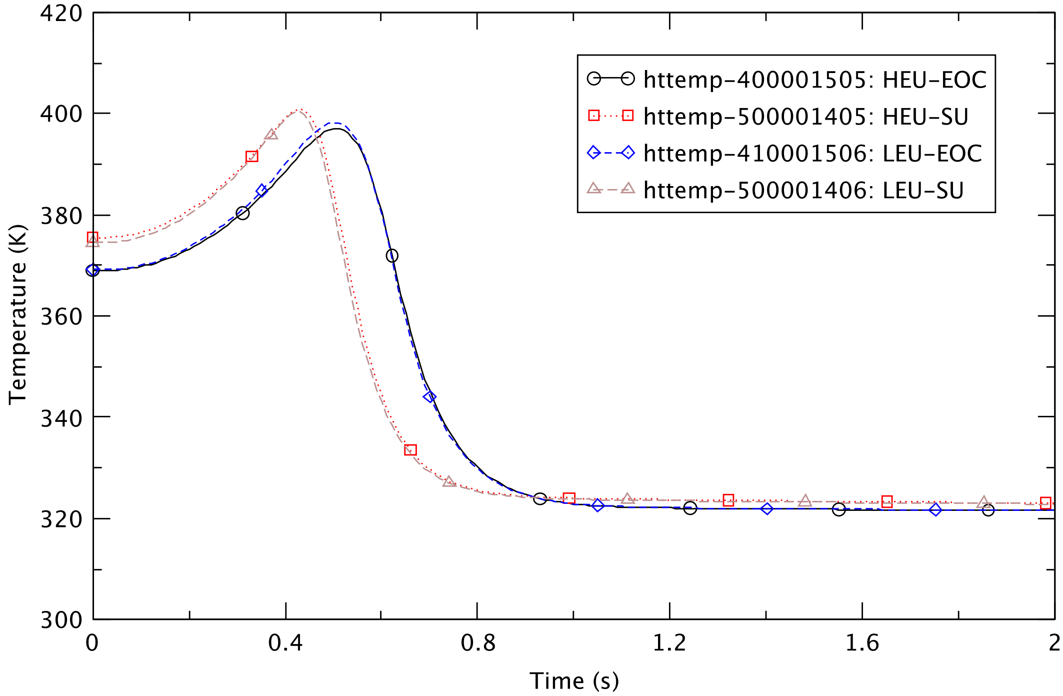


Figure 6 Cladding Temperature in Maximum Reactivity Insertion Accident

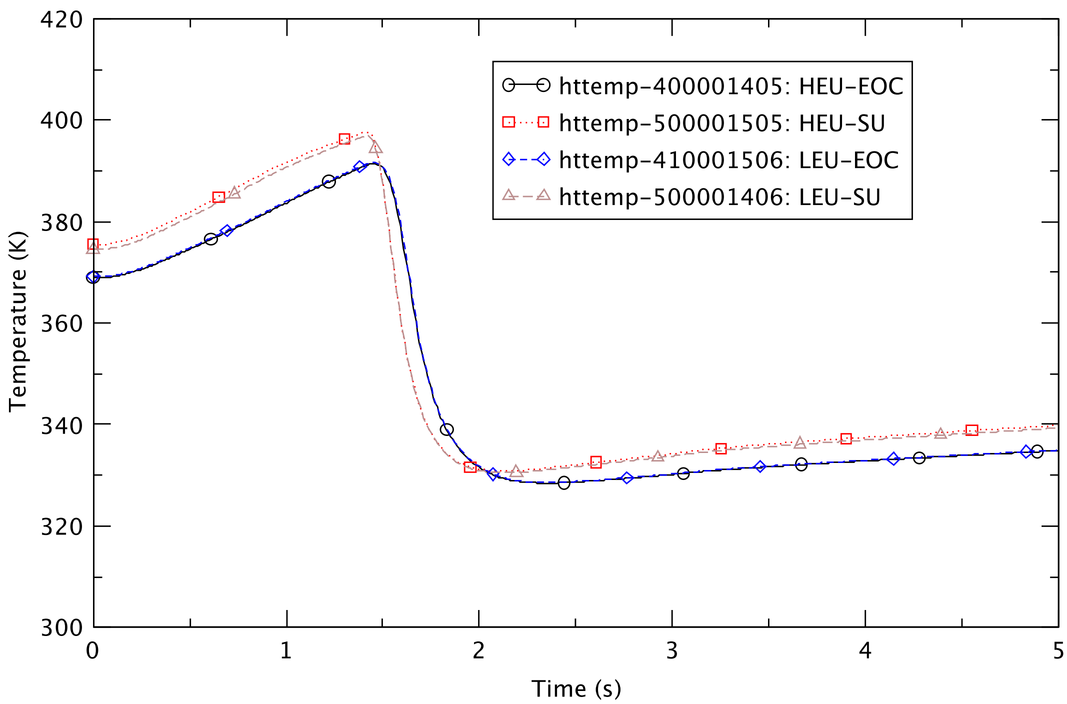


Figure 7 Cladding Temperature after Loss of Offsite Power

- Critical heat flux ratio has been evaluated using the Sudo-Kaminaga correlation. The results show that the minimum CHFR is high enough so that CHF is precluded with probability greater than 99.9% in all accidents with HEU or LEU fuel at SU and EOC. When the Sudo-Kaminaga correlation is not applicable (e.g., with low flowrates), the integrity of fuel elements has been assured by observing that the predicted peak clad temperatures are all much less than the expected blister temperatures.
- The Saha-Zuber criteria are used to evaluate onset-of-flow-instability ratio. The results show that minimum OFIR is high enough so that OFI is precluded with a probability greater than 99.9% in all accidents with HEU or LEU fuel at SU and EOC.

Operation at low power without forced circulation was analyzed. Even though a significant natural circulation flow is not established through the primary system in the simulation of natural circulation cooling at 100 kW operation, it is predicted that the safety of the reactor core is maintained because of the large coolant inventory in the reactor pressure vessel.

From these results it can be concluded that the NBSR reactor with either HEU or LEU fuel is safe under postulated accident conditions and satisfies applicable thermal criteria to assure fuel element integrity.

5. References

[1] A. Hanson and D. Diamond, "Calculation of design parameters for an equilibrium LEU core in the NBSR," BNL Technical Report, Brookhaven National Laboratory, Upton, NY, 2011 September 29.

[2] A. Hanson, and D. Diamond, "A neutronics methodology for the NIST research reactor based on MCNPX," in 19th International Conference on Nuclear Engineering (ICONE19), Chiba, Japan, 2011b May 16-19.

[3] J.S. Baek, L.-Y. Cheng, A.L. Hanson, A. Cuadra, N.R. Brown and D.J. Diamond, "Accident analysis for the NIST research reactor before and after fuel conversion – Revision 2," BNL-98524-2014-IR-R2, Brookhaven National Laboratory, Upton, NY, 2014 July.

[4] NIST, "Technical specifications for the NIST test reactor (NBSR)," Appendix A to License No. TR-5, National Institute of Standards and Technology (NIST), Gaithersburg, MD, 2010.

[5] A.L. Hanson and D.J. Diamond, "Calculation of kinetics parameters for the NBSR," BNL-97007-2012, Brookhaven National Laboratory, Upton, NY, 2012 March.

[6] L-Y Cheng et al., "Physics and safety analysis for the NIST research reactor," BNL-NIST-0803, Rev. 1, Brookhaven National Laboratory, Upton, NY, 2004 April.

[7] A. Cuadra and L-Y. Cheng, "Statistical hot channel analysis for the NBSR," BNL-105288-2014-IR, Brookhaven National Laboratory, Upton, NY, 2014 May 23.

[8] M. Kaminaga, K. Yamamoto and Y. Sudo, "Improvement of critical heat flux correlation for research reactors using plate-type fuel," *J. Nucl. Sci. Technol. 35[12]*, 943-951, 1998.

[9] P. Saha and N. Zuber, "Point of net vapor generation and vapor void fraction in subcooled boiling," Proceedings of the 5th International Heat Transfer Conference, Vol. IV, p. 175, Tokyo, Japan, 1974 September 3-7.

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