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Progress with the Conversion of the NIST Research Reactor

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ABSTRACT

Recent progress on conversion of the 20 MW D_2O-moderated research reactor (NBSR) at the National Institute of Standards and Technology (NIST) has been made with respect to 1) accident analysis, 2) a draft "Reactor Description" (Chapter 4) for a conversion Safety Analysis Report (SAR), and 3) spent fuel analysis. The accident analysis was done with RELAP5 for reactivity-initiated and loss-of-flow accidents. Results were obtained for both the current high enriched uranium (HEU) fueled core and the proposed low enriched uranium (LEU) core. The SAR Chapter 4 was for the LEU equilibrium core and compared results with those for the HEU core. It is a draft as some information on fuel qualification is not yet available. The LEU spent fuel analysis enables calculations of activity and decay heat to easily be made for application to spent fuel casks. These efforts have all been recently documented.

1. Introduction

Planning is underway to convert the NIST research reactor (NBSR) from using high-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel. Analysis has been completed to determine the core neutronic parameters for an equilibrium LEU core [1] which will provide the users of the NBSR with the same cycle length as exists for the current HEU-fueled reactor although with some loss of neutrons at radial beam tube locations. The work this year has focused on using those results to carry out accident analysis [2] and to use those results (and others) to write a template for what would be Chapter 4, Reactor Description, of a Conversion Safety Analysis Report (SAR) [3]. In addition, an algorithm was developed to calculate the activity and decay heat of the spent LEU fuel [4]. Each of these activities is summarized in the sections below.

2. Accident Analysis

The objective of the recent accident analysis is to provide that portion of the safety analysis that requires the use of the systems analysis (thermal-hydraulic) code RELAP5. The code simulates the time dependent behavior after different initiating events. Ancillary objectives are to compare these results to those obtained for the current HEU-fueled reactor (using the same methodology) and assure that the results are acceptable in terms of safety margin. The study is for the equilibrium LEU core; transition cores will be treated in a separate study.

A detailed three-dimensional Monte Carlo model for both MCNPX and MCNP5 [1, 5] was used to calculate physics parameters for the HEU- and LEU-fueled cores for use in the accident analysis at startup (SU) and end-of-cycle (EOC); namely, power distributions, neutron kinetics parameters, and the reactivity worth of the shim arms. The model includes a plate-by-plate representation of each fuel element, the water gap at the axial mid-plane, beam tubes, shim arms, regulating rod, axial and radial reflectors, cold neutron sources, and other structures internal to the NBSR.

The RELAP5 model includes the primary piping from vessel inlet to outlet, primary pump and shut-down pump flow paths, heat exchanger, fuel element geometry and flow area, and flow channels for the six inner and twenty-four outer fuel elements. The initial operating parameters (flows, temperatures, power level and distribution, etc.) were assumed to be at their most limiting values or at the Limiting Safety System Setpoints (LSSSs). The NBSR reactor protection system logic was modeled and able to initiate a reactor trip upon reaching a setpoint and after the appropriate instrumentation response delay. Fuel temperature is calculated to assure that no fuel damage can take place. This is done by comparison with the blister temperature (an assumed value for the LEU fuel). In addition, the critical heat flux ratio (CHFR) and onset of flow instability ratio (OFIR) have been evaluated as supplementary parameters to examine integrity of fuel elements. The former is calculated using the Sudo-Kaminaga correlation [6] for CHF and the latter is calculated using the Saha-Zuber criteria [7] for OFI. The criteria for whether or not the resulting CHFR and OFIR are acceptable are based on a complementary statistical analysis [8].

Calculations have been performed for accidents involving

1) excessive positive reactivity insertions (startup withdrawal of rods and rapid withdrawal of large reactivity worth experiments), and

2) power-cooling mismatches (loss of electrical power for primary pumps, pump seizure, closure of throttling valve and loss of both shutdown coolant pumps).

An example of the results for one of the events considered is shown in Figure 1 for the HEU and LEU cores at SU and EOC. It shows the reactor power as a function of time after the ramp insertion of 0.005 Δk/k in 0.5 s. This amount of reactivity is the Technical Specification limit for the reactivity of any experiment.

The resulting clad temperature, CHFR, and OFIR all show that there is sufficient margin to fuel damage. Figure 2 shows CHFR during the event. The minimum value for all four cases is 2.19 which assures that there is no CHF with greater than 99.9% probability.

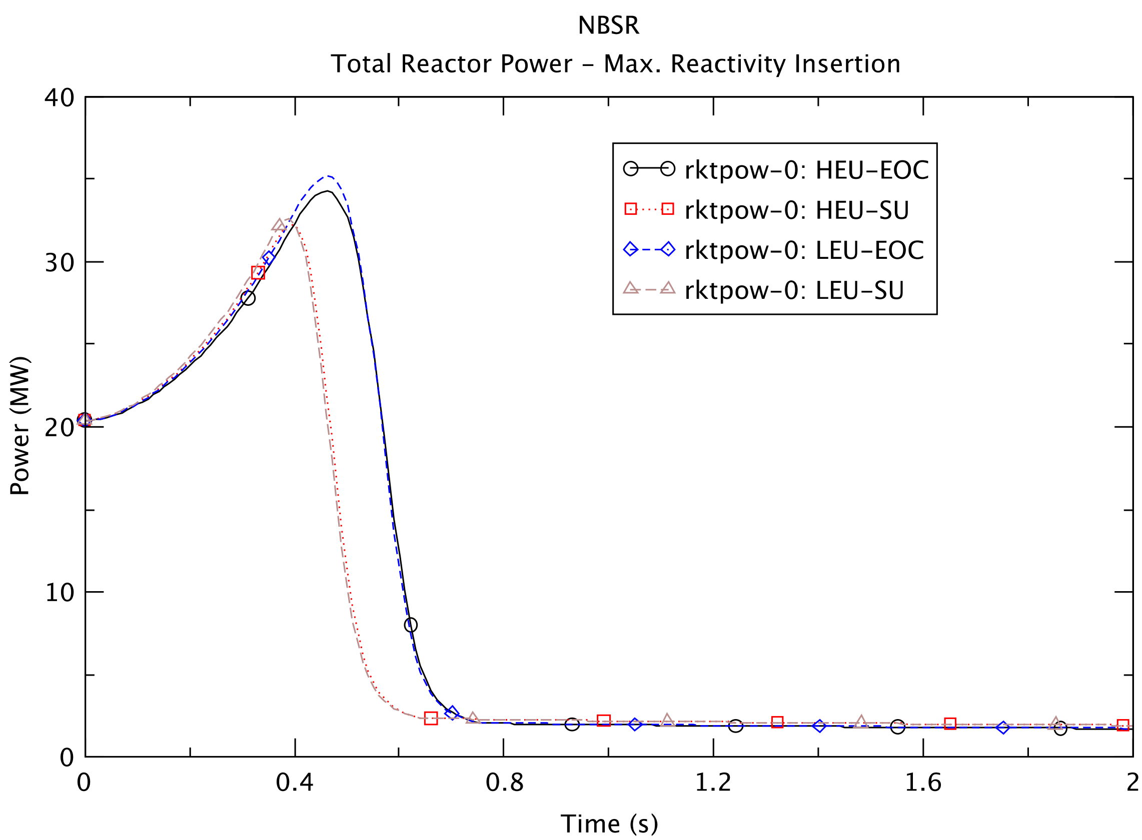

Figure 1 Reactor Power Response to Maximum-Reactivity Insertion Accident

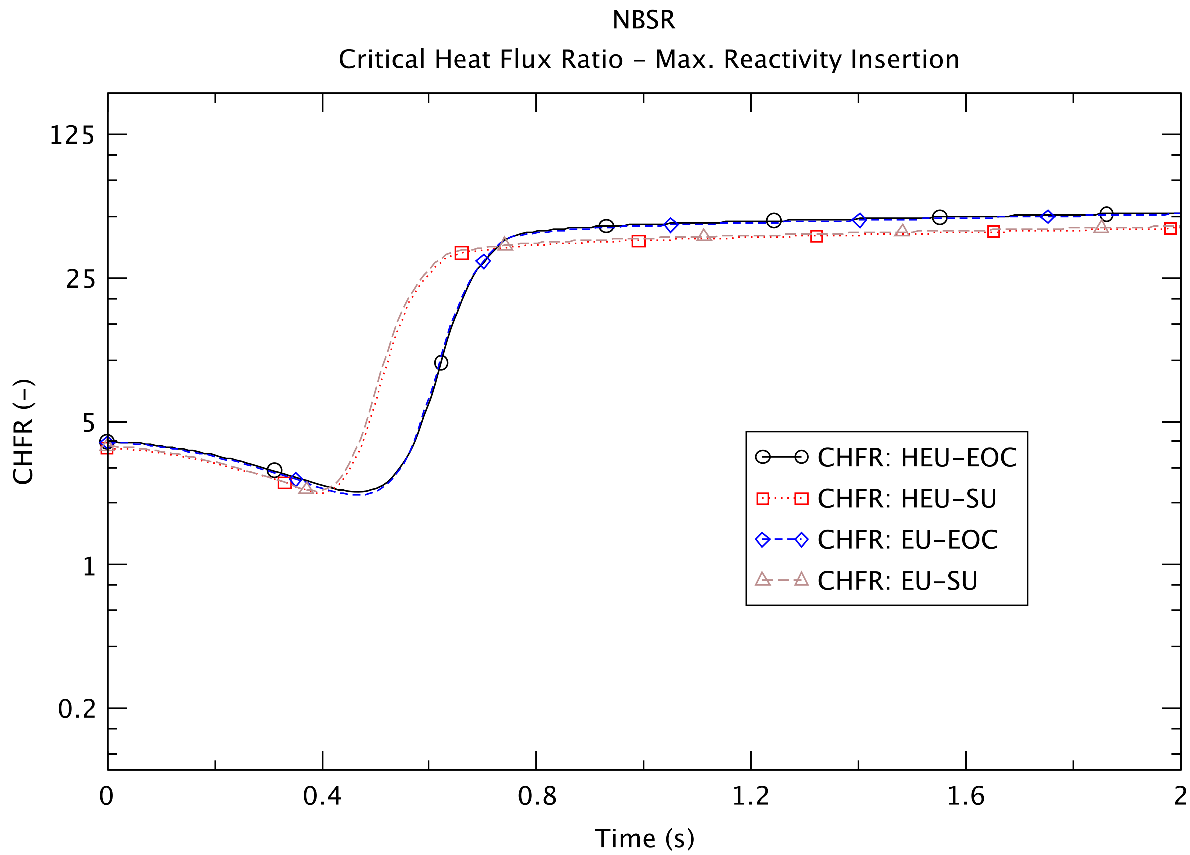

Figure 2 Critical Heat Flux Ratio During Maximum-Reactivity Insertion Accident

All events calculated at SU and EOC conditions show that the LEU and HEU cores yield similar results. Furthermore, they show that both cores have sufficient margin to safety limits.

3. Conversion Safety Analysis Report – Chapter 4

The conversion SAR must show that safety margins are not compromised by the LEU fuel and that the change in important safety parameters is acceptable. The latter means that the SAR must contain comparisons of results for both the HEU and LEU cores. It also should show what changes to the Technical Specifications will be necessitated by the conversion.

The current version of Chapter 4 of the SAR is a draft as a final version would refer to fuel qualification documents that do not yet exist. The draft includes a complete description of the reactor with the emphasis on what is being changed; namely, the fuel. It includes sections on nuclear design and thermal-hydraulic design. The former includes:

- Neutronic and Burn-up Model of the NBSR
    - NBSR Modeling with MCNPX
    - Burn-up Model of the NBSR
- Reactivity Calculations
    - Excess Reactivity and Shutdown Margin
    - Moderator Dump
    - Reactivity Worth of the Shim Safety Arms and Regulating Rod
    - Moderator Temperature Reactivity Coefficient
    - Void Reactivity Coefficient
    - Beam Tube Flooding
    - Light Water Ingress
- Power Distribution and Energy Spectra Calculations
    - Radial Power Distribution
    - Axial and Plate-wise Power Distributions
    - Energy Spectra
    - Fuel Misloading Accident
- Reactor Kinetics Parameters
    - Delayed Neutron Parameters
    - Photoneutron Contribution to the Delayed Neutrons
    - Prompt Neutron Lifetime

Examples of the results for power distributions are shown in Figures 3 and 4 for a fresh fuel element at SU. Figure 3 gives the relative plate-wise power for each of the 17 plates in a fuel element and Figure 4, the axial power for the entire fuel element. As can be seen, the HEU and LEU results are very close.

The section on thermal-hydraulic design includes:

- Design Basis
    - Flow Distribution in the Core
    - Power Distribution in the Core
- Determination of Limiting Conditions
    - Critical Heat Flux
    - Onset of Flow Instability Correlation
    - Statistical Analysis of Thermal-Hydraulic Parameters
- Shutdown Cooling
- Operation with Natural Convection

The steady state values for CHFR and OFIR (obtained using the correlations referred to in Section 2 above) are provided and show that there is a very large safety margin during normal operation. Results are similar for the HEU and LEU cores.

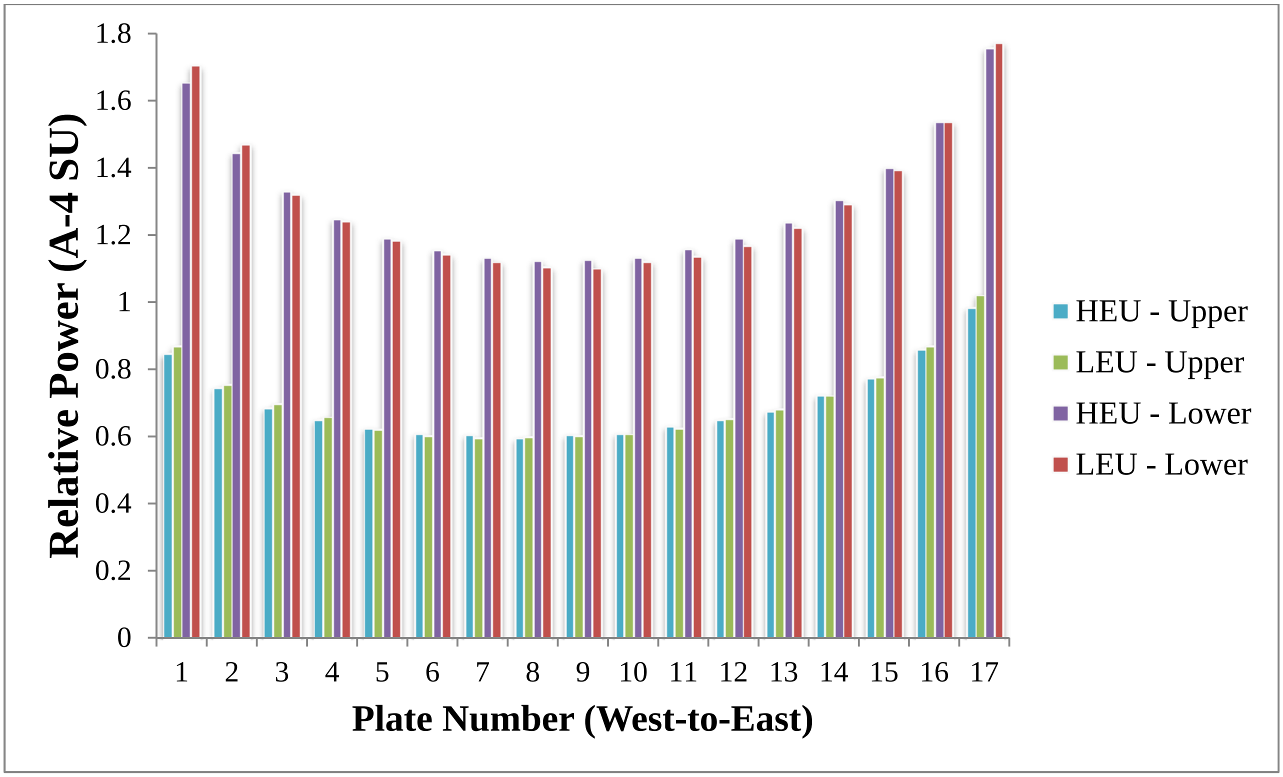

Figure 3 Plate-wise Power for Fresh Fuel Element at SU

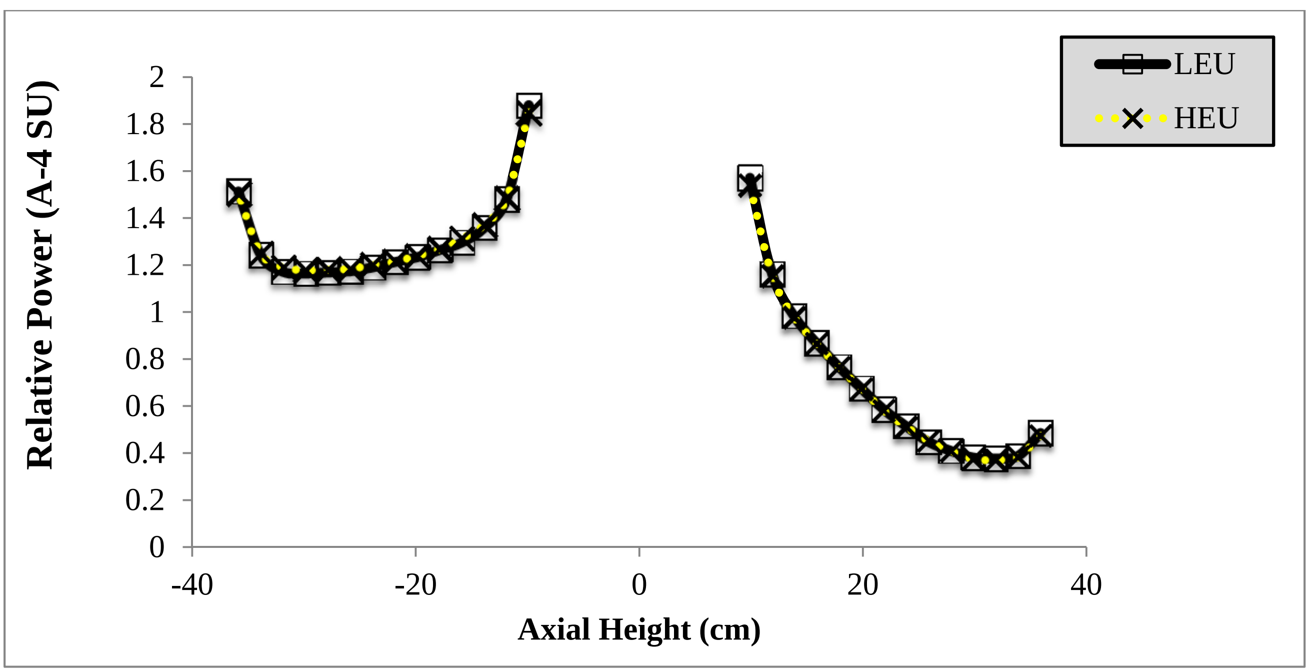

Figure 4 Axial Power Distribution for Fresh Fuel Element at SU

4. Spent Fuel Isotopics and Radiation Source Terms

In this study, spent fuel inventories were utilized from an LEU equilibrium core model of the NBSR [1]. The spent fuel inventories were decayed for three months (the earliest time at which they might be shipped) with MCNPX/CINDER'90 [9] and then utilized in an ORIGEN-S point model to solve the Bateman equations in a radioactive decay problem [10]. The inventories of four spent fuel elements (eight half elements) were calculated for a period of 10 years. This allows for calculation of activity and decay heat during this interval.

An example of the type of information available from this model is shown in Figure 5 which shows the activity (in kCi) for all fission products in a half-element and for selected fission products.

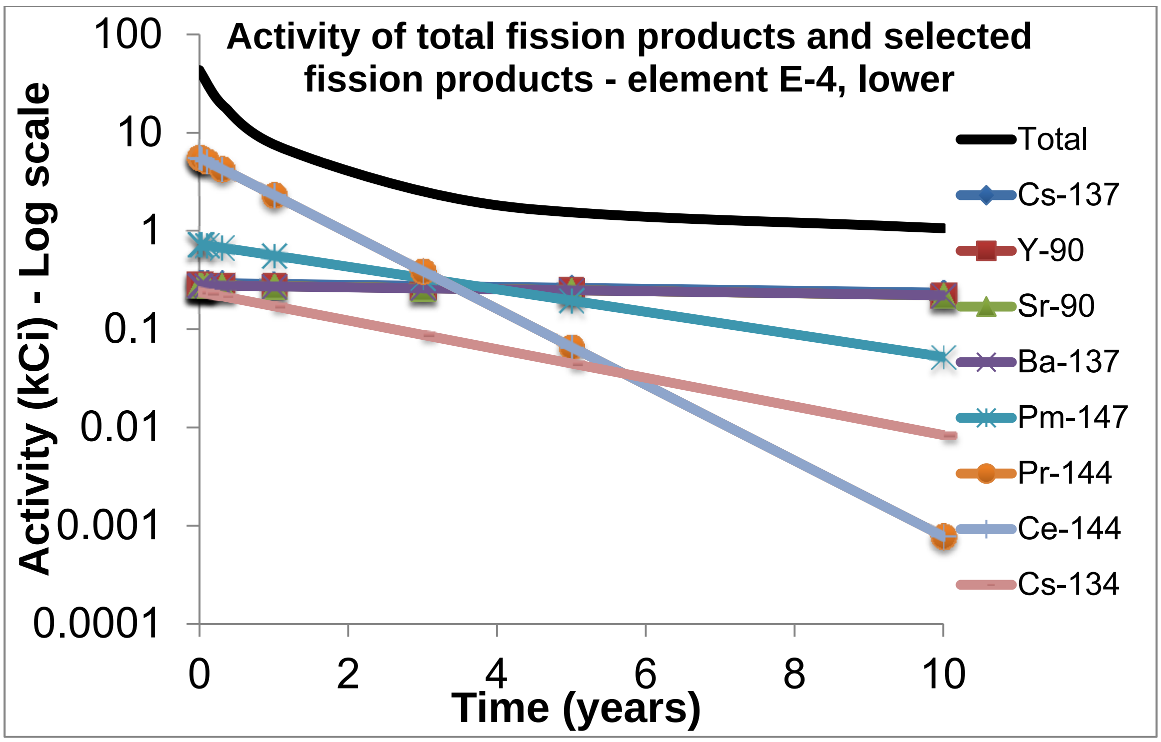

Figure 5 Radioactivity of Fission Products in Lower Half of LEU Fuel Element

5. References

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