

Applying New Methods to Research Reactor Analysis

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Abstract — Detailed reactor physics and safety analyses are being performed for the 20 MW D₂O-moderated research reactor at the National Institute of Standards and Technology (NIST). The analyses employ state-of-the-art calculational methods and will contribute to an update to the Final Safety Analysis Report (FSAR). Three-dimensional MCNP Monte Carlo neutron and photon transport calculations are performed to determine power and reactivity parameters, including feedback coefficients and control element worths. The core depletion and determination of the fuel compositions are performed with MONTEBURNS to model the reactor at the beginning, middle, and end-of-cycle.

The time-dependent analysis of the primary loop is determined with a RELAP5 transient analysis model that includes the pump, heat exchanger, fuel element geometry, and flow channels. A statistical analysis used to assure protection from critical heat flux (CHF) is performed using a Monte Carlo simulation of the uncertainties contributing to the CHF calculation. The power distributions used to determine the local fuel conditions and margin to CHF are determined with MCNP.

Evaluations have been performed for the following accidents: (1) the control rod withdrawal startup accident, (2) the maximum reactivity insertion accident, (3) loss-of-flow resulting from loss of electrical power, (4) loss-of-flow resulting from a primary pump seizure, (5) loss-of-flow resulting from inadvertent throttling of a flow control valve, (6) loss-of-flow resulting from failure of both shutdown cooling pumps and (7) misloading of a fuel element.

These analyses are significantly more rigorous than those performed previously. They have provided insights into reactor behavior and additional assurance that previous analyses were conservative and the reactor was being operated safely.

I. INTRODUCTION

The Final Safety Analysis Report (FSAR) for the NIST research reactor (NBSR) was documented in April 1966 [1] and supported the NBSR initial criticality in December 1967. The FSAR provided the required assurance that the consequences of reactor operation and postulated accidents did not result in undue risk to the public. The 1966 analysis was updated in a 1980 Addendum-1 [2] to support the increase in NBSR power level from 10 MW to 20 MW, which occurred in May of 1985. Further updates occurred in 1994 and 1998 [3, 4] that documented several major improvements to the NBSR physical plant and experimental facilities.

The analysis described in this paper provides an update to the previous analysis using state-of-the-art calculational methods. It provides additional detail for the accident scenarios and quantification of the conservatism in the original evaluations. The new analysis also takes into account recent changes in design of the cold source (although this does not make a significant change to any

safety parameters.) In this updated analysis, detailed three-dimensional MCNP Monte Carlo neutron and photon transport [5] calculations were performed to determine the behavior of the key safety parameters. The core depletion and determination of the fuel compositions were performed with MONTEBURNS [6]. This replaces analyses based on simple diffusion theory calculations of the core.

The time-dependent analysis of the primary loop is determined with the RELAP5 transient analysis code [7] for loss of flow accidents. A statistical analysis is used to assure protection from critical heat flux (CHF) for steady state and accident conditions based on the RELAP5 thermal-hydraulic conditions. The CHF ratio (CHFR) is calculated using a Monte Carlo simulation of the uncertainties contributing to the CHF calculation. This is more realistic than the previous FSAR analysis since it allows the variation of the individual uncertainty components to be random and independent, rather than making the extremely conservative assumption that all

components are at their maximum (most conservative) values.

For reactivity accidents the RELAP5 point kinetics model is used, although this is expected to yield similar results to the point kinetics analysis that had been done in the past.

In the following two sections a description of the new reactor physics and thermal-hydraulics analysis is presented. This includes both a brief description of the modeling and sample results of the analysis. More details can be found elsewhere [8].

II. REACTOR PHYSICS EVALUATION

II.A. MCNP Reactor Physics Model

The calculation of the reactor physics parameters is performed with the MCNP Monte Carlo code. The three-dimensional model used in the evaluation is a modified version of a model originally developed at NIST[9]. Figure 1 depicts the reactor cross section at the mid-plane, as modeled in MCNP. The boxes in the reactor core show the unfueled gap region of the fuel elements, consisting of the aluminum frames that are always filled with D_2O . (Above and below the gap region the fuel is modeled as parallel plates.)

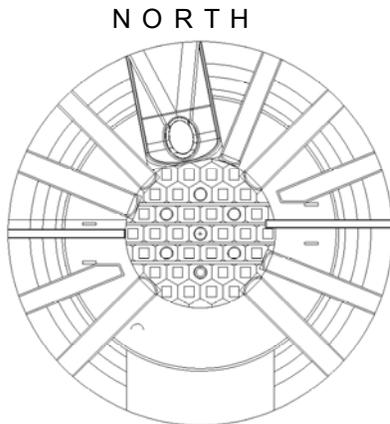


Fig. 1. Cross-Sectional View of the Reactor at the Mid-Plane.

Surrounding the boxes are hexagonal lines, which are computational divisions that do not represent any physical structures. Some of the hexagons are intersected by straight lines running east and west. These lines represent computational boundaries for the travel of the shim arms used for reactivity control, and these areas are filled with the shim arms and D_2O . The core region is surrounded by an outer D_2O region, followed by the reactor vessel region and then the biological shielding region. In the outer D_2O region, four rectangles represent the segments of shim

arms that intersect the mid-plane gap region for the particular position being modeled.

Within the reactor core there are seven circular regions. The southernmost region represents an aluminum tube containing the solid aluminum regulating rod that displaces D_2O as it is inserted. The other six circular regions represent the experimental thimbles, which are aluminum tubes filled with D_2O . There are also nine radial beam tubes and two pneumatic beam tubes in the gap region. There is one pneumatic beam tube that is located in the lower section of the core and two tangential beam tubes located below the core. The beam tubes and pneumatic tubes are all modeled as aluminum structures filled with a vacuum. North of the reactor core is the Cold Neutron Source (CNS), modeled as an aluminum structure filled with a combination of D_2O and liquid H_2 , as well as H_2 gas (which is modeled as a vacuum).

Because of the high burnup and short, 38-day fuel cycle, an accurate determination of the fuel isotopic inventory is required. Four core models with specific material inventories were developed for the analyses: the startup core (SU), the beginning-of-cycle equilibrium core (BOC), the middle-of-cycle equilibrium core (MOC), and the end-of-cycle equilibrium core (EOC). Fresh fuel elements contain 350 g of ^{235}U . During each cycle, the fuel element inventory is reduced by approximately 30 g of ^{235}U . At the end of each cycle, the NBSR rotation scheme requires that four fuel elements are removed, the remaining 26 are moved to different locations, and four fresh, unirradiated elements are inserted. Consequently, after the initial core load, there is a mix of fuel elements that have received different levels of irradiation and burnup.

II.B. Sample Analysis

Steady state power distributions are required to evaluate the heat flux and fuel temperatures at the limiting core locations. There are several distributions that are applicable. These are obtained from MCNP analysis for each of the different times during the fuel cycle that are modeled:

- Core radial power distribution. This is the fuel assembly relative power averaged axially for the 30 assemblies.
- Fuel element plate-wise power distribution. This is the fuel plate relative power for selected fuel assemblies.
- Fuel plate transverse power distribution. This is the power distribution along the lateral direction of the fuel plate (i.e., horizontal). It is obtained

by modeling each element with reflective boundary conditions, and dividing each plate into 17 equally spaced segments. The fission rate is calculated in each of the 17 segments in each of 17 plates.

- Fuel element axial power distribution. This is determined by dividing selected fuel elements into 16 equally spaced axial segments (8 in the upper section and 8 in the lower section).

The axial power distribution for one fuel element at EOC conditions is shown in Figure 2. It shows that the power distribution increases in the segments closest to the gap at the core midplane and at the top and bottom of the fuel element next to the reflector. The axial peak power occurs at an axial segment either immediately above or below the central unfueled gap.

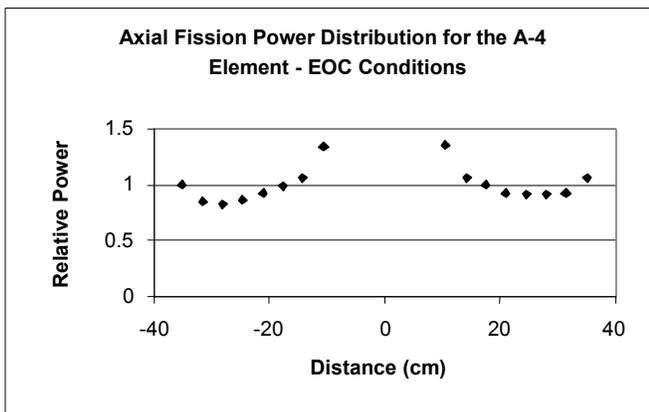


Fig. 2. Relative Axial Power Distribution for the Fuel Element in the A-4 Position in the Equilibrium Core at End-of-Cycle.

There are several reactivity parameters that are also obtained from MCNP analysis:

- Void reactivity worth. This is calculated to assure that the introduction of voids during an accident reduces reactivity. The calculations consider voiding of the water in the irradiation thimbles, in the coolant channels within the fuel elements along with the mid-reactor gap region, and voiding of the moderator region outside of the fuel assemblies. In each case the effect is negative. The void reactivity coefficients are not taken into account in the transient analysis thereby adding some conservatism (albeit small).
- Beam tube and cold neutron source (CNS) reactivity due to flooding. For the beam tubes,

the assumption is made that void is replaced with D_2O . This flooding results in a positive reactivity insertion. The reactivity worth of the CNS is determined by replacing the liquid hydrogen and hydrogen gas with D_2O . The calculations are carried out for various combinations of accident assumptions to show that the reactivity insertion is less than that assumed in the accident analysis.

- Reactivity of light water contamination. In these calculations, the amount of light water in the heavy water coolant was increased and the core k_{eff} was calculated. The results of the calculations demonstrate that light water contamination of the coolant results in a substantial negative reactivity insertion.
- Regulating rod worth. This reactivity worth is small and does not enter into any accident analysis.
- Shim arm worth. The reactivity worth of the shim arms as a function of position must be known at different times in the cycle in order to do the accident analysis; in particular both the startup accident and the reactivity insertion accident are limited by the effect of reactor trip.

The shim arm worth for the EOC core is shown in Figure 3. When the shim arms are fully inserted, the core is significantly shut down.

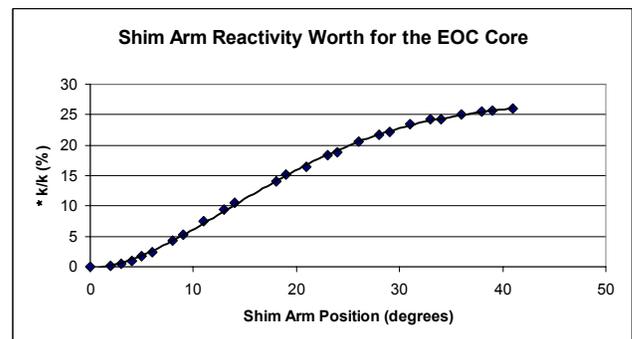


Fig. 3. Shim Arm Worth as a Function of Angular Position in the Equilibrium Core at End-of-Cycle.

III. THERMAL-HYDRAULIC ANALYSIS

III.A. RELAP5 Thermal-Hydraulic Model

The RELAP5/MOD3.2 model of the NBSR simulates the transport of heat and coolant in the primary system. The reactor vessel is divided into a number of interconnected

hydrodynamic volumes as shown in Figure 4. In addition, heat structures with internal heat generation are used to model the fuel plates. The inner six fuel elements are modeled as an inner group while the outer 24 fuel elements are modeled as an outer group. Each group is divided into three different channel types, each with a different heating rate and flow area. The three types of channels are the hot stripe, hot element, and average element. The hot stripe and the average element channel are similar in their composition of hydraulic volumes that constitute the flow path for the coolant in a fuel element. The hot element has two parallel flow paths in the upper and lower core. This arrangement is to simulate the effects of coolant mixing in the gap, the common flow area of a fuel element.

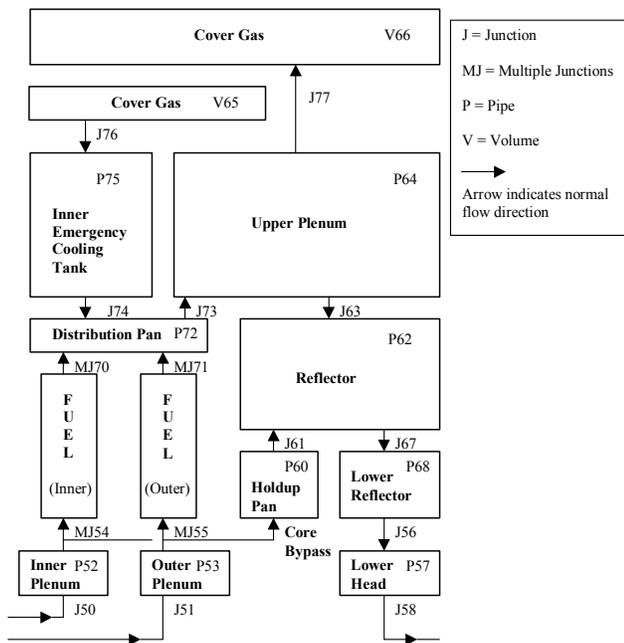


Fig. 4. Node Diagram of NBSR Reactor Vessel

The secondary cooling loop is modeled simply as a once through circuit. At one end a source supplies the cooling water to the primary heat exchangers. After the heat exchangers the secondary coolant (light water) flows to a sink.

III.B. Sample Analysis

The hypothetical accidents analyzed have been selected to represent a wide range of frequency of occurrence, as well as to span a range of probability of core damage and potential releases of radioactive isotopes to the environment. For all of these accident scenarios, the reactor is assumed to be operating with all critical parameters at the most unfavorable extreme value of their

normal range. This assures that the analysis for each accident scenario uses the worst-case initial conditions that might be anticipated, within the normal limits of operation.

The events analyzed with RELAP5 are:

- Loss of flow accidents. These include loss of offsite power, loss of electrical power feed to the primary pumps, seizure of one primary coolant pump, throttling of coolant flow to the inner/outer plenum, and loss of both shutdown coolant pumps.
- Maximum reactivity insertion accident. The accident assumes a large amount of reactivity is inserted due to changes in the configuration of experiments.
- Startup accident. This event assumes that contrary to operating procedures and all previous training and experience, the operator withdraws the shim arms without stopping. The accident is terminated by reactor trip. Typical results are shown in Figure 5 for EOC conditions. The resulting excursion energy above 20 MW is 8.09 MJ, which is within the acceptance criterion for the event.

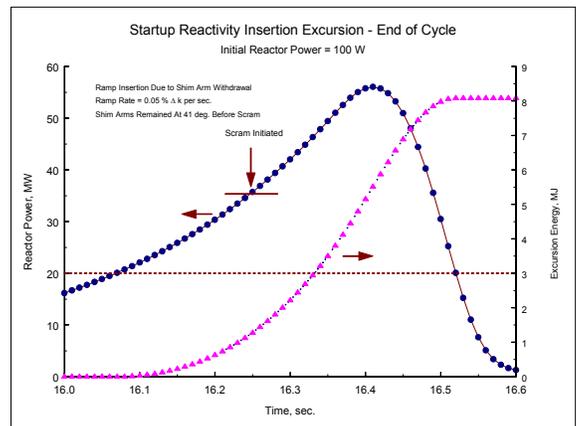


Fig. 5. Startup Reactivity Insertion Excursion – EOC

The CHF is obtained using results from the RELAP5 analysis for the loss of flow accidents and for steady state operation. A statistical analysis is performed to account for the effects of uncertainties in the CHF and to determine the probability of exceeding local limits. The analysis accounts for the uncertainty in each parameter used in the determination of the CHF, either by a random sampling of the parameter or by applying a conservative bias to the parameter. The probability

distributions were determined using a direct Monte Carlo simulation of the uncertainty propagation. The Bernath correlation [10] was used to determine the CHF. The parameter uncertainties were sampled from normal distributions having standard deviations based on estimates of the uncertainty in the individual parameters. For each hot channel variable, the probability distribution function was used to determine the limiting value such that there was a 95% or 99.9% probability of not exceeding this value. For example, the 95% (99.9%) limit (expressed as a ratio of the nominal value to the random (or limit) value) determined by the statistical analysis for full power operating conditions is 1.44 (1.91).

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