INL/CON-23-71104-Revision-0

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Pressurized Water Reactor Control Rod Ejection Analysis Using PARCS, RELAP5-3D, and BISON for High Burnup Fuel

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INTRODUCTION

In order to prevent cladding failure and the subsequent release of fission products into the coolant system, Pressurized Water Reactor (PWR) fuel is placed under a variety of regulatory limits, including limitations on uranium enrichment and fuel utilization. Currently, NRC regulations limit uranium enrichment levels to less than 5 wt% ^{235}U, which, in conjunction with individual reactor operating licenses, effectively limits rod average burnup levels to 62 GWd/MTU [1]. It would be economically beneficial for utility providers to be able to extend the life of the fuel for longer cycles. However, in order to meet the design goals listed by the US Nuclear Regulatory Commission (NRC) fuel standard review plan (SRP), it may be necessary to increase the burnup and thus enrichment limits.

This work targets the performance of high-burnup fuel (HBF) during loss of coolant accidents (LOCA), reactivity-initiated accidents (RIA), and anticipated operational occurrences (AOOs) for Southern Nuclear's Vogtle Electric Generating Plant (VEGP) through a series of safety assessments. A goal of this project is to create a multiphysics modeling approach using RELAP5-3D, PARCS, and BISON and use it in combination with experimental studies (Figure 1) to accurately model the progression of these events. In support of the project, this paper specifically seeks to identify the performance of a HBF pin during steady state conditions and during a control rod ejection (CRE) scenario using a loose coupling of PARCS, RELAP5-3D, and BISON.

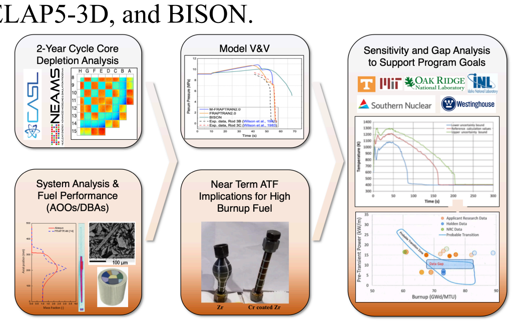

Fig. 1. Relationship of RELAP5-3D, PARCS, BISON, and experimental studies for the analysis of high burnup fuel during a PWR two-year cycle.

BACKGROUND

RIA Acceptance Criteria

A CRE, a type of RIA, is a type of Design Basis Accident (DBA). It is important to delineate that during an AOO, the fuel must be able to return to service as this occurrence is expected at some point in the reactor lifetime. A DBA, however, is not anticipated to occur during normal operation, and, as such, it is not required that the fuel be able to return to service following the event. Thus, under DBA conditions fuel failures can result.

For this work, three important aspects are evaluated under RIA conditions: fuel cladding failure, core coolability, and fission product inventory. The fuel cladding failure criteria is dependent on the radial average enthalpy of the fuel and the stresses which develop as a result of pellet-cladding mechanical interaction (PCMI). Fuel cladding failure is presumed if the cladding hoop stress exceeds a temperature dependant failure stress, if the local heat flux exceeds the thermal design limits, or if the change in radial average fuel enthalpy is greater than the corrosion-dependent limit described in section 4.2 of the SRP, NUREG-800. When considering core coolability, the peak radial average fuel enthalpy must remain below 230 cal/g, the peak fuel temperature must remain below fuel melting conditions, and there must be no loss of coolable geometry due to fuel pellet and cladding fragmentation or fuel rod ballooning [2].

IFBA Fuel

A consequence of increasing fuel enrichment levels is excess reactivity in fresh fuel. Typically, excess reactivity is accommodated for using burnable poisons such as boron and gadolinium. The core design used in this analysis utilizes Westinghouse's Integral Fuel Burnable Absorber (IFBA) fuel design [3]. IFBA fuel uses a thin layer of boron on the fuel pellet to absorb neutrons and suppress reactivity. When ^{10}B absorbs a neutron, lithium and helium are generated and released into the gap and plenum region of the fuel. This reaction contributes to increased plenum pressures. The initial plenum pressure for IFBA fuel pins is estimated to be 0.7 MPa, while typical fill pressures are 1.9 MPa. The initial pressure is decreased in order to offset the production and release of lithium and helium to the plenum and gap regions. This allows the fuel rod to maintain similar cladding creep down behavior due to the pressure differential across the cladding.

METHODOLOGY

PARCS

PARCS is a deterministic 3D spatial kinetics core simulator code that is designed for predicting the responses of a reactor for steady-state and transient conditions. During RIAs, the core flux distribution and peaking factors can vary significantly and PARCS is suited well for analyzing such conditions. It also has capabilities to construct individual pin power information, which is valuable when performing evaluations for safety analysis purposes. To generate the few-group cross sections needed for PARCS to conduct deterministic transport and fuel depletion calculations, the Polaris lattice physics code [4] was used.

RELAP5-3D

The Reactor Excursion and Leak Analysis Program (RELAP5-3D) is a nuclear systems analysis code developed at INL that is used to model the thermal-hydraulics (TH) of the postulated reactor transient scenarios for LWR systems and descriptions of the code capabilities may be found in the literature [5]. For this work, RELAP5-3D was used to model the TH system response of the core during a hot full power CRE after three 2-year cycles, using the linear heat rate (LHR) and axial peaking factor (APF) behaviors determined in PARCS. The transient TH behavior, in addition to the LHR and APF behavior, of a selected high burnup fuel pin was then provided to BISON for fuel performance analysis. For this analysis, coolant pressure and temperature profiles were the transient TH boundary conditions used to perform BISON calculations.

BISON

BISON is a finite element code for nuclear fuel performance and safety [6]. The BISON model employed in this work is a 2D azimuthally symmetric (R-Z), smeared-mesh fuel pin model developed by the NEAMS program, representative of the VEGP AP1000 PWR fuel design[7]. The fuel model developed in BISON consists of 3.67 m of solid pellet enclosed in a zirconium cladding with a specified gap. A rod plenum region extends beyond the fuel height in order to accommodate fuel elongation and the release of fission gas from the fuel. The fuel enrichment varies between the pins across the core, but the pin discussed in this work is enriched to 6.2 wt% ^{235}U. The mesh is partitioned into 500 axial elements and 15 radial cladding elements. Of the 15 radial elements, 12 are fuel and 3 are cladding. This simulations utilized BISON's built-in mesh generation and material property libraries, specifically those for light water reactor fuel performance analysis. For the fuel elements, thermal conductivity, elasticity, plastic deformation, relocation, fission gas production and release, thermal expansion, and creep models were utilized. For the cladding elements, thermal conductivity, elasticity, plastic deformation, creep, thermal expansion and irradiation growth models were utilized.

RESULTS

Steady-State Analysis

Fuel behavior during end of cycle scenarios is heavily dependent on the full operating history of the pin. It was determined during the PARCS analysis that the fuel pin that experiences the highest burnup in the core is pin A11-1-1. Assembly A11 is circled in Fig. 2, which shows the starting and ending average burnup for the quadrant of the core considered in this analysis. This assembly is a thrice-burned assembly, starting towards the center of the core in the first two cycles and moving to the periphery for the last fuel cycle.

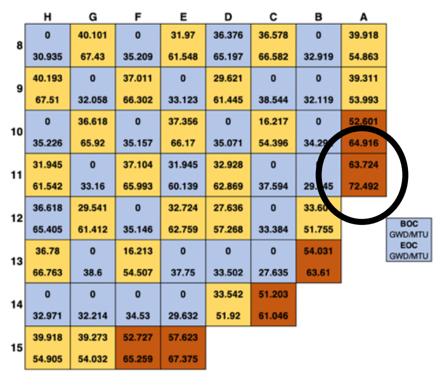

Fig. 2. Reactor core quadrant with assembly average burnup levels at the beginning and end of cycle. Blue indicates fresh fuel, yellow indicates once burnt fuel, and orange indicates twice burnt fuel at beginning of cycle.

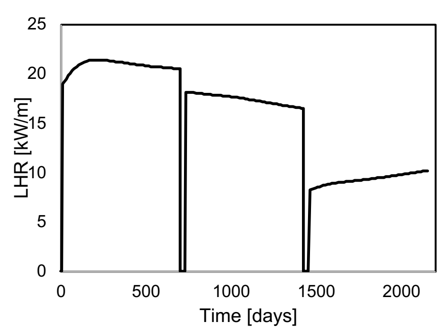

Fig. 3. Steady state LHR history for pin A11-1-1.

The LHR history for the addressed pin is shown in Fig. 3. For the fuel performance analysis, PARCS provided the LHR and APF histories, while coolant pressure and temperature histories were developed using operational constants.

When considering fuel performance behavior, the maximum fuel and cladding temperatures, plenum pressure and hoop strain were evaluated, shown in figures 4, 5, and 6 respectively. The primary driver for fuel thermal response is pin power, which is determined by the LHR and APF. As such, the fuel thermal response follows a similar pattern to the LHR with modifications due to the changing fuel and gap conductivity along with fission product release to the gap/plenum.

Heat transfer properties through the gap region and the outer region of the fuel degrade over the life of the fuel due to build up of fission products and morphological changes to the fual as burnup increases. This degradation causes the fuel centerline temperature to remain in a similar range as the first fuel cycle during the second fuel cycle, even though the LHR decreases. When the fuel is shuffled to the core periphery in the third cycle, the fuel temperature decreases by approximately 200 K. Plenum pressure increases with burnup due to solid fission product accumulation and fuel thermal expansion. During the first 20 GWd/MTU of burnup, the layer of boron on the fuel pellet is consumed, releasing helium and lithium into the gap region, causing a rapid initial increase in plenum pressure. The increase then slows after all boron has been depleted. The cladding hoop strain initially decreases due to cladding creepdown under the coolant system pressure. Compressive hoop strain is observed in the first fuel shuffle sequence as a result of the decrease in temperature. During the second and third fuel cycles, the gap region is closed, and an expanding hoop strain is observed. As the fuel thermally expands in the radial direction, strain increases. Similar to the first fuel shuffle sequence, a decrease in cladding hoop strain is observed during the second fuel shuffle sequence due to the decrease in fuel temperature.

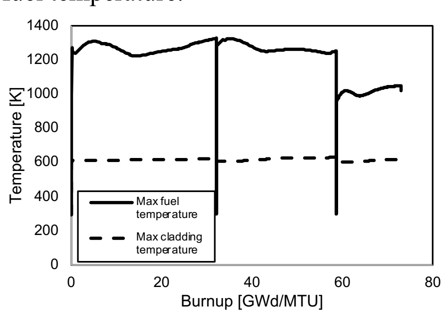

Fig. 4. Steady state maximum fuel and cladding temperatures for pin A11-1-1.

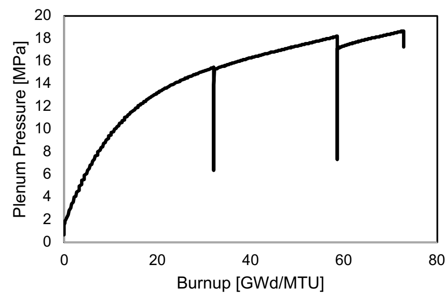

Fig. 5. Steady state plenum pressure for pin A11-1-1.

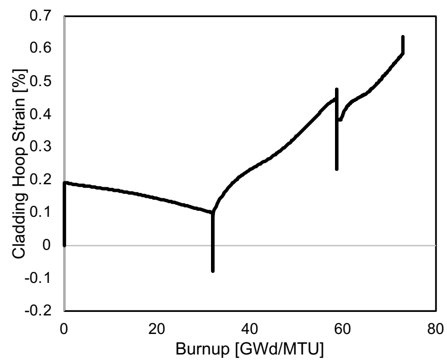

Fig. 6. Steady state cladding hoop strain for pin A11-1-1.

Transient Analysis

For the transient portion of the analysis, PARCS provided the LHR (figure 7) and APF boundary conditions. Because the analyzed pin was in the periphery region of the core, this specific pin does not experience an aggressive LHR increase during a CRE, as the ejected rod is closer to the center of the core. While coolant temperature and pressure were determined using operational constants in the steady-state portion of this analysis, RELAP5-3D was used to develop the TH boundary conditions for the transient analysis. Figure 8 shows the maximum fuel and cladding temperatures throughout the transient. Similar to the steady-state case, there is a corresponding increase and decrease in maximum temperatures to the increase and decrease in LHR. The maximum fuel and cladding temperatures throughout the transient were 1,138 K and 642 K, respectively. The fuel temperature is below the fuel melting point, 2,700 K, by a significant margin. The maximum cladding temperature is limited to 1,478 K by coolability constraints (as indicated by the SRP), which is not approached in this scenario. There is a small increase in plenum pressure during the transient. Because the plenum pressure is higher than the system pressure, 15.5 MPa, the radial average enthalpy is limited to 150 cal/g, and the maximum radial average enthalpy observed was 50.4 cal/g.

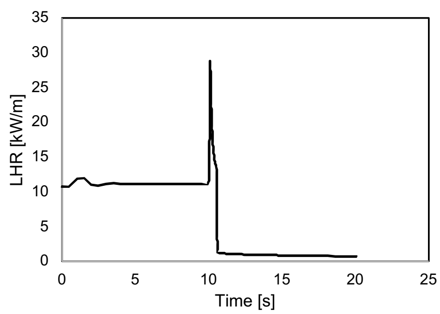

Fig. 7. A11-1-1 CRE LHR profile.

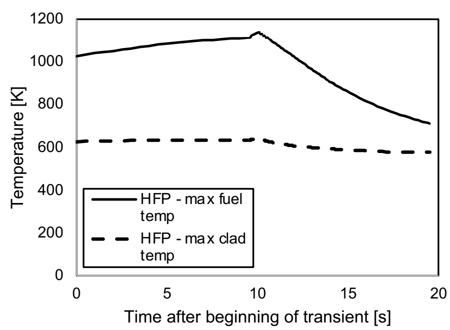

Fig. 8. A11-1-1 maximum cladding and fuel temperature during CRE transient.

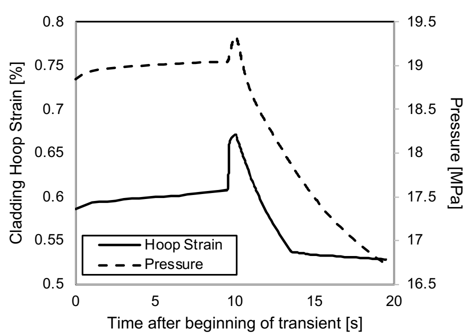

Fig. 9. A11-1-1 plenum pressure and cladding hoop strain during CRE.

Cladding deformation can be evaluated using the cladding hoop strain, where high cladding hoop strain indicates thermal expansion in the fuel region. There is an increase of .09% hoop strain during the transient, which is less than observed during reactor startup in the steady-state analysis.

When considering the fuel failure criteria outlined in the SRP, the fuel cladding is not expected to fail during this scenario. Cladding failure due to PCMI is dependent on oxide thickness and fuel enthalpy rise. In the case of the high oxide thickness, the fuel can experience a rise in enthalpy of 60 cal/g before failure. The maximum radial average enthalpy was 50.4 cal/g, during which a rise of only 6.2 cal/g was experienced. The 50.4 cal/g maximum radial average enthalpy is also below the 230 cal/g coolability limit. Fuel melt is not experienced, and low hoop strains indicate there was not ballooning that would limit coolability.

CONCLUSIONS

The primary goal of this work was to analyze high burnup fuel behavior during a CRE scenario. The results indicate fuel failure is not expected to occur under the conditions presented. Future work in this area could include an analysis of a fuel pin towards the center of the core, which would experience a more aggressive power pulse as a result of a CRE.

ACKNOWLEDGEMENT

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