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Impact of fuel thermal conductivity degradation on Doppler feedback during rod ejection accident

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HIGHLIGHTS

- Burnup dependent fuel thermal conductivity model was implemented in CTF/TORT-TD.
- Impact of fuel thermal conductivity degradation on Doppler feedback during REA was investigated.
- Modeling fuel thermal conductivity degradation with burnup is important for LWR safety analyses.

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ABSTRACT

This paper discusses the importance of the fuel thermal conductivity degradation modeling for accurate predictions of the Doppler feedback during reactivity insertion transients. The impact of the fuel thermal conductivity degradation model, recently implemented in the coupled sub-channel thermal-hydraulic/time-dependent neutron transport code system CTF/TORT-TD, on Doppler feedback predictions during a control rod ejection accident was investigated. The rod ejection was simulated for a 4×4 pressurized water reactor pin array, extracted from the Purdue University MOX (mixed oxide) benchmark, starting at both hot zero power and hot full power conditions with the control rod being half-inserted before the ejection. The two scenarios were simulated with CTF/TORT-TD and the effect of the fuel thermal conductivity degradation on the Doppler feedback was analyzed. The results were compared with existing reference calculations performed with the coupled sub-channel thermal-hydraulic/time-dependent neutron transport/fuel performance code system CTF/TORT-TD/FRAPCON-FRAPTRAN.

The power pulse, the time evolution of average fuel temperature, and the peak enthalpy rise during the transient were examined. It was confirmed that the impact of the fuel thermal conductivity degradation is more significant when the control rod is ejected at hot full power conditions. If the fuel conductivity degradation was not taken into account, less conservative CTF/TORT-TD predictions for the transient power response were obtained. For the selected 4×4 pin array, the coupled code calculated 13 MW higher power pulse when modeling degradation effects on fuel conductivity. The difference in the power response is due to the less negative prompt fuel temperature (Doppler) coefficient at elevated temperatures. Lower thermal conductivity will lead to higher fuel pellet temperatures, and subsequently to a less negative Doppler coefficient, which will result in a stronger power pulse. The maximum fuel enthalpy rise during the hot full power rod ejection accident was found to be 60 cal/g (251,208 J/kg).

1. Introduction

A thermal conductivity degradation (TCD) model was implemented in the Reactor Dynamics and Fuel Modeling Group (RDFMG) version of the sub-channel thermal-hydraulics code COBRA-TF (CTF) (Salko and Avramova, 2013). The new capability was also incorporated in the coupled sub-channel thermal-hydraulic/time-dependent neutron transport code system CTF/TORT-TD (Magedanz et al., 2015). The model takes into account the fuel thermal conductivity degradation with burnup and its dependence on the content of Gadolinium burnable poison for both UO_2 (uranium dioxide) and MOX (mixed oxide) nuclear fuels. The modified Nuclear Fuel Industries (NFI) (Lusher and Geelhood,

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2010) model for UO_2 fuel rods and the Duriez/Modified NFI (Lusher and Geelhood, 2010) model for MOX fuel rods were implemented in CTF.

A rod ejection accident (REA) was simulated starting at both hot zero power (HZP) and hot full power (HFP) conditions. A 4×4 pressurized water reactor (PWR) pin array from the Purdue MOX benchmark (Kozlowski and Downar, 2007) was used in the simulations assuming that the control rod was half-inserted before the ejection. Both scenarios were analyzed with and without fuel TCD modeling in CTF/TORT-TD. The obtained results were compared to already available CTF/TORT-TD/FRAPCON-FRAPTRAN (Magedanz et al., 2015) predictions. In this work, the CTF/TORT-TD/FRAPCON-FRAPTRAN coupled sub-channel thermal-hydraulic/ time-dependent neutron transport/fuel performance calculations were considered as reference solutions: TORT-TD provided a time-dependent neutron transport solution on a pin-by-pin homogenized level; FRAPTRAN accounted for the thermo-mechanical changes within the fuel rod (pellet-gap-cladding) during the transient; the steady state conditions were obtained with FRAPCON; CTF solved for time-dependent coolant flow conditions. The power pulse, the average fuel temperature, and the peak enthalpy rise during the transient were investigated to evaluate the impact of the fuel TCD on the Doppler feedback.

First, the HZP REA was simulated and results were compared to the same cases calculated with CTF/TORT-TD/FRAPTRAN. Sensitivity studies were performed to investigate the impact of the in-pellet radial power distribution (RPD) and the gap conductance models on: (1) the magnitude and the timing of the power pulse; and (2) the average fuel temperature time history during the transient.

Next, the REA calculations were started at HFP conditions. The power pulse, the increase in the average fuel temperature, and the enthalpy rise were obtained using the CTF dynamic gap conductance model and FRAPTRAN-based in-pellet RPDs. To investigate the Doppler feedback dependence on the fuel TCD, the core reactivity and the power response obtained with CTF/TORT-TD with fuel TCD were compared to the same cases with no fuel TCD effects being modeled.

2. Background

The rod ejection accident is accepted as a design basis reactivity initiated accident for PWRs U.S. Atomic Energy Commission (1975); Diamond et al. (2002); USRNC (2007); NEA State-of-the-art Report (2010). It might occur due to mechanical failure of the control rod drive mechanism or the pressure housing unit. The control rod and drive tube are ejected due to pressure difference between the reactor coolant system and the containment. The positive reactivity, promptly inserted in the fuel, will cause a rapid (milliseconds) core power increase which may result in a fuel rod damage due to fuel pellet overheating and thermal expansion. Once the prompt criticality threshold is exceeded, a super prompt critical state is reached and the fuel experiences an extensive heat-up, which could result in a pellet-cladding mechanical interaction and possible cladding failure by the increased local stress and strain.

During such transients, the negative fuel temperature (Doppler) reactivity coefficient will inherently and immediately reduce the reactor core power. In a typical low-enriched light water moderated reactor, Doppler coefficient is always negative: increasing fuel temperature makes the nuclei vibrate more rapidly within their lattice structure that effectively broadens the energy range of neutrons that might be absorbed in the fuel resonances; the resonance broadening will increase the total number of neutrons absorbed by the large resonances of fuel heavy uranium material such as U-238 while also reducing the resonance escape probability. This will reduce the neutron multiplication factor as illustrated by the six-factor formula definition (Lamarsh and Baratta, 2001). Such mechanism will introduce a negative reactivity into the system, which will eventually compensate for the initial positive reactivity insertion by the control rod ejection. The Doppler feedback effect is prompt and much more important than the delayed moderator feedback effect. This is due to the fact that the fuel temperature quickly increases following the reactor power increase. The moderator feedback, on another hand, is delayed three to five seconds depending on the time constant associated with the heat transfer from the fuel to the coolant, and therefore cannot immediately contribute to overcome the power increase. However, the fuel (Doppler) feedback effect injects negative reactivity into system immediately. In other words, the generated energy by the power rise is stored quickly in the fuel and then released to the rest of the system. The time needed for the heat deposited in the fuel to be transfer across the pellet to the coolant will depend on how effective is the heat conduction, which is primary determined by the fuel thermal conductivity.

Models for degradation of the fuel thermal conductivity with burnup already exist in the fuel performance codes such as FRAPCON and FRAPTRAN-3.4 Geelhood et al. (2010a,b); whereas, the most of the thermal-hydraulics codes continue to use simplified fuel rod models often along with the 1979 MATPRO-11 material properties of non-irradiated UO_2. Modeling of the fuel TCD is of high importance for an accurate prediction of the Doppler feedback and thus for the reactor safety evaluations.

The modified Nuclear Fuel Industries (NFI) model for UO_2 fuel rods, and the Duriez/Modified NFI model for MOX fuel rods had been previously implemented in the standalone CTF; the code predictions for fuel centerline temperature were validated with the Halden experimental data and were benchmarked against FRAPCON-3.4 predictions Yilmaz (2014). It was found that the new burnup dependent fuel thermal conductivity model significantly improves the CTF predictions: the error band for the ratio of predicted versus measured fuel centerline temperatures was reduced from more than 20% to less than 5% (Fig. 1).

The modified NFI model for UO_2 fuel rods and the Duriez/Modified NFI model for MOX fuel rods were also added to the CTF/TORT-TD coupled code system, and the predictions for the fuel centerline temperature, the fuel surface temperature, and the average fuel temperature were compared to reference results obtained with the coupled code system CTF/TORT-TD/FRAPCON-FRAPTRAN for a 4×4 PWR pin array at HFP steady state conditions. Fig. 2 illustrates the consistency between the CTF/TORT-TD and the reference CTF/TORT-TD/FRAPTRAN calculations, demonstrating that CTF with the new fuel thermal conductivity model can predict the fuel rod temperature distribution as accurately as fuel performance codes; and therefore, increasing the confidence that CTF/ TORT-TD with the thermal conductivity degradation model can be utilized for Doppler feedback simulations instead of the more computationally expensive CTF/TORT-TD/FRAPTRAN.

3. Methodology

3.1. Multi-channel fuel configuration

A 4×4 PWR fuel bundle configuration from the Purdue MOX benchmark (Kozlowski and Downar, 2007), as shown in Fig. 3 (Magedanz et al., 2015), was used to assess the performance of the new CTF thermal conductivity model for multi-rod multi-channel configurations. The 4×4 array consisted of fifteen PWR pins and one control rod guide tube. There were four types of fuel rods within the array: low burnup MOX, high burnup MOX, low burnup UO_2 and high burnup UO_2.