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BWR fuel rod behavior evaluation for preconditioning power ramps with FEMAXI-V

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ABSTRACT

The licensing authorities around the world usually set a limit value to the operation LHGR, as a function of burnup. Such limit provides a bound state to a steady state operation, but also prevents against some thermal and mechanical phenomena that could occur during overpowered transients. In particular, in some countries, the PCI limit is set based on experimental ramp tests and directly related to the LHGR limit value. Thus, to avoid violating the PCI limit, fuel conditioning procedures are still required for both barrier and non-barrier fuel. Simulation of the power ramp procedures to be performed by the reactor operator during startup or power increase maneuvers is advisable as a preventive measure of possible overpower consequences on the fuel thermomechanical behavior.

The thermomechanical behavior of BWR fuel rod is analyzed for fuel preconditioning procedures. Five different preconditioning computations were performed with the FEMAXI-V code, each with three different ascending linear power rate ramps. The starting point of the ramps was taken from data of the Unit 1 from the Laguna Verde Nuclear Power Plant, located in MEXICO. The top limit of the ramps was the threshold linear power at which failure by PCI could occur, as a function of burnup.

1. Introduction

Because of safety and economic reasons, setting adequate thermal and mechanical limits for the operation of a nuclear power reactor depends on several aspects, as reactor type, fuel rod composition, power generated, etc. Particularly, power increase during reactor startup, or because higher demand of electrical power to the network, makes mandatory to consider the changes in the thermomechanical properties of the specific fuel rod type used in a fuel assembly. For example, in a BWR, the density change of the coolant and moderator along a fuel pin causes different thermomechanical stresses and oxidation levels of the fuel rod cladding at the different axial sections of the fuel pin.

Although the number of failed rods is still low, in comparison to the number of fuel elements in the core of the operating power nuclear reactors, the recent advantages in competitiveness of nuclear energy can be challenged by public opinion, and thus forcing the regulatory entities to restrict the use of the new core management strategies, particularly on power peaking and linear heat generation rate (LHGR) operation limits.

An increasing emphasis in economic revenue has leaded the utilities to apply for or perform power up-rates, pursue longer operating cycles, and introduce innovative fuel reload patterns. These current more aggressive operation strategies have as result improved plant capacity factors, leading the nuclear industry to reach its lowest electricity production costs in many countries. In the last few years, however, the BWR fuel failure rate has presented a new and noticeable increase (Yang et al., 2004). The cause is considered to be a combination of very diverse areas, as water chemistry, new cladding materials and manufacturing procedures, and higher fuel duty.

During normal steady state operation of a nuclear power reactor, the gap and fuel thermal conductivities are the main physical properties dominating the thermal behavior of a fuel rod. On the other hand, during transient events, the heat capacity of the fuel is the ruling physical property of the thermal behavior. In both cases, if a fast power increase occurs, thermal expansion of the fuel pellet could lead to pellet–cladding interaction, which is a primary type of defect that could lead to further clad degradation, and eventually cause clad failure. If the power ramp rate to which the fuel rod is subjected is appropriately limited, the dimensional changes of the fuel pellet and cladding may be moderate, and thus creep, and relaxation can alleviate the consequences of PCI mechanism. Appropriate slopes for the power ramps can be outlined from the results of thermomechanical fuel behavior computer codes.

Fuel conditioning is the physical mechanism that includes all the local thermomechanical phenomena that help limiting the consequences of power transients in fuel elements. Fuel densification and stress relaxation are examples of the physical phenomena occurring during fuel conditioning that reduces the contact pressure between the fuel pellet and cladding and reopens the gap. Fuel

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conditioning may take from hours to days. Once equilibrium between cladding creep and pellet swelling is reached, a new steady state condition at a higher power level is established. Fuel de-conditioning, contrary to fuel conditioning, is the phenomena that aggravate PCI, such as fuel swelling, by increasing the contact pressure and reducing the gap size.

The term conditioning power level is defined as the rod power level at a typical reference stress, when cladding creep and pellet swelling equilibrate each other (Ito et al., 1983). This conditioning power level is also known as the conditioning LHGR, which is the limit where neither conditioning nor de-conditioning occurs (NEA, 2003). That is, contact pressure between pellet and cladding is moderate and constant. If fuel conditioning occurs, then the rod power level needs to increase to reach the conditioning LHGR. On the contrary, if fuel de-conditioning occurs, the rod power level needs to decrease to reach the conditioning LHGR. Once a new steady state is reached, the conditioning LHGR asymptotically approaches the current value of the fuel rod LHGR.

One measure normally taken in a nuclear power reactor operation to avoid the failure mechanism due to PCI is to establish a procedure to limit the number and types of sudden power increases that could reach the levels at which clad failure by PCI occurs. Many countries still require using such procedures. This is so because it is necessary to moderate the consequences of the fuel conditioning and de-conditioning phenomena described above. The operational procedures used to reduce the probability of such type of clad failure are known as fuel preconditioning operations. The preconditioning is a controlled and constant power increase that follows a previously set ascending ramp. This process is considered at nodal level, and not the average power of the whole fuel rod.

The initial point of the ramp is a reference power level, and the ending point is the nominal power at which reactor operation is desired. Preconditioning rules are normally applied during reactor startup or after a control rod blade pattern change. By following an appropriate preconditioning power ramp the possibility of fuel damage is greatly reduced, and it also helps the fuel to better assimilate further and faster power changes, below the preconditioned envelope. However, even at this controlled conditions, it is necessary to perform thermomechanical analysis of the fuel rods to ensure that clad failure by PCI will not occur during the preconditioning action, or to determine the linear heat generation rate value at which the failure could occur.

In this paper, the thermomechanical behavior of BWR fuel rod is analyzed for fuel preconditioning procedures. Five different preconditioning computations were performed with the FEMAXI-V code (Suzuki, 2000; Suzuki and Saitou, 2001), each with three different ascending linear power rate ramps. The starting point of the ramps was taken from data of the Unit 1 from the Laguna Verde Nuclear Power Plant. The top limit of the ramps was the threshold linear power at which failure by PCI could occur, as a function of burnup.

2. Fuel rods description

The fuel rod has natural uranium at both top and bottom extremes. While in the middle part, the fuel rod has two regions with ^{235}U enrichment of 4.90 $^{w}/_{0}$ and 4.40 $^{w}/_{0}$, respectively. The total active length of fuel rod was 381.0 cm. Table 1 presents the design dimensions for fuel rod. Also, shown the test conditions typical of BWRs. Fig. 1 shows the axial distribution of the ^{235}U enrichment for fuel rod.

The FEMAXI-V geometry model used consisted of 10 axial nodes, 10 radial segments in the fuel pellet, one for the gap, and two segments for the cladding. Although the maximum number of axial nodes allowed by FEMAXI-V is 12, only 10 were used because this is the same number of axial nodes used in the RODBURN (Uchida and Saito, 1993; RODBURN, 1999) code for the computation of the power distribution. The top and bottom nodes represented the natural uranium areas of the actual fuel rods. The middle nodes were all assumed to have the average ^{235}U enrichment corresponding to the fuel rod.

Table 1
Fuel rod specifications and test conditions.

Parameter	Value
Cladding	
O.d. (cm)	1.0262
I.d. (cm)	0.8941
Material	Zircaloy 2
Fuel pellet	
D (cm)	0.8763
Height/diameter ratio	1
% TD (UO_2)	96.5
Rod	
Plenum volume (cm^3)	1.08
Fill gas initial pressure (MPa)	1.013
Active length (cm)	381
Test system conditions	
Coolant inlet temperature (K)	560
Reactor pressure (MPa)	7.14
Coolant mass flux (kg/cm^2 s)	0.166

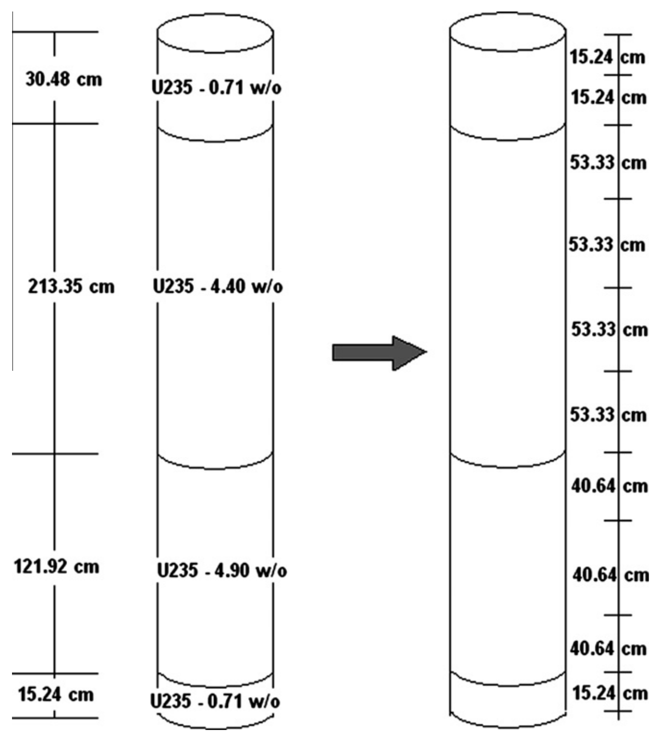

Fig. 1. Axial distribution enrichment of ^{235}U in fuel rod.

3. Computation procedure

For the computations, it was firstly assumed that the fuel assemblies containing the fuel rod were the hottest assemblies in the core (the hot channel). The considered reactor operating conditions corresponded to the nominal steady-state operation, see Table 1.

Fuel preconditioning is burnup dependent, since the threshold linear power for possible clad failure also changes as a function of burnup. A threshold power is thus previously set, as the