Description and Assessment of Deformation and Temperature Models in the FRAP-T6 Code

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Summary

The FRAP-T6 code was developed at the Idaho National Engineering Laboratory (INEL) for the purpose of calculating the transient performance of light water reactor fuel rods during reactor transients ranging from mild operational transients to severe hypothetical loss-of-coolant accidents. An important application of the FRAP-T6 code is to calculate the structural performance of fuel rod cladding. During a reactor transient, the cladding may be overstressed by mechanical interaction with thermally expanding fuel, overpressurized by fission and fission gases, embrittled by oxidation, and weakened by temperature increase. If the cladding does not crack, rupture or melt, the cladding has achieved the important objective of containing radioactive fission products.

A combination of first principle models and empirical correlations are used to solve for fuel rod performance. The incremental cladding plastic strains are calculated by the Prandtl-Reuss flow rule. The total strains and stresses are solved iteratively by Mendelson's method of successive substitution. A circumferentially nonuniform cladding temperature is taken into account in the modeling of cladding ballooning. Iodine concentration and irradiation are taken into account in the modeling of stress corrosion cracking. The gap heat transfer model takes into account the size of the fuel-cladding gap, fission gas release, fuel-cladding interface pressure, and the incomplete thermal accommodation of gas molecules to surface temperature. The heat transfer at the cladding surface is determined using a set of empirical correlations which cover a wide range of heat transfer regimes.

The capabilities of the FRAP-T6 code are assessed by comparisons of code calculations with the measurements of several hundred in-pile experiments on fuel rods. The results of the assessments show that the code accurately and efficiently models the structural and thermal response of fuel rods.

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a. Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research under DOE Contract No. DE-AC07-76ID01570.
1. Introduction

The cladding on light water reactor (LWR) fuel rods serves to contain the radioactive fission products produced by fissioning of the uranium fuel. If the cladding does not crack, rupture, or melt, the radioactive fission products remain contained in the fuel rod and the cladding has performed its duty. During a reactor transient or hypothetical accident, however, the cladding may be overstressed by mechanical interaction with fuel, weakened by temperature increase, overpressurized because of excessive fission gas release or reduced coolant pressure, attacked by volatile fission products, and embrittled by oxidation. These events alone or in combination can cause cracking or rupture of the cladding and release of the radioactive fission products to the coolant.

Most reactor operational transients and hypothetical accidents will adversely influence the performance of the fuel rod cladding. For example, during an operational transient such as a turbine trip without steam bypass [1] (boiling water reactor), the reactor power will increase by a factor of 5 for a half second. This power transient causes an increase in the thermal expansion of the fuel, which may result in the mechanical interaction of the fuel and cladding and the overstress of the cladding. During a loss-of-coolant accident (LOCA) [2], the heat generated by the radioactive decay fission products cannot be completely removed because of insufficient cooling. Because of cladding temperature increase and coolant depressurization, the cladding weakens and may possibly balloon. In addition, the cladding may oxidize and become embrittled. A complete list of reactor transients that may adversely influence the cladding is given in Reference 3.

The FRAP-T6 computer code [4] has been developed to calculate the influence on LWR fuel rods of a broad range of reactor transients, including the turbine trip and LOCA transients described above. Under sponsorship of the U.S. Nuclear Regulatory Commission, the development of the code began in 1972 and has now been completed. The code is proven to have capable models for all the phenomena that influence the transient performance of a fuel rod. The code is programmed in FORTRAN-IV and is operational on CDC computers. The code can be obtained from the National Energy Software Center at the Argonne National Laboratory or from the Program Library at the INEL. The acronym FRAP-T6 stands for Fuel Rod Analysis Program-Transient Mod 6.

This paper describes the structural and thermal models in the FRAP-T6 code and the assessment of these models. The structural and thermal models are described in Sections 2 and 3, respectively. The material properties package is summarized in Section 4. The method of solution is outlined in Section 5. The experimental data used to assess the models is described in Section 6.

2. Structural Models

In analyzing the structural response of fuel rods, two physical situations are encountered. The first situation occurs when the fuel pellets and cladding are not in contact. Here, the problem of a cylindrical shell (the cladding) with specified internal and external pressures and a specified cladding temperature distribution must be solved. This situation is called the "open-gap" regime.

The second situation encountered is when the fuel pellets (which are considerably hotter than the cladding) have expanded enough to be in contact with the cladding. Further heating of the fuel results in cladding stress because of the fuel thermal expansion. This situation is called the "closed-gap" regime and results in pellet-cladding mechanical...
interaction (PCM). Alternatively, this closed-gap regime can occur because of the collapse of the cladding onto the fuel pellets due to an elevated cladding temperature and a high coolant pressure.

Two models are available for calculating the global structural response of the fuel and cladding. The first model, named FRACAS-I, includes the effects of the thermal expansion and relocation of the fuel and thermal expansion, plasticity and high temperature creep of the cladding. The model neglects the stress-induced deformation of the fuel. The second model, named FRACAS-II, includes the stress-induced deformation of the fuel but does not consider ballooning or high temperature (>900 K) deformation of the cladding.

The FRACAS-I model is based upon the following theories and assumptions: (a) incremental theory of plasticity, (b) Mendelson's [5] method of successive elastic solutions, (c) Prandtl-Reuss flow rule, (d) isotropic work-hardening, (e) thin wall cladding, (f) no axial slippage at fuel-cladding interface when fuel and cladding are in contact, (g) axisymmetric loading, and (h) no permanent deformation of fuel. In the open-gap regime, the model considers the cladding to be a thin cylindrical shell with specified internal and external pressures and a prescribed temperature. The fuel is considered to be a solid cylinder with radial cracks. In the closed-gap regime, the model considers the cladding to be a thin cylindrical shell with prescribed external pressure and a prescribed radial displacement at its inner surface. The prescribed displacement is obtained from a calculation of the amount of fuel thermal expansion. Furthermore, since no slippage is assumed to take place when the fuel and cladding are in contact, the axial expansion of the fuel is transmitted directly to the cladding. Hence, the change in axial strain in the shell is also prescribed.

The FRACAS-II model is used to calculate deformation when stress effects on fuel deformation become important. This model calculates elastic and permanent strains in both the fuel and cladding. Generalized plane strain deformation is assumed. The outside portion of the fuel is assumed to have radial cracks. Stresses and strains are solved using the transfer matrix approach and the method of successive elastic solutions. The transfer matrices depend upon whether the fuel is cracked axially or radially or both and whether the cladding and fuel are in contact. A set of three equations is simultaneously solved for the stresses at the center of the fuel. The stresses and strains at all other locations are determined from recursion relations.

Two additional models are used for calculating the localized structural response of the cladding. The first model, named BALON2 [6], is put into action when the cladding strain has increased to the point that the cladding can no longer maintain a cylindrical shape. For the local region at which this instability occurs, the BALON2 model calculates the nonaxisymmetric ballooning of the cladding and takes into account axial and circumferential temperature variations. The second model, named CRACK [7], calculates the propagation of cracks through the cladding wall. The model considers the influence on crack propagation rate of stress, fast neutron fluence, and iodine surface concentration.

3. Temperature Models

The temperature models apply the laws of heat transfer and thermodynamics to calculate the temperature distribution throughout the fuel rod. First, the local coolant conditions (pressure, quality, and mass flux) are determined, either by a one-dimensional homogeneous transient fluid flow model or from an input coolant boundary condition tape. Next, the
heat generation in the fuel is found by interpolation in the user-input tables of fuel rod power distribution and power history. Next, the circumferentially averaged gap conductance is calculated taking into account the release of fission gases, the circumferentially varying fuel-cladding gap size for the case of an open gap, and the fuel-cladding interfacial pressure for the case of a closed gap. Next, the surface temperature of the cladding is calculated. This calculation includes a determination of the mode of convective or boiling heat transfer and an evaluation of the surface heat transfer coefficient. Finally, the temperature distribution from the fuel center to the cladding surface is determined by simultaneous solution of a set of heat conduction equations.

4. Material Properties

The NATPRO-II, Revision 2 computer code [8] is combined with the FRAP-T6 code so that the code user does not have to supply any material properties. NATPRO-II, Revision 2 contains a library of 43 subroutines dealing with the thermal and mechanical properties of uranium oxide and zircaloy-4 cladding at temperatures varying from room temperature to melting temperature. The code also calculates the conductivity and viscosity of the fill fission gases inside a fuel rod. Each material property is programmed as a separate unit so that individual parts may be used without necessitating inclusion of the whole code.

5. Method of Solution

The structural and temperature models and other models must be combined together to calculate the transient performance of fuel rods. In FRAP-T6, the models are combined as shown in Figure 1. First, the temperature distribution in the fuel and cladding is determined using past time step or past iteration values for the fuel-cladding gap. Next, the structural response of the fuel and cladding is calculated using the most recent temperature distribution and past time step or iteration value for internal gas pressure. This calculation results in updated values of the fuel-cladding gap. Finally, the pressure of the gas inside the fuel rod is calculated. This sequence of calculations is cycled until essentially the same temperature distribution is calculated by two successive cycles. To reduce the number of iterations, the method of Newton is used to predict converged values of gap conductance and internal gas pressure. Convergence almost always occurs within five cycles. After the temperature and structural response calculations have converged, cladding oxidation and fission gas release are calculated. Time is then incrementally advanced, and the complete sequence of calculations is repeated to obtain the values of the fuel rod variables at the advanced time.

The code user has four means of controlling the accuracy and cost of solution. This control is exerted through input specification of (a) nodalization, (b) convergence criteria, (c) time-step size, and (d) inclusion or exclusion of sophisticated models. Typically, the solution for one fuel rod for one time step is performed in 0.5 s on a CDC 176 computer using 300K octal of central memory and 65K decimal of extended core memory. The time-step sizes vary from 0.001 s for the power pulse period of a reactivity initiated accident to 5 s for the reflood period of a LOCA.

6. Assessment

A wide range of fuel rod tests have been performed in the past 10 years. At the Power Burst Facility (PBF) [9] in Idaho for example, tests have been performed that measure both the steady state and transient response of fuel rods. The variables measured in these tests include fuel centerline temperature, cladding surface temperature, internal fuel rod pressure,
and cladding outer diameter. At the Halden Boiling Water Reactor (HBWR), [10] tests have been performed that measure fuel centerline temperature, cladding diameter, and elongation as a function of fuel rod power and burnup. Tests to measure the ability of fuel rods to withstand power ramps have been performed at the Studsvik reactor in Sweden [11]. At the National Reactor Universal (NRU) in Canada, tests have been performed that measure the cladding temperature history during the reflood stage of a LOCA [12]. The wealth of data obtained from these various test facilities allows a complete assessment and verification of computer codes such as FRAP-T6. The tests used to assess FRAP-T6 are listed in Table 1. A summary description of many of these tests is given in Reference 13. An overall assessment of FRAP-T6 is given in Reference 23. An assessment focusing on the code's capabilities during LOCAs is given in Reference 24.

Because the temperature models strongly influence the structural models, they must first be assessed to verify that they are not distorting the calculations of the structural models. The assessment of the temperature models is performed by modeling with FRAP-T6 fuel rod tests that measure fuel centerline and cladding surface temperature. Measured and calculated temperatures are then compared. Figure 2 through 6 show comparisons for tests that are particularly descriptive. The good agreement of calculations and measurements show that most of the temperature models in FRAP-T6 are performing accurately. As shown in Figure 3 for the fuel rod filled with xenon, some inaccuracies exist in the modeling of fuel relocation and gap conductance. These inaccuracies lead to a 0 to 5% overprediction of temperature for helium-filled rods and a 5 to 10% overprediction of temperature for xenon-filled rods.

The assessment of the structural models is based upon fuel rod tests that measure cladding deformation and observe cladding failure. For tests in which cladding ballooning occurs, the transient cladding diameter has not been measured. So for these tests, the fuel rod internal pressure is measured. The reciprocal relation of gas volume to internal pressure allows an indirect measure of the amount of cladding ballooning.

Plots of data used to assess the cladding deformation models are shown in Figures 7 and 8. FRAP-T6 calculated values of the measured variables are also shown. These plots show that the cladding deformation models are reasonably accurate. The time of cladding rupture is correctly calculated. The calculated cladding rupture strain is in agreement with the measured rupture strain. As exhibited in Figure 7, the abrupt decrease in internal pressure, at a time of 15 s, gives evidence that for cladding temperature in the neighborhood of 750 K improvements may be possible to the model of the cladding stress-strain relation. In the neighborhood of 750 K, the cladding stress shifts from being primarily a function of strain to being primarily a function of strain rate.

7. Conclusions

The assessment of the FRAP-T6 code has shown that the code is capable of analyzing fuel rod performance during operational transients and hypothetical reactor accidents. The capabilities of the code have been demonstrated by comparing code calculations against measured values for a wide range of fuel rod tests. The proven capabilities of the code include the modeling of fuel stored energy, fuel rod cooling, and cladding ballooning. The code accurately calculates the extent of cladding ballooning and time of cladding rupture. This accurate calculation is attributed to modeling of local cladding thinning and including the effects of fuel rod heating and pressurization rates. The code can be expected to
overpredict fuel temperature by 0 to 5% for helium-filled rods and by 5 to 10% for xenon-filled rods. The assessment has shown that improvement may be possible to models of fuel relocation and the cladding constitutive equation for temperatures in the neighborhood of 750 K.

8. References


Figure 1. Combination of models.

Figure 2. Comparison of measured and calculated fuel centerline temperature for helium-filled Rod 3 of PFBR Test LOC-11C.

Figure 3. Comparison of measured and calculated fuel centerline temperature for xenon-filled Rod 501 of PFBR Test GC 2-1.
Figure 4. Comparison of measured and calculated cladding surface temperature for Rod 12 of TREAT Test FRP-2.

Figure 5. Comparison of measured and calculated fuel centerline temperature for Rod 3 of PBF Test RIA 1-1.

Figure 6. Comparison of measured and calculated fuel centerline temperature for Rod 15 of PBF Test PCM-4.
Figure 7. Comparison of measured and calculated fuel rod plenum pressure for TREAT Test FRP-2.

Figure 8. Comparison of measured and calculated cladding permanent hoop strain for Rod 12 of TREAT Test FRP-2.
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