Advancements in the Behavioral Modeling
of Fuel Elements and Related Structures

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ABSTRACT

An important aspect of the design and analysis of nuclear reactors is the ability to predict the behavior of fuel elements in the adverse environment of a reactor system. By understanding the thermomechanical behavior of the different materials which constitute a nuclear fuel element, analysis and predictions can be made regarding the integrity and reliability of fuel element designs. The SMiRT conference series, through the division on fuel elements and the post-conference seminars on fuel element modeling, provided technical forums for the international participation in the exchange of knowledge concerning the thermomechanical modeling of fuel elements. This paper discusses the technical advances in the behavioral modeling of fuel elements presented at the SMiRT conference series since its inception in 1971. Progress in the areas of material properties and constitutive relationships, modeling methodologies, and integral modeling approaches was reviewed and is summarized in light of their impact on the thermomechanical modeling of nuclear fuel elements.

INTRODUCTION

The SMiRT Conference series began at a time when nuclear power was perceived to have come of age. No major industry-wide problems existed, and the system performed as expected. The first generation thermal reactor technology was almost relegated to a text-book status, clearing the way for intensified development in the fast breeder technology as a natural extension and a necessary second generation system. However, at the start of the seventies decade, this orderly progression was not to continue uninterrupted as public opposition to nuclear power began to develop, encouraged by unfavorable trends in nuclear power economics. This coincided with the emergence of performance problems in light water reactor (LWR) fuel when increased incidents of pellet-clad interaction (PCI) failures forced industry to re-evaluate their fuel design methods and to develop fuel research programs to better understand fuel behavior.

Prior to that time, LWR fuel design was governed by physical and chemical considerations which placed significant constraints on the fuel's thermal performance. For example, the initial thickness of the fuel-cladding gap had to be increased, thereby penalizing the gap heat transfer, in order to accommodate end-of-life fuel swelling. The mechanical design was far less important, almost a loose end that needed to be addressed only to satisfy conventional design requirements which, in some cases, were irrelevant. For example, the cladding thickness was initially selected to satisfy such benign condition as the contact stresses caused by the spacer grid spring forces. Because of this elementary nature with which the mechanical behavior of fuel elements was viewed, structural mechanics aspects of fuel elements were not considered to be proper topics for technical forums. It was not until the first SMiRT conference that a major international conference provided a

technical forum to discuss the mechanical behavior of fuel elements. It was almost visionary that fuel element mechanics, perceived at the time to be technically routine, was given a prominent position in a major technical conference, for not too long after SMiRT-1 the thermomechanical performance of reactor fuel elements became a dominant topic in the experimental and analytical literature. The SMiRT conferences which followed played a major role in this technical exchange.

Unlike other reactor components that are dealt with in the SMiRT conferences, fuel elements occupy a unique position in the nuclear literature. They are the common grounds for such major disciplines as reactor physics, nuclear chemistry, materials science, heat transfer and structural mechanics. Furthermore, because of the central role of the fuel in the nuclear steam supply system (NSSS), fuel element models of various types are incorporated in the NSSS codes. These codes, arranged in ascending order of complexity of the fuel element models contained in them, include system thermal hydraulics codes, core simulation codes, loss of coolant accident codes, and fuel performance codes. Because of this universality of the role of fuel elements in the NSSS performance, the modeling of fuel element behavior is not the exclusive domain of SMiRT. Despite the usual difficulties of coordinating multi-disciplinary activities, the SMiRT division coordinators have succeeded in securing the participation of well known researchers in the general field of fuel element behavior. Thus, the SMiRT conferences and their adjunct post-SMiRT seminar series on fuel element modeling contributed, along with other nuclear industry forums, to the present level of understanding of fuel element behavior. Although it would not be possible to isolate SMiRT's specific contribution to this endeavor, we will attempt to describe in this paper the progress made in those areas of fuel element modeling that were emphasized in fuel element and assembly division of the SMiRT conference.

In this division, concepts have been presented which dealt with a wide spectrum of behavioral modeling topics on fuel elements and assemblies for Light Water Reactors (LWRs), Liquid Metal Fast Reactors (LMFRs) and High Temperature Gas Reactors (HTGRs). The scope of this discussion will be to present an overview of the advances in key technical areas regarding nuclear fuel elements and assemblies which were presented during the first twenty years of the SMiRT conferences. Because we expect the readership of this review to include non-specialists, we will provide general descriptions and background information where appropriate.

BACKGROUND

A nuclear fuel element is composed of fissionable material in the form of a uranium based ceramic or metal. Reactor fuel elements appear to be rather simple structures. Generally, they are long cylindrical rods with diameters in the range of 5-12 mm and lengths in the range 1-4 m. A reactor core consists of a number of assemblies of these fuel rods, each assembly containing rods in either a square or hexagonal array with spacer wires or grids allowing space between the rods for axially flowing coolant [water for LWRs, liquid Na for LMFRs or He for HTGRs]. The production of hazardous by-products in the fissioning process requires that some type of barrier or cladding be used to contain these products. This outer shell of each fuel element - the cladding - is generally a thin-walled tube of structural material (e.g., zircaloy or stainless steel) with diameter-to-thickness ratios in the range of 10-25. The fuel inside the cladding is either a ceramic [e.g., UO_2 , $(\text{U,Pu})\text{O}_2$, $(\text{U,Pu})\text{C}$, $(\text{U,Pu})\text{N}$] in sintered pellet or sphere-pac form or a metallic slug (e.g., U-Fs, U-Pu-Zr alloys) with a fuel column length of 3-4m for LWRs and 0.3-1m for LMFRs. Between the as-fabricated fuel and cladding is a thermal bond (e.g., He or liquid Na) to allow for differential fuel/cladding expansion. Above the fuel column is a gas plenum space for accommodating fission-product gases. Thus, in simple mechanical engineering terms, a fuel element consists of an internal heat generating material surrounded by a cylindrical pressure vessel.

The thermomechanical behavior of a nuclear fuel element is a complex system of interacting and competing processes as a consequence of the high thermal power

densities and neutron flux environment. In a ceramic fuel element, the low thermal conductivity produces steep radial temperature gradients across the fuel pellets. This results in significant tensile thermal stresses in the fuel material, causing pellet cracking and redistribution. Fuel restructuring also takes place in the high temperature regions, due to the migration of porosity, fuel constituents, and fission products. The neutron flux and fission process induces changes in the material properties of both the cladding and fuel material. Swelling as a result of fission products, both gaseous and solid, and atomic displacements due to the neutron flux influence the dimensional changes and mechanical properties of these materials. Mechanical interaction between the fuel and the cladding can produce stresses and deformations in the cladding and the chemical interactions between the fuel and cladding and the cladding and coolant can increase the potential for failure. These processes have made the problem of predicting the nuclear, thermal, and mechanical performance of fuel elements quite challenging. However, detailed thermomechanical modeling of the fuel and cladding is required to assure the integrity of the cladding during both normal and abnormal conditions and to understand the basic phenomenon taking place within a fuel element. Considerable nuclear, thermal, structural, materials, and chemical expertise has been devoted to developing verified and validated theories, methodologies, computer models and codes for predicting fuel element performance during the past 20 years. Yet, by knowing the fundamental concepts of fuel element behavior, design characteristics can be incorporated which decrease the propensity for cladding failure. An example of this is the development of zirconium lined cladding for LWRs which decrease the susceptibility of pellet-cladding interaction (PCI) failures. The mechanisms of PCI failures were identified through the detailed mechanical modeling of fuel element behavior as will be discussed later.

The technical concepts presented at the SMiRT conferences have provided for the advancement of methodologies, material properties and constitutive relationships in the modeling and understanding of the complexities of fuel element thermomechanical behavior. The level of sophistication expressed in the technical sessions have allowed for in-depth views and discussions on the current state-of-the art in fuel element analysis. As the evolutionary process of technical advancement brought in new ideas and phenomena, additional sessions were added and viewpoints broadened to encompass these issues.

Several key areas related to the understanding of fuel element modeling showed improvements or advancements throughout the SMiRT conferences. As a result, the SMiRT conference provided a platform for the progressive development of these issues. The purpose of this discussion is to present a brief overview of a number of the issues which showed significant advancements and development within the SMiRT conference in fuel element and assembly modeling.

Before launching into this review, the authors would like to acknowledge past overview and review articles. Y. R. Rashid's (1973) paper (D1/1) presented at SMiRT-2 in West Berlin on mathematical modeling of the structural performance of fuel rods is cited for detailing the forms of the material constitutive relations to be used for plasticity, creep, swelling and hot pressing. It was an inspiration to modeling efforts within both the fast and thermal reactor communities. R. W. Weeks (1978) presented an excellent overview of the capabilities and limitations of fuel performance codes in his Principal Division Lecture D/O at SMiRT-4 in San Francisco. Finally, K. Lassmann (1980) is cited for his excellent discussion and categorization of fuel element codes presented in paper D3/1 at SMiRT-5 in West Berlin.

The following section will discuss the improvements in fuel and cladding material properties, fuel and cladding mechanical interaction (PCI), and integral fuel rod modeling codes.

The thermomechanical modeling of a nuclear fuel element consists of determining the behavior of the fuel and the cladding, separately, and linking them by the gap between these two components. Because of its thermal feedback properties, the correct modeling of the gap is critical to the modeling of the fuel element overall behavior. The deformation states in the fuel and cladding determine the amount of gap closure. The thermal environment in the fuel material causes the cracking of the fuel pellets and further fuel movement into the gap. The SMiRT conferences have continually dealt with the problem of fuel cracking, and several approaches have been presented and utilized in the nuclear fuel modeling community [Rashid (1973) and Lassman (1977)].

Cladding deformations under the pressure of the coolant also reduces the fuel-cladding gap. Creep of the cladding, either thermally or irradiation induced can cause the cladding to contact the fuel. Also, work hardening and irradiation hardening influence the behavior of the cladding. As a result of the numerous phenomena affecting the mechanical behavior of the cladding, a substantial amount of work has been devoted to the development of detailed constitutive relationships which incorporate these different factors [Liu (1977), Murty (1977), Miller (1977) and Senski (1979)].

Lastly, a major factor in determining the mechanical behavior of the fuel and cladding is to predict the amount of gap closure which occurs during irradiation. The gap width influences the temperature profile in the fuel rod and defines the onset of pellet-cladding interaction. The modeling of this interaction is very important in determining the amount of elongation, the ridge height formations and the failure mechanisms of the cladding. The modeling of PCI requires a thorough understanding of the different mechanisms involved, including the fuel, cladding and fission products and the interplay of these different components. Not only are cladding deformations due to PCI important in the mechanical analysis of fuel elements, but ultimately a prediction of the amount of damage or type of failure mode is desired. Several different cladding failure modes are possible which vary from gross cladding breach as a result of molten fuel in LMFRs to stress corrosion cracking PCI failures in LWRs. Except for cladding ballooning in LWR LOCA transients, the cladding failure mechanisms are the result of fuel and cladding interaction. Accurate predictions of the deformations in both the fuel and cladding are required to determine the stress levels generated in the cladding which in turn determine the amount of damage incurred during power ramps under normal operation. The analysis of the fuel and cladding deformations require the ability to predict the behavior of the material during in-reactor operation in which severe thermal gradients, the fission process, and a high neutron flux are present.

Fuel and Cladding Material Properties

The fundamental basis for modeling the mechanical behavior of the fuel and cladding is a well described system of material properties and constitutive relationships. The elastic, plastic and creep behavior of the fuel and cladding during in-reactor operation define the type of mechanical interaction. The material properties and constitutive relations depend on the type of irradiation environment due to the material changes induced by the neutron flux. Dimensional changes caused by irradiation induced swelling and densification and thermal expansion also influence the fuel and cladding interaction. As a result of the intimate relationship between the material properties and constitutive relations and the mechanical behavior of a nuclear fuel rod, these factors need to be well defined in the temperature, neutron flux, and stress regimes which could be possible in a nuclear reactor.

The scientists and engineers participating in the SMiRT conferences over the years realized this and devoted a significant portion of the discussions to the topics concerning the material properties of the fuel and cladding. For example, the hexagonal closed packed (hcp) crystalline structure of zircaloy results in a material which is highly textured with nonrandomly oriented grains [Dressler (1973)]. The limited number of slip planes in zircaloy and

preferred orientation of the grains causes the mechanical behavior to be anisotropic. The anisotropy of Zircaloy is dependent on the heat treatment and fabrication variables [Beauregard (1977)]. Studies have also revealed that the anisotropic behavior of zircaloy is temperature and strain-rate dependent [Kallstrom (1971) and Murty (1985)]. The characterization of the mechanical anisotropy of the cladding material is important in predicting the dimensional changes of the cladding during irradiation and the subsequent pellet-cladding mechanical interaction. The anisotropic behavior of zircaloy was studied by several authors during the SMiRT conferences in a wide spectrum of views and applications [Dressler (1973) - Beauregard (1977)].

As the importance of the gap width was realized, the constitutive relationship of the irradiated cladding was investigated to improve the calculation of fuel and cladding interaction. Extensive discussions have been presented which concerned the elastic, plastic, and creep behavior of irradiated cladding materials [Liu (1977) - Bohner (1987)]. Experimental results of creep tests on unirradiated and irradiated cladding using various manufacturing techniques were reported. Both phenomenological and empirical formulations were used to develop relationships of thermally activated and irradiation induced strain rates to stress, temperature, and neutron fluence. The thermal creep behavior was identified to be controlled by diffusion of dislocations [Murty (1977)]. The stress dependence of the creep rate varied between arguments and was normally represented by either an exponential function or a power law. The irradiation induced creep is the dominant component in the cladding deformations at the temperatures observed during in-reactor service [Senski (1979)]. Several studies were presented which characterized the creep behavior of irradiated material. Irradiation hardening and work hardening terms were also added to the constitutive relationships to improve the predictions of cladding deformations [Miller (1977)]. The importance of the manufacturing processes on the material properties and the behavior of the cladding material during irradiation was also identified in numerous SMiRT sessions [Garzarolli (1981) and Bohner (1987)]. The complete description of the behavior of the cladding during in-reactor service greatly improved the predictability of fuel element performance. The SMiRT conference assisted in disseminating these advances in experimental data and modeling methods in cladding constitutive relationships.

Deformations in the fuel also influence the fuel-cladding gap. Fuel deformations are caused by the thermal expansion of the material, swelling as a result of fission products, and densification or removal of a percentage of the as-manufactured porosity. Fuel pellet cracking and redistribution is one of the major contributors to the fuel dimensional changes [Lassman (1977), Appelhans (1981) and Guha (1977)]. Fuel cracks also play an important role in fuel-cladding interaction and could lead to cladding failure [Gittus (1971)]. Due to their importance in determining the integrity of the cladding material, fuel cracking behavior was studied throughout the SMiRT conferences. Various treatments in 1, 2 and 3-dimensional analyses have been conducted to quantify the cracking behavior of UO_2 [Rashid (1973), Levy (1973) and Lassman (1977)]. Secondly, upon gap closure the cladding imposes a compressive force on the fuel. A portion of the crack strains are recovered during this process. This type of recovery is called the fuel crack compliance [Willford]. Also hot-pressing and plastic deformation takes place in the fuel and cladding. This results in stress relaxation in the fuel and cladding. The material properties which define the compliance of the fuel during plastic deformation and hot pressing are influenced by the type of irradiation environment present and the amount of fuel restructuring which occurs such as grain growth and pore and fuel constituent migration [Rashid (1973) and Olsen (1979)]. Fuel swelling is also an important component to the fuel deformations.

Pellet-Cladding Interaction (PCI)

The geometrical changes induced by in-reactor service ultimately results in the closure of the fuel and cladding gap. This interaction leads to permanent deformations to occur in the fuel and cladding as a result of the contact pressure which develops. In addition to the PCI induced deformations, cladding failure can occur due to the interaction. These failure modes vary from gross cladding breach as a result of molten fuel to low probability

intergranular stress corrosion cracking (ISCC) failures. Molten fuel-cladding interaction can occur in LMFBRs during severe loss of flow transients. In overpower transients, two fuel-cladding type failures can take place depending on the operating conditions: cladding rupture or stress corrosion cracking. In the first case, the thermal expansion of the fuel increases at such a rate due to a power ramp, that the yield stress in the cladding is exceeded and rupture failure of the material occurs. The ramp rate of the fuel rod power must be fast enough to limit the amount of stress relaxation and compliance of the fuel. In stress corrosion cracking a threshold stress level must be maintained for a certain incubation period to induce corrosion assisted intergranular cleavage and crack formation. The liberated fission products or carburization of stainless steel in LMFBR carbide fuels enhance the cracking of the cladding material due to corrosive attack on the grain boundaries [Miller (1981), Peehs (1981) and Levine (1983)]. Experimental and theoretical evidence indicate that the mechanism by which the cladding stress is increased to the threshold level is by the radial cracks in the fuel opening at high powers and causing localized strain concentrations after gap closure. This is brought about by the friction between the fuel and cladding.

After contact occurs, a contact pressure is established between the fuel and cladding. Frictional forces are also produced due to the difference in relative motion between the fuel and cladding. During a power ramp, the thermal expansion of the fuel material causes a net outward force to be applied to the cladding. This force results in stresses in the cladding which, if the stresses are large enough, can cause rupture. At lower powers, the force on the cladding from the fuel is not large and does not appreciably deform the cladding. However, the outward expansion of the fuel will attempt to open the radial cracks present along the periphery of the fuel pellet. The frictional forces at the fuel-cladding interface creates localized straining of the cladding in the vicinity of the radial fuel cracks. The localized straining results in high stress levels in this area of the cladding. This behavior coupled with the presence of fission products can cause cladding cracks to nucleate and propagate through the wall thickness, producing cladding failure.

Gittus identified the fundamental phenomena of PCI in some preliminary work presented at the 1st SMIRT conference [Gittus (1971)]. In his formulation, a simple model was developed which incorporated the fuel-cladding interfacial effects in determining the local cladding strains. By using elastic thin shell theory and assuming a symmetric distribution of N fuel pellet cracks, Gittus was able to obtain the average cladding strain, $\bar{\epsilon}$, across the cladding thickness as a function of the interfacial friction coefficient, m , and the polar angle, q , measured from the fuel pellet crack. Originally, this simplified model provided insight into the fundamental characteristics of localized PCI effects, however, prediction of cladding failure was not possible. Several other investigators also began to develop models to represent PCI, based on the original approach of Gittus [Ranjan (1977)].

A more general and comprehensive approach was present by Rashid (1973) using basic constitutive relations for the fuel and cladding materials which takes into account the elastic-plastic behavior as well as the creep and restructuring behavior of the fuel. This method was applied through a two-dimensional, axisymmetric and plane strain formulation within the framework of the finite element method. Special gap contact elements are used at pellet-pellet and pellet-cladding interfaces that contain the properties necessary to represent friction or full bonding and appropriate gap conductance values. The modeling approach of Rashid provided the guidelines in which detailed analysis of the PCI phenomena was studied by following scientists [Hsu (1979)].

COMPUTER PROGRAMS

To predict the overall behavior of the fuel and cladding during irradiation required the development of sophisticated computer programs which integrated the constitutive models of the fuel and cladding with numerical representation of the various phenomena present and advanced numerical methods. These

computer programs allowed for detailed examination of the controlling mechanisms in fuel element behavior. The progresses in the development and verification of several computer programs used in both LWR and LMFR fuel element analysis were reported during the SMiRT conferences. The following will be a review of the important issues regarding integrated fuel performance codes.

Mechanistic Fuel Performance Codes

Tables 1 and 2 summarize the fuel performance codes (as reported in SMiRT proceedings) developed specifically for, or used for, analyses of fast and thermal reactor fuel elements. The codes are listed along with the specific SMiRT conferences, paper numbers, and a general characterization in terms of steady-state (SS), transient (T), or steady state and transient (SST), as well as dimensionality. In this context, transient refers specifically to a transient heat transfer capability and time-dependent deformation models. In sorting through the many codes reported at SMiRT conferences, an attempt was made to identify (and include in Table 1 and 2) only "mechanistic" computer codes which contained both thermal and mechanical analyses of reactor fuel elements. What was considered to be the deciding factor in this distinction was that the code either contain mechanistic models or have a framework general enough to allow the incorporation of mechanistic models. Fast-running "correlations" codes for mechanical and/or thermal analyses are discussed in a separate section.

In surveying the number of SMiRT papers for each code, it is clear that the fast reactor codes LIFE, FRUMP, and URANUS and the thermal reactor codes WAFER, FREY, and FRAP have endured through the years. Within the context of SMiRT, it is appropriate then to focus on the evolution and impact of these codes within the reactor community.

Integral Fast Reactor Computer Codes

The LIFE-II code was introduced at SMiRT-1 in 1971. At that time, there were several codes in the USA (e.g., BEHAVE, CYGRO, FMODEL, and LIFE) under consideration by the then Atomic Energy Commission for further development as the reference USA fast reactor code. Since that time, it has been used for pre- and post-test analyses of fast reactor test pins irradiated in EBR-II, FFTF, and the transient test reactor TREAT; to guide materials testing programs; and for fuel element analysis of reactor design concepts, including the Clinch River Breeder Reactor (CRBR) project, the Gas Cooled Fast Reactor (GCFR) project, the Advanced Fuels (e.g., mixed carbide and mixed nitride) Reactors, and currently the all metallic core Integral Fast Reactor (IFR) design concept.

The LIFE-II code reported at SMiRT-1 was developed specifically for the steady-state and quasi-steady (e.g., load following) analysis of LMFR fuel with Na coolant, stainless steel cladding, He thermal bond, and mixed-oxide fuel. The fuel structural analysis was based on the assumptions of small strain and generalized plane strain for up to nine axial fuel nodes and one plenum node. However, the number of radial rings was restricted to 3 for the fuel (corresponding to columnar, equiaxed, and unrestructured zones) and 1 for the cladding. At SMiRT-2, Jankus presented Paper D2/5 detailing his multiregion formulation generalizing the radial zones to up to 15 rings for the fuel and the remainder of the maximum number of 20 total rings for the cladding. This major change, along with the inclusion of a finite strain formulation for fuel and cladding presented by Jankus at SMiRT-4 (D1/4), led to the introduction of LIFE-III. Also at SMiRT-4, the mixed-carbide and mixed-nitride version of LIFE, UNCLE, was discussed by Billone at the Post Conference Seminar on Mathematical/Mechanical Modeling (MMM) of Reactor Fuel Elements. Between SMiRT-4 and SMiRT-5, LIFE code development focused on the inclusion of cladding options for advanced, low-swelling ferritic and austenitic alloys, ^{an}a GCFR version, ~~an~~ LIFE-4, with transient heat transfer and advanced cladding options, was described by Harbourne et al. (D1/1) at SMiRT-5. Liu et al. presented the corresponding mixed-carbide and mixed-nitride version, LIFE-4CN, at the MMM post conference. This latter subject was also addressed by Billone at SMiRT-6 (D4/1). Sample calculations from LIFE-GCFR were also presented by Billone at SMiRT-6 (D4/4).

The current focus of LIFE development, LIFE-METAL, was alluded to, but not described in detail, at SMiRT-9 (vol. C, p.11). With the introduction of the Integral Fast Reactor (IFR) all-metallic-core design concept in 1984, there arose a need within the fast reactor community to model the performance of U-Pu-Zr fuel alloys, as well as the U-Fs EBR-II driver fuel. LIFE-4CN, which included an option for Na thermal bond, as well as (U,Pu)C fuel which is closer to metallic fuel in performance than (U,Pu)O₂, was used as the base code to generate LIFE-METAL. Presently, LIFE-METAL is being used for pre-test data packages and post-test interpretation of experimental irradiations and as the main fuel-element design tool in the USA for the PRISM design concept. While transient capabilities are present in LIFE-METAL, high temperature phenomena (e.g., fuel melting, fuel/cladding eutectic formation) encountered under accident conditions are not yet modeled. These transient calculations are performed with FPIN (see Table 1) which uses LIFE-METAL steady-state predictions to establish initial conditions for the transient.

The FRUMP code was introduced to the SMiRT community at SMiRT-2 by Wilmore and Haynes (D1/8) and Matthews (D2/2) of the UK. At the time, it was basically a steady-state code for stainless-steel-clad, mixed-oxide, fast-reactor fuel elements. Throughout its development, there has been an emphasis on fundamental modeling, particularly in the area of fission product swelling and release. Since the time of its introduction, FRUMP has had a similar status within the UK fast-reactor community that LIFE has had within the USA fast-reactor community. Haynes presented some FRUMP results at SMiRT-3 (D1/12) comparing fuel performance in fast test reactors to that in thermal test reactors. The most detailed description of FRUMP was given by Matthews at SMiRT-4 (MMM). By this time a transient heat transfer capability had been implemented. FRUMP's major applications were in parametric surveys to aid pin design, analysis of experiments in DFR and material test reactors, determining power rise policy for PFR, loss of flow simulations in the SCARABEE loop, and pre-test calculations to define conditions for the CABRI experiments. Between SMiRT-2 and SMiRT-4, mixed-carbide and mixed-nitride options were developed for FRUMP. At SMiRT-5 (D7/1), Matthews focused his paper on transient performance predictions.

Progress on the development of the URANUS code of the Federal Republic of Germany (FRG) has been widely reported at SMiRT Conferences. Lassmann presented the first SMiRT papers on URANUS at SMiRT-4 (C1/3, D1/2). Structural analysis models for fuel and cladding were emphasized. The steady-state Integral Version of URANUS was completed around the time of SMiRT-4. Shortly after that, both the Statistical Version and the fast-running Design Version became available. Some of this work was alluded to in Lassmann's excellent review of the structure of fuel element modeling codes at SMiRT-5 (D3/1). At this time, URANUS was essentially a steady-state/load-following code. Preusser presented the details of the carbide version of URANUS at SMiRT-6 (D4/2). This effort began in 1978. The focus of the effort then shifted to modeling the transient behavior of both oxide and carbide fuel between SMiRT-5 and SMiRT-6. At SMiRT-7 (C4/3, C4/4), Preusser and Lassmann presented the most detailed description of the models and performance of the SST version of URANUS for oxide and carbide fuel. By this time, URANUS had been used for analysis in several major projects (e.g., CAPRI, MOL-7C, and Halden). At SMiRT-8 (C2/8) Preusser and Schiffer presented results for URANUS calculations evaluating different fuel-element failure criteria under FBR hypothetical accident conditions.

In summary, the three mechanistic fuel element modeling codes most reported at SMiRT conferences - LIFE, FRUMP, and URANUS - have been widely used throughout the fast reactor community. All three codes have steady-state and transient heat transfer and deformation analytical capabilities. They are 1-1/2D whole fuel-element codes with detailed radial analyses and varying degrees of coupling in the axial direction. They are distinguished from many other fuel element codes in their degree of mechanistic models, verification of algorithms, and validation of thermal-mechanical predictions. All three codes have options or versions for more than 1 fuel type (e.g., oxide, carbide, nitride, metal). Also, all three codes have been used for the analysis of LWR fuel rod performance, as well. URANUS started out as an LWR code and

continues to be used to analyze LWR fuel as evidenced by paper C1/3 at SMiRT-8. FRUMP was used to analyze LWR fuel throughout its development. An LWR version of LIFE was released in 1976. However, the COMETHE-III-J code was selected by the U.S. Electrical Power Research Institute (EPRI) over LIFE-THERMAL for LWR fuel performance analysis.

Currently, the focus of LIFE development efforts is on modeling the performance of U-Pu-Zr alloy fuel for the IFR design concept. LIFE-METAL is unique in this application for quasi-steady analysis, as is FPIN for transient analysis of IFR metallic fuel. URANUS continues to be used for SST analysis of both FBR and LWR fuel-rod performance.

Integral Thermal Reactor Computer Codes

The WAFER-1 code was introduced at SMiRT-3 in 1975 by Kjaer-Pedersen. WAFER-1 was developed to calculate the local stress and strain history throughout a clad segment corresponding to a one half-pellet geometry. A unique method of superposition of an r-q and r-z mechanical analysis of the fuel pellet resulted in a quasi-three dimensional approach, taking into account both radial and transversal cracking. Shell theory was used in the cladding to determine the elastic behavior and a detailed mesh solution was used for the creep analysis. The WAFER-1 code provided for a method to analyze a single pellet to determine the interaction kinetics of the fuel and cladding to determine cladding ridge formation.

At SMiRT-4, Kjaer-Pedersen presented the improvements made to the WAFER code and the release of the new version, WAFER-2. The improvements implemented in WAFER-2 were designed to increase reliability and accuracy and decrease storage requirements and processing time. The pellet cracking model was improved to allow for more versatility in both radial and transversal cracks. This improvement provided for improved creep rates to be calculated for the fuel. The pellet hourglassing and cladding ridging model was upgraded by the more thorough representation of transversal cracks and axial interaction between the fuel and cladding. New models for inpile densification, gaseous swelling and fission gas release were added to improve the overall capabilities of the WAFER-2.

With the prediction of fission gas release becoming more important, a full length version of WAFER was introduced at SMiRT-5 by Kjaer-Pedersen. WAFER-3 also contained a first-principles fission gas release model which accounted for local variable influence on gas release. The full size version allowed for the use of time varying axial flux distributions. The axial mechanical interaction was also improved to allow for the effects of a trapped stack phenomena as a consequence of contact above or below a reference point. The pellet cracking pattern was modified to contain a finite number of radial and transversal fuel cracks to provide for stress build-up between cracks. Fuel rod failure integrals were added to WAFER-3 to provide for accumulated damage in the cladding for failure analyses. The improvements made to WAFER-3 resulted in a more versatile code which could be applied to a wide spectrum of fuel performance analyses.

For the WAFER code, the SMiRT conference provide an avenue to present the underlying theories to the unique mechanical solutions used to calculate the fuel-cladding interaction. The evolution of the code from a very specific pellet model to a full-length code was established via the SMiRT conferences.

The FRAP series of codes was introduced at the SMiRT-4 conference by Berna and at the MMM Post-conference seminar associated with SMiRT-4 by Dearien. FRAP-S and FRAP-T are a pair of codes developed for the US NRC to predict the behavior of nuclear fuel rods during both steady state and transient conditions. Each code is a full length thermal-mechanical analysis code. Detailed mechanical analysis is carried out encompassing a wide spectrum of elastic, plastic and creep behavior in both small strain and large strain (FRAP-T) deformations. Fuel rod failure predictions are considered from a wide range of failure mechanisms. The FRAP series of codes represents an integrated approach to predicting both the steady state and transient analysis of nuclear fuel rods. The design and implementation of experiments conducted

by the United States in the LOFT, PBF, and TREAT facilities were assisted by the FRAP series of codes. Extensive post-test analyses were also conducted to determine the fundamental behavior of the test rods.

A description of the advanced version, FRAP-T6, was presented by Seifken at SMiRT-7. FRAP-T6 represented the final version of the NRC transient fuel performance code. A thorough overview of the capabilities of the code was presented along with an assessment of these capabilities. Shah discussed at SMiRT-7 an overview of the cladding axial elongation models used in FRAP-T6. These models represented four different mechanisms: (a) axial PCMI, (b) trapped fuel stack, (c) fuel relocation and (d) effective fuel thermal expansion. The first two models are mechanistically based and the fuel relocation and effective thermal expansion are empirically based. The axial PCMI model is formulated on the basis that axial PCMI occurs at a lower power than radial PCMI due to random pellet stacking, pellet cocking, cladding ovality, etc. The trapped stack phenomena effect occurs when pellet-cladding interaction traps the lower portion of the pellet column between the bottom of the rod and the lowest pellet experiencing PCMI. The relocation and effective thermal expansion models are empirical relationships used to describe the amount of fuel relocation into the gap and the differential axial expansion between the fuel and cladding. The axial thermal expansion of a fuel stack depends on the type of pellets used as a result of different pellet lengths and geometries (e.g., flat end, dashed, chamfered, etc). This influences the effective thermal expansion and the model utilizes an empirical factor which depends on the type of pellet. The paper by Shah demonstrated the mechanical capabilities of FRAP-T6 for use in calculating fuel rod failures due to PCMI.

Rashid presented the FREY transient fuel performance code at SMiRT-6. FREY currently represents a unique method for calculating the thermomechanical behavior of LWR fuel rods. FREY utilizes a fully coupled 2-dimensional formulation which calculates the radial and axial interaction in an axisymmetric analysis. A unique 2-dimensional r - θ capability exists to study the effects of azimuthal variations. Both the thermal and continuum mechanics solution are solved by the finite element formulation. The mechanical representation is based on a first principles approach using well described constitutive relationships for both the fuel and cladding. The presentation at SMiRT-6 provided the fundamentals used to develop FREY.

FREY is sponsored by the Electric Power Research Institute to provide the utilities the capability to conduct transient licensing calculations and detailed phenomenological fuel element modeling. FREY has been applied in the area of licensing transient calculations, transient experimental programs conducted in the PBF and TREAT facilities, and PCI analysis of both international programs and specific problems related to operating reactors.

A summary of the applications of FREY were presented at SMiRT-7 and SMiRT-9 by Rashid. The transient capabilities were presented against experimental data for such tests as FRF-1 conducted in the TREAT facility and LOCA-3 conducted in the PBF facility to demonstrate the mechanical performance of FREY. At the 5th MMM post-conference seminar Rashid presented the application of a transient failure model for zircaloy cladding with FREY. The state of the art transient failure model was developed to account for the important variables which influence zircaloy failure. Such parameters as heating rate, strain rate, temperature, and fluence were incorporated into a comprehensive model for the prediction of the accumulated cladding damage during a transient. The model was implemented into FREY and tested against other code predictions and experimental data.

FREY represents a state of the art fuel performance code used for the prediction of the thermomechanical behavior of fuel rods during transients and normal operation. The SMiRT conferences have provided for the fuel modeling community to gain exposure to the FREY code through presentation of the basic formulation, applications and phenomenological models.

To summarize, the progress in development, advancement and verification of the three LWR fuel performance codes were reported during the span of the nine

SMiRT conferences and associated MMM Post-Conference Seminars. These codes have been used to analyze the technical issues associated with LWR fuels such as fission gas release, hourglassing and ridging during PCMI, PCI stress corrosion cracking failure, and cladding behavior during simulated LWR transients. The integral codes WAFER, FREY, and FRAP-T, represent various methods and procedures in the thermomechanical modeling of nuclear fuel rods and have demonstrated the success of detailed representation in the prediction of fuel rod behavior. Of the three codes, FREY continues to be an evolving code which incorporates the advances made in our understanding of modeling fuel and cladding behavior.

Correlation Fuel Performance Codes

In some applications, it is desirable to sacrifice detail and accuracy for fast running time. Specific examples are probabilistic analyses, whole-core accident analyses, and production codes to qualify every fuel element in a reactor core. As discussed by Weeks (1973), this has given rise to a number of codes which use correlations based on data and/or parametric versions of more complicated models, rather than the fundamental models themselves. While these codes are useful, they are restricted to interpolation within a specific set of operating conditions.

Table 3 lists some of the "correlations" codes reported at SMiRT conferences. The term "correlations" is used quite loosely here. For example, fast running mechanics modules (e.g., PIN-M, DEFORM) are listed under this category because they do not contain a heat transfer analysis and do not have detailed fuel microstructural models and constitutive equations. However, they play an important role in accident analysis codes. Codes which are better termed as "Correlations codes" are the steady-state mixed-oxide codes NUKER, OLYMPUS, and SWELL and the metallic fuels code BEMOD. These have been replaced by the more detailed mechanistic codes for fuel-element analyses.

Fuel Model Codes

Several codes reported at SMiRT conferences are designed to solve particularly aspects of fuel behavior (e.g., gap conductance, fission product behavior). These models, listed in Table 4, or faster-running parametric versions of these models, have been used in mechanistic fuel performance codes.

DIGRAS (mixed oxide fuel), UNCLE-T-BUBE (mixed carbide and nitride fuel) and UNCSWELL (mixed carbide) were all developed at the University of California at Los Angeles (UCLA) in the USA to describe fission gas release and swelling. Both UNCLE-T-BUBE and UNCSWELL were formally coupled to the UNCLE fuel element modeling code for performance predictions for (U,Pu)C and (U,Pu)N fuel elements with stainless steel, advanced austenitic, and advanced ferritic claddings. The FRAS family of codes (including PFRAS, a fast-running parametric version) were developed at Argonne National Laboratory (ANL) in the USA for (U,Pu)O₂, (U,Pu)C, (U,Pu)N, U-5Fs, and U-Pu-Zr fast reactor fuels. The development of FRAS has paralleled FPIN and has provided guidance on fission gas release and swelling models. The focus has been on fission gas release during transients. A later code, STARS, was developed by this same ANL group to describe the steady state performance of metallic fuels and to provide guidance to the metallic fuels version of FPIN.

The development of the GRASS family of codes has paralleled the LIFE code development of ANL. Initially GRASS was intended as a detailed, mechanistic model for fast-reactor (U,Pu)O₂ fuel. The steady-state and transient version of GRASS (GRASS-SST) was incorporated formally into LIFE-GCFR. Other developmental versions of LIFE have included the faster-running FASTGRASS version for mixed-oxide fuel. However, the major impact of GRASS has been in the area of describing the behavior of solid, gaseous, and volatile fission products for LWR UO₂ fuel under steady-state, load-following, design-basis transients, and accident conditions. It is recognized by the U.S. Nuclear Regulatory Commission as the state-of-the-art code in this area. It has been coupled to the FRAP and SCDAP codes for LWR fuel analysis.

Two gap conductance models, GAPCON and GAPRS, are also mentioned in Table 4. Initially GAPCON was developed for fast-reactor, fuel/cladding gap conductance. However, because of the sensitivity of gap conductance to fuel restructuring, cracking, and deformation, as well as cladding deformation, it evolved into an integrated fuel-element modeling code. The gap conductance model from GAPCON is used in all of the LIFE codes. The integrated code (GAPCON-THERMAL) is used to predict the performance of LWR fuel elements. GAPRS has been used to analyze experiments with (U,Pu)C fuel.

Cladding Performance Codes

In addition to the fuel performance codes and fuel models discussed at SMiRT conferences, there are a whole host of codes dealing with fast-reactor cladding behavior for isolated cladding, cladding/wrap interaction, cladding/coolant interaction, bowing, pin-to-pin contact, probabilistic failure analysis, etc. Table 5 lists some of these codes. However, a detailed discussion of these codes is beyond the scope of this section.

Summary

A number of mechanistic, fast and thermal reactor fuel rod performance codes, which have been reported at SMiRT conferences and Post Conference Seminars, were reviewed. It was found that the three fast reactor codes (LIFE, FRUMP, and URANUS) with the most papers written about them have had significant impact on fast-reactor analysis within their respective countries (USA, UK, and FRG). All three have been used to model the behavior of several fuel types (e.g., mixed-oxide, carbide, and -nitride fuel) and several cladding types (e.g., stainless steel, low-swelling advanced austenitics, and low-swelling advanced ferritics). Of the three, both LIFE and URANUS are still being actively developed. The current focus of the LIFE effort is on developing LIFE-METAL to predict the performance of U-Pu-Zr and U-Fs alloys. URANUS continues to be developed and used for steady-state and transient analysis of both LMFBR and LWR fuel.

Of the large number of thermal reactor codes discussed at the SMiRT conferences, the three codes WAFER, FREY, and FRAP contained the most number of presentations. These codes have played an important part in determining the thermomechanical performance of LWR fuel. The FRAP-T6 code continues to be utilized by both the NRC and INEL to analyze transient fuel behavior. As severe core damage analysis gains momentum, the FRAP-T6 code has been implemented in the SCDAP/RELAP methodologies to provide detailed behavioral predictions. The FREY code continues to evolve as the US utility transient fuel behavior code and is currently being prepared for NRC review as a licensing tool. However, the application of the code spans a wide spectrum of fuel behavior problems and is not limited to transient licensing issues. Fuel failure analyses associated with PCI have been studied with FREY and continued applications regarding cladding defects due to corrosion and manufacturing flaws are being pursued.

Other codes reviewed in less depth than the mechanistic fuel element performance codes are: "correlations" codes, fuel models, cladding models, and general purpose codes. It was found that a number of these models and codes (e.g., GRASS and GAPCON) have been used as stand-alone models and/or subroutines for analyzing performance under steady-state and transient conditions for both LMFBR and LWR fuel.

None of this discussion is intended to imply that SMiRT was instrumental in the development of these codes and models or that development was first reported at SMiRT conferences. Rather the point of this review has been that SMiRT has been an effective avenue for sharing results on an international scale and that developers of major codes respect the high quality of standards of SMiRT enough to contribute to the proceedings.

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Table 1. Summary of Mechanistic Fast Reactor Fuel Performance Codes Reported at SMiRT Conferences and/or at Post Conference Seminars on Mathematical/Mechanical Modeling (MMM) of Reactor Fuel Elements

Code	Country	SMiRT #	Ref.	SS,T,SST ^(a)	Dimen. ^(b)
ACTIVE	Japan	4	MMM	SS	1.5
BEHAVE-2	USA	4	D/O	SS	1.5
		5	D7/1	SST	1.5
COMETHE-III	Belgium	1	C2/1	SS	1.5
		4	MMM	SST	1.5
CYGRO-F	USA	4	D/O	SS	1
FMODEL	USA	1	C3/4	SS	1.5
FPIN	USA	4	MMM	T	1.5
		5	D7/1	T	1.5
		9	Vol. C, p.3	T	1.5
FRUMP	UK	2	D1/8,D2/2	SS	1
		3	D1/12	SST	1
		4	MMM	SST	1.5
		5	D7/1,MMM	SST	1.5
IAFETIN	FRG	6	D4/6	SST	2
		7	C5/3	SST	2
IAMBUS	FRG	4	C4/12	SS	1.5
ISUNE-5	USA	7	C4/7	SS	1
JANE/KRASS	USA	5	D6/4	SST	1.5
LIFE-II	USA	1	C4/1	SS	1.5
-II		2	D2/5	SS	1.5
-III		4	D/O,D1/4	SS	1.5
-UNCLE		4	D/O,MMM	SST	1.5
-4		5	D1/1	SST	1.5
-4CN		5	MMM	SST	1.5
-GCFR		6	D4/4	SS	1.5
-4CN		6	D4/1	SST	1.5
-METAL		9	Vol.C, p.11	SST	1.5
NERFS	UK	2	D2/2	SS	1
NUFROD	USA	5	D7/3	T	1.5
SATURN-1	FRG	1	C4/2	SS	1
		4	D1/6	SS	1
		6	C1/2	SS	1
SLEUTH	UK	4	MMM	SS	1.5

Table 1 Cont'd

Code	Country	SMiRT #	Ref.	SS,T,SST (a)	Dimen. (b)
TEXDIF-P	FRG	6	C1/2	SS	2
URANUS	FRG	4	C1/2,/D1/2	SS	1.5
		5	D3/1	SS	1.5
		6	D4/2	SST	1.5
		7	C4/3,C4/4	SST	1.5
		8	C2/3	SST	1.5

(a) Steady state (SS), transient (T), or steady-state-and-transient (SST). The transient capability includes load-following, transient heat transfer, and time-dependent deformation models.

(b) One-dimensional codes include a radial analysis only. 1.5-D codes also include a provision for dividing the fuel rod into axial segments with different operating conditions. Coupling among axial segments varies from code-to-code.

Table 2. Summary of Thermal Reactor Fuel Performance Codes Report at SMiRT Conferences and/or at Post Conference Seminars on Mathematical/Mechanical Modeling (MMM) of Reactor Fuel Elements

Code	SMiRT #	Ref.	SS,T,SST (a)	Dimen. (b)
WAFER -1	3	D1/4	SS	2
-2	4	D1/3	SS	2
-3	5	D1/5	SS	2
CYGRO -4	4	D1/1	SS	1.5
-5	7	D5/5	SS	1.5
FREY	6	D2/1	SST	2
	7	C2/2	SST	2
-01	9	Vol.D, p.147	SST	2
-01	5	MMM	SST	2
COCCINEL-TITUS	6	D1/6	SS	2
	7	C5/6	SS	2
BACO	5	D1/2	SS	1
	7	C1/7	SS	1
FROST	5	D1/4	SS	1.5
VERDON	5	D1/9	SS	1
SENATOR	6	D1/5	SS	1
SAMURA	6	D2/4, D2/5	SST	1.5
SAFENR	7	C3/6	SS	2
TRANS	4	C2/4	SS	1
	7	C3/5	SS	1
RODEX/RAMPEX	4	D1/5	SS	1
FRAP -S	4	D1/9	SS	2
-T	4	D1/9	SST	2
-T6	7	C3/1, C3/2	SST	2
-T6	8	D3/3	SST	2
ISUNE -2	3	D1/3	SS	1.5
FEMAXI-1V	5	MMM	SS	1.5/2
COMETHE	4	C2/8	SS	1.5
ELESIM	5	C1/3	SS	1
CARO	5	D2/1	SS	1.5

Table 2 Cont'd

Code	SMiRT #	Ref.	SS,T,SST(a)	Dimen. (b)
GAPCON-THERMAL-3	5	D6/3	SST	1.5
FUMAC-84	8	C3/4	SS	2
INTERPIN	9	P53	SST	1.5

(a) Steady state (SS), transient (T), or steady-state-and-transient (SST). The transient capability includes load-following, transient heat transfer, and time-dependent deformation models.

(b) One-dimensional codes include a radial analysis only. 1.5-D codes also include a provision for dividing the fuel rod into axial segments with different operating conditions. Coupling among axial segments varies from code-to-code.

Table 3. Summary of Correlations Fast Reactor Fuel Element Performance
Codes Reported at SMiRT

Codes	Country	SMiRT #	Ref.	Comment
BEMOD	USA	4	D/O	U-5Fs, SS
BREDA	FRG	2	D2/4	Mechanics
-2, IIIa		5	C1/8	module coupled of RADYVAR accident analysis code
DEFORM	USA	1	C3/7	Mechanics
-II		2	C3/1	module coupled to SAS accident analysis code
NUKER	USA	4	D/O	(U,Pu)O ₂ , SS
OLYMPUS	USA	4	D/O	(U,Pu)O ₂ , SS
PINM	Europe	8	C2/6	Mechanics module of EAC accident analysis code
SWELL	USA	4	D/O	(U,Pu)O ₂ , SS

Table 4. Summary of Fast Reactor Fuel Model Codes Reported at SMiRTConferences and/or at Post Conference Seminars on Mathematical/Mechanical Modeling (MMM) of Reactor Fuel Elements

Codes	Country	SMiRT #	Ref.	Comment
DIGRAS	USA	5	C1/2	Fission gas release and swelling
FASTGRASS	USA	6	C1/7	Parametric version of GRASS
FRAS3	USA	9	Vol. C,p.3	Fission gas release and swelling
GAPCON	USA	4	D/O	Gap conductance
GAPRS	FRG	5	D2/2	Gap conductance
GRASS-SST	USA	4	D/O	Fission gas release and swelling
		5	C1/6,MMM	
STARS	USA	9	Vol. C,p.3	Metallic fuels fission gas release and swelling
UNCLE-T-BUBE	USA	5	C1/10	(U,Pu)C/N fission gas release and swelling
UNCSWELL	USA	8	C1/2	(U,Pu)C fission gas release and swelling

Table 5. Summary of Fast Reactor Cladding Performance Codes
Reported at SMiRT Conferences and/or at Post
Conference Seminars on Mathematical/Mechanical
Modeling (MMM) of Reactor Fuel Elements

Codes	Country	SMiRT #	Ref.	Comment
BEACON	JAPAN	5	D5/9	3-D Bowing
BOW	India	5	D5/6	Bowing
CMOT	FRG	3	C3/6	Melting/ Plugging
CRASH	Belgium	1	C2/1	Stress/Strain
DISCO	UK	8	C4/1	Growth/ Distortion
DISKUS	FRG	4	C4/6	Clad/Na
DOT-A	FRG	4	D2/4	Bowing/Pin-Pin
ETIOLE	Japan	7 8 9	C7/5 C4/3 Vol.D,p.25	Bundle/Duct
GOGO	USA	2	C2/4	2-D Stress/ Strain/Failure
IAMBIC	UK	3	D1/7	Cluster Bowing
KASTEN	KfK	4	D2/5	3-D Bundle/Duct
MEOS	UK	2	C3/5	Stress/Strain
PECTCLAD	USA	4	C4/3	Stress/Strain/ Failure
PECS-III	USA	4	C4/9	Probabilistic
PYSURA	France	7	C4/2	Failure
RODSWELL	Italy	7	C2/3	Cladding Balloon Subchannel Block.
RUPTURE	USA	2	D2/6	Failure
SHADOW	Japan	4	D2/4	Pin-Pin
STRAW	USA	5	D5/8	Thermal bowing
THEBO	UK	3	D1/7	Bowing
TRIAMBIC	UK	4	D2/1	Bowing
XEBRA	USA	7	C7/4	Bundle/Duct