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ROLES AND DIRECTION OF MATERIAL SCIENCE IN NUCLEAR TECHNOLOGY

NUCLEAR TECHNOLOGY AND MATERIALS SCIENCE

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Abstract:

Current and expected problems in the materials of nuclear technology are reviewed. In the fuel elements of LWRs, cladding waterside corrosion, secondary hydriding and pellet-cladding interaction may be significant impediments to extended burnup. In the fuel, fission gas release remains a key issue. Materials issues in the structural alloys of the primary system include stress-corrosion cracking of steel, corrosion of steam generator tubing and pressurized thermal shock of the reactor vessel. Prediction of core behavior in severe accidents requires basic data and models for fuel liquefaction, aerosol formation, fission product transport and core-concrete interaction. Materials questions in nuclear waste management and fusion technology are briefly reviewed.

KEYWORDS : nuclear materials, fuel elements, uranium dioxide, cladding, Zircaloy, severe fuel damage.

The term Nuclear Materials generally refers to the structural alloys and ceramic fuel that constitute the nuclear steam supply system of an electrical power plant. The performance requirements for these materials contrast starkly to conventional structures, such as bridges and buildings, or electronic materials for integrated circuits. Materials for nuclear power applications must withstand, for periods of years or decades, a hostile environment of high temperatures and temperature gradients, large and perhaps cyclical stresses, and the intense radiation fields from fast neutrons, gamma rays, and fission fragments. Prior industrial experience offers practically no guidance to the prediction of the response of solids to these extreme conditions. Indeed, the pace of nuclear power development is dictated by the rate of solution of problems involving the materials behavior in the unique environment of a nuclear power system.

Nuclear materials concerns were brought to the fore during the development of the liquid metal fast breeder reactor decades ago. Entirely new materials phenomena, such as void swelling and irradiation creep in metals and temperature-gradient-driven restructuring in ceramics were discovered during this era. The environmental conditions of light water reactors are not as severe as those encountered in the LMFBR, and the nuclear materials problems in LWRs tend to be those that appear after long irradiations.

Fuel Element Materials

A nuclear reactor fuel element is remarkable for its simplicity. Its design is simple : pellets of a ceramic oxide held in a long tube of corrosion-resistant metal. Its materials are simple: the fuel is a homogeneous crystalline form of the refractory oxide UO_2 mixed, perhaps, with other oxides such as those of plutonium or the rare earths. The most popular cladding tube is still Zircaloy, a zirconium-based alloy with small quantities of tin and the transition metals for strength. Like the gasoline engine for automobiles, the design of a fuel element was established decades ago; unlike the internal combustion engine, it has undergone remarkably little modification in 40 years of use. Moreover, the standard fuel element appears to have no competition. With minor variations, it is used in virtually all water-cooled reactors throughout the world.

The simplicity and tenacity of LWR fuel element design does not mean that there have been or are not now problems associated with its functioning. The consequences of the ingrowth of fission products in the fuel due to irradiation and the degradation of the cladding due to physical and chemical changes in the reactor environment are terribly complex processes that have yet to be modeled satisfactorily. Figure 1 illustrates the interrelations of elementary processes which taken together govern fuel element performance. This chart is for liquid metal fast breeder reactor (LMFBR) fuel, but similar if not identical problems arise in water reactors. In one aspect, materials degradation in water is more severe than in the sodium coolant of LMFBRs. Sodium is quite benign towards structural metals as long as its oxygen content is low. Water, on the other hand, always reacts with structural metals in accord with thermodynamic dictates. It is simply a matter of slow kinetics that makes this coolant acceptable.

There have been notable successes in solving some of the early problems in water reactor fuel materials. For example, internal hydriding of Zircaloy cladding by water in the fuel, which led to a rash of fuel failures in the 1970s, was solved by the simple means of adding a mild drying step in the fabrication process. Densification of fuel due to radiation-enhanced removal of residual porosity in the oxide was not as simply eliminated. However, a well-supported research program in the U. S. and in Germany led to a theoretical understanding of the phenomenon and relatively inexpensive measures for correcting it.

Many materials problems in LWR fuel elements have not been as amenable to theoretical understanding and technical solution as those mentioned above. Despite enormous efforts involving in-pile and laboratory experiments and extensive computer code modeling, prediction of fission product release from even normally-operating fuel is still a black art. Many analyses of this phenomenon rely on purely empirical correlations or models with an apparently theoretical basis, such as the Booth model, which is currently used in several major fuel behavior codes. This model replaces the polycrystalline structure of the UO_2 fuel by a collection of independent spheres from which fission product release occurs by classical solid-state diffusion. This model has a dubious physical basis; the size of the "equivalent spheres" can be anywhere between the size of a grain to the size of the fuel pellet and cannot be predicted a priori. The Booth model, however, offers some practical advantages. It requires only one temperature-dependent effective diffusion coefficient to use; it correctly predicts the effect of fission product half life on release rates. However, it predicts that the fractional release of a fission product should be independent of the size of the specimen, a consequence that is patently false. The Booth model cannot adequately deal with the complex microstructure generated in high burnup fuel, in which inter- and intragranular bubbles and resolution of bubbles by fission fragment impact are important. In reality, the Booth model, even when applied to the simple microstructure of fresh fuel, is a basically empirical method of accounting for the simultaneous effects of lattice diffusion in the grains and grain boundary diffusion between the grains. A physically-based model must account for these different phenomena explicitly rather than lump them together into an equivalent radius of a hypothetical sphere.

The phenomenon of grain growth in the fuel greatly influences the retention of fission products. Many fuel behavior codes attempt to account for this effect by allowing all fission products in the solid volume swept by a moving grain boundary to be irreversibly trapped by the grain boundary. This is a perfectly reasonable supposition since the fission gases are essentially insoluble in the fuel and the grain boundary acts as a perfect sink for them. However, these codes then add a diffusional release term derived from the solution of the diffusion equation for stationary boundaries. As in the case of the Booth model, two coupled phenomena are incorrectly decoupled. The correct approach is to solve the diffusion equation in a sphere with a moving boundary, which is a far more complex mathematical problem and for which the first solution has appeared in the literature just last fall.

The phenomenon of pellet-cladding interaction, commonly called PCI, has been the source of numerous fuel failures in operating LWRs. In this process, Zircaloy, normally a tough, ductile metal, is penetrated by a sharp crack of the type normally associated with brittle fracture. Although not fully understood, reactor operators have learned how to avoid PCI by treating the fuel as delicate glassware, avoiding rapid, power changes and "conditioning" fuel by stepwise power increases during startup. These measures have greatly reduced cladding breaches ascribable to PCI. but at the cost of reduced flexibility in operation. Understanding PCI is complicated by the difficulty in modeling stress corrosion cracking in the exotic high-temperature environment inside a reactor fuel element. As is widely recognized from its common occurrence in aqueous systems, stress corrosion cracking requires the confluence of three conditions: a tensile stress state, a susceptible alloy, and a corrosion-assisting species in the environment.

Calculation of stresses in fuel and cladding during operation is hampered by the incredibly complex interrelation of chemical, physical, and mechanical phenomenon that occur during irradiation. Some, such as fuel cracking, are inherently stochastic and difficult to incorporate into deterministic analyses. The identity of the chemical agent responsible for the process is in dispute: iodine is the favored villian, but radiation decomposition of its stable form, cesium iodide, must be invoked to allow it to operate inside a fuel rod.

This brings us to another contradiction in the way that materials in nuclear fuel elements are analyzed. On the one hand, radiolytic decomposition needs to be invoked to explain iodine-assisted PCI. On the other hand, chemical thermodynamic analysis quite adequately predicts the stable chemical and physical states of variety of fission products in the fuel without the added feature of irradiation influence on the thermochemistry.

Another pair of examples illustrates the same dichotomy. Secondary hydriding failure of the cladding is due to entry of coolant into the fuel-cladding gap through a tiny crack. The water turns to steam which proceeds to oxidize the fuel and/or the inside wall of the cladding. As the gas moves from the site of the initial flaw, it becomes richer in hydrogen and eventually reacts with the zirconium to produce brittle hydrides in the metal. A second failure of the tubing occurs at a location far removed from the original crack. Researchers who have examined this process carefully have concluded that its occurrence cannot be explained unless water radiolysis by fission fragments passing through the fuel-cladding gap is accepted as the principal source of hydrogen. Radiolysis also produces hydrogen peroxide which is undoubtedly more reactive towards the metal and the fuel than steam. These reactions have never been studied in controlled laboratory experiments.

On the outside or "waterside" wall of the cladding of PWRs, corrosion proceeds at a rate that is quite adequately predicted by out-of pile experiments in which radiation is absent. Why is it that waterside corrosion is not radiation-assisted whereas internal corrosion is? It may be that fission fragments recoiling from the fuel are present in the fuel-cladding gap but not in the coolant. If fission fragments are so crucial in the phenomena of PCI and secondary hydriding, why do they not exert a comparable influence on the high-temperature equilibria that govern the physical and chemical states of fission products and fuel during irradiation?

There is a world-wide effort to increase the life of the fuel in order to improve fuel-cycle economics. Goals for the end of the decade are fuel burnups of about 50% longer than current practice. The principal fuel materials concerns in the extended burnup programs for both types of LWRs are waterside corrosion of the cladding, increased susceptibility of the cladding to PCI, and fission gas release.

Research and development programs for extended burnup include modifications of both the fuel and the cladding. Slight alterations in the minor-element content of Zircaloy is expected to be sufficient for extended burnup use. For PWR cladding, the tin content of the Zircaloy will be slightly reduced in order to improve oxidation resistance, and niobium may be added to increase resistance to hydriding of the cladding. Advanced fuel may contain small additions of niobium oxide or other fuel-soluble oxides to the uranium dioxide to permit larger grain sizes to be achieved during fuel fabrication. The larger grain size oxide reduces fission gas release during operation as well as decreases grain boundary adhesion, which promotes fuel plasticity. This is shown in Fig. 2, which is the result of work at NFD. Reducing the creep strength of the fuel reduces stress on the cladding when it interacts mechanically with the fuel, thereby ameliorating pellet-cladding interaction effects.

Balance of Plant Materials

It is fair to say that most of the critical materials problems in LWRs involve structures outside of the fuel, or in the balance of plant. The problem areas that are currently of most concern include the integrity of the reactor pressure circuit, particularly of the thick ferritic steel pressure vessel and the nozzles and piping constructed principally of austenitic stainless steel. These components conduct cooling water between the reactor pressure vessel and the turbine in boiling water reactors (BWRs) or the steam generator in pressurized water reactors (PWRs). The concern is that tiny flaws on the inner surfaces, in the presence of impurities in the water, stress, and accumulated radiation embrittlement of the metal, may cause a catastrophic break leading to a loss of coolant accident.

Less of a safety concern but more of a long-term reliability and maintainability question are the corrosion processes engendered by the chemical reactivity of hot(300℃), high-pressure (7-15 MPa) water with the steam generators in PWRs. There appears to be a virtually unlimited number of corrosion modes in PWR steam generators, which in the U. S. have led to considerable reactor downtime for corrective actions, and in some cases, to costly replacement of this $100m component.

Intergranular stress corrosion cracking, called IGSCC, of austenitic stainless steel piping of the primary coolant circuit has been (and still is) a major problem for BWRs in the U. S. Between 1975 and 1987, the reported loss of capacity factor of a reactor plant due to all reactor coolant system problems was about 13% in the U. S. but only 0.4% in Japan. When the problem first arose in the mid seventies, the Japanese response was to develop an IGSCC-resistant steel which is low in carbon but high in nitrogen to maintain strength. This new material is used in new reactors, but the piping in plants already in operation can be treated by a variety of methods aimed at reducing tensile residual stresses in the metal near welds. Typical IGSCC countermeasures require removal of affected welds and replacing them with components fabricated with corrosion-resistant cladding and using the techniques of heat-sink welding and solution heat treatment. An situ treatment that is effective in some cases involves induction heating stress relief of existing welds.

Even if an IGSCC crack penetrates the entire wall thickness of a pipe, it need not lead to catastrophic rupture and trigger a loss-of coolant accident. There is evidence to suggest that such a crack would slowly leak coolant rather than break, the so-called "leak-before-break" concept, and thus be more readily detectable. This concept has been tested experimentally and found to be correct.

A complex interaction of corrosion-fatigue, irradiation embrittlement and thermal shock during a rapid cooling event may cause catastrophic failure of older reactor pressure vessels in PWRs. This problem has threatened a number of older American reactors with derating before the end of their 30-year license period. Because of the size of a reactor pressure vessel, there is little that can be done to rectify this problem in an existing system. However, limited experience in the former USSR suggests that annealing of the entire vessel to remove radiation damage may be useful, if not cost-effective. Alternatively, replacement of the standard linear elastic fracture mechanics analysis of this phenomenon by more realistic elastic-plastic fracture mechanics principles may be sufficient to demonstrate that the problem is purely hypothetical.

Current corrosion research emphasizes the synergistic effects of two simultaneous corrosion mechanisms, such as corrosion-fatigue in conjunction with stress corrosion cracking. Another combined-effect corrosion phenomenon involves stress corrosion cracking assisted by irradiation.

All of the above problems involve progression of cracks in the stressed metal assisted by aggressive chemical species in the water environment. Consequently, considerable attention is paid to the purity of the water and to additives that may mitigate crack propagation. "Water chemistry" is a term used to describe all measures used to control the concentrations of dissolved ions and gases in coolant water with the objective of minimizing corrosion of the containing structures.

Until recently, U. S. BWRs utilized essentially pure, neutral water. The persistent problem of IGSCC of the reactor coolant piping led the addition of molecular hydrogen to the water in order to suppress this form of corrosive attack by reducing the dissolved oxygen content by an order of magnitude. Instead of hydrogen addition to the coolant, oxygen is injected into the coolant in Japanese BWRs. That this seemingly contradictory procedure is beneficial is an illustration of the complexity of water chemistry in a reactor coolant system. The added oxygen serves to passivate metal surfaces and thus reduce corrosion. Water chemistry is also controlled by fabricating piping of corrosion-resistant steels and by using advanced in-line water purification systems and by in situ measurement of coolant pH and electrochemical potential.

The buildup of radioactive corrosion product deposits on the walls of the primary coolant circuit is a source of radiation exposure to personnel and must be minimized. A major source of this radioactivity is cobalt-60 formed by neutron capure in natural cobalt. This species is a minor component in many structural steels and is a major constituent in stellite, which is used as hard facing for bearing surfaces such as valve seats. Stellite is being replaced by high-nickel alloys, which greatly reduces radioactive contamination of the primary circuit.

Corrosion problems extend as far away form the reactor core as the turbine. A persistent problem with turbine rotors is stress corrosion cracking of the joint between the shaft and the disk, a phenomenon called "keyway corrosion". This problem can be eliminated by the use of monoblock rotors, in which the disk-shaft combination is fabricated from a single large forging.

Materials Behavior in Severe Reactor Accidents

The accident at the Three Mile Island reactor in 1979 focussed attention on the need for deeper understanding of the behavior of reactor materials at temperatures far above those of normal operation. One of the main results of this event was the realization that the behavior of fission products and fuel predicted by then-current methodology was incorrect by orders of magnitude, fortunately in the conservative direction. Furthermore, the deficiencies in basic knowledge were found to be mainly chemical in nature.

The complexity of the events that occur during a severe fuel damage accident is illustrated in Fig. 3. The accident is considered to begin with loss of liquid water in the core, which is then blanketed in steam with poor heat removal capacity. Decay heating by the fission products first leads to cladding burst followed at about 1300℃ by rapid oxidation of the cladding. This provides a potent source of additional heat and of hydrogen. Left unchecked, the cladding melts and dissolves fuel at temperatures well below the melting point of pure UO_2. At this stage, massive release of volatile fission products takes place, as well as formation of aerosols that can transport normally nonvolatile radioactive species from the core.

The molten mass falls to the lower reaches of the vessel where it can block emergency cooling water, and may contact the lower head of the pressure vessel. This massive structure can be breached by meltthrough or by creep failure due to the high temperature. The final stage of the accident is spillage of the core debris on the concrete basemat. Interaction of the core debris with concrete can lead to further release of fission products, particularly the rare earths.

To fill the need for information relevant to severe fuel damage accidents, governments and industrial organizations initiated two types of research efforts. The first was the development of detailed computer codes to simulate various accident scenarios in a mechanistic fashion. Chemical reactions, coolant flows, fission product releases, and physical disintegration of the core are modeled, despite the daunting complexity of the process. These analyses are hampered by the loss of regular geometry during core degradation, which requires estimating a phenomenon not unlike wax dripping down a candle. Other codes attempt to model fission product transport in the primary system of the reactor, which consists of hundreds of tons of steel upon which reactive fission products can condense, and because of fission heating, revaporize.

It is generally agreed by all bodies that have evaluated this situation that the database supporting the codes is weak, particularly concerning the physicochemical behavior of the fuel, cladding and structural materials with the released fission products. Basic data and model needs include fission product release from degraded fuel, speciation of the fission products in steam, their interaction with steel, and the formation and transport of aerosols.

In tandem with code development, large integral experiments were either established or existing programs accelerated. These tests, which include intentional destruction of fuel bundles at facilities at the Idaho National Engineering Laboratory, The NSRR experiments at JAERI, and the Marviken aerosol tests in Sweden, were intended either to validate the codes or to provide missing data for them. Significantly, very little support was provided for small-scale laboratory experiments to provide the fundamental data and physical and chemical models that are needed by the large codes.

Materials Concerns in Nuclear Waste Management

Materials issues in the long-term storage of nuclear wastes is a relatively recent addition to the technical areas in the nuclear industry requiring immediate attention. Many of the problems here are not new, nor in other contexts, would be considered particularly difficult to deal with. Corrosion of metallic wasteforms by ground water in the repository is not basically different from a myriad of other corrosion phenomena in modern technology. However, corrosion of the ceramic or glass wasteform proper is a topic unique to nuclear waste management. The question of stability of minerals dissolved in water is an established area of geochemistry. Leaching and speciation of the acitinides in ground water has been extensively studied, although not with reference to the transuranium elements. Diffusion, advection and sorption of aqueous species transported by groundwater through a variety of rock formations has been studied extensively in the past and has general application in hazardous waste management.

However, geologic nuclear waste disposal introduces several unique demands on these established disciplines. Nuclear waste generates heat as well as provides a source of undesirable chemical species. The response of the rock around the hot waste involves potential dryout of the moisture or water movement towards the wasteform. Heating also results in thermal stresses in the medium that affects clack geometry and accelerates physical and chemical processes beyond what would be expected in isothermal environments.

Perhaps the most difficult aspect of the analysis of nuclear waste behavior is the regulatory demand for accurate performance prediction for periods of a minimum of 1000 years. Experimental verifiction of materials behavior for such long times is obviously impossible, and it is difficult to see how even the most rigorous analyses based on the best short-term data can survive the regulatory challenges that are sure to accompany licensing of a waste repository.

Fusion Materials

Despite the fundamental differences in the nuclear reactions that provide energy, fission and fusion exhibit considerable overlap in materials issues. Fusion reactor designs are based on lifetime fast neutron radiation doses to critical components that are not much different from those specified for fast breeder reactors. The fusion reactor analog of the cladding of a fission reactor fuel element is the first wall, a 1-cm thick barrier between the near-vacuum plasma and the high-pressure coolant. At critical locations, the first wall is protected from plasma disruptions and electromagnetic radiations emanating from the plasma by refractory materials such as beryllium, graphite, or tungsten.

The integrity of the first wall is more critical to operation of a fusion reactor than for the cladding of a fission reactor fuel element. A fission power plant can operate safely with a significant fraction of defective fuel elements. The first wall of a fusion reactor, on the other hand, must remain intact over its entire design lifetime. Even a single small breach in the first wall can cause unacceptable contamination of the plasma and quench it by excessive radiation cooling or by loss of vacuum. Immediate corrective action is necessary. This is a serious constraint on the maintainability of a commercial fusion reactor. Moreover, replacing a defective portion of the first wall is a far more complicated matter than simply replacing a defective fuel element during a regular outage in a fission reactor.

The D-T fuel of a fusion reactor is many orders of magnitude less dense than that of a fission reactor, which accounts in part for the absence of a"meltdown" accident scenario and release of massive quantities of radioactivity. Related phenomena due to activation of structural metals are a safety concern in fusion reactor technology. However, the need to maintain essentially absolute integrity of the pressure boundary in a fusion reactor produces new materials concerns. Any process that can produce plasma contamination by the solids that compose the plasma-facing components of the internal structures must be minimized. Leakage of high-energy ions and neutral species from the hot plasma leads to sputtering and blistering of the containing solids and to degradation of the electrical properties of insulators. Melting and vaporization of plasma-facing materials requires use of refractory first-wall and protection structures. Void swelling and radiation creep can compromise the very strict dimensional tolerances and mechanical loading limits of the structures subjected to the flux of 14 MeV neutrons originating from the plasma. The periodic nature of the plasma burn. ash-flush cycle that characterizes most fusion devices means that metal fatigue due to thermal stress is perhaps the major life-limiting failure mechanism.

Although fusion does not produce highly radioactive fission products, the handling of the tritium fuel has required re-examination of the classical field of hydrogen in metals. The complex processes involving tritium in the first wall include permeation, retention, and recycling to the plasma. In situ removal of tritium from the breeder material for production of fuel is a significant materials issue.

Contrary to the well-established fuel element designs of a fission reactor, fuel and structural materials selection, and substantial database for materials performance analysis, the design specifications of fusion reactor components are not yet fixed. Coolants ranging from helium, water, liquid metals to molten salts are under consideration. Potential first wall structures include advanced steels for minimum void swelling, and refractory metals such as molybdenum and vanadium. The breeder material must include lithium, but whether the form is a liquid metal or a solid oxide is not yet fixed.

All in all, the materials issues in fusion reactor designs are much broader in scope and considerably less well understood than those in the more mature fission reactor technology. The lack of a 14-MeV test reactor facility that could permit long-term observation of radiation effects, tritium behavior and mechanical response is a serious impediment to development of this type of nuclear power source. This effort can only succeed by an international project such as the International Tokamak Experimental Reactor (ITER).

Conclusions

From the point of view of new research areas and research activity, the past decade has seen a steady erosion in the interest in and the support for the speciality labeled nuclear materials. In the United States in the 1970s, the field was primarily stimulated by the materials imperatives of the oxide fast breeder program. This program was actively supported by the Department of Energy, with substantial projects at the national laboratories and in the laboratories of the reactor vendors. Anticipation of the construction of a demonstration LMFBR, The Clinch River Breeder Reactor, and the Fast Flux Test Facility kept the field of nuclear materials broadly visible to the nuclear technology community as well as to the general public. Student interest in the challenges of the materials aspects of nuclear reactors was high and active programs existed at several major U. S. universities. It was during this period that seminal advances in the understanding of radiation damage physics, high temperature thermochemistry of the actinides, and mechanical behavior of structural alloys under irradiation were made and phonomena such as void swelling and irradiation creep were discovered.

The breeder reactor represented a large technological endeavor in which materials were an essential component. With the demise of the U. S. Breeder program, nuclear materials as a research area began a decline in which it still finds itself. Nuclear materials specialists found their funding sources disappearing and many moved into other areas of modern technology to apply their expertise. Student interest in nuclear materials diminished. The number of papers in technical journals on nuclear materials dropped. The quantity of papers submitted to regular meetings of the American Nuclear Society declined to an extent that the Materials Science and Technology Division of this organization considered offering sessions only in alternate meetings. The most recent meeting of the ANS offered only a handful of research papers devoted to nuclear materials.

The advent of advanced light water reactors has not stimulated renewed interest in nuclear materials. The fuel is still Zircaloy-clad UO_2. Anticipation of extended burnup of the fuel in light water reactors has stimulated activity in modifications of cladding and fuel to improve long-term performance. However, these concerns, important as they are to fuel vendors and to utilities, are not in themselves sufficient to rejuvenate the field of nuclear materials. They are minor concerns when compared to the major questions in nuclear thermal hydraulics and system design which the new generation of light water reactors will entail.

Materials concerns have moved away from the core of the reactor to the balance of plant, where steam generator corrosion, pipe cracking by environmentally-assisted corrosion and pressure vessel embrittlement remain stubborn problems. However, the further removed from the core, the less are the materials problems of a strictly nuclear type. These technological concerns can be and are being addressed by scientists and engineers who do not have a nuclear orientation.

Fuel damage during severe accidents is an area of great concern at present. However, the utilities and the vendors are loathe to deal with these problems lest it bring new regulations upon them. The Department of Energy does not consider severe fuel damage research as one of its primary missions. This field is the province of the U. S. Nuclear Regulatory Commission, which, because of its shrinking research budget and predilection for large codes, restricts its activities to few national laboratories.

A disturbing world-wide tendency has been the increased reliance on large computer codes to analyze the behavior of materials in both normal operation and severe fuel damage events. The limitations of these codes are often forgotten in the satisfyingly detailed predictions of materials behavior they purport to provide. When these codes are examined in detail, as they have on several occasions, the gaps and uncertainties in the basic materials database becomes apparent. Despite decades of effort, a code that can reliably predict fission product release during normal operation is still an elusive goal. Phenomena unique to severe accidents such as fuel liquefaction and fuel shattering are treated with gross simplifications or simplistic application of empirical models in situations in which they are unlikely to apply. Detailed first principles fuel behavior codes contain so many unknown physical and chemical properties that by judicious selection, can be made to fit any set of data.

What can the nuclear materials community do to reverse this decline? Much of the future is not in our hands, but rather will depend on increased public acceptance and even demand for nuclear power. However, within our capabilities as researchers and engineers, we should concentrate our efforts on topics that are essential to safe nuclear technology. The effort should be on fundamental data and models, not on expensive integral tests or development of large computer codes. Governmental funding sources should be urged to focus support on critical questions, and not dissipate funding on projects such as the gaseous core reactor, the molten salt reactor, or even space reactors. These are frills which cannot be afforded in the present climate. We should not accept a technology that relies on imperfect databases, that utilizes models with no physical basis, or seeks to model phenomena that are inherently unmodelable.

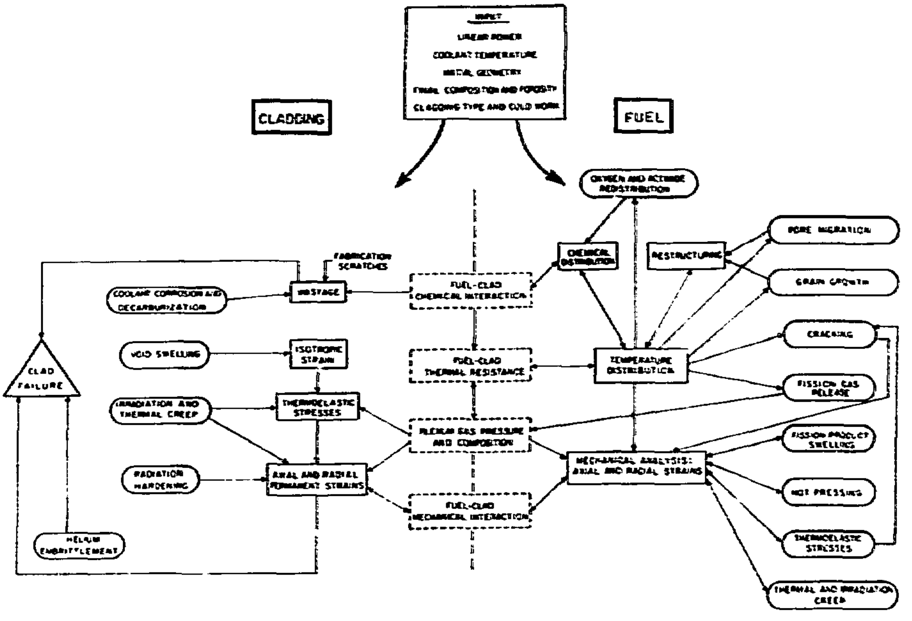

Fig.1 Interrelation of mechanical, physical and chemical processes in an irradiated LMFBR fuel element

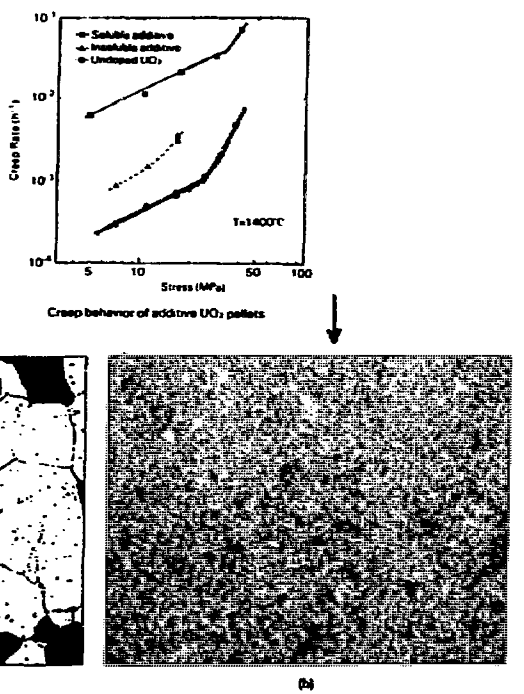

Fig.2 Microstructure of (a) soluble additive-doped and (b) undoped UO_2 pellets. (Courtesy NFD)

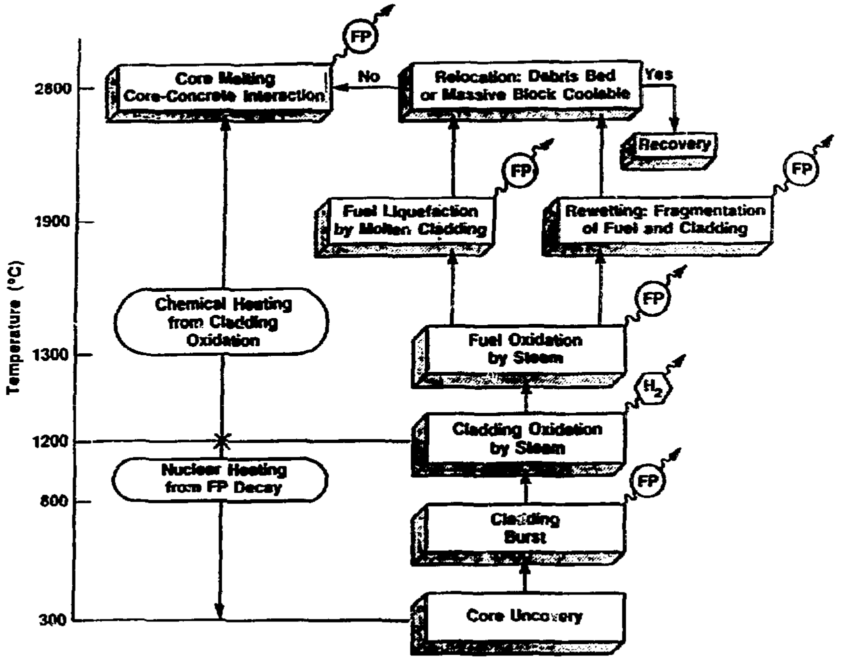

Fig.3 The course of a severe fuel-damage accident