could have been avoided by a different index notation, e.g., $k_{rr}$, $k_{rz}$, etc., instead of $k_{11}$, $k_{13}$, etc.

Although the reviewer has not read everything with a magnifying glass nor checked every equation in this book, nevertheless, during his extensive use of the book, he has not found an error or misprint. This attests to the excellent editing and printing that went into this work. The book is of the highest quality both in content and presentation of the material, and it can be highly recommended as a textbook and as a reference for the practicing engineer and scientist.

It is unavoidable to compare this book with the classic work of Carslaw and Jaeger, *Conduction of Heat in Solids*. To a large extent both books cover the same material. However, there are differences. For one, the mathematical notation in the classic book is antiquated and unfamiliar to today's readers. Second, the book by Carslaw and Jaeger stresses mathematical rigor and elegance; whereas, Özisik adopts a didactic approach in presentation. As a result, if a student or engineer needs to solve a particular heat conduction problem, Özisik shows him the way to accomplish this. On the other hand, the classic work of Carslaw and Jaeger may serve in this context more as a compendium of analytical solutions, and it requires a certain amount of skill to discover and extract the particular one that solves the problem at hand.

As a last point, Özisik has included many problems and exercises, which are an indispensible aid in learning how to solve heat conduction problems. There are no exercises in Carslaw and Jaeger. In the final evaluation, the reviewer decided that he will want to have both books on his shelves.

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**Safety Aspects of Fuel Behaviour in Off-Normal and Accident Conditions**

(Proceedings of a CSNI Specialists Meeting, Finland, September 1-4, 1980)

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| *Publisher* | OECD Nuclear Energy Agency with the cooperation of the International Atomic Energy Agency |
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| *Reviewer* | A. R. Wazzan |

This is the *third* specialists meeting on the "Safety Aspects of Fuel Behavior in Off-Normal and Accident Conditions" to be organized by the Organization for Economic Cooperation and Development Nuclear Energy Agency (Nuclear Safety Division) with the cooperation of the International Atomic Energy Agency. The first was held in France in 1973 and the second in Norway in 1976. The present meeting assumed particular importance because of the Three Mile Island-2 (TMI-2) accident. The meeting was hosted by the Technical Research Centre of Finland.

The proceedings of the conference held in Espoo, Finland, September 1980, provide a snapshot of the current status of knowledge and the kind of analytical and experimental programs in progress with regard to fuel element behavior and associated thermal-hydraulic conditions during transients and accidents in water-cooled power reactors. Many of the accident scenarios considered appear, to a large degree, motivated by the TMI-2 accident. The bulk of the papers was presented by authors working in European establishments, although there was some representation from Japan, Canada, India, Argentina, and the United States. There were four sessions of contributed papers, plus one session that consisted of a review paper on the largely "inferred" fuel element behavior during the TMI-2 accident.

The proceedings provide a series of modest-length papers together with a record of the discussions and a final summary panel. The papers mostly represent reports of current progress rather than long-lived reviews and evaluations on major original contributions.

*Session I*

As a result of the TMI-2 accident, due emphasis is now being placed on the understanding of thermal-hydraulic system effects during transients and small loss-of-coolant accidents (LOCAs). The session provided an opportunity for the interaction between specialists in thermal hydraulics and fuel element behavior. Papers presented in this session dealt mostly with specific typical events. Various core uncovery scenarios were investigated. Computations are performed with many computer codes, NCRCOOL-1, RELAP4, LOOP7, BRUSEK. For example, the RELAP4-MOD6 code is used to determine the response of the primary circuit to a small break transient with system pressure higher than accumulator pressure. Other studies included fuel element response to turbine trip caused by loss of condenser vacuum with and without scram, rupture of a main steam line, double-ended guillotine break in various positions, and a host of cold-leg break sizes. A limited number of experiments are performed in support of various aspects of these transients. Results vary with postulated modes of cooling and clad deformation. It appears much progress in the understanding of fuel element response to severe and modest transients has and can be further attained by closer cooperation between code developers and thermal-hydraulic experimentalists.

*Session II*

The second session included a dozen papers dealing with theoretical and experimental studies in Zircaloy deformation and rupture and aimed at the development of *universal rupture criteria* for Zircaloy cladding in light water reactors (LWRs). Out-of-pile tests involving single-rod geometry and multirod environment demonstrated the importance of temperature, pressure, heating rate, and azimuthal and axial temperature distributions to ballooning and rupture of Zircaloy clad under conditions typical of off-normal operation and accidents, e.g., LOCA, anticipated transient without scram, reactivity-initiated accidents (RIA), etc. Local disturbances, such as spacer grids and control guide tubes, play an important role in the development of the axial and azimuthal temperature variations. The studies encompassed investigation of the behavior of Zircaloy cladding under the combined chemical and mechanical loads that occur during power increases and transients under accident conditions. It is determined that stress corrosion cracking requires a critical iodine concentration, which is a function of the temperature, and depends significantly on the availability of iodine on the cladding surface. In this domain, a paper by Pickman of the United Kingdom provided a review of the papers presented at the "Fifth International Conference on Zirconium in the Nuclear Industry," Boston, August 4–7, 1980. The individual papers and the review suggest that a good understanding exists of the chemical effects of fission products, oxygen, hydrogen, and volatiles, particularly with regard to oxidation kinetics and embrittlement of Zircaloy. Criteria are given for a pressurized water reactor (PWR) such that extended deformation of Zircaloy, which can impede the operation of the emergency core cooling system, would not occur under conditions of the reflood phase of a large-break LOCA. Substantial efforts are under way to adapt a universal strain criterion (the life fraction rule using the Monkman-Grant life fraction relationship, the strain fraction rule, and a burst test criterion) for Zircaloy failure under LOCA conditions. Tests (including temperatures at failure as well as different temperature transients) are under way to determine which of these criteria is the best. A discussion is also given of the need for a statistical and/or probabilistic failure (burst) criterion.

These developments are based, largely, on out-of-pile tests, and, although in-pile single-rod tests in the heavy water research reactor (FR2, Germany) and the power burst facility (PBF, United States) confirm comparable out-of-pile tests, the in-pile behavior of assemblies in power reactors is expected to deviate from out-of-pile observations. For example, a comparison of Westinghouse LOCA burst test results with Oak Ridge National Laboratory and other program results suggests the effect of heatup rate on Zircaloy behavior in nuclear reactors will be different from those found in most simulation tests. Also, most of the current in-reactor fuel behavior tests are essentially separate effects tests. Studies presented here are not conclusive as to whether or not the behavior of multirod systems is predictable from the behavior of the constituent parts. Therefore, appropriate multirod in-reactor tests to determine fuel behavior in power reactors are urged. Such data can be used to check out and calibrate a strain criterion for failure.

### *Session III*

The third session featured 11 papers and dealt with in-reactor fuel damage experiments. Related computations were also presented. Extensive in-pile data from PBF and the Nuclear Safety Research Reactor (NSRR, Japan) are reported. In particular, fuel behavior in the following accident types were considered: Power-Cooling-Mismatch (PCM), RIAs, and LOCAs.

The original Special Power Excursion Reactor Test Program and transient reactor test facility data available before 1974 indicated that 280 cal/g $UO_2$ was a conservative radial average *total energy deposition* limit to insure minimal core damage and maintenance of short- and long-term coolable core. And, although radial average *peak fuel enthalpy* is *less* than the associated radial average total energy deposition (because of heat transfer from the fuel and the relatively large fraction of the total energy, which is due to delayed fissions), the U.S. Nuclear Regulatory Commission (NRC) required that LWRs operated in the United States should be so designed that a worst case RIA should not result in a radial average *fuel enthalpy* greater than 280 cal/g $UO_2$ at any axial location in any fuel rod. In addition, any fuel rod that departs from nucleate boiling in a PWR, or is subjected to a radial average peak fuel enthalpy of 170 cal/g $UO_2$ or above in a boiling water reactor, is assumed to have failed for the purposes of population dose calculations. In retrospect then, the NRC criterion for RIA appears not to be a conservative one. In fact, the present U.S. and Japanese experience indicates that a more appropriate criterion would be that for the worst RIA, the radial average fuel enthalpy should not exceed 230 cal/g $UO_2$. However, since computations, albeit limited, showed that a control rod ejection accident, e.g., in presently operating LWRs results in low radial average fuel enthalpy at the axial flux peak, it was speculated at the meeting that because of "low control rod worth," an RIA would pose no real safety issues. But in any event, PBF and NSRR data indicate the NRC criterion for RIA should be reexamined. The Japanese RIA program considered four different modes of failure: cladding melting, fuel pellet melting, high temperature cladding burst, and low temperature cladding burst. This experience showed that the radial average peak fuel enthalpy should not exceed 170 cal/g $UO_2$ during a reactivity initiated abnormal transient. Also, water-logged fuel rods were found to fail at low energy depositions (100 to 130 cal/g $UO_2$). Some PBF tests suggest that low pressure irradiated LWR fuel rods, when subjected to a radial average peak fuel enthalpy of 140 cal/g $UO_2$ during an RIA, exhibit high strain rates and the fuel rod may fail before departing from nucleate boiling.

The PBF power cooling mismatch tests indicate the NRC criteria appear to be quite conservative and ensure coolable core geometry. Within the limits of these criteria, energetic fuel-coolant interactions and/or fuel rod failure are not likely. In-pile as well as out-of-pile data strongly suggest that coolable core fuel rods are obtained with LOCA conditions.

Some of the papers in this session described future test programs (examples are TRIBULATION, continued PBF, FLASH 01, PHEBUS) that are of great value to safety considerations of LWRs. For example the TRIBULATION program will examine the response of fuel irradiated to 20 to 40 GWd/t to power transients. Some of this fuel will be reirradiated, post transient, to ~70 GWd/t. The program is intended to elucidate the effects of irradiation history as well as fuel rod design parameters on the response of fuel elements to transient and accident conditions.

In-pile data (burst temperature, pressure, and strain) on fuel rod behavior under LOCA conditions in the Karlsruhe FR-2 reactor, using fresh rods and rods irradiated from 2500 to 35 000 MWd/t, were found to be in general agreement with corresponding out-of-pile data. Cladding deformation was found independent of burnup. Tests performed with irradiated fuel rods exhibited pellet fragmentation and relocation of the fragments into the ballooned region. The reviewer must point out however, that these results are limited (applicable) to single rods and *not* to rod bundles. These results could not be generalized or accepted as universal but are rather limited to the test conditions and geometry. For example, some characteristics of these test rods are not typical of PWR rods. There are also indications that test conditions have led, e.g., to a relatively uniform azimuthal cladding temperature distribution, and hence the burst data are not typical of in-pile rod bundle behavior. Also, some FR-2 (German) and FLASH 01 (French) data were not consistent.

Finally, this session also featured papers that presented limited data for fission products, particularly volatiles, released when a fuel rod bursts upon exposure to a LOCA-type accident sequence. However, cladding and LOCA conditions as may occur in a power reactor were not well reproduced in these tests.

*Session IV*

A tentative discussion is given of the core damage at TMI-2 based on a postulated accident progression made by the Nuclear Safety Analysis Center as deduced from a limited amount of recorded data. This analysis suggests that: fuel rod failure began some 30 min after the start of core uncovery; a first core disruption occurred after the core quench resulting from the brief restart of a coolant pump; 40 to 70% of the entire core inventory of volatile fission products was released in <50 min; no reliable data exist on rate or timing of hydrogen production; a new core disruption was noted $\frac{1}{2}$ h after restart of high pressure injection; local temperatures remained below the melting point of $UO_2$, Zircaloy, and cladding fuel eutectic solutions.

To date the consequences of TMI-2 do not appear to have precipitated any plans for major modifications to current U.S. fuel design.

*Session V*

This session on fuel behavior models and codes included ten papers and dealt with the modeling of specific phenomena and with a description of integral codes. These include: RAPTA-1, SPARA, CARATE, COMETHE, TESOA, CANSWEL-2, and MABEL 2A. Two approaches are used: the deterministic model uses a single-rod model with corrections for bundle effects; the second employs statistical methods. Neither approach yet appears quite capable of modeling the whole core.

The code RAPTA-1 (Soviet Union) is designed to model LWR fuel element behavior for a wide range of transients, including fuel element behavior up to the point of cladding failure in a LOCA. The code considers two modes of cladding deformation—tensile stress induced ballooning and compressive stress induced collapse. Computations for transients (LOCA, PCM, etc.) in VVER- and RBMK-type reactors showed the fuel element response is strongly affected by the rod geometry, strength anisotropy, oxygen dissolution, and oxidation of the cladding. Cladding by Zr–1% Nb is found not to be superior to Zircaloys.

The SPARA code (Italy), which is calibrated against in-pile data on center fuel temperatures and fission gas release in the CIRENE elements and diametral plastic strains measured on internally pressurized cladding tubes, is used to analyze fuel element behavior in the CIRENE prototype reactor under LOCA conditions. The code can also treat degenerate accidents, such as channel blockage.

Experiments tend to confirm that, in general, LWR Zircaloy cladding failure is a strong function of the cladding azimuthal temperature and stress distributions. To treat these effects, the code CARATE (Germany) used azimuthal nodalization combined with locally applicable creep and burst models. However, although experiments suggest azimuthal cladding failure strain is a very strong function of the cladding azimuthal temperature distribution, this code shows failure strain is strongly dependent on both the temperature ramp and the azimuthal temperature distribution. Also, the strong influence of oxygen uptake on cladding deformation and burst are not explicitly modeled in the code. The code is targeted for modification in the near future to include axial nodalization, whereby axial extension and the form of ballooning defects can be studied. This should further enhance the current state of the art on the modeling of cladding failure in LWRs during LOCA, RIA, etc. conditions.

A review is given of modifications made to the COMETHE code since its presentation in 1976 at the Committee on the Safety of Nuclear Installations specialists' meeting in Spätind, Nord Torpa, Norway. This important and well benchmarked LWR code now allows for: grain growth, fission gas release models, tunneling effect, axial fuel/cladding interaction, variable fuel plasticity temperature, and Zircaloy cladding failure criteria. The extensive experience with this code is used to identify important characteristics of cladding, fuel pellet, fuel rods, and power history, as relevant to safety-related fuel behavior.

Finally, the CANSWEL-2 code is designed to model the effects of hot spots on the deformation of Zircaloy cladding as pertains to conditions in a PWR LOCA. It treats azimuthal nonuniformities in clad thickness and temperature and models rod interaction with nearest neighbors.

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## Radiation Chemistry of Hydrocarbons

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| *Editor* | G. Földiak |
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The radiation chemistry of hydrocarbons has been the subject of hundreds of scientific articles, but until this year