SAFETY RESEARCH IN A COMPETITIVE WORLD

Jack R. Strosnider, Jr., Deputy Director
Ralph O. Meyer, Senior Technical Advisor
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC, USA

1. Introduction

The U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research, has several important responsibilities in maintaining nuclear safety. These include (1) providing support for licensing decisions, (2) performing anticipatory research, and (3) maintaining centers of technical excellence. According to our approach in the U.S., supporting licensing decisions generally involves confirmatory work to independently check a licensee's submittal and establish greater confidence in the decisions the agency makes. Such confirmatory work on fuel behavior is currently being conducted for several postulated reactor accidents, new fuel cladding materials, spent fuel transportation, and storage in dry casks. Anticipatory research is performed for new reactor concepts and new fuel designs that are being planned by the industry because safety related codes and data may take a long time to develop before licensing decisions are even considered. At the present time, planning is underway for designs that are based on light-water-reactor technology as well as several that are not. Maintaining a center of technical excellence involves a highly competent technical staff, up-to-date analytical tools, and crucial experimental facilities. To maintain these skills and tools, engineers and scientists must remain actively employed in projects that are both relevant and unique to the nuclear industry. In the fuels area, this not only means maintaining computer codes and skills of personnel at NRC headquarters, but it also means maintaining specialized facilities that will be needed from time to time. Such facilities include hot cells and materials test reactors.

2. Cooperation to Reduce Costs

Twenty years ago, NRC had a large research budget. Such budgets were typical in the early years of the nuclear industry and permitted the NRC to support a full range of safety-related research. At that time, within a broad research program we were operating nuclear test facilities like the Loss of Fluid Test Reactor (LOFT) for testing fuel under loss-of-coolant-accident (LOCA) conditions and the Power Burst Facility (PBF) for testing fuel under conditions of reactivity-initiated accidents (RIAs). Operation of those facilities alone cost a large fraction of NRC's total research budget today. Consequently, those and many other large facilities cannot be supported and have been shut down for a long time now. Therefore, in the 1990s, when new safety issues related to LOCA and RIA were identified for high-burnup fuel, a different research paradigm had to be defined. It was no longer possible for the NRC to address all of the issues in its own dedicated programs. Industry and regulatory agencies around the world found themselves in the same situation. Cooperative research with shared funding was a way out of this situation and was pursued aggressively by NRC. One principle was followed to avoid conflicts of interest between the regulators and the regulated. Research cooperation would be limited to collecting data, but would not include interpretation and engineering judgments that support regulatory decisions. These would continue to be done independently.

3. Research to Support Licensing Decisions

From the mid 1990s, the NRC has been addressing a number of high-burnup fuel issues to establish greater confidence in decisions that had been made to permit burnups to 62 GWd/t (average for the peak rod) in operating reactors. Specific technical issues were raised regarding high-burnup fuel operation, and the NRC undertook an aggressive research program to address these issues. Some of this work has been completed, some will be completed in the near future, and some will take awhile longer. These specific issues are discussed in the following paragraphs.

3.1. Criteria and Analysis for Reactivity Accidents

The particular accidents of concern are the rod-ejection accident in a PWR and the rod-drop accident in a BWR. For these postulated accidents, the NRC has historically used one criterion to ensure that fuel rods remain coolable and that fuel particles are not dispersed into the coolant (280 cal/g peak fuel enthalpy) and other criteria to indicate the occurrence of cladding failure (DNB, MCPR, 170 cal/g peak fuel enthalpy) for the purpose of dose calculations. Recent test results have shown that cladding damage in high-burnup Zircaloy fuel occurs in a partially brittle manner, as a result of the mechanical expansion of the pellets, rather than by dryout and overheating of the cladding as addressed by the current criteria.[1] Cladding failure and fuel dispersal thus occur at significantly lower energies than previously thought, and the current criteria need to be revised.

No test program of this kind had been in operation in the U.S. for over 15 years, so the NRC entered into formal agreements with France (Cabri test reactor), Japan (NSRR test reactor), and Russia (IGR and BIGR test reactors) to obtain data from current programs. These programs have now produced a substantial amount of new data, as can be seen in Fig. 1. The staff reached a preliminary conclusion that it is likely that peak fuel enthalpies in LWRs will remain below a new cladding failure threshold, although a quantitative analysis has not yet been completed. Our own calculations and others show that LWR fuel enthalpies will not exceed about 40 cal/g during a worst-case rod ejection (PWR) or rod drop (BWR) accident.[2] Therefore, we continue to believe that this accident will result in little or no fuel damage and that we can wait for a well documented assessment to resolve this issue. Our assessment, which will be based on these new test data, mechanical properties, and analysis, is now scheduled for completion at the end of this year.

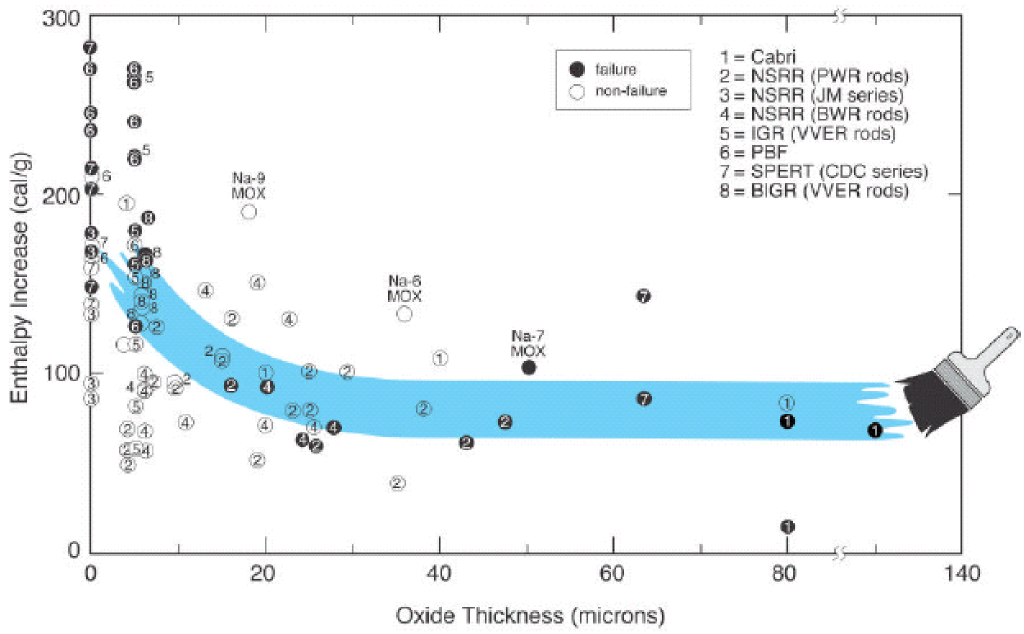

Fig. 1. Data base for reactivity-initiated accidents

3.2. Criteria and Analysis for Loss-of-Coolant Accidents

For these postulated accidents, the NRC uses cladding embrittlement criteria (2200°F peak cladding temperature, 17% cladding oxidation) to ensure that coolable geometry is not lost. Related evaluation models must be used in safety analyses for ballooning, rupture, flow blockage, oxidation (heat generation and embrittlement) and temperature of the cladding to demonstrate that long-term cooling is maintained. The criteria, models, and analyses being used today were based on data from unirradiated cladding, yet the burnup process will likely have an effect. High-burnup fuel rods can accumulate heavy oxide coatings (corrosion) during normal operation. Two consequences of corrosion are reduced metal thickness and absorption of hydrogen into the cladding. These corrosion consequences will contribute to the further reduction of ductile metal thickness and hydrogen absorption during a LOCA transient. It is thus possible that the criteria and models for LOCA analysis will be affected at high burnup, although it is not clear that high-burnup fuel will become limiting. It is also possible that the criteria and models for LOCA analysis will be somewhat different for newer cladding materials, which may have different rates of hydrogen absorption and altered mechanical properties.

In 1998, the NRC issued an information notice on the importance of pre-accident oxidation. Further, in late 1999, the NRC clarified its position that total oxidation thickness for the purpose of comparison with the 17% limit should include the pre-accident oxidation.[3] This clarified definition of total oxidation is now being used by all U.S. licensees and the NRC believes that the inclusion of pre-accident oxidation will accommodate burnup effects on the embrittlement criteria. Burnup may also affect the ballooning process and the oxidation kinetics, but early test results from NRC's research do not indicate large effects. The effects of high burnup on the embrittlement criteria and on the fuel-related analytical models are being addressed in a program at Argonne National Laboratory. The program is sponsored by NRC with cooperation from the Electric Power Research Institute and the U.S. Department of Energy. Framatome and Westinghouse have recently joined a portion of that program that deals with unirradiated M5 and ZIRLO alloys. NRC also participates in related (and coordinated) programs in Norway (Halden) and Russia (Kurchatov), and the staff has access to other related international work.

3.3. Criteria and Analysis for BWR Power Oscillations (ATWS)

NRC regulations contain requirements to prevent or terminate power oscillations that may be associated with anticipated transients without scram (ATWS) in BWRs. This regulation does not contain any criteria to ensure coolable fuel geometry as in the LOCA rule, but it is tacitly assumed that fuel behavior would be benign if the plant conditions of the rule are met. In the most recent industry assessment, however, cladding temperatures were found to exceed 2200°F in some cases and an argument was presented that the 280 cal/g fuel enthalpy limit for reactivity accidents should be applied.[4] However, the staff recently discussed this accident sequence with a group of international fuel experts who were conducting phenomenon identification and ranking tables (PIRTs) for this accident sequence.[5] Based on those discussions, the staff now believes that high-temperature LOCA-like cladding embrittlement would be the most likely cause of loss of coolable geometry for this type of accident, and this embrittlement could occur in a short time after only a few oscillations. Therefore, some of these ATWS sequences may exceed the LOCA cladding temperature limit, and coolable geometry might not be assured. Nevertheless, ATWS events require the assumption of multiple failures and are, therefore, not design-basis events which have to meet certain criteria in a safety analysis. Thus, licensing bases are not affected. However, this research is useful in performing risk assessments that include beyond-design-basis events.

To try to understand these events more thoroughly, a small research effort is being maintained at NRC on this subject. The staff has access to recent repeated-pulse tests in the Japan Atomic Energy Research Institute’s NSRR test reactor and to tests on high-temperature ATWS-like excursions that might be performed in the Halden reactor. Eventually, however, resolution of this issue will have to rely on analysis. This may require a significant upgrade of dryout and rewet models for oxidized fuel cladding. Cooperative work with the Radiation and Nuclear Safety Authority (STUK) in Finland appears to offer some of the improvements that are needed. Toward this end, the Finnish single-channel GENFLO thermal-hydraulics code was coupled with FRAPTRAN and was recently installed on NRC computers. Calculations of peak cladding temperatures and total cladding oxidation are being planned at this time, and results will be compared with the same criteria that are used for LOCA analysis.

3.4. Source Term and Core Melt Progression

During a severe accident, the progression of the accident sequence is strongly dependent on the way molten material develops in the core. Radiological releases, in turn, are determined by the progression of the accident. Estimated releases for a spectrum of severe accidents have been used to develop the recent NUREG-1465 source term.[6] However, that report noted that the source term in the report (particularly gap activity) may not be applicable for fuel irradiated to high burnup levels (in excess of about 40 GWd/t). It is known that at higher burnups the gap inventory will increase, fuel particle behavior will be different, and the isotopics will shift. It is also known that cladding becomes more brittle at higher burnups, potentially resulting in earlier cladding failure and fuel relocation during a severe accident.

Recently, the staff organized a panel of source term experts to evaluate the applicability of the revised source term to reactors with a maximum assembly burnup of 75 GWd/t. The panel considered data from recent international tests, discussed physical phenomena affecting the source term for high burnup fuel, and identified and prioritized source term research.[7] NRC's staff has reviewed the panel's assessment and concluded that the NUREG-1465 source term is generally applicable for high-burnup fuel. Nevertheless, the source term panel members recommended acquiring certain international data on fission product releases for high-burnup fuels. Source term tests on higher burnup fuel specimens are being made by the French Institute for Radiological and Nuclear Safety (IRSN) in its VERCORS and PHEBUS programs and by the Japan Atomic Energy Research Institute in its VEGA program. Assessment of those data will be performed as they become available.

3.5. Transportation and Dry Storage

Nuclide inventory and long-term cladding integrity are two aspects of transportation and dry storage of spent fuel that might be affected by high burnups. The nuclide inventory affects shielding, heat sources, and potential releases of activity. As in reactors, the spent fuel cladding is the first barrier for retention of fission products. The cladding's integrity affects potential releases of fission products and the ability of licensees to safely retrieve the spent fuel for ultimate disposal. Recently, work was initiated to investigate spent-fuel cladding behavior and to measure nuclide inventories in high-burnup fuel. Based on preliminary information from that work, the staff developed interim staff guidance for cask approval. Consequently, reviews are being conducted of transportation and storage casks to approve their use for fuel with burnups up to 68 GWd/t (average for the peak rod), which is sufficient to accommodate currently operating fuel.

Work is underway at Argonne National Laboratory on medium-burnup and high-burnup fuel rods to (a) measure isotopic compositions, (b) measure creep rates under storage conditions, (c) determine mechanical properties in relation to expected accident loads, and (d) examine the general metallography of spent fuel cladding. This work is also being done in cooperation with the U.S. industry and the Department of Energy. Results for fuel rods with Zircaloy-2 and Zircaloy-4 cladding will be completed in 2003, and similar results for advanced cladding alloys will be obtained subsequently to update regulatory guidance for cask reviews

4. Anticipatory Research

Regulatory agencies like NRC will need a research infrastructure of expertise, analytical tools, methods, and facilities to address the technology that forms the basis for the new designs. Seven new designs are on the horizon, not including the Generation-IV designs.

Table 1. Possible Near-term New Reactor Designs

Manufacturer	Reactor
Westinghouse	Advanced PWR (AP-1000)
Westinghouse	International Reactor Innovative and Secure (IRIS)
General Atomic	Gas Turbine Modular Helium Reactor (GT-MHR)
Eskom	Pebble Bed Modular Reactor (PBMR)
General Electric	Simplified Boiling Water Reactor (ESBWR)
Atomic Energy of Canada	Advanced CANDU Reactor (ACR-7000
Framatome	Advanced Boiling Water Reactor (SWR-1000)

For designs that are based on light-water-reactor technology for which the NRC staff has substantial expertise, we are confident that our analytical tools will be ready to support the review process. Nonetheless, the staff will need to expand its analytical models to capture unique designs such as the IRIS and ACR-700 reactors. For the non-LWR designs, the staff will require a longer-term commitment to establish a research infrastructure to support licensing applications.

Critical to the safety of gas-cooled reactors are the coated fuel particles that provide the principal safety barrier and primary containment function against fission product release. Because of the high cost of fuel testing, the NRC is seeking to enter into cooperative agreements with the U.S. Department of Energy and various international organizations, including the European Commission, to assess fuel performance and identify parameters that could affect fuel quality during fabrication. Additionally, the staff is developing plans to implement cooperative agreements on development and validation of analytic tools for assessing fuel particle behavior and fission product release. These tools will aid the NRC staff in understanding margins to failure and radiological source terms.

In the areas of nuclear materials and waste safety, research will focus on out-of-reactor safety issues associated with the front-end of the fuel cycle, as well as spent fuel reactor storage, transport, and disposal at the back-end of the fuel cycle. Differences in the nuclear materials area include criticality safety, radionuclide inventories, decay heat, radiation sources, shielding, and detection. Differences in the nuclear waste area primarily result from differences in long-lived radionuclide inventories and associated source term releases, waste forms (ceramic), higher fuel burnup and enrichment parameters, increased storage volumes of spent fuel and materials (e.g., graphite), and transportation of an increased amount of advanced reactor spent fuel. The staff is pursuing a number of international cooperative opportunities in these areas.

5. Maintaining a Center of Technical Excellence

5.1 Technical Expertise

Maintaining technical capabilities is not easy. For about ten years, during the 1980s and early 1990s, NRC's technical expertise in the fuels area atrophied as attention was shifted from fuel behavior to core damage following the TMI-2 accident. Rebuilding that expertise has been difficult and limited. The emergence of real issues, as discussed in Section 3, has caused us to dedicate staff and resources to achieving resolutions, and this has led to the recovery we have experienced. As these issues are resolved, we will have to substitute interesting, but perhaps lower priority, work to challenge the staff and laboratories and keep their skills sharp. This will become increasingly difficult in our competitive world because funding is not always available for lower priority work. Yet reactor fuels technology is so unique to the nuclear industry that expertise cannot be pulled off the shelf from the outside.

5.2. Fuel Rod & Neutronic Computer Codes for Analysis

NRC uses FRAPCON-3, a steady-state fuel behavior computer code, to audit similar vendor codes that calculate LOCA stored energy, end-of-life rod pressure, gap activity, and perform other licensing analyses. FRAPTRAN, a transient code, is also used by NRC for special calculations and to interpret test results. For reactor power calculations, neither the industry nor the NRC was, as a rule, using 3-D neutronics codes. Postulated accidents like the rod ejection in a PWR, the rod drop in a BWR, and the BWR ATWS power oscillations are very localized in nature and cannot be analyzed well without 3-D kinetics codes. Recently, FRAPTRAN was updated to install the high-burnup thermal models that had been developed for the FRAPCON-3 code, which had been updated earlier.[8] Further, NRC's new 3-D capability with the coupled PARCS code is also available now. Therefore, we believe that the code issues are now largely resolved.

Nevertheless, continued maintenance and improvements in these codes are necessary in order to resolve the other issues described in previous sections. We know from experience that maturity of these codes comes only with heavy use and constant attention to small details such that corrections and improvements are made. In addition, the experimental programs are providing new mechanical properties data for Zircaloy cladding on high-burnup fuel and for the newer niobium-bearing cladding alloys. These data need to be modeled, or correlated, and incorporated into the MATPRO materials properties library. At the present time, fixed schedules for further improvements in FRAPCON-3, FRAPTRAN, and MATPRO have not been established, but such improvements along with additional peer review and software quality assurance for these codes are being planned and carried out. Because of the need to maintain independence in our regulatory judgments, which often involve code audits, NRC's code development work is not done in the same cooperative manner as experimental data collection.

5.3. Facilities

At the present time, important fuel research is going on in facilities around the world, and these are often jointly funded cooperative programs. Some of these noteworthy facilities are the materials test reactor at Halden, Norway; pulse reactors in France, Japan, and Russia; hot cells at Argonne, Illinois, and Studsvik, Sweden; and others. Many facilities have been closed, and many of those still in operation are ageing and require significant upkeep. Obtaining funding to operate all these facilities has become increasingly difficult.

We can probably conclude that, as long as there is nuclear power, there will be a need to examine some irradiated fuel. This requires the existence of good functioning hot cells, which have to be located around the world in regions with nuclear power. We can also probably conclude that there will be a continuing need for a materials test reactor to test new fuel materials and new operating environments. The situation is somewhat different for the more specialized facilities. In the near future, a group of experts under the auspices of OECD will address the issue of test facilities and the need to preserve them.

6. Summary.

Today, in the fuels area, the NRC operates only one indigenous experimental program, which is testing high-burnup fuel rods under LOCA conditions and studying the behavior of irradiated cladding. This program is being run with cooperation from the Electric Power Research Institute, the U.S. Department of Energy, Framatome, and Westinghouse. Testing of fuel under RIA conditions is no longer being done in the U.S., and the NRC formally participates in pulse reactor test programs in France, Japan, and Russia. The NRC is also a long-time member of the Halden Project, where we get data from a materials test reactor. While the NRC does have cooperative arrangements in fuel analysis with the regulatory agencies in Finland and Spain, we maintain our own independent analytical capabilities, which provide us with the uncompromised ability that we need to interpret data, apply engineering judgment, and make regulatory decisions. Together, these programs provide confirmatory support for licensing decisions and they are maintaining skills and facilities that will be needed in the future of safety regulation for the industry.

7. References

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