NRC ACTIVITIES RELATED TO HIGH BURNUP, NEW CLADDING TYPES, AND MIXED-OXIDE FUEL

Ralph O. Meyer, U.S. Nuclear Regulatory Commission, Washington, DC 20555, rom@nrc.gov

ABSTRACT

Many NRC decisions about high-burnup fuel and new cladding types were made before there was an understanding of changes in cladding oxidation and pellet microstructures that could affect regulatory criteria. Some of these early decisions are reviewed along with the NRC's high-burnup program plan that identified confirmatory research that would address these effects. Issues are also identified that are likely to arise in licensing the use of mixed-oxide fuel from excess weapons material. The ensuing research program is outlined that will lead to the resolution of these issues and to the reduction of some unnecessary burdens on the industry.

INTRODUCTION

By the mid 1980s, NRC had de-emphasized its work on reactor fuels and shifted most of those resources to severe accidents in the wake of the accident at Three Mile Island. Yet this was a time when the industry was moving into new operating regimes with fuel and developing new fuel designs. It is not surprising, then, that we look back on regulatory activities of that period and find some areas where confirmatory work is needed. In this presentation, some of those activities will be reviewed along with the agency's program plan that put the confirmatory work in motion. In a similar way, NRC's current and near-term future activities related to reactor fuel will also be reviewed.

HIGH-BURNUP FUEL APPROVED BY NRC

In response to a request from NRC, all five fuel vendors in the U.S. submitted licensing topical reports requesting approval to apply their safety analysis methods to high-burnup fuel. These reports were given NRC approval in 1985 and 1986[1-5] for various burnup levels up to 60 GWd/t average burnup for the peak rod, and the regulatory burnup limit was later raised to 62 GWd/t. It is interesting to look at some of the findings of those early NRC reviews in the area of the postulated accidents, which are routinely analyzed to ensure that core damage does not occur.

□ The embrittlement criteria used for analyzing a loss-of-coolant accident (17% cladding oxidation and 2200°F peak cladding temperature[6]) were concluded to be unaffected by extended burnup operation without a discussion of a basis but with the thought in mind that beginning-of-life conditions would be bounding.

□ The fuel enthalpy limit for reactivity accidents (280 cal/g[7]) was said to be acceptable for extended burnup application although at that time there were only four related tests above 6 GWd/t (two around 13 GWd/t and two around 32 GWd/t).

□ Cladding ballooning models used for LOCA analysis[8] with fresh fuel were believed to be more conservative at high burnup because it was thought that cladding oxidation at extended burnup levels may result in reduced cladding strains.

□ Fuel assembly structural analysis[9] using properties for unirradiated material was thought to apply for seismic and LOCA loads at high burnup because yield strength would increase with irradiation and the corresponding decrease in cladding ductility was believed to be small or negligible. This was based on the conclusion that the ductility decreases that occur with increasing fluence will saturate around 8-12 GWd/t burnup.

□ Steady-state fuel rod codes, which are used by the vendors for LOCA input (and other calculations related to normal operation), were updated to include burnup-dependent properties such as stored energy and fission gas release.

□ Source terms, which are used to assess the radiological release to the environment from an accident with core melting, were not discussed.

During the 15 years since those approvals were given, we have learned that break-away cladding oxidation can approach the 17% LOCA limit during normal operation in this burnup range and that the associated hydride accumulation can embrittle underlying metal. High concentrations of burnable poisons, which are needed to achieve high burnups, delay the occurrence of peak power such that the worst case is no longer at beginning of life. Therefore, oxidation and hydrogen pickup raise questions about the adequacy of the LOCA embrittlement criteria at higher burnups. Further, more recent testing under reactivity accident conditions of fuel with burnups around 60 GWd/t has demonstrated that cladding failure and fuel dispersal can occur well below the 280 cal/g level, thus calling into question the fuel enthalpy criteria being used for these events.[10-11]

Changes in pellet properties at high burnup, along with the rim structure that develops, also alter the thermal performance of fuel rods. Radial power profiles are altered and centerline temperature is increased, thus increasing the important stored energy input for LOCA analysis. Changes such as the reduction in the delayed neutron fraction, which results from the buildup of plutonium isotopes, also affect neutron kinetics codes. These changes necessitated the modification of some fuel rod and kinetics codes used in licensing analysis.

NRC PROGRAM PLAN FOR HIGH-BURNUP FUEL

Recognizing these changes in fuel behavior at high burnup, the NRC developed an Agency Program Plan for High-Burnup Fuel in 1998.[12] The program plan identified nine issues related to high-burnup fuel.

1. Cladding Integrity and Fuel Design Limits
2. Control Rod Insertion Problems
3. Criteria and Analysis for Reactivity Accidents
4. Criteria and Analysis for Loss-of-Coolant Accidents
5. Criteria and Analysis for BWR Power Oscillations
6. Fuel Rod & Neutronic Computer Codes for Analysis
7. Source Term and Core Melt Progression
8. Transportation and Dry Storage
9. High Enrichments (>5%)

It was concluded that the first two issues were being satisfactorily addressed by industry activities and that the last two were related to future actions. Attention was thus focused on the remaining five issues (Nos. 3-7), the postulated accidents, which are routinely analyzed to ensure that core damage does not occur.

The regulatory criteria for reactivity accidents, specifically the PWR rod-ejection accident, were found to be non-conservative in light of the test data from France and Japan. Nevertheless, it was concluded that no action was necessary pending the outcome of confirmatory research because of the low probability of the accident and because of generic calculations that implied that energy inputs would remain below the relevant test data failure levels.[13]

The regulatory criteria for loss-of-coolant accidents would be affected by enhanced cladding oxidation and related effects (hydriding), according to the program plan, but current criteria

are conservative for fresh fuel and may prove to be adequate at high burnup, provided that the oxide accumulation prior to the accident is taken into account. Thus no action was thought to be needed unless the confirmatory research demonstrated a need for change.

The same 280 cal/g enthalpy criterion that was used for reactivity accidents was also being used for BWR power oscillations. Based on the test results for the reactivity accidents, the conservatism in this application was also questioned. However, it was believed that the power oscillations would be slower and probably less damaging than the sharp pulses in the tests and that this did not necessarily imply unacceptable fuel damage for the BWR power oscillations. Again, it was concluded that there was no need to change the approved burnup levels unless the confirmatory research demonstrated a need for change. It should be noted that attention was given to BWR power oscillations that are related to anticipated transients without scram (ATWS) rather than to the BWR rod drop accident because the perceived risk from the oscillations was greater. This switch in emphasis for these reactivity accidents was a direct result of using risk information to set priorities.

The need for modifications to NRC's steady-state fuel rod code (FRAPCON) was recognized earlier and this had been rectified by the time the program plan was issued.[14] Similar changes were in progress for NRC's transient fuel rod code (FRAPTRAN), and needed modifications were also underway in Purdue University's PARCS kinetics code, which the NRC is using.

The staff then argued that it was unlikely that high burnup would have a significant effect on source terms or core melt progression. The argument was based on the facts that there would be less unoxidized metal in the core, that gap activity which might be increased is only a small fraction of a source term, that fragments from small grain sizes would not get into the atmosphere as aerosols, and that the release fractions would not be affected by the isotope shifts. Upon review, the Advisory Committee on Reactor Safeguards did not agree with the staff on this conclusion,[15] and consequently the NRC decided to pursue an understanding of burnup effects on source terms in its research program.

Finally, an important licensing and research strategy was described in the program plan and was stated as follows. In the past, the NRC has always performed the research needed to define regulatory criteria, and the industry has performed research to develop methods of demonstrating compliance with those criteria. In recent years, NRC's research budget has declined to a level that the NRC can no longer support such research. Thus, if the industry wants further burnup extensions, it will have to develop a data base for revised (or confirmed) regulatory criteria. The staff will make it clear to the industry that such research must be non-proprietary, to ensure that resulting criteria are fully scrutable, and the NRC staff must have full access to those research programs. If NRC resources are available, the NRC will actively participate in those research programs; however, the industry will be expected to take the lead in this work.

NEW CLADDING TYPES APPROVED BY NRC

In 1990, Westinghouse submitted a licensing topical report on their Vantage+ fuel assembly in which they introduced ZIRLO cladding. ZIRLO is a zirconium alloy that contains less tin than Zircaloy and has some niobium added to achieve improvements in corrosion resistance and dimensional stability under irradiation. The compositions of ZIRLO, Zircaloy, and several other alloys used in reactor fuel designs are shown in Table 1. ZIRLO was approved by the NRC for burnups to 60 GWd/t about a year later.[16]

Table 1. Composition of zirconium alloys used in reactor fuel design

Element	Zircaloy-4	ZIRLO[17]	E635[18]	M5[19]	E110[18]
Nb (wt%)	--	0.9-1.3	0.95-1.05	0.8-1.2	0.95-1.05
Sn (wt%)	1.2-1.7	0.9-1.2	1.20-1.30	--	--
Fe (wt%)	0.18-0.24	0.1	0.34-0.40	0.015-0.06	0.006-0.012
Cr (wt%)	0.07-0.13	--	--	--	--
Zr	Balance	Balance	Balance	Balance	Balance

Note: Oxygen (~0.125 wt%) is also considered an alloying element in these alloys.

If one looks at the issues that were discussed above in connection with the high-burnup approvals, one finds the following for the ZIRLO approval.

- The Zircaloy embrittlement criteria used for analyzing a loss-of-coolant accident (17% cladding oxidation and 2200°F peak cladding temperature) were concluded in NRC's safety evaluation report to be applicable to ZIRLO, although there was no discussion of a basis for the conclusion. The Westinghouse report did not comment on the applicability of these criteria to ZIRLO. About a year after the approval of the Westinghouse report, the regulations were amended to include ZIRLO along with Zircaloy, and the Federal Register notice[20] addressing this amendment leads back to the NRC's safety evaluation[16] for the technical basis.

- The fuel enthalpy limit (280 cal/g) for reactivity accidents was not impacted by the fuel design changes according to NRC's safety evaluation report, although no reasons were presented. There was also no discussion in the Westinghouse report of the applicability of these criteria to ZIRLO.

- Cladding ballooning and rupture models that are used for LOCA analysis were modified for ZIRLO based on single-rod burst tests performed by Westinghouse with unirradiated ZIRLO tubes. Beginning-of-life conditions were thought to remain limiting and no effects of irradiation were discussed.

- Fuel assembly structural analysis for seismic and LOCA loads was concluded to be the same as for similar Westinghouse Zircaloy fuel. While the basis for this conclusion was not described explicitly, irradiation hardening and ductility of ZIRLO were assumed to be similar to Zircaloy in other parts of the safety evaluation report.

- The Westinghouse steady-state fuel rod code, PAD, which had already been upgraded for high-burnup operation, was further modified to account for the different creep behavior of ZIRLO. Comparisons were made with data taken from lead test assemblies to validate the code modification.

- Source terms, which should not be affected by the cladding alloy, were not discussed.

In 1997, Framatome Cogema Fuels submitted a licensing topical report on their cladding and structural material, M5. M5 is a zirconium alloy that contains niobium, but no tin (see Table 1). The NRC safety evaluation of this report and its conclusions[21] were quite similar to those for ZIRLO with the following exceptions.

- The Zircaloy embrittlement criteria used for analyzing a loss-of-coolant accident (17% cladding oxidation and 2200°F peak cladding temperature) were concluded to be applicable to M5 based on quench tests with M5 cladding. Although the test specimens were unirradiated and were not subjected to ballooning deformation, the criteria were said to be acceptable up to currently approved burnup levels (i.e., 62 GWd/t).

- The safety evaluation report noted that recent testing on the fuel enthalpy limit (280 cal/g) for reactivity accidents has indicated that fuel expulsion and fuel failure may occur before the 280 cal/g limit and the onset of departure from nucleate boiling (DNB), respectively. It was concluded that further testing and evaluation are needed and the limits may decrease in the future, but the current limits will continue to be accepted. It was also stated that there is little impact on the use of M5 cladding on fuel expulsion and failure (compared to the use of Zircaloy-4) as long as the cladding remains ductile, although there was no further discussion of this point.

- Fuel assembly structural analysis methodology for seismic and LOCA loads was unchanged, but would require strength values for M5, should M5 be used for guide tubes or thimble tubes.

MIXED-OXIDE FUEL IN THE U.S.

Mixed-oxide (MOX) fuel, containing initial quantities of plutonium in addition to uranium, is not currently used in the U.S. although significant testing and irradiation in power reactors had been done in earlier years. Just after completing a generic environmental statement on mixed oxides (GESMO[22]) in 1976, the U.S. abandoned plans for recycling plutonium from spent reactor fuel. Mixed-oxide fuel is, of course, used in Europe and its use in Japan is planned.

However, in September 1998, the U.S. and Russia issued a joint statement of principles wherein each pledged to remove 50 metric tons of plutonium from their weapons program.The U.S. Department of Energy, which is responsible for disposing of the U.S. material, is planning to use two methods to dispose of it: (1) reconstituting the plutonium into mixed-oxide fuel rods and burning it in current light water reactors, and (2) immobilizing the plutonium in glass logs. Fuel fabrication for the former method is to take place at the Savannah River site in South Carolina, and burning of the fuel would take place in nearby Westinghouse-type PWRs (North Anna 1 & 2, McGuire 1 & 2, and Catawba 1 & 2).

In 1999, the NRC staff informed their Commissioners of these plans to introduce mixed-oxide fuel in licensed facilities and summarized the technical issues that need to be considered.[23]

- Control rod and absorber worths are reduced compared with uranium, and this can reduce reactor shut-down margin.

- The coolant void coefficient of reactivity is different for MOX fuel than for UO_2 fuel.

- The neutron energy spectrum is harder, and higher neutron energies could enhance irradiation damage in the reactor pressure vessel.

- Reactor kinetics computer codes used in safety analyses need to be modified for MOX fuel to account for the larger cross sections, changes in the energy dependence of the cross sections, smaller delayed neutron fraction, increased energy per fission, and other basic neutronic parameters that are altered by the plutonium isotopes in weapons-grade MOX fuel.

- For a given fuel rod power, MOX fuel rods operate with higher centerline temperatures compared with UO_2.

- Inhomogeneous plutonium clusters in MOX fuel may affect fuel behavior during reactivity accidents, especially at high burnups.

- Plutonium from nuclear weapons contains some gallium, and its effects on fuel and cladding behavior have not yet been fully assessed.

- Fuel rod computer codes used in safety analyses will have to be modified for MOX fuel to account for altered physical properties such as thermal conductivity, thermal expansion, and creep rates.

- Fission product and actinide concentrations are somewhat different in MOX fuel and UO_2 fuel, and this could affect the consequences of a severe accident.

- Criticality analysis will be altered for MOX fuel fabrication, transportation, and storage because of the different isotopes and material forms.

- Material control and accountability measures will be more difficult for MOX fuel because of the need for remote handling during fabrication and secure transportation of weapons-usable material.

- For cooling times of one year or greater, the decay heat is higher for MOX fuel because of its larger inventory of actinides.

The NRC staff observed that experience in Europe with these technical issues suggests that they can be resolved in the U.S. as well. Nevertheless, some of the technical issues will be unique because of the difference between weapons-grade and reactor-grade mixed oxides, and at least one of the issues concerning plutonium inhomogeneities and their effects on reactivity accidents has not been resolved in Europe. At the present time, the NRC staff are preparing a detailed program plan to address these issues in anticipation of mixed-oxide fuel loading in 2007.

NRC'S FUEL BEHAVIOR RESEARCH PROGRAM

NRC's current fuel behavior research program is focused on the behavior of high-burnup fuel during two types of design-basis accidents, and the program will be expanded as necessary to cover new cladding types and mixed-oxide fuel. The accident conditions being addressed are for the loss-of-coolant accidents in PWRs and BWRs, the rod-ejection accident in PWRs, and the ATWS-related power oscillations in BWRs. The program also includes code developments on two fuel rod codes and one neutron kinetics code that are needed to analyze these accidents.

NRC's research budgets are a small fraction of what they were in the 1970s and early 1980s, so some areas of coverage are obtained by participation in international programs rather than from indigenous programs. The mix of home-grown and international programs that comprise NRC's research effort are shown in the following list.

a. Argonne National Laboratory (NRC program): hot cell LOCA tests of fuel rods and mechanical properties of cladding

b. Pacific Northwest National Laboratory (NRC program): steady-state and transient fuel rod codes and analysis

c. Brookhaven National Laboratory (NRC program): neutron kinetic codes and analysis of plant transients

d. Halden Reactor Project (Norway): tests of fuel rods in steady state and mild transients

e. Cabri Test Reactor (France): reactivity accident tests of fuel rods and related programs

f. Nuclear Safety Research Reactor (Japan): reactivity accident tests of fuel rods and related programs

g. Impulse Graphite Reactor (Russia): reactivity accident tests of fuel rods and related programs

h. Grenoble Research Center (France): high temperature fission product release tests

This set of research programs addresses the issues as described in the NRC program plan discussed above. It also corresponds to a risk-informed, performance-based approach by focusing on events that can lead to core damage rather than addressing routine operational problems. To be risk significant, an event must have the potential to cause fuel melting because only then could there be a large fission product release and

significant consequences (hence risk). While there may be many pathways leading to such events, there are only two ways to melt fuel. One is to lose the coolant and the other is to get excessive power in the fuel. Selected design-basis accidents are postulated to serve as bounding examples of these kinds of events, and fuel damage limits are used to ensure that coolable core geometry is not lost, thus avoiding significant consequences. These selected design-basis accidents are the ones that were identified in the NRC program plan and are being addressed in the research program.

On the other hand, there are a large number of regulatory criteria that address normal operation, and these are the so-called specified acceptable fuel design limits or SAFDLs. These fuel design limits arise from one of the General Design Criteria,[24] whose purpose is to ensure integrity of the first fission product barrier -- the fuel cladding -- during normal operation, including the effects of anticipated operational occurrences. These fuel design limits cover such properties as design stress, uniform strain, strain fatigue, internal gas pressure, and overheating of the cladding related to critical heat flux. These regulatory criteria are not being investigated in the NRC research program and are left for the industry to address because of their lower risk significance.

Recently, as part of an effort to make risk-informed changes to the regulations, a proposal was made to eliminate most of the specified acceptable fuel design limits (except for those related to critical heat flux) without changing the intention of the General Design Criteria to ensure cladding integrity. The basis for this proposal was that (a) low cladding failure rates of 1-2 rods per core are being maintained by aggressive industry action and clearly meet the intention of the regulation, and (b) few if any of the recent failure causes bear a relationship to the specified acceptable fuel design limits such that those limits are no longer contributing to this record of compliance. This proposal is still under consideration.

CONCLUSION

In conclusion, the NRC is addressing the important issues related to high burnups, new cladding alloys, and mixed oxides, and is performing research that will lead to the resolution of these issues and reduce unnecessary burdens on the industry.

REFERENCES

1. C. O. Thomas, "Acceptance for Referencing of Licensing Topical Report NEDE-22148-P, 'Extended Burnup Evaluation Methodology,'" NRC letter to J. S. Charnley (General Electric), August 13, 1985.

2. E. J. Butcher, (in reference to Combustion Engineering's report CENPD-269-P), NRC letter to A. E. Lundvall (Baltimore Gas & Electric), October 10, 1985.

3. C. O. Thomas, "Acceptance for Referencing of Licensing Topical Report WCAP-10125 (P), 'Extended Burnup Evaluation of Westinghouse Fuel,'" NRC letter to E. P. Rahe (Westinghouse), October 11, 1985.

4. H. N. Berkow, "Acceptance for Referencing of Licensing Topical Report BAW-10153P, 'Extended Burnup Evaluation,'" NRC letter to J. H. Taylor (Babcock & Wilcox), December 3, 1985.

5. C. E. Rossi, "Acceptance for Referencing of Licensing Topical Report XN-NF-82-06 (P) Rev. 1, 'Qualification of Exxon Nuclear Fuel for Extended Burnup,'" NRC letter to G. N. Ward (Exxon), July 18, 1986.

6. *Code of Federal Regulations*, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," 10 CFR Part 50.46.

7. *Regulatory Guide 1.77*, "Assumptions used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," U. S. Nuclear Regulatory Commission.

8. *Code of Federal Regulations*, "ECCS Evaluation Models," Appendix K to 10 CFR Part 50.

9. *Standard Review Plan*, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," Appendix A to Section 4.2, U. S. Nuclear Regulatory Commission.

10. J. Papin, M. Balourdet, F. Lemoine, F. Lamare, J. M. Frizonnet, and F. Schmitz, "French Studies on High-Burnup Fuel Transient Behavior Under RIA Conditions," *Nuclear Safety*, Vol. 37, October-December 1996, p. 289-327.

11. T. Fuketa, F. Nagase, K. Ishijima, and T. Fujishiro, "NSRR/RIA Experiments with High-Burnup PWR Fuels," *Nuclear Safety*, Vol. 37, October-December 1996, p. 328-342.

12. L. J. Callan, "Agency Program Plan for High-Burnup Fuel," NRC memorandum to the Commissioners, July 6, 1998.

13. *Proceedings of the CSNI Specialist Meeting*, "Transient Behaviour of High Burnup Fuel," Cadarache, France, September 12-14, 1995, OCDE/GD(96)197.

14. D. D. Lanning, C. E. Beyer, C. L. Painter, G. A. Berna, K. L. Davis, "FRAPCON-3," U. S. Nuclear Regulatory Commission, NUREG/CR-6534, December 1997.

15. R. L. Seale, "NRC Reactor Fuels Research Program," ACRS letter to Chairman S. A. Jackson (NRC), June 15, 1998.

16. A. C. Thadani, "Acceptance for Referencing of Topical Report WCAP-12610 'VANTAGE+ Fuel Assembly Reference Core Report,'" NRC letter to S. R. Tritch (Westinghouse), July 1, 1991.

17. R. Comstock, G. Schoenberger, G. Sobol, "Influence of Processing Variables and Alloy Chemistry on the Corrosion Behavior of ZIRLO Nuclear Fuel Cladding," *Zirconium in the Nuclear Industry: Eleventh International Symposium*, ASTM STP-1295, November 1996, p. 712.

18. A. Nikulina et al., "Zirconium Alloy E635 as a Material for Fuel Rod Cladding and Other Components of VVER and RBMK Cores," *Zirconium in the Nuclear Industry: Eleventh International Symposium*, ASTM STP-1295, November 1996, p. 787.

19. J-P. Mardon, D. Charquet, J. Senevat "Influence of Composition and Process on Out-of-Pile and In-Pile Properties of M5 Alloy," *Zirconium in the Nuclear Industry: Twelfth International Symposium*, ASTM, to be published.

20. *Federal Register*, Nuclear Regulatory Commission, Reducing the Regulatory Burden on Nuclear Licensees, Regulatory Analysis, Vol. 57, No. 169, August 31, 1992, p. 39356.

21. S. A. Richards, “Acceptance for Referencing of FCF Topical Report BAW-10227P: ‘Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel,’” NRC letter to T. A. Coleman (Framatome Cogema Fuels), December 14, 1999.

22. “Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors,” NUREG-0002, August 1976.

23. W. D. Travers, “Mixed-Oxide Fuel Use in Commercial Light Water Reactors,” NRC memorandum to the Commissioners, April 14, 1999.

24. *Code of Federal Regulations*, “General Design Criteria for Nuclear Power Plants,” Criterion 10, Reactor Design, Appendix A to 10 CFR Part 50.